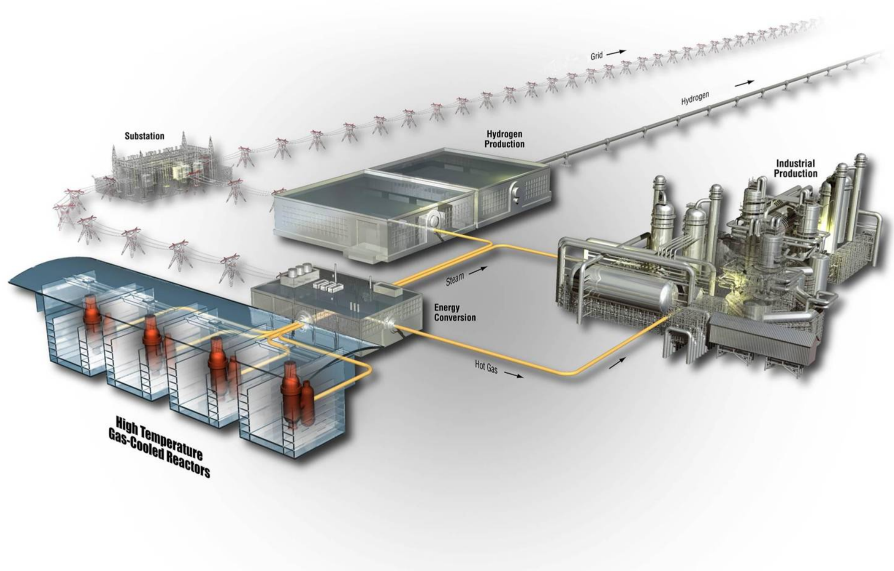
INL/EXT-12-26895

COL Application Content Guide for HTGRs: Revision to RG 1.206, Part I – Status Report

August 2012



The INL is a
U.S. Department of Energy
National Laboratory
operated by
Battelle Energy Alliance

Idaho National Laboratory

INL/EXT-12-26895

COL Application Content Guide for HTGRs: Revision to RG 1.206, Part I – Status Report

August 2012

Idaho National Laboratory
VHTR Program
Idaho Falls, Idaho 83415

http://www.inl.gov

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

VHTR Program

COL Application Content Guide for HTGRs: Revision to RG 1.206, Part I – Status Report

INL/EXT-12-26895

August 2012

Approved by:

Wayne L. Moe   8/22/12

Wayne L. Moe
NGNP Regulatory Affairs

Date

Jim C. Kinsey   8-22-12

Jim C. Kinsey
NGNP Regulatory Affairs Director

Date

SUMMARY

Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants" (LWR Edition), provides guidance to nuclear power plant license application writers when developing a light water reactor (LWR) license application acceptable for review by the staff of the U.S. Nuclear Regulatory Commission (NRC). Revision to this guidance was initiated to provide similar format and content guidance to writers of high temperature gas-cooled reactor (HTGR) license applications. This report describes the status of HTGR license application guidance development efforts.

Of the 19 chapters contained in RG 1.206, seven were revised to form an initial HTGR combined license application Content Guide. Conversion was done by replacing LWR technology discussions with HTGR descriptions, addressing the inherent and passive safety features unique to the HTGR design, and incorporating a risk-informed performance-based licensing approach in lieu of a deterministic approach. Modified chapters were; Chapter 1, "Introduction and General Description of the Plant," Chapter 3, "Design of Structures, Systems, Components, and Equipment," Chapter 4, "Reactor System," Chapter 5, "Helium Pressure Boundary and Connecting Systems," Chapter 6, "Engineered Safety Features," Chapter 9, "Auxiliary Systems," and Chapter 15, "Transient and Accident Analysis" (Table of Contents only). These chapters contained some of the largest technical and licensing framework challenges when developing an application format and content guide for HTGR technology. They were also selected because they defined many basic presumptions that must be referenced in other chapters.

Extensive interaction with NRC staff was necessary to support development of the HTGR Content Guide. However, the Guide itself has not yet been reviewed by the NRC. Major interactions included submission of numerous topical white papers on a proposed HTGR licensing approach, responses to NRC requests for additional information, three NRC working group white paper assessment reports, response to requests for additional information related to the assessment reports, and a series of public meetings which are still ongoing. Regulatory interactions will continue for the remainder of 2012 and are expected to culminate in NRC Staff and Commission policy decisions that define a regulatory framework supportive of HTGRs. The results of these interactions are reflected in the draft HTGR Content Guide to the extent that this framework has been established.

Key portions of the HTGR Content Guide have been drafted on the basis of available HTGR design information and current understanding of the affiliated regulatory framework. These chapters, along with all other sections of RG 1.206, have been entered into a database that will support development of a complete HTGR licensing guide. As the HTGR licensing framework continues to evolve and additional details of advanced design become known, commensurate progress can be made using the database to complete remaining chapters of the HTGR Content Guide, thereby allowing the document to become a common source of guidance for future applicants and NRC staff.

CONTENTS

ACRONYMS

ACRS	NRC Advisory Committee on Reactor Safeguards
AGR	advanced gas reactor
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulation
COL	combined license
FSAR	final safety analysis report
GDC	general design criteria
HTGR	high temperature gas-cooled reactor
INL	Idaho National Laboratory
LWR	light water reactor
NGNP	Next Generation Nuclear Plant
NRC	U.S. Nuclear Regulatory Commission
NUREG	U.S. Nuclear Regulatory Commission report
PDC	principle design criteria
RG	regulatory guide
RIPB	risk-informed performance-based
SMR	small modular reactor
SRP	standard review plan
SSC	structure, system and component

COL Application Content Guide for HTGRs: Revision to RG 1.206, Part I – Status Report

1. INTRODUCTION

An application for license is required by the Nuclear Regulatory Commission (NRC) for all nuclear plants. A combined license (COL) application must contain a final safety analysis report (FSAR) that describes the facility, presents the design basis and limits on operation, and presents a safety analysis of the structures, systems, and components (SSCs) of the facility as a whole. The application must include information prescribed by 10 CFR 52.79, “Contents of Applications; Technical Information in Final Safety Analysis Report.” An applicant for a modular high temperature gas-cooled reactor (HTGR) is required to develop and submit for NRC review and approval a COL application that conforms to these requirements.

The information needed for the NRC staff to evaluate the acceptability of a COL application and resolve all safety issues related to a proposed facility is detailed and extensive. To support a comprehensive safety review, Regulatory Guide (RG) 1.206, “Combined License Applications for Nuclear Power Plants” (LWR Edition), was developed.[1] The guidance contained in RG 1.206 was published to assist applicants in identifying and properly formatting information necessary for NRC staff review. However, RG 1.206 was written based on light water reactor (LWR) technology and may not fully encompass important issues associated with alternative reactor technologies.

While much of the guidance and information contained in RG 1.206 is technology neutral and therefore applicable to non-LWR designs, important variances can exist as a function of individual design attributes and the safety case associated with a particular advanced reactor type. These discrepancies can be significant and make large portions of the LWR guidance contained in RG 1.206 unsupportive of non-LWR license application development. Specifics of an individual reactor design can also require different licensing approaches from those previously used, particularly when dealing with uniquely inherent and passive safety features.

The modular HTGR is an advanced reactor that employs design and operations features significantly different than those of a standard LWR.[2] Technological differences are great enough in some areas as to render portions of RG 1.206 guidance inconsistent or not applicable to an HTGR COL application. Also, some design and operations features and topical areas native to HTGR technology may be entirely unaddressed by the LWR perspective of RG 1.206.

With respect to future modular HTGR licensing actions, a RG 1.206 equivalent that addresses HTGR technology would benefit COL application writers and reviewers by providing a readily available source of accepted format and content guidance. As a result, the Idaho National Laboratory’s (INL) Next Generation Nuclear Plant (NGNP) program has initiated revision to RG 1.206 that incorporates a generic modular HTGR design. This report outlines the status of that effort.

2. TASK DESCRIPTION

Portions of RG 1.206, Part I, "Standard Format and Content of Combined License Application," were updated to provide timely guidance to future COL application writers charged with preparing an FSAR that integrates the unique technology attributes and safety functions associated with modular HTGR technology.[2] Moreover, the update sought to incorporate the inherent and passive safety and operations features uniquely associated with generic modular HTGR technology within a licensing framework predicated on a robust risk-informed, performance-based (RIPB) licensing approach.

Work activities focused on creating a new HTGR COL application format and content guide (HTGR Content Guide) derived from RG 1.206, Part I. This was done by modifying existing RG 1.206 language to reflect the technology licensing strategy and SSCs of a modular HTGR.

2.1 Scope

To be useful to future HTGR COL application writers, all 19 chapters of RG 1.206, Part I, must be evaluated and amended as necessary to be consistent with the technology. However, certain chapters presented a greater challenge to revision than others because they required complete redefinition of key technology attributes and novel underlying approaches in explaining new safety elements when compared to a LWR specification. Each chapter of RG 1.206, Part I, was examined on the basis of:

1. Scope of change necessary to incorporate appropriate technology descriptions and address fundamental HTGR design features that are absent or inaccurate in RG 1.206
2. The level of modular HTGR (conceptual) design information currently available to support revision
3. Need to identify and discriminate between elements important to safety that are applicable to HTGR technology as opposed to the LWR technology currently described
4. Opportunities to incorporate a RIPB framework into the HTGR licensing approach
5. Opportunities to clarify how existing licensing guidance for LWR technology can be extended and/or adapted into guidance for modular HTGR technology.

Those chapters of RG 1.206 offering the greatest challenge and opportunity in accurately portraying HTGR technology within a RIPB licensing framework were selected for early revision. The following chapters were revised to form the core of the new HTGR Content Guide:

Chapter 1, Introduction and General Description of the Plant

Chapter 3, Design of Structures, Systems, Components, and Equipment

Chapter 4, Reactor System

Chapter 5, Helium Pressure Boundary and Connecting Systems

Chapter 6, Engineered Safety Features

Chapter 9, Auxiliary Systems

Chapter 15, Transient and Accident Analysis (Table of Contents only).

Although the chapter and section titles contained in RG 1.206 may have been modified to more accurately reflect ensuing HTGR-oriented text, the original structure and configuration of RG 1.206 was followed as closely as practical.

2.2 Related Regulatory Interactions

In September 2011, NGNP completed an HTGR regulatory gap analysis and issued a report, *NGNP Project Regulatory Gap Analysis for Modular HTGRs*.[3] This report documented analysis results and identified regulatory gaps between current NRC licensing requirements and guidance pertaining to LWR designs, and the requirements and guidance needed to license a modular HTGR. Insights derived from the analysis were incorporated into the HTGR Content Guide text as necessary to indicate regulatory applicability.

NGNP also developed and submitted a series of white papers to the NRC that presented HTGR technical information and methodologies associated with key prelicensing technical and policy issues. The objectives of NGNP prelicensing white papers were also incorporated into HTGR Content Guide text, as applicable, to outline a HTGR RIPB licensing approach useful to writers of a COL application.

Technical expertise and licensing perspectives were gathered from the NGNP Licensing Working Group, an assembly of experienced power reactor licensing representatives from three HTGR design firms and one nuclear owner/operator organization. These experts periodically reviewed and provided feedback on the HTGR Content Guide as chapters were drafted such that the end product generally represents the input of the domestic HTGR “fleet”.

The HTGR Content Guide also reflects extensive discussions with NRC staff concerning the development of an overall HTGR regulatory framework. Formal interactions regarding that framework began with the NGNP prelicensing white papers submitted to the NRC. These submissions were followed by a series of NRC requests for additional information and NGNP responses, issuance of NRC working group assessment reports on selected prelicensing white papers, and public meetings (still ongoing) to address assessment report preliminary findings and outstanding prelicensing issues.

2.2.1 NGNP White Papers

Eleven NGNP white papers on key HTGR prelicensing topics were provided to the NRC. Together, these documents outline a proposed approach to establishing a regulatory framework suitable for effective licensing of a generic modular HTGR facility. However, certain white papers (along with subsequent interactions derived from them), had great bearing on HTGR Content Guide development. The NGNP white papers most significant in this regard were:

1. *Next Generation Nuclear Plant Defense-in-Depth Approach* (INL/EXT-09-17139)[4]
   Submitted to NRC: December 9, 2009
   NRC Public Meeting: March 8, 2010

2. *NGNP High Temperature Materials White Paper* (INL/EXT-09-17187)[5]
   Submitted to NRC: June 25, 2010
   NRC Public Meeting: September 1, 2010

3. *NGNP Fuel Qualification White Paper* (INL/EXT-10-18610)[6]
   Submitted to NRC: July 21, 2010
   NRC Public Meetings: September 2, 2010 and October 19, 2011

4. *Mechanistic Source Terms White Paper* (INL/EXT-10-17997)[7]
   Submitted to NRC: July 21, 2010
   NRC Public Meetings: September 2, 2010 and October 19, 2011

5. *Next Generation Nuclear Plant Licensing Basis Event Selection White Paper* (INL/EXT-10-19521)[8]
   Submitted to NRC: September 16, 2010
   NRC Public Meeting: November 2, 2010

6. *Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification White Paper,* (INL/EXT-10-19509)[9]
   Submitted to NRC: September 21, 2010
   NRC Public Meeting: November 2, 2010

7. *Modular HTGR Safety Basis and Approach* (INL/EXT-11-22708)[2]
   Submitted to NRC: September 6, 2011
   NRC Public Meeting: None

8. *Next Generation Nuclear Plant Probabilistic Risk Assessment White Paper* (INL/EXT-11-21270)[10]
   Submitted to NRC: September 20, 2011
   NRC Public Meeting: None

2.2.2 NRC Assessment Reports

On February 15, 2012, NGNP received two NRC working group assessment reports that outlined NRC staff opinions and unresolved concerns about HTGR safety. These reports were preliminary in nature and were identified as:

- *Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source Terms*
- *Assessment of White Paper Submittals on Defense in Depth, Licensing Basis Event Selection, and Safety Classification of Structures, Systems and Components*

The NRC transmittal letter for these assessment reports identified four important areas for policy development attention during the remainder of 2012. These were:

- Licensing basis event selection
- Radiological source terms
- Containment functional performance
- Emergency planning.

These four issues comprise the nucleus of the RIPB licensing approach for modular HTGRs. Progress in developing the licensing framework in response to these issues proved instrumental in creating the HTGR Content Guide.

Supplemental to the February 15, 2012, assessment reports was an additional NRC working group assessment released on May 9, 2012:

- *Assessment of White Paper Submittal on High Temperature Materials*

This transmittal provided NGNP with:

- NRC feedback regarding the content of the referenced white paper
- The identification of critical policy and technical issues concerning use of new and novel materials in HTGR design and construction
- Information needed to revise the white paper to reflect outstanding NRC concerns regarding the use of materials in an HTGR.

Criteria derived from the working group assessment reports were incorporated into the HTGR Content Guide as appropriate to portray the RIPB licensing framework.

2.2.3 Public Meetings

The licensing technical and policy issues identified in the two February 15, 2012, assessment reports were further developed through a series of public meetings held with NRC Headquarters staff. To date, the major interactions consisted of:

- April 16, 2012 – RIPB Workshop
  - Establish common understanding for further discussion of RIPB assessment report issues as identified by the NRC
  - Clarify and resolve stated objectives of the Defense-in-Depth white paper
  - Address the licensing basis event selection process and clarify consequence assessment criteria
- April 17, 2012 – Technical discussion on fuel qualification and mechanistic source terms white papers
  - Initiate dialog for resolving about 45 technical issues relating to white paper assessment report
  - Establish path to resolve questions on advanced gas reactor (AGR) test plan, fuel quality, air/moisture ingress, fission transport phenomena, fuel fabrication quality control, and other related issues.
- May 16, 2012 – Identify key issues on licensing basis event selection
  - Agree on approach to definition of licensing key event sequences
  - Clarify and agree on definitions related to event frequency
  - Identify resolution path for licensing basis event selection
- July 10, 2012 – Event selection, SSC classification, probabilistic risk assessment
  - Further define proposed approach to licensing basis event selection, the use of probabilistic risk assessment, and methods of classifying SSCs
  - Discuss NRC staff opinions and position options on these topics.
- July 11, 2012 – Functional containment performance
  - Provided NRC staff a description of NGNP's functional containment arrangements and relate them to regulatory performance goals
  - Outline NGNP capabilities to facilitate NRC review of NGNP licensing framework proposals in this area.
  - Present draft HTGR principle design criteria (PDC) derived from the general design criteria (GDC) of 10 CFR 50, Appendix A.
- July 19, 2012 – Response to request for additional information
  - Transmit letter CCN 227958, "Next Generation Nuclear Plant Submittal – Additional Information in Support of NRC Assessment Report Follow-Up Items Regarding Fuel Qualification/Mechanistic Source Terms – NRC Project # 0748" in response to NRC requests made during the April 17, 2012 public meeting.
- July 24. 2010 – Follow-on to April 17, 2012, on fuel qualification and mechanistic source term
  - Continue discussion of remaining open assessment report items related to particle fuel qualification and mechanistic source terms
  - Agree on an approach for particle fuel proof testing

- August 22, 2012 – Follow-on to July 10, 2012 public meeting
  - Confirm NRC positions concerning licensing basis event selection

Additional public meetings are being scheduled and will continue for the remainder of 2012. A summary briefing before the NRC Advisory Committee on Reactor Safeguards (ACRS) concerning the results of these interactions is tentatively planned for early 2013.

3. HTGR CONTENT GUIDE

The revised chapters for the HTGR Content Guide are provided in Appendixes A through H. These documents must be considered working drafts, in part because the modular HTGR still resides at the conceptual stage of design. Many important safety decisions remain to be made as the design continues to advance. As a result, numerous specifications and SSC performance attributes important to licensing cannot be fully addressed in the HTGR Content Guide at this time. The details normally available in later stages of design remain to be incorporated with appropriate specificity in the HTGR Content Guide.

Similarly, the NRC is still actively engaged in the process of establishing RIPB policy expectations that satisfy existing safety goals relative to future HTGR licensing actions. As NRC working group feedback is transformed into formal NRC staff positions and Commission policy, the text of the HTGR Content Guide will require appropriate re-examination and update to reflect those decisions.

Extrapolation of RG 1.206 into a functional HTGR Content Guide is still very much a work in progress. Although six chapters plus one chapter Table of Contents has been revised thus far, these chapters must be re-examined and updated to maintain currency with the still evolving HTGR regulatory framework. Furthermore, if the guidance is to be cohesive as well as comprehensive, the entire document must be periodically reviewed and refined as individual chapters are revised. Developing HTGR license application guidance is a highly iterative process and on this basis no portion of the document written thus far should be considered “final” or appropriate for unconditioned use by a COL applicant.

Having recognized the developmental phases of the HTGR Content Guide, the current status of this task is:

1. Regulatory Guide 1.206 (Part I) consists of separate 19 chapters, each corresponding to a chapter of a nuclear plant FSAR. Text in six of the most technically challenging chapters was revised in order to define the fundamental and essential technical and informational safety approach differences between HTGRs and LWRs. These are the chapters that call upon the applicant to characterize basic HTGR technology features and incorporate a RIPB licensing approach into the COL application. Working drafts of the revised chapters are provided in Appendixes B through G.

2. One RG 1.206 chapter deals exclusively with the nuclear facility event selection and analysis of the facility’s response to those events. The HTGR licensing framework concerning this topic is currently undergoing discussion with the NRC and is not yet adequately defined to support revision in the HTGR Content Guide. Thus revision to Chapter 15, “Transient and Accident Analysis,” was limited to updating the Table of Contents and is provided in Appendix H. Chapter 15 is highly supportive of the RIPB safety approach and should be modified once the NRC finalizes its positions concerning RIPB licensing basis event selection.

3. All 19 RG 1.206 chapters have been entered into an Access database optimized to support HTGR Content Guide development. The database resides with the INL NGNP office and is packaged for use by future Guide developers and users. Outstanding and unresolved issues concerning HTGR design specifications and licensing policy have been entered into the database pending eventual resolution.

4. RG 1.206 presents guidance that conforms to the NRC requirements, acceptance criteria, and review standards for a COL application as set forth in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (SRP). The SRPs are predicated on a typical large LWR design and prescribe many explicit expectations in that regard. Although certain considerations may be made to modify the SPR for use on alternative reactor designs (such as development of small modular reactor (SMR) design review guides), it is uncertain whether the NRC will revise the SRPs in support of HTGR COL application reviews.

To bridge inconsistencies a future HTGR COL application writer may encounter when referring to an SRP, the HTGR Content Guide database was designed to incorporate NRC SRP requirements and review acceptance criteria once they are established within the HTGR licensing framework. Acceptance criteria can be added to chapters of the HTGR Content Guide using the database at any future time should they become a necessary supplement to the SRPs.

5. Numerous references to NGNP prelicensing white papers are currently embedded in the working draft of the HTGR Content Guide. As the HTGR licensing framework becomes clearer and issues pertaining to NGNP white papers are resolved, the resulting guidance must be appropriately inserted into the HTGR Content Guide as replacement for the white paper citation. This guidance should reflect final NRC policy/position decisions concerning the topic being discussed.

6. Issues such as identification of safety-related SSCs and specific material failure mechanisms will be confirmed in a later stage of HTGR design. The Content Guide has addressed these items based on currently available conceptual design information. This early information must be confirmed and possibly adjusted to accurately represent the end state of HTGR technology.

7. Codes and standards for HTGR applications from organizations like the American Society of Mechanical Engineers (ASME) are still under development and must be endorsed by the NRC prior to use. The HTGR Content Guide has cited contemporary codes and standards as they now exist but progress in this area must be routinely integrated into the Guide. Similarly, as the various NRC regulatory documents (e.g., regulatory guides, NUREG documents, generic communications, policy statements, etc.) now cited in the Content Guide are revised, newly issued or withdrawn, their applicability must be reexamined with respect to HTGR licensing guidance.

8. Appendix A to 10 CFR 50 contains general design criteria (GDC) that must be applied to LWR design, fabrication, construction, testing, and performance. As stated in Appendix A, the GDCs provide guidance in establishing principal design criteria (PDC) for other reactor types like the HTGR. While some technology-neutral GDCs can be applied to non-LWR technologies, others require adaptation. Early in the HTGR Content Guide development process, a list of preliminary HTGR PDCs (modeled after the GDCs) was drafted and parenthetically cited in working draft HTGR Content Guide text. As HTGR licensing framework discussions move to decisions concerning application of LWR GDCs to modular HTGRs, the results of those decisions must be reflected in the Guide. It has been presumed that the draft HTGR PDCs will eventually be adopted and formally replace the LWR GDCs now cited in the Guide.

4. REFERENCES

1. RG 1.206, *"Combined License Applications for Nuclear Power Plants, (LWR Edition)"*, Part I: Standard Format and Content of Combined License Applications, U.S. Nuclear Regulatory Commission, June 2007.

2. INL/EXT-11-22708, *"Modular HTGR Safety Basis and Approach,"* Next Generation Nuclear Plant Project, Idaho National Laboratory, August 2011.

3. INL/EXT-11-23216, "*NGNP Project Regulatory Gap Analysis for Modular HTGRs,"* Rev. 0, Next Generation Nuclear Plant Project, Idaho National Laboratory, September 2011.

4. INL/EXT-09-17139, *"Next Generation Nuclear Plant Defense-in-Depth Approach,"* Next Generation Nuclear Plant Project, Idaho National Laboratory, December 2009.

5. INL/EXT-09-17187, *"NGNP High Temperature Materials White Paper,"* Next Generation Nuclear Plant Project, Idaho National Laboratory, June 2010.

6. INL/EXT-10-18610, *"NGNP Fuel Qualification White Paper,"* Rev. 0, Next Generation Nuclear Plant Project, Idaho National Laboratory, July 2010.

7. INL/EXT-10-17997, *"Mechanistic Source Terms White Paper,"* Rev 0, Next Generation Nuclear Plant Project, Idaho National Laboratory, July 2010.

8. INL/EXT-10-19521, *"Next Generation Nuclear Plant Licensing Basis Event Selection White Paper,"* Next Generation Nuclear Plant Project, Idaho National Laboratory, September 2010.

9. INL/EXT-10-19509, *"Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification White Paper,"* Next Generation Nuclear Plant Project, Idaho National Laboratory, September 2010.

10. INL/EXT-11-21270, *"Next Generation Nuclear Plant Probabilistic Risk Assessment White Paper,"* Next Generation Nuclear Plant Project, Idaho National Laboratory, September 2011.

Appendix A

Acronym List

Appendix A
Acronym List

AE	anticipated events
AISC	American Institute of Steel Construction
ALWR	advanced light-water reactor
AOO	anticipated operational occurrences
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BDBEs	beyond design basis events
BOL	beginning-of-life
Btu	British thermal unit
BWR	boiling water reactor
CFR	Code of Federal Regulations
COL	combined license
CRDM	control rod drive mechanism
CRDS	control rod drive system
CS	core support
DBA	design basis accident
DBE	design basis event
DCD	design certification document
DID	defense-in-depth
EAB	exclusion area boundary
ELWR	evolutionary light-water reactor
EOL	end-of-life
EPA	U.S. Environmental Protection Agency
ESF	engineered safety features
ESP	early site permit
FDA	Food and Drug Administration
FHS	fuel handling system
FIV	flow-induced vibration
FPP	fire protection program
FR	Federal Register
FSAR	final safety analysis report
GDC	general design criterion/criteria

GRMS	ground motion response spectrum
GSI	generic safety issue
HEPA	high efficiency particulate air
HPB	helium pressure boundary
HPS	helium purification system
HTGR	high temperature gas-cooled reactors
HTS	heat transport system
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	intergranular stress-corrosion cracking
IHX	intermediate heat exchanger
INPO	Institute of Nuclear Power Operations
ISI	inservice inspection
IST	inservice testing
ITAAC	inspections, tests, analysis, and acceptance criteria
ITP	initial test program
LANL	Los Alamos National Laboratory
LBB	leak-before-break
LBE	licensing basis events
lbm	pound (mass)
LCO	limiting condition of operation
LOOP	loss of offsite power
LWR	light-water reactor
MOV	motor operated valve
N/A	not applicable
NDE	nondestructive examination
NGNP	Next Generation Nuclear Plant
NRC	U.S. Nuclear Regulatory Commission
NSRST	non-safety related with special treatment
NUREG	U.S. Nuclear Regulatory Commission Regulation report
OBE	operating-basis earthquake
OM	operation and maintenance
PDC	principal design criteria

PRA	probabilistic risk assessment
PSD	power spectral density
PSHA	probabilistic seismic hazards assessment
PSID	preliminary safety information document
PWR	pressurized water reactor
QA	quality assurance
RB	reactor building
RCCS	reactor cavity cooling system
RCS	reactor coolant system
RG	regulatory guide
RIPB	risk-informed performance-based
RIS	regulatory issues summary
rpm	revolutions per minute
RPS	reactor protection system
RPV	reactor pressure vessel
RTNDT	reference temperature for nil ductility transition
RTNSS	regulatory treatment of non-safety systems
SBO	station blackout
SCS	shutdown cooling system
SDD	system design description
SECY	Office of the Secretary of the Commission
SER	safety evaluation report
SRM	staff requirements memorandum
SRP	standard review plan
SSC	structure, system and component
SSE	safe shutdown earthquake
SSI	soil-structure interaction
TBD	to be determined
TGSCC	transgranular stress-corrosion cracking
TLRC	top-level regulatory criteria
TMI	Three Mile Island
TS	technical specification
UHS	ultimate heat sink
USIs	unresolved safety issues

Appendix B

Chapter 1. Introduction and General Description of the Plant

Appendix B
Chapter 1. Introduction and General Description of the Plant

Modified Section/Title: C.I.1; Introduction and General Description of the Plant
Original Section/Title: C.I.1; Introduction and General Description of the Plant
In accordance with Subpart C, “Combined Licenses,” of 10 CFR Part 52, COL applicants may reference designs that have been certified according to Subpart B, “Standard Design Certifications,” of 10 CFR Part 52 and ESPs that have been certified according to Subpart A, “Early Site Permits,” of 10 CFR Part 52. The guidance in Section C.I of this regulatory guide applies to COL applicants who reference neither a certified design nor an ESP, but provides a design for a complete facility on a specified site (i.e., a custom design). For COL applicants who reference a certified design, Section C.III.1 of this regulatory guide furnishes additional guidance. For COL applicants who reference a certified design and an ESP, Section C.III.2 of this regulatory guide offers additional guidance.

The first chapter of the FSAR should include an introduction to the report and a general description of the plant and its safety design bases. This chapter should provide the reviewer or reader with a basic understanding of the overall facility without needing to refer to subsequent chapters. The review of the subsequent detailed chapters can then be accomplished with a better perspective and recognition of the relative safety-significance of each individual item in the overall plant design.

Modified Section/Title: C.I.1.1; Introduction
Original Section/Title: C.I.1.1; Introduction
In this section, the COL applicant should briefly discuss the principal aspects of the overall application, including the type of license requested, an overview of the modular HTGR plant design, including the number of reactor modules, a brief description of the proposed plant location, the type of energy conversion system and its designer, the core thermal power levels (both rated and design), the corresponding energy conversion for total net electrical/thermal output for each modular reactors power level, and the scheduled completion date and anticipated commercial operation date of each unit. The following subsections address these aspects of the application.

Modified Section/Title: C.I.1.1.1; Plant Location
Original Section/Title: C.I.1.1.1; Plant Location
The COL applicant should provide plant location information, such as the State and county in which the site will be located, as well as one or more maps showing the site location and plant arrangement within the site, including the extent (if any) to which the plant is collocated and/or interfaces with an existing industrial facility or licensed existing nuclear power plant (i.e., one that is currently located within the existing exclusion area boundary [EAB]).

Modified Section/Title: C.I.1.1.2; Functional Containment Description
Original Section/Title: C.I.1.1.2; Containment Type
The COL applicant should provide a summary-level description of the functional containment for modular reactor type. The discussion should include: how the functional containment performance standards are going to be meet; how the LBEs will be considered for functional containment design decisions; and, how series of barriers provide the functional containment to ensure that radionuclides are retained and that regulatory requirements and plant design goals on the release of radionuclides are met at the exclusion area boundary.

Modified Section/Title: C.I.1.1.2.1; Co-generation
Original Section/Title: NEW; NEW
The COL applicant should provide a summary-level description of the co-generation system model, including a description of the major components and a simplified diagram.

Modified Section/Title: C.I.1.1.2.2; Modular Design
Original Section/Title: NEW; NEW
Provide a summary description of each modular reactor design, including a simplified diagram of the site layout of the nuclear island showing the relationship between the reactor modules and other important support buildings (e.g., reactor auxiliary buildings, electrical services building, operations center/control room, etc.). Identify interfaces that exist between the individual reactor modules and how the reactor modules interface with the energy conversion portion of the facility.

Modified Section/Title: C.I.1.1.3; Reactor Type
Original Section/Title: C.I.1.1.3; Reactor Type
The COL applicant should specify the HTGR reactor system model and identify the reactor plant designer. Additionally, the COL applicant should provide a summary-level description of the modular reactor type. If multi modular reactor designs are planned, a brief description should be provided for each modular reactor type.

Modified Section/Title: C.I.1.1.4; Power Output
Original Section/Title: C.I.1.1.4; Power Output
The COL applicant should provide the approximate net output of the energy conversion unit (for information only) and the core thermal power levels (both rated and design) for each of the HTGR reactor modules.

Modified Section/Title: C.I.1.1.5; Schedule
Original Section/Title: C.I.1.1.5; Schedule
The COL applicant should provide estimated schedules for the completion of construction and the start of commercial operation (estimates may be specified in duration, rather than calendar dates, based on the application submittal date). As an alternative, COL applicants may include a commitment to provide the construction and startup schedules after issuance of the COL once the licensee has made a positive decision to construct the plant.

Modified Section/Title: C.I.1.1.6; Format and Content
Original Section/Title: C.I.1.1.6; Format and Content
The Regulatory Guide (RG) 1.206, "Combined License Applications For Nuclear Power Plants (LWR Edition)," should be used as a reference document in developing the format and content of the COL applications. The approach for development of the application is similar to what is described in the RG 1.206.

The COL applicant should provide information on the following aspects of the format and content of its application:

1.1.6.1 This section should discuss conformance with the format and content guidance.
1.1.6.2 This section should discuss conformance with NUREG-0800 (or its equivalent NUREG for HTGR Design) in effect 6 months before the application submittal date(i.e., the applicant should evaluate the differences in the design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria).

1.1.6.3 This section should provide the format, content, and numbering of text, tables, and figures included in the application and discuss their use.
1.1.6.4 This section should discuss the format for page numbering.
1.1.6.5 This section should discuss the method used to identify and reference proprietary information.
1.1.6.6 This section should list the acronyms used in the FSAR. Documents that are not part of the FSAR, but are part of the application should include their own list of acronyms.

Modified Section/Title: C.I.1.2; General Plant Description

Original Section/Title: C.I.1.2; General Plant Description

In this section, the COL applicant should summarize the principal characteristics of the site and provide a concise description of the facility. The facility description should include a brief discussion of the principal design criteria, operating characteristics, and safety consideration for the facility: engineered safety features (ESF) and emergency systems; instrumentation, control, and electrical systems; power conversion system; fuel handling and storage systems; cooling water and other auxiliary systems; and radioactive waste management system. The applicant should indicate the general arrangement of major structures and equipment by using plan and elevation drawings, furnished in sufficient number and detail to provide a reasonable understanding of the general layout of the plant. The applicant also should identify those features of the plant that are likely to be of special interest because of their relationship to safety. In addition, the COL applicant should highlight items such as unusual site characteristics, solutions to particularly difficult engineering and/or construction considerations (e.g., modular construction techniques or plans), and significant extrapolations in technology represented by the design.

Modified Section/Title: C.I.1.3; Safety Basis and Approach

Original Section/Title: NEW; NEW

The safety basis and approach description should include a brief discussion of the safety considerations for the facility, including the facility safety objectives; inherent and passive safety features; the barriers to radioactive releases; description of the fuel particle kernel, fuel particle coatings, and core graphite/carbonaceous materials; description of the helium pressure boundary and the reactor building; a description of the functional safety approach to address removal of core heat, control of heat generation, and control of chemical attack; and a discussion of the risk-informed performance-based safety approach to include use of PRA, selection of licensing basis events, safety classification of SSCs, and the approach for demonstrating defense in depth. The applicant also should identify those features of the plant that are likely to be of special interest because of their relationship to safety. (Refer to NGNP paper entitled, Modular HTGR Safety Basis and Approach, INL/EXT-11-22708, dated August 2011)

Modified Section/Title: C.I.1.3.1; Safety Objectives

Original Section/Title: NEW; NEW

The COL applicant should provide primary safety objectives of the design. This should include the licensing and safety basis for the plant that is being developed using the risk-informed performance-based (RIPB) based process, top-level regulatory criteria (TLRC) that are identified from NRC regulations and guidance that establish dose limits on consequences from licensing basis events (LBEs) to assure public safety.

Discussion of the TLRC should be provided with the following objectives:

1. Provide direct public health and safety acceptability limits in terms of individual radiological consequences
2. Be independent of HTGR reactor type and site
3. Provide well-defined, quantifiable risk criteria.

Modified Section/Title: C.I.1.3.2; Intrinsic And Passive Safety Features

Original Section/Title: NEW; NEW

The COL applicant should provide a brief discussion of the intrinsic and passive safety features of the HTGR design. The discussion should include: high temperature characteristics of TRISO-coated fuel particles, graphite moderator, and helium coolant, along with passive heat removal capability that will assure sufficient core residual heat removal under loss-of-forced cooling or loss-of-coolant-pressure conditions.

Modified Section/Title: C.I.1.3.3; Radionuclide Release Barriers

Original Section/Title: NEW; NEW

The COL applicant should provide an overview of the integrated barriers to release of radionuclides provided by the HTGR design. The discussion should include the barriers to radionuclide release that form a functional containment for modular HTGRs and the relative effectiveness of these barriers in controlling radionuclides.

Modified Section/Title: C.I.1.3.3.1; Fuel Particle Kernel

Original Section/Title: NEW; NEW

The COL applicant should provide a description of the fuel particle kernel and its inherent characteristics that affect the release of fission products from the fuel kernel. The discussion should include condition of the fuel particle kernel under normal operating conditions, accident conditions, and transient conditions. The release of fission gases, metallic fission products, and other noble metals released from the fuel kernels at normal and elevated temperatures should also be described.

Modified Section/Title: C.I.1.3.3.2; Fuel Particle Coatings

Original Section/Title: NEW; NEW

The COL applicant should provide a description of the fuel particle coatings (including the buffer layers) and how each coating (silicon carbide and pyrocarbon coatings) affects the release of particular types of fission products from the fuel particle. The discussion should include transport of fission products during normal operating conditions and accident conditions.

Modified Section/Title: C.I.1.3.3.3; Core Graphite And Carbonaceous Materials

Original Section/Title: NEW; NEW

The COL applicant should provide a description of the core graphite and carbonaceous materials to include their effects on the release of fission products from the reactor core. The discussion should include release of fission products during normal operating conditions and accident conditions. Also, the discussion should include the effects of the graphite materials due to ingress of air or moisture into the reactor.

Modified Section/Title: C.I.1.3.3.4; Helium Pressure Boundary

Original Section/Title: NEW; NEW

The COL applicant should provide a brief description of the helium pressure boundary to include its contribution to the retention of fission products. The discussion should include: how the chemical impurities in the helium are controlled and the efficiency of helium purification system (HPS) for removal of both gaseous and metallic fission products from the helium. The discussion should also include circulating, dust, and plateout activities in the primary circuit. If steam generator(s) is connected to the system, then issues such as moisture ingress should be discussed. Other mechanisms that can potentially result in the removal and subsequent environmental release of the primary circuit plateout activity (e.g., steam-induced vaporization and washoff) should be discussed.

Modified Section/Title: C.I.1.3.3.5; Reactor Building

Original Section/Title: NEW; NEW

The COL applicant should provide a brief description of the functional requirements for the reactor building including its contribution to the retention of fission products. The discussion should include design of the reactor building including its retention capability for radionuclides releases during normal, transient and accident conditions. The transport behavior of radionuclides during core heat-up accidents should also be discussed.

Modified Section/Title: C.I.1.3.4; Functional Safety Approach

Original Section/Title: NEW; NEW

The COL applicant should provide an overview of the HTGR functional safety approach to include the methods for removal of core heat, the control of heat generation, and control of chemical attack. Subsequent subsections will provide details related to each aspect of the HTGR functional safety approach.

Modified Section/Title: C.I.1.3.4.1; Remove Core Heat

Original Section/Title: NEW; NEW

The COL applicant should provide a brief description of core heat removal capability for HTGR design. The discussion should include design functions of the reactor cooling system, shutdown cooling system and any passive cooling from the core through the reactor vessel to the reactor cavity cooling system (RCCS). In addition, any passive mode of operation for removing residual core heat under normal, transient and accident conditions should be discussed. Discuss the implications of helium coolant pressure loss and measures necessary to ensure that core temperature are sufficiently maintained to ensure safety.

Modified Section/Title: C.I.1.3.4.2; Control Heat Generation

Original Section/Title: NEW; NEW

The COL applicant should provide a brief description of how the control heat generation is accomplished for HTGR design. The discussion should include description of the normal and backup shutdown systems for maintaining the reactivity control including consequences associated with loss of normal and backup shutdown systems.

Modified Section/Title: C.I.1.3.4.3; Control Chemical Attack

Original Section/Title: NEW; NEW

The COL applicant should provide a brief description of control of chemical attack on fuel particles and on the graphite core structure due to air or moisture ingress into the primary system. The discussion should include how impurity ingress is managed during normal operations, anticipated operational occurrences (AOO) and accident sequences. If steam generator(s) and/or other water cooler(s) and heat exchanger(s) are installed in the system, discuss the likelihood of water entering the primary system and the detection and mitigation capabilities to limit the potential for chemical attack. Also, provide a discussion of detection and mitigation capabilities due to a failure of the helium pressure boundary for which air ingress becomes a concern.

Modified Section/Title: C.I.1.3.5; Risk-Informed Performance-Based Safety Approach

Original Section/Title: NEW; NEW

The COL applicant should provide an overview of the risk-informed performance-based safety approach that will be used to select events to be analyzed. Subsequent subsections will provide discussions related to use of probabilistic risk assessment (PRA), selection of licensing basis events, safety classification of SSCs, and the methods for demonstrating defense-in-depth.

Modified Section/Title: C.I.1.3.5.1; Use of Probabilistic Risk Assessment

Original Section/Title: NEW; NEW

The COL applicant should provide a brief description of technical approach to use of PRA for HTGR design. The discussion should include HTGR PRA model structure that includes specific end-state frequencies that correspond with the LBEs. The PRA model should account for risk of multiple modules where multiple reactors are to be located at the same site.

Reference: RG 1.174 and RG 1.200

Modified Section/Title: C.I.1.3.5.2; Licensing Basis Event Selection

Original Section/Title: NEW; NEW

The COL applicant should provide a brief description of licensing basis event (LBE) selection based on set of event sequences that form the basis for plant analysis and that represent the plant's characteristic performance in all analyzed frequency and consequence ranges. The discussion should include conditions of normal, including AOOs, design basis events (DBEs), and beyond design basis events (BDBEs) that inform the deterministically selected design basis accidents (DBAs). Since LBE selection is an integral part of the overall design process, discuss the process to be used for each phase of the detailed design development, including the process for use of the PRA.

Modified Section/Title: C.I.1.3.5.3; Structures, Systems, and Components Safety Classification

Original Section/Title: NEW; NEW

The COL applicant should provide a brief description of safety classification process for structures, systems and components (SSCs). The discussion should include risk-informed and performance based licensing approach that includes categories of safety classification for SSCs.

Modified Section/Title: C.I.1.3.5.4; Defense-In-Depth

Original Section/Title: NEW; NEW

The COL applicant should provide a brief description of how the principles of defense-in-depth (DiD) are applied in the design, construction, and operation of HTGR. The discussion should include risk-informed and performance-based framework for DiD.

Modified Section/Title: C.I.1.4; Comparison with Other Facilities

Original Section/Title: C.I.1.3; Comparison with Other Facilities

The COL applicant should provide a comparison with other facilities of similar design and comparable power level.

Modified Section/Title: C.I.1.5; Identification of Agents and Contractors

Original Section/Title: C.I.1.4; Identification of Agents and Contractors

In this section, the COL applicant should identify the primary agents or contractors for the design, construction, and operation of the nuclear power plant. The applicants should note the principal consultants and outside service organizations (such as those providing audits of the QA program). The applicant also should delineate the division of responsibility among the reactor/facility designer, architect-engineer, constructor, and plant operator.

Modified Section/Title: C.I.1.6; Requirements for Additional Technical Information

Original Section/Title: C.I.1.5; Requirements for Additional Technical Information

In this section, COL applicants who do not reference a certified design should provide information to demonstrate the performance of new safety features for nuclear power plants or use simplified, intrinsic passive, or other innovative means to accomplish their safety functions. The requirement to provide this information is part of 10 CFR Part 52 and is necessary to ensure that (1) these new safety features will perform as predicted in the applicant's FSAR, (2) the effects of system interactions are acceptable, and (3) the applicant provides sufficient data to validate analytical codes. The design qualification testing requirements may be met with either separate effects or integral system tests; prototype tests; or a combination of tests, analyses, and operating experience. These requirements implement the Commission's policy on proof-of-performance testing for all advanced reactors (Volume 51, page 24643 of the Federal Register (51 FR 24643), dated July 8, 1986), as well as the Commission's goal of resolving all safety issues before authorizing construction.

The COL applicant who does not reference a certified design as part of the application must provide design information for the entire proposed facility, including a level of detail necessary to resolve all safety issues (i.e., the same level of detailed design information as that supplied in a certified design). Although a COL applicant who does not reference a certified design must furnish sufficient design information for a complete facility, the NRC expects that it may need additional technical information (beyond that in the application), including items such as verification of unique design concepts (e.g., concepts that may require tests and/or additional verification analyses for the first plant, the first three plants, and so forth).

The COL applicant is responsible for providing a complete design for its proposed facility to identify any requirements for additional technical information in its application, including an estimated schedule for furnishing the additional technical information that may be necessary for issuance of a COL.

Modified Section/Title: C.I.1.7; Material Referenced

Original Section/Title: C.I.1.6; Material Referenced

In this section, the COL applicant should tabulate all topical reports that are incorporated by reference as part of the application. In this context, topical reports are defined as reports that reactor designers and manufacturers, architect-engineers, or other organizations have prepared and filed separately with the NRC in support of this application or of other applications or product lines. For each topical report, this tabulation should include the report number and title, the date that the report was submitted to the NRC, and the sections of the COL application that reference the report. For any topical reports that have been withheld from public disclosure as proprietary documents pursuant to 10 CFR 2.390(b), this tabulation should also reference nonproprietary summary descriptions of the general content of each such report. This section should also include a tabulation of any documents submitted to the Commission in other applications that are incorporated in whole or in part into the application by reference. If any information submitted in connection with other applications is incorporated by reference into the application, the applicant should summarize such information in appropriate sections of the application, as necessary, to provide clarity and context.

The applicant may submit results of test and analyses as separate reports. In such cases, this section should reference these reports, which the appropriate sections of the FSAR should summarize.

Modified Section/Title: C.I.1.8; Drawings and Other Detailed Information

Original Section/Title: C.I.1.7; Drawings and Other Detailed Information

The COL applicant should provide a tabulation of all instrument and control functional diagrams and electrical one-line diagrams cross-referenced to the related application sections, including legends for electrical power, instrument and control, lighting, and communication drawings.

In addition, the COL applicant should furnish a tabulation of system drawings (e.g., piping and instrumentation diagrams) and system designators that are cross-referenced to the related sections of the application. This information should include the applicable drawing legends and notes.

Modified Section/Title: C.I.1.9; Interfaces (with Standard Designs and Early Site Permits)

Original Section/Title: C.I.1.8; Interfaces (with Standard Designs and Early Site Permits)

COL applicants who do not reference a certified design as part of the application must provide design information for a complete facility (i.e., not limited in scope such as a certified design), including a level of detail necessary to resolve all safety issues (i.e., the same level of detailed design information as that provided in a certified design). By definition, there are no interface requirements between standard designs and site-specific designs for a complete facility design. The expectation is that all interfaces, such as those that may exist between certified designs, ESPs, and a COL application that references a certified design and/or ESP, will be integral to a COL application that provides a complete facility design. COL applicants who reference a certified design and/or ESP are the only applicants who will have interface requirements.

COL applicants who do not reference a certified design will need to submit design information on the entire facility and should not include any conceptual design information for the facility. To facilitate the NRC staff review of previous applications for design certification, conceptual designs were included in the design control documents (DCDs) to offer a comprehensive design perspective. However, the conceptual design portions of the DCDs were not (and were not intended to be) certified by the NRC. Rather, these conceptual designs typically included portions of the balance-of-plant. Thus, the NRC expects that COL applicants who do not reference a certified design will provide complete designs for the facility without reliance on conceptual designs.

Modified Section/Title: C.I.1.10; Conformance with Regulatory Criteria

Original Section/Title: C.I.1.9; Conformance with Regulatory Criteria

Modified Section/Title: C.I.1.10.1; Conformance with Regulatory Guides

Original Section/Title: C.I.1.9.1; Conformance with Regulatory Guides

The requirements of 10 CFR 52.79(a)(4)(I) specify that the content of a COL application must include information on the design of the facility, including its principal design criteria. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes minimum requirements for the principal design criteria (PDC)for water-cooled nuclear power plants that are similar in design and location to plants for which the Commission has previously issued construction permits. Appendix A also provides guidance to applicants for use in establishing PDC for other types of nuclear power units. In general, regulatory guides describe methods that the NRC staff considers acceptable for implementing the general design criteria (GDC) specified in Appendix A to 10 CFR Part 50. Additionally, the PDCs developed specific for Modular HTGR should be considered since it establishes the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. Thus, COL applicants should provide an evaluation of conformance with the guidance in NRC regulatory guides in effect 6 months before the submittal date of the COL application. That evaluation should also include an identification and description of deviations from the guidance in the NRC regulatory guides as well as suitable justifications for any alternative approaches proposed by the COL applicant.

COL applicants should furnish an evaluation of conformance with the following groups of regulatory guides:

- Division 1, Power Reactors
- Division 4, Environmental and Siting (applies to the environmental report and should be discussed therein)
- Division 5, Materials and Plant Protection (applies to the security plan and should be discussed therein)
- Division 8, Occupational Health

Modified Section/Title: C.I.1.10.2; Conformance with Standard Review Plan and/or other Applicable Review Guidance Document

Original Section/Title: C.I.1.9.2; Conformance with Standard Review Plan

The COL applicants should evaluate the COL applications against the guidance provided in 10 CFR 52.79(a)(41) so that the facility is evaluated against the NRC's application and review guidance in effect 6 months before the docket date of the application. The evaluation required by this section must include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the facility and those corresponding features, techniques, and measures in the acceptance criteria in the application and review guidance. If differences exist, the evaluation should discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding acceptance criteria.

Modified Section/Title: C.I.1.10.3; Generic Issues

Original Section/Title: C.I.1.9.3; Generic Issues

The requirements of 10 CFR 52.79(a)(20) specify that a COL application must include the proposed technical resolutions for those unresolved safety issues (USIs) and medium- and high-priority generic safety issues (GSIs) that (1) are identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current on the date up to 6 months before the docket date of the application and (2) are technically relevant to the design.

Since the inception of the generic issues program in 1976, the NRC has identified and categorized reactor safety issues. The NRC grouped these issues into R (TMI) action plan items, task action plan items, new generic items, human factors issues, and Chernobyl issues, collectively calling them GSIs. Section C.IV.8 of this regulatory guide provides additional guidance for addressing the USIs and medium- and high-priority GSIs that NUREG 0933 identifies.

Modified Section/Title: C.I.1.10.4; Operational Experience (Generic Communications)

Original Section/Title: C.I.1.9.4; Operational Experience (Generic Communications)

The requirements of 10 CFR 52.79(a)(37) specify that the COL application must include information to demonstrate how operating experience insights from generic letters and bulletins issued after the most recent revision of the applicable SRP (and/or other applicable review guidance document) and 6 months before the docket date of the application, or comparable international operating experience, have been incorporated into the plant design.

To ensure that the knowledge base for reviewers and applicants captured the operational experience described in generic letters and bulletins from decades of nuclear power plant operation in the United States, the NRC staff incorporated the insights from these generic letters and bulletins into the updates to applicable SRPs. To ensure that the operational experience in these SRP updates is considered, applicants with plant designs that are based on, or are evolutions of, plants that have operated in the United States are required by 10 CFR 52.79(a)(41) to evaluate their facility designs against the review guidance (i.e., SRPs and/or other applicable review guidance document) in effect 6 months before the docket date of the application. In addition, applicants are required to demonstrate how the operating experience insights from generic letters and bulletins issued after the review guidance update (i.e., in or about March 2007) have been incorporated into the plant design (i.e., address those generic communications not incorporated in the SRP update). The significance of limiting this review to generic letters and bulletins is that these documents pertain to issues that rose to a level of safety significance such that responses and resolutions from nuclear operating plant licensees were required. Other forms of generic communications have included circulars, information notices, and regulatory information summaries (RIS); however, as these types of generic communications do not require responses or actions on the part of licensees, COL applicants need not address them. In addition, the issues discussed in these types of communications are generally of a more specific (rather than generic) nature.

Alternatively, COL applicants with a plant design that is not based on, or is not an evolution of, plants that have operated in the United States should demonstrate how they have incorporated comparable international operating experience into the plant design. Nuclear industry regulators or owners groups in countries that include nuclear reactor vendors and/or nuclear power plants (e.g., Canada, France, Germany, Japan) may track, maintain, and/or issue operating experience bulletins or reports similar to the NRC generic letters and bulletins. The COL applicant should address how it assessed and/or incorporated the applicable operating experience into the plant design. In addition, COL applicants should consult organizations such as the Institute of Nuclear Power Operations (INPO) or the World Association of Nuclear Operators for applicable comparable international operating experience.

Modified Section/Title: C.I.1.10.5; Advanced and Evolutionary Reactor Design Issues

Original Section/Title: C.I.1.9.5; Advanced and Evolutionary Light-Water Reactor Design Issues

COL applicants who do not reference a certified design should provide sufficient information on the complete design of the proposed facility, including those portions of the facility design that are typically provided by reactor vendors or applicants for reactor design certification in accordance with Subpart B of 10 CFR Part 52. Therefore, COL applicants should address the licensing and policy issues developed by the NRC and documented in the Office of the Secretary of the Commission (SECY) documents listed below and the associated staff requirements memoranda (SRM) for advanced and evolutionary designs that apply to the proposed facility design. The following SECY documents provide guidance to applicants on issues that they should consider and, as appropriate, address in a COL application that does not reference a certified design (i.e., a custom design); however, this list may not be comprehensive, and some of the references may not apply to all potential COL applicants:

SECY 88-203, "Key Licensing Issues Associated with DOE Sponsored Advanced Reactor Designs"

SECY 90-241, "Level of Detail Required for Design Certification under Part 52"

SECY 90-377, "Requirements for Design Certification under 10 CFR Part 52"

SECY 90-0341, Staff Study on Source Term Update and Decoupling Siting from Design"

SECY 91-074, "Prototype Decisions for Advanced Reactor Designs"

SECY 91-178, "ITAAC for Design Certifications and Combined Licenses"

SECY 91-210, "ITAAC Requirements for Design Review and Issuance of FDA"

SECY 93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs"

SECY 93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements"

SECY 94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems"

SECY 95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs"

SECY 97-0020, "Results of Evaluation of Emergency Planning for Evolutionary and Advanced Reactors"

SECY-97-171, "Consideration of Severe Accident Risk in NRC Regulatory Decisions"

SECY 98-0144, "White Paper on Risk-Informed and Performance-Based Regulation (Revised)"

SECY 99-0186, “Staff Plan for Clarifying How Defense-in-Depth Applies to the Regulation of a Possible Geologic Repository at Yucca Mountain, Nevada”

SECY 00-0198, “Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR Part 50.44 (Combustible Gas Control)”

SECY 01-0070, “Plan for Preapplication Activities on the Pebble Bed Modular Reactor (PBMR)”

SECY 02-0057, “Update to SECY-01-0133, ‘Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)”

SECY 02-0076, “Semi-Annual Update of the Future Licensing and Inspection Readiness Assessment”

SECY 02-0139, “Plan for Resolving Policy Issues Related to Licensing Non-Light Water Reactor Designs”

SECY 03-0047, “Policy Issues Related to Licensing Non-Light Water Reactor Designs”

SECY 03-0059, “NRC’s Advanced Reactor Research Program”

SECY 05-0006, “Second Status Paper on the Staff’s Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing”

SECY 05-0138, “Risk-Informed and Performance-Based Alternatives to the Single-Failure Criterion”

SECY 06-0217, “Improvement to and Update of the Risk-Informed Regulation Implementation Plan”

SECY 08-0019, “Licensing and Regulatory Research Related to Advanced Nuclear Reactors”

SECY 09-0056, “Staff Approach Regarding a Risk-Informed and Performance-Based Revision to Part 50 of Title 10 of the Code of Federal Regulations and Developing a Policy Statement on Defense-in-Depth for Future Reactors”

SECY 10-0034, “Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs”

Modified Section/Title: C.I.1.11; Hazards Posed by Construction to Operating Modules

Original Section/Title: NEW; NEW

The COL applicant should provide a brief description of potential hazards posed by construction to operation modules. The discussion should include those potential risks and controls that will be established for construction of a new module next to an operating module(s) for a multi-module site.

Appendix C

Chapter 3. Design of Structures, Systems, Components, and Equipment

Appendix C
Chapter 3. Design of Structures, Systems, Components, and Equipment

Modified Section/Title: C.I.1; Introduction and General Description of the Plant
Original Section/Title: C.I.1; Introduction and General Description of the Plant
In accordance with Subpart C, "Combined Licenses," of 10 CFR Part 52, COL applicants may reference designs that have been certified according to Subpart B, "Standard Design Certifications," of 10 CFR Part 52 and ESPs that have been certified according to Subpart A, "Early Site Permits," of 10 CFR Part 52. The guidance in Section C.I of this regulatory guide applies to COL applicants who reference neither a certified design nor an ESP, but provides a design for a complete facility on a specified site (i.e., a custom design). For COL applicants who reference a certified design, Section C.III.1 of this regulatory guide furnishes additional guidance. For COL applicants who reference a certified design and an ESP, Section C.III.2 of this regulatory guide offers additional guidance.

The first chapter of the FSAR should include an introduction to the report and a general description of the plant and its safety design bases. This chapter should provide the reviewer or reader with a basic understanding of the overall facility without needing to refer to subsequent chapters. The review of the subsequent detailed chapters can then be accomplished with a better perspective and recognition of the relative safety-significance of each individual item in the overall plant design.

Modified Section/Title: C.I.3; Design of Structures, Systems, Components, and Equipment
Original Section/Title: C.I.3; Design of Structures, Systems, Components, and Equipment
Chapter 3 of the FSAR should identify, describe, and discuss the principal architectural and engineering design of those SSCs, and equipment that are important to safety.

Modified Section/Title: C.I.3.1; Conformance with U.S. Nuclear Regulatory Commission General Design Criteria
Original Section/Title: C.I.3.1; Conformance with U.S. Nuclear Regulatory Commission General Design Criteria
The applicant should discuss the extent to which plant SSCs important to safety meet the HTGR Principal Design Criteria modeled after the NRC's criteria in 10 CFR Part 50, Appendix A. For each criterion, the applicant should provide a summary showing how the principal design features meet the PDC. The discussion of each criterion should identify the sections of the FSAR that present more detailed information to demonstrate compliance with the PDC.

Modified Section/Title: C.I.3.2; Classification of Structures, Systems, and Components
Original Section/Title: C.I.3.2; Classification of Structures, Systems, and Components

Modified Section/Title: C.I.3.2.1; Seismic Classification
Original Section/Title: C.I.3.2.1; Seismic Classification
The applicant should identify those SSCs important to safety that are designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. Plant features, including foundations and supports, that are designed to remain functional in the event of a SSE (see FSAR Section 2.5) or surface deformation should be designated as seismic Category I. Specifically, the plant features of interest are those necessary to ensure the following characteristics:

- Alterations to nominal HPB integrity do not adversely affect the safety-related function of any plant SSC

- Capability to shut down the reactor and maintain it in a safe, stable condition according to design conditions and requirements
- Capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1) and 10 CFR 52.79.

RG 1.29, ***"Seismic Design Classification,"*** contains guidance for identifying seismic Category I SSCs. The applicant should provide a list of all seismic Category I items and indicate whether it has followed the recommendations of RG 1.29. If only portions of structures and systems are seismic Category I, the applicant should list them and, where necessary for clarity, show the boundaries of the seismic Category I portions on piping and instrumentation diagrams. The applicant should also identify portions of SSCs not required to continue functioning, but the failure of which could reduce the functioning of any seismic Category I plant feature to an unacceptable safety level or could result in incapacitating injury to control room occupants. The SSCs should be designed and constructed so that the SSE would not cause such failure. The applicant should identify any differences from the recommendations of RG 1.29 and discuss the proposed classification.

RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," provides recommendations for determining the seismic design of SSCs of radioactive waste management facilities. The applicant should identify the radioactive waste management SSCs that require seismic design considerations and discuss differences from the recommendations of RG 1.143.

RG 1.151, "Instrument Sensing Lines," offers recommendations for determining the seismic design of instrument sensing lines. The applicant should identify the instrument sensing lines that require seismic design considerations and discuss differences from the recommendations of RG 1.151.

The applicant should list or otherwise clearly identify all SSCs or portions thereof that are designed for an operating-basis earthquake (OBE).

Modified Section/Title: C.I.3.2.2; System Quality Group Classification

Original Section/Title: C.I.3.2.2; System Quality Group Classification

The applicant should identify those fluid systems or portions thereof that are important to safety, as well as the applicable industry codes and standards for each pressure retaining component. See NGNP white papers ***"Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification,"*** INL/EXT-10-19509 and ***"Next Generation Nuclear Plant Probabilistic Risk Assessment,"*** INL/EXT-11-21270.

The regulations at 10 CFR 50.55a, "Codes and Standards," specify quality requirements for component groups, and RG 1.26, ***"Quality Group Classification and Standards for Water, Steam, and Radioactive Waste-Containing Components of Nuclear Power Plants,"*** describes a quality group classification system and relates it to industry codes for water- and steam containing fluid systems. RG 1.143, ***"Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light Water-Cooled Nuclear Power Plants,"*** provides recommendations regarding system quality group classification and/or standards for radioactive waste management systems, and RG 1.151, ***"Instrument Sensing Lines,"*** provides this same information for instrument sensing lines. The applicant should indicate the extent to which it has followed the recommendations of RG 1.26, RG 1.143, and RG 1.151. The applicant should identify any differences between the recommendations and its application and justify each proposed quality group classification in terms of the reliance placed on those systems that perform any of the following functions:

- Prevent or mitigate the consequences of accidents and malfunctions originating within the fission product barriers
- Permit reactor shutdown and maintenance in the safe, stable condition
- Contain radioactive material.

For such systems, the applicant should specify the proposed design features and measures that it would apply to attain a quality level equivalent to the level of the RG 1.26, RG 1.143 and RG 1.151 classifications (as applicable), including the QA programs that would be implemented. The applicant should discuss the group classification boundaries of each safety related system. The classifications should be marked/noted on drawings at valves or other appropriate locations in each fluid system where the respective classification changes in terms of appropriate designation of the NRC safety classification and special treatment for non-safety related components or, alternatively, in terms of corresponding classification notations that can be referenced with those classification groups in RG 1.26, RG 1.143, and RG 1.151, as functionally applicable to HTGR technology.

Modified Section/Title: C.I.3.3; Wind and Tornado Loadings
Original Section/Title: C.I.3.3; Wind and Tornado Loadings

Modified Section/Title: C.I.3.3.1; Wind Loadings
Original Section/Title: C.I.3.3.1; Wind Loadings
To define the design-basis wind loadings of seismic Category I structures, the applicant should provide the following:

1. Design wind velocity and its recurrence interval, the importance factor, and the exposure category

2. Methods used to transform the wind velocity into an effective pressure applied to surfaces of structures and present the results in tabular form for plant SSCs, as well as current references for the basis, including the assumptions.

The applicant should provide information showing that the failure of the facility structures or components not designed for wind loads will not affect the ability of other structures to perform their intended safety functions.

Modified Section/Title: C.I.3.3.2; Tornado Loadings
Original Section/Title: C.I.3.3.2; Tornado Loadings
The applicant should define the design-basis tornado loadings on structures that must be designed to withstand tornadoes. It should include the following specific information in the description:

1. Design parameters applicable to the design-basis tornado, including the maximum tornado velocity, the pressure differential and its associated time interval, and the spectrum and pertinent characteristics of tornado generated missiles

2. The methods used to transform the tornado loadings into effective loads on structures
    a. Methods used to transform the tornado wind into an effective pressure on exposed surfaces of structures, including consideration of geometrical configuration and physical characteristics of the structures and the distribution of wind pressure on the structures
    b. If venting of a structure is used, the methods employed to transform the tornado generated differential pressure into an effective reduced pressure
    c. The methods used to transform the tornado generated missile loadings, which are considered impactive dynamic loads, into effective loads

d. The various combinations of the above individual loadings that will produce the most adverse total tornado effect on structures.

The applicant should provide information showing that the failure of any structure or component that is not designed for tornado loads will not affect the ability of other structures to perform their intended safety functions.

Modified Section/Title: C.I.3.4; Water Level (Flood) Design

Original Section/Title: C.I.3.4; Water Level (Flood) Design

Modified Section/Title: C.I.3.4.1; Internal Flood Protection

Original Section/Title: C.I.3.4.1; Internal Flood Protection

The applicant should describe the internal flood protection measures for all SSCs whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity. The information provided in this section of the FSAR should be consistent with the information provided in FSAR Sections 2.4 and 2.5 for safe shutdown ground motion, as well as FSAR Section 3.8.4 for seismic design, which should be referenced as appropriate. The applicant should:

1. Identify and evaluate the SSCs that are safety related and must be protected against internal floods and flood conditions

2. Identify the location of safety related SSCs in relation to the internal flood levels in various areas that house safety-related SSCs

3. Identify and evaluate SSCs, if any, that may be potential sources of internal flooding (e.g., pipe breaks and cracks, tank and vessel failures, backflow through drains)

4. If flood protection is required, discuss the adequacy of techniques such as enclosures, pumping systems, drains, internal curbs, and watertight doors used to prevent flooding of safety-related systems or components. The application should identify the above mentioned techniques by using plant arrangements, layout drawings or any other acceptable method

5. Discuss the measures taken to assess the potential flooding of SSCs important to safety due to the operation of the fire protection systems and the postulated failure of piping in accordance with Section 3.6.2 of this guide. Postulated failures of non-seismic and non-tornado protected piping, tanks, and vessels should be assessed. For the purposes of the flood analysis, for each analyzed area, the assumption of the rupture of the single, worst case pipe (or non-seismic tank/vessel) can be made. For moderate energy piping that is not, seismically supported full circumferential ruptures, not just cracks, should be considered. Ways to mitigate the consequences of potential internal flooding to safety-related systems, such as drains and sump pumps should be considered in this assessment. If the postulated break occurs in a non-seismically supported system, then only seismically qualified systems should be assumed available to mitigate the effects of the analyzed break (a seismic event may have caused the initial break).

6. Discuss the risk assessment for external and internal flooding to identify potentially significant vulnerabilities to flooding. This will include an analysis of flooding during shutdown conditions. Determine if flooding consequences that result from failures of liquid-carrying systems in the proximity of essential equipment will not preclude the required functions of safety systems with a failure mode and effects analysis.

7. Identify and evaluate those safety-related systems or components, if any, that are capable of normal functions while completely or partially flooded.

8. Determine, if any safety-related equipment or components (on plant arrangement and layout drawings) are located within individual compartments or cubicles which may function as positive barriers against potential means of flooding, and if barriers or other means of physical separation are used between redundant safety-related trains. Evaluate the adequacy of such barriers. Identify potential flow paths from connected non-safety related areas to rooms that contain safety related SSCs.

9. Identify and describe the design features that will be used to mitigate the effects of internal flooding (adequate drainage, sump pumps, etc.) These features should be safety-related to ensure adequate time to bring the reactor to a safe, stable condition. Only seismically qualified systems may be assumed to be available to mitigate the effects of the flooding from non-seismic systems.

The applicant should describe the flood protection of any safety-related structure dependent on a permanent dewatering system from the effects of ground water. It should:

1. Provide a summary description of the dewatering system, including all major subsystems. The dewatering systems should be designed as a safety-related system and meet the single failure criterion requirements.

2. Describe the design bases for the functional performance requirements for each subsystem, along with the bases for selecting the system operating parameters

3. Demonstrate the system satisfies the design bases, the system's capability to withstand design basis events, and its capability to perform its safety function assuming a postulated accident with the loss of offsite power (LOOP). It should evaluate the protection against single failure in terms of piping arrangement and layout, selection of valve types and locations, redundancy of various system components, redundancy of power supplies, redundant sources of actuation signals, and redundancy of instrumentation and demonstrate that the dewatering system is protected from the effects of pipe breaks and missiles.

4. Describe the testing and inspection to be performed to verify that the system has the required capability and reliability, as well as the instrumentation and controls necessary for proper operation of the system.

Modified Section/Title: C.I.3.4.2; Analysis Procedures

Original Section/Title: C.I.3.4.2; Analysis Procedures

The applicant should describe the methods and procedures by which the static and dynamic effects of the design basis flood or groundwater conditions identified in Section 2.4 of the FSAR are applied to seismic Category I structures that are designated as providing protection against external flooding. For each seismic Category I structure that may be affected, the applicant should summarize the design-basis static and dynamic loadings, including consideration of hydrostatic loadings, equivalent hydrostatic dynamically induced loadings, coincident wind loadings, and the static and dynamic effects on foundation properties (see Section 2.5 of the FSAR).

The applicant should describe any physical models used to predict prototype performance of hydraulic structures and systems. RG 1.125, “Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants,” provides guidance if a safety-related source of cooling water is necessary. RG 1.125 addresses physical models of structures intended to protect safety-related SSCs against the effects of hydraulic forces such as floods, seiches, wave runup, etc.

Modified Section/Title: C.I.3.5; Missile Protection
Original Section/Title: C.I.3.5; Missile Protection

Modified Section/Title: C.I.3.5.1; Missile Selection and Description
Original Section/Title: C.I.3.5.1; Missile Selection and Description

Modified Section/Title: C.I.3.5.1.1; Internally Generated Missiles (Outside Reactor Building)
Original Section/Title: C.I.3.5.1.1; Internally Generated Missiles (Outside Containment)
The applicant should identify all structures, systems (or portions of systems), and components that are to be protected against damage from missiles generated internally or from co-located industrial facilities. These are the SSCs necessary to perform functions required to attain and maintain a safe shutdown condition or to mitigate the consequences of an accident. RG 1.117, "Tornado Design Classification," provides guidance on the SSCs that should be protected. The applicant should consider missiles associated with overspeed failures of rotating components (e.g., motor-driven pumps and fans), failures of high pressure system components, and gravitational missiles (e.g., falling objects resulting from a nonseismically designed SSC during a seismic event). The design bases should consider the design features provided for either continued safe operation or shutdown during all operating conditions, operational transients, and postulated accident conditions.

The applicant should provide the following information for those SSCs outside the reactor building that require protection from internally generated missiles:

1. Locations of the SSCs

2. Applicable seismic category and quality group classifications (information may be referenced from FSAR Section 3.2)

3. Sections of the FSAR where the items are described, including applicable drawings or piping and instrumentation diagrams

4. Missiles to be protected against, their sources, and the bases for their selection for analysis

5. Missile protection provided.

Applicants should evaluate the ability of the SSCs to withstand the effects of selected internally generated missiles. Examples of missiles to be considered are noted above. For protection against low trajectory turbine missiles, the protection provided should meet the guidance of Regulatory Position 3 of RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," Rev. 2.

Modified Section/Title: C.I.3.5.1.2; Internally Generated Missiles (Inside the Reactor Building)
Original Section/Title: C.I.3.5.1.2; Internally Generated Missiles (Inside Containment)
The applicant should identify all plant SSCs inside the reactor building that should be protected from internally generated missiles. These are the SSCs whose failure could lead to offsite radiological consequences or those required for safe plant shutdown. The applicant should identify credible missiles associated with overspeed failures of rotating components (e.g., pumps, fans, compressors), primary and

secondary failures of high pressure system components (e.g., reactor vessel, steam generator, intermediate heat exchanger, piping), gross failure of a control rod drive mechanism (CRDM), and gravitational effects (e.g., falling objects resulting from the movement of a heavy load or a non-seismically designed SSC during a seismic event, secondary missiles caused by a falling object striking a high-energy system).

For those SSCs important to safety inside the reactor building and that need to be protected against internally generated missiles, the applicant should provide the following information:

- Location of the SSCs
- Missiles to be protected against, their sources, and the bases for their selection for analysis
- Missile protection provided (identify SSCs protected by physical barriers and, for those protected by redundancy, demonstration of the separation and independence)
- An evaluation demonstrating the ability of the SSCs to withstand the effects of selected internally generated missiles.

Modified Section/Title: C.I.3.5.1.3; Turbine Missiles

Original Section/Title: C.I.3.5.1.3; Turbine Missiles

The applicant should provide the information listed below to demonstrate that SSCs important to safety have adequate protection against the effects of potential turbine missiles. (RG 1.117 describes examples of SSCs important to safety that should be protected.) It should provide:

1. Indication of whether the orientation of the turbine is favorable or unfavorable relative to the placement of the reactor building and other SSCs important to safety. Favorably oriented turbine generators are located such that the reactor building and all, or almost all, SSCs important to safety located outside the reactor building are excluded from the low-trajectory hazard zone described in RG 1.115, ***"Protection Against Low-Trajectory Turbine Missiles."*** This section should include the following information to justify the turbine’s orientation (information provided in other sections may be referenced as appropriate):
    a. Dimensioned plant layout drawings (plan and elevation views) with the turbine and reactor buildings clearly identified
    b. Barriers, including structural wall material strength properties and thickness
    c. SSCs important to safety in terms of location, redundancy, and independence
    d. All turbine generator units (present and future) in the vicinity of the plant being reviewed
    e. A quantitative description of the turbine generator in terms of rotor shaft, wheels/buckets/blades, steam valve characteristics, rotational speed, and turbine internals pertinent to turbine missile analyses
    f. Postulated missiles in terms of missile size, mass, shape, and exit speed for design over-speed and destructive overspeed in postulated turbine failures (description of the analysis used in estimating the missile exit speeds and identification of the direction of rotation for each turbine generator under consideration).

2. The methods, analyses, and results for the turbine missile generation probability calculations

3. Description of the inservice inspection (ISI) and testing program that will be used to maintain an acceptably low probability of missile generation

4. Demonstration of the structural capability of any barriers (or structures used as barriers) that protect SSCs to withstand turbine missiles in the event of a turbine failure.

Modified Section/Title: C.I.3.5.1.4; Missiles Generated by Tornadoes and Extreme Winds

Original Section/Title: C.I.3.5.1.4; Missiles Generated by Tornadoes and Extreme Winds

The applicant should identify all missiles generated as a result of high-speed winds such as tornadoes, hurricanes, and any other extreme winds. For selected missiles, the applicant should specify the origin (including height above plant grade), dimensions, mass, energy, velocity, trajectory, and any other parameters required to determine missile penetration. RG 1.76, ***"Design Basis Tornado and Tornado Missiles for Nuclear Power Plants,"*** provides guidance for selecting the design-basis, tornado-generated missiles.

Modified Section/Title: C.I.3.5.1.5; Site Proximity Missiles (Except Aircraft)

Original Section/Title: C.I.3.5.1.5; Site Proximity Missiles (Except Aircraft)

The applicant should identify all missile sources resulting from accidental explosions in the vicinity of the site, based on the nature and extent of nearby industrial, transportation, and military facilities (other than aircraft) identified in Sections 2.2.1–2.2.3 of the FSAR. The applicant should consider the following missile sources with respect to the site:

- Train explosions (including rocket effects)
- Truck explosions
- Ship or barge explosions
- Industrial facilities (where different types of materials are processed, stored, used, or transported)
- Pipeline explosions
- Military facilities.

The applicant should identify the SSCs listed in Section 3.5.2 of the FSAR that have the potential for unacceptable missile damage and estimate the total probability of the missiles striking a vulnerable critical area of the plant. If the total probability is greater than an order of magnitude of 10^{-7} per year, a specific missile description, including size, shape, weight, energy, material properties, and trajectory, should accompany the description of the missile effects on the SSCs.

Modified Section/Title: C.I.3.5.1.6; Aircraft Hazards

Original Section/Title: C.I.3.5.1.6; Aircraft Hazards

The applicant should provide an aircraft hazard analysis for each of the following:

- Federal airways, holding patterns, or approach patterns within 3.22 km (2 miles) of the nuclear facility
- All airports located within 8.05 km (5 statute miles) of the site
- Airports with projected operations greater than $193d^2$ ($500d^2$) movements per year located within 16.10 km (10 statute miles) of the site and greater than $386d^2$ ($1000d^2$) outside 16.10 km (10 statute miles), where d is the distance in km (statute miles) from the site
- Military installations or any airspace usage that might present a hazard to the site (for some uses, such as practice bombing ranges, it may be necessary to evaluate uses as far as 32.19 km (20 statute miles) from the site).

Hazards to the plant may be divided into accidents resulting in structural damage and accidents involving fire. These analyses should be based on the projected traffic for the facilities, the aircraft accident statistics provided in Section 2.2, and the critical areas described in Section 3.5.2 of the FSAR. The aircraft hazard analysis should provide an estimate of the total aircraft hazard probability per year. The plant design should consider aircraft accidents that could lead to radiological consequences in excess of the exposure guidelines of 10 CFR 52.79 with a probability of occurrence greater than an order of magnitude of 10^{-7} per year. The applicant should provide and justify the aircraft selected as the design-basis impact event, including its dimensions, mass (including variations along the length of the aircraft), energy, velocity, trajectory, and energy density. Section 3.5.3 of the FSAR should provide the resultant loading curves on structures.

All parameters used in these analyses should have an explicit justification. Wherever a given parameter has a range of values, this should be plainly indicated and the most conservative value used. The applicant should state a clear justification for all assumptions.

Modified Section/Title: C.I.3.5.2; Structures, Systems, and Components to Be Protected from Externally Generated Missiles

Original Section/Title: C.I.3.5.2; Structures, Systems, and Components to Be Protected from Externally Generated Missiles

The applicant should identify the SSCs that should be protected from externally generated missiles. These are the SSCs necessary for safe shutdown of the reactor facility and those whose failure could result in a significant release of radioactivity. Structures (or areas of structures), systems (or portions of systems), and components should be protected from externally generated missiles if such a missile could prevent the intended safety function. If a missile impact on a non-safety related system whose failure could degrade the intended function of a safety-related system, that system is designated for special treatment classification. The SSC under this category needs adequate separation from safety-related SSCs to prevent any failure of a non-safety-related SSC from preventing a safety-related SSC from performing its intended functions. Guidance on the SSCs that should be protected against externally generated missiles appears in Regulatory Position 2 of RG 1.13, "Spent Fuel Storage Facility Design Basis," Rev. 2; Regulatory Position C.1 of RG 1.115, ***"Protection Against Low-Trajectory Turbine Missiles,"*** Rev. 2; and Regulatory Positions 1–3 and the appendix to RG 1.117, ***Tornado Design Classification,"*** Rev.1.

Modified Section/Title: C.I.3.5.3; Barrier Design Procedures

Original Section/Title: C.I.3.5.3; Barrier Design Procedures

The applicant should provide the following information concerning the design of each structure or barrier to resist the missile hazards previously described:

- Methods used to predict local damage in the impact area, including estimation of the depth of penetration
- Methods used to estimate barrier thickness required to prevent perforation
- Methods used to predict concrete barrier potential for generating secondary missiles by spalling and scabbing effects
- Methods used to predict the overall response of the barrier and portions thereof to missile impact, including assumptions on acceptable ductility ratios and estimates of forces, moments, and shears induced in the barrier by the impact force of the missile.

Modified Section/Title: C.I.3.6; Protection against Dynamic Effects Associated with Postulated Rupture of Piping

Original Section/Title: C.I.3.6; Protection against Dynamic Effects Associated with Postulated Rupture of Piping

The applicant should describe design bases and design measures used to ensure that the reactor building and all essential equipment inside or outside the reactor building have been adequately protected against the effects of blowdown jet and reactive forces and pipe whip resulting from postulated rupture of piping located either inside or outside of the reactor building.

Modified Section/Title: C.I.3.6.1; Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside of the Reactor Building

Original Section/Title: C.I.3.6.1; Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside of Containment

The applicant should describe the plant design for protection against high- and moderate-energy fluid system piping failures outside the reactor building to ensure that such failures would not cause the loss of needed functions of systems important to safety and ensure that the reactor can be brought to a safe, stable condition in the event of such failures. Recommended actions include the following:

1. Identification of systems or components important to plant safety or shutdown that are located near high or moderate-energy piping systems and that are susceptible to the consequences of failures of these piping systems
    a. Relating the identification to predetermined piping failure locations in accordance with Section 3.6.2 of this guide and providing drawings indicating typical piping runs with failure points
    b. Identifying those conditions under which the component may still operate; and
    c. Indicate the design approach taken to protect the systems and components identified above.

2. Providing a list of high and moderate-energy lines, which includes:
    a. A description of the layout of all piping systems where physical arrangement of the piping systems provides the required protection
    b. A description of the design basis of structures and compartments used to protect nearby essential systems or components
    c. A description of the arrangements to ensure the operability of safety features where neither separation nor protective enclosures are practical.

3. Providing a failure mode and effects analysis to verify that the consequences of failures of high and moderate energy lines do not affect the ability to the plant to a safe, stable condition, including:
    a. Identification of the locations and types of failures considered (e.g., circumferential or longitudinal pipe breaks, through-wall cracks, leakage cracks) and the dynamic effects associated with the failures (e.g., pipe whip, jet impingement). The potential effects of secondary missiles should also be considered.
    b. An explanation of the assumptions made in the analyses with respect to the following:
    c. Availability of offsite power
    d. Failure of single active components in systems used to mitigate the consequences of the piping failure
    e. Special provisions applicable to certain dual-purpose systems
    f. Use of available systems to mitigate the consequences of the piping failure
    g. A description of the effects of piping failures in systems not designed to seismic Category I standards on essential systems and components, assuming postulated accident conditions
    h. A description of the environmental effects of pipe rupture (e.g., temperature, humidity, pressure, spray-wetting, flooding), including potential transport of the steam environment to other rooms or compartments, and the subsequent effects on the functional performance of essential electrical equipment and instrumentation
    i. A description of the effects of postulated failures on habitability of the control room and access to areas important to safe control of post-accident operations.

Modified Section/Title: C.I.3.6.2; Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Pipe Breaks Inside and Outside of the Reactor Building

Original Section/Title: C.I.3.6.2; Determination of Rupture Locations and Dynamic Effects Associated with the Postulated

The applicant should describe the criteria for determining the location and configuration of postulated breaks and cracks in high- and moderate-energy piping inside and outside of the reactor building; the methods used to define the jet thrust reaction at the break or crack location and the jet impingement loading on adjacent safety-related SSCs; and the design criteria for pipe whip restraints, jet impingement barriers and shields, and guard pipes. If requested as-built information is not available at the time of the application, the applicant should provide current design information, representative, or bounding information. The applicant should in the application propose an appropriate method (e.g., ITAAC, license condition, FSAR) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

Modified Section/Title: C.I.3.6.2.1; Criteria Used to Define Break and Crack Location and Configuration

Original Section/Title: C.I.3.6.2.1; Criteria Used to Define Break and Crack Location and Configuration

The applicant should provide the criteria used to determine the location and configuration of postulated breaks and cracks in those high- and moderate-energy piping systems for which separation or enclosure cannot be achieved. In the case of reactor building penetration piping, in addition to the material requested above, the applicant should provide details of the reactor building penetration identifying all process pipe welds, access for ISI of welds, points of fixity, and points of geometric discontinuity. The applicant should discuss the implementation of criteria for defining pipe break and crack locations and configurations and provide the resulting number and location of design-basis breaks and cracks. The discussion should also include the postulated rupture orientation (such as circumferential and/or longitudinal break) for each postulated design-basis break location.

Modified Section/Title: C.I.3.6.2.2; N/A

Original Section/Title: C.I.3.6.2.2; Guard Pipe Assembly Design Criteria

Modified Section/Title: C.I.3.6.2.3; Analytical Methods to Define Forcing Functions and Response Models

Original Section/Title: C.I.3.6.2.3; Analytical Methods to Define Forcing Functions and Response Models

The applicant should describe the analytical methods for defining the forcing functions to be used for the pipe whip dynamic analyses. This description should include direction, thrust coefficients, rise time, magnitude, duration, and initial conditions that adequately represent the jet stream dynamics and the system pressure differences. Pipe restraint rebound effects should be included if appropriate. The applicant should provide diagrams of typical mathematical models used for the dynamic response analysis and present and justify all dynamic amplification factors to be used. The discussion should cover the implementation of the methods used for the pipe whip dynamic analyses to demonstrate the acceptability of the analysis results, including the jet thrust and impingement functions and the pipe whip dynamic effects.

Modified Section/Title: C.I.3.6.2.4; Dynamic Analysis Methods to Verify Integrity and Operability

Original Section/Title: C.I.3.6.2.4; Dynamic Analysis Methods to Verify Integrity and Operability

The applicant should describe the analytical methods, including the details of jet expansion modeling, that it will use to evaluate the jet impingement effects and loading effects applicable to nearby SSCs resulting from postulated pipe breaks and cracks. In addition, the applicant should provide the analytical methods used to verify the integrity and operability of these impacted SSCs under postulated pipe rupture loads. In the case of piping systems that include pipe whip restraints, the applicant should provide loading combinations and design criteria for the restraints along with a description of the typical restraint configuration to be used. The applicant should discuss the implementation of the dynamic analysis methods used to verify the integrity and operability of the impacted SSCs and should demonstrate the design adequacy of these SSCs to ensure that pipe whip or jet impingement loading will not impair their design intended functions to an unacceptable level of integrity or operability.

Modified Section/Title: C.I.3.6.2.5; Implementation of Criteria Dealing with Special Features

Original Section/Title: C.I.3.6.2.5; Implementation of Criteria Dealing with Special Features

The applicant should discuss the implementation of criteria dealing with special features, such as an augmented ISI program or use of special protective devices (such as pipe whip restraints). The discussion should include diagrams showing the final configurations, locations, and orientations of the special features in relation to break locations in each piping system.

Modified Section/Title: C.I.3.6.3; Leak-before-Break Evaluation Procedures

Original Section/Title: C.I.3.6.3; Leak-before-Break Evaluation Procedures

The applicant should describe the analyses used to eliminate from the design basis the dynamic effects of certain pipe ruptures and demonstrate that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping. The applicant should give adequate consideration to direct and indirect pipe failure mechanisms and other degradation sources that could challenge the integrity of piping. Information to be provided includes the following:

1. List of the piping systems included in the leak-before-break (LBB) evaluation, including:
   a. Identification of the types of as-built materials and material specifications used for base metal, weldments, nozzles, and safe ends;
   b. The material properties, including the following:

2. Toughness (J-R curves) and tensile (stress-strain curves) data at temperatures near the upper range of normal plant operation long-term effects attributable to thermal aging yield strength and ultimate strength
   a. The welding process/method (e.g., submerged arc welding) used in the weld(s).

3. If the as-built materials and material specifications are not available at the time of the application, representative and bounding materials and associated specifications may be used in the LBB analysis to be submitted with the application. The applicant should in the application propose an appropriate method (e.g., ITAAC, license condition, FSAR) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

4. Discussion of the design-basis loads for each piping system, including:
   a. Provision of as-built drawing(s) of pipe geometry (e.g., piping isometric drawings). Identify locations of support and their characteristics (such as gaps) and of the analysis nodal points. If as-built drawings are not available at the time of the application, design piping isometric drawings may be submitted. The applicant should in the application provide an appropriate method (e.g., ITAAC, license condition, FSAR) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.
   b. Locations and weights of components such as valves
   c. Snubber reliability
   d. The sources (e.g., thermal, deadweight, seismic, and seismic anchor movement), types (e.g., forces, bending and torsional moments), and magnitudes of applied loads and the method of combination

5. If as-built drawings, weight of components, and analysis loads are not available at the time of the application, design piping isometric drawings and analysis may be submitted. The applicant should in

the application propose an appropriate method (e.g., ITAAC, license condition, FSAR) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

6. Deterministic fracture mechanics analysis. This analysis should identify the locations that have the least favorable combination of stress and material properties for base metal, weldment, and safe ends and should postulate a through-wall leakage flaw at these locations. The analysis should demonstrate that the leakage flaw has sufficient safety margin with respect to the critical crack size under various loading combinations, that leakage flaw growth would be stable, and that the final flaw size would be limited such that a double-ended pipe break would not occur.

7. Leak-rate evaluation to demonstrate that there is sufficient margin between the leak rate from the leakage flaw and the detection capability of the leak-rate detection systems. This evaluation should demonstrate that the leak-rate detection systems are sufficiently reliable, redundant, and sensitive to provide adequate margin on the detection of unidentified leakage.

8. Evaluation of creep and creep-fatigue and demonstration that the piping material is not susceptible to brittle cleavage-type failure over the full range of system operating temperatures

9. Identification of measures taken to improve the corrosion resistance of piping that may be subject to intrusion or ingress of impurities

10. Demonstration that the piping systems under LBB evaluation do not have a history of fatigue cracking or failure including:
    a. A showing that the potential for pipe rupture attributable to thermal and mechanical induced fatigue is unlikely
    b. A demonstration that there is adequate mixing of high- and low-temperature fluids so that there is no potential for significant cyclic thermal stresses
    c. A showing that there is no significant potential for vibration-induced fatigue cracking or failure

11. Demonstration that the following indirect failure mechanisms (as defined in the FSAR) are remote causes of pipe failure:
    a. Seismic events
    b. System overpressurization attributable to accidents resulting from human error
    c. Fires
    d. Flooding causing electrical and mechanical control systems to malfunction
    e. Missiles from equipment failure
    f. Damage from moving equipment
    g. Failures of SSCs in proximity to the piping

12. Description of any inspection programs developed for piping systems that are qualified for LBB

13. Demonstration that the piping and weld materials are not susceptible to stress-corrosion cracking, intergranular stress-corrosion cracking (IGSCC), transgranular stress-corrosion cracking (TGSCC), and other degradation mechanisms specific to the helium environment.

Modified Section/Title: C.I.3.7; Seismic Design

Original Section/Title: C.I.3.7; Seismic Design

Modified Section/Title: C.I.3.7.1; Seismic Design Parameters

Original Section/Title: C.I.3.7.1; Seismic Design Parameters

The applicant should discuss the seismic design parameters (design ground motion, percentage of critical damping values, supporting media for seismic Category I structures) that are used as input parameters to the seismic analysis of seismic Category I SSCs for the OBE and SSE.

Modified Section/Title: C.I.3.7.1.1; Design Ground Motion

Original Section/Title: C.I.3.7.1.1; Design Ground Motion

The applicant should specify the earthquake ground motion (GRMS and/or ground motion time histories) exerted on the structure or the soil-structure interaction (SSI) system based on seismicity and geologic conditions at the site, expressed such that it can be applied to dynamic analysis of seismic Category I SSCs. The earthquake ground motion should consider the three components of design ground motions, two horizontal and one vertical, for the OBE and SSE. For the SSI system, this ground motion should be consistent with the free-field ground motion at the site. Additional guidance is provided in RG 1.208, ***"A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion."***

Modified Section/Title: C.I.3.7.1.1.1; Design Ground Motion Response Spectra

Original Section/Title: C.I.3.7.1.1.1; Design Ground Motion Response Spectra

The applicant should provide design GRMS for the OBE and SSE, which are consistent with those defined based on the guidelines in Section 2.5 of the FSAR. In general, these response spectra are developed for 5 percent damping. If the ground response spectra are different from the generic ground response spectra, such as the response criteria provided in RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," the applicant should describe the procedures to calculate the response spectra for each damping ratio to be used in the design of seismic Category I SSCs and the procedures for the development of target power spectral density (PSD). The applicant should also provide bases to justify its choices to apply the response spectra either at the finished grade in the free field or at the various foundation locations of seismic Category I structures.

For COL plants, the applicant should develop a site specific ground spectrum from a PSHA in accordance with RG 1.208, ***"A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,"*** or the equivalent. (However, NUREG/CR 6728, "Technical Basis for Regulatory Guidance on Design Ground Motions: Hazard and Risk Consistent Ground Motion Spectra Guidelines," issued in 2001, does not specify spectrum matching for damping other than 5 percent and target PSD enveloping. Another option, is to develop the target PSD for ground response spectra other than the RG 1.60 ground response spectra, including ground response spectra developed from PSHA.)

Modified Section/Title: C.I.3.7.1.1.2; Design Ground Motion Time History

Original Section/Title: C.I.3.7.1.1.2; Design Ground Motion Time History

The applicant should describe how it selected or developed the earthquake ground motion time history (actual or synthetic). For the time history analyses, the applicant should provide the response spectra derived from actual or synthetic earthquake time-motion records. For each of the damping values to be used in the design of SSCs, this description should include a comparison of the response spectra obtained in the free field at the finished grade level and at the foundation level (obtained from an appropriate time history at the base of the SSI system) with the design response spectra. Alternatively, if the design response spectra for the OBE and SSE are applied at the foundation levels of seismic Category I structures in the free field, the applicant should provide a comparison of the free field response spectra at the foundation level (derived from an actual or synthetic time history) with the design response spectra for each of the damping values to be used in the design. If the seismic analysis is using the synthetic time history (three components), the applicant should demonstrate that (1) the cross-correlation coefficients between the three components of the design ground motion time histories are within the criteria of Section

3.7.1 of this guide, and (2) the PSD calculated from these three components envelop the target PSD developed based on the guidance in Section 3.7.1 of this guide. Also, the applicant should identify the period intervals at which the spectra values were calculated.

For COL plants, the discussion of PSHA in Section 3.7.1 applies.

Modified Section/Title: C.I.3.7.1.2; Percentage of Critical Damping Values
Original Section/Title: C.I.3.7.1.2; Percentage of Critical Damping Values
The applicant should specify the percentage of critical damping values used for seismic Category I SSCs and soil for both the OBE and SSE (e.g., damping ratios for the type of construction or fabrication). Also, the applicant should compare the damping ratios assigned to SSCs with the acceptable damping ratios provided in RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," and include the bases for any proposed damping ratios that differ from those given in RG 1.61 for the proposed soil damping.

Modified Section/Title: C.I.3.7.1.3; Supporting Media for Seismic Category I Structures
Original Section/Title: C.I.3.7.1.3; Supporting Media for Seismic Category I Structures
For each seismic Category I structure, the applicant should describe the supporting media, including foundation embedment depth, depth of soil over bedrock, soil layering characteristics, dimensions of the structural foundation, total structural height, and soil properties of each soil layer, such as shear wave velocity, shear modulus, soil material damping, and density. The applicant should use this information to evaluate the suitability of either a finite element or lumped soil-spring approach for modeling soil foundation in the SSI analysis.

Modified Section/Title: C.I.3.7.2; Seismic System Analysis
Original Section/Title: C.I.3.7.2; Seismic System Analysis
The applicant should discuss the seismic system analyses applicable to seismic Category I SSCs.

Modified Section/Title: C.I.3.7.2.1; Seismic Analysis Methods
Original Section/Title: C.I.3.7.2.1; Seismic Analysis Methods
For all seismic Category I SSCs, the applicant should identify and describe the applicable seismic analysis methods (e.g., response spectrum analysis, modal time history analysis, direct integration time history analysis, frequency domain time history analysis, and equivalent static load analysis). The discussion should address how the dynamic system analysis method covers foundation torsion, rocking, and translation. The applicant should indicate which analysis method it will use for seismic Category I and non-seismic Category I (seismic Category II and non-seismic) SSCs. Seismic Category II SSCs are defined as SSCs that perform no safety-related function and the continued function of which is not required. However, the design of these SSCs should ensure that the SSE does not cause unacceptable failure of or interaction with seismic Category I items. The applicant should describe the types of soil-structure system models to be analyzed and which analysis methods it will use. The applicant should also indicate the manner in which the seismic dynamic analysis considers the maximum relative displacement among supports.

The applicant should indicate other significant effects accounted for in the seismic dynamic analysis, such as hydrodynamic effects and non-linear response. If the applicant uses tests or empirical methods in lieu of analysis for any seismic Category I SSCs, it should provide the testing procedure, load levels, and acceptance bases. If these tests or empirical methods are not complete at the time the application is filed, the applicant should describe the implementation program, including milestones. These tests or empirical methods should be submitted to staff for review and approval prior to issuance of license. When a nonlinear analysis is performed, the applicant should provide specific information regarding consideration of inelastic/non-linear behavior of SSCs.

Modified Section/Title: C.I.3.7.2.2; Natural Frequencies and Responses

Original Section/Title: C.I.3.7.2.2; Natural Frequencies and Responses

When the applicant performs modal time history analyses and/or response spectrum analyses, it should provide the modal properties (natural frequencies, participation factors, mode shapes, modal masses, and percentage of cumulative mass). For all seismic system analyses performed (modal time history analyses and response spectrum analyses), the applicant should provide seismic responses (maximum absolute nodal accelerations, maximum displacement relative to the top of foundation mat, and maximum member forces and moments) for major seismic Category I structures. Also, the applicant should include the in-structure response spectra at major seismic Category I equipment elevations and points of support, generated from the system dynamic response analyses.

Modified Section/Title: C.I.3.7.2.3; Procedures Used for Analytical Modeling

Original Section/Title: C.I.3.7.2.3; Procedures Used for Analytical Modeling

The applicant should describe the types of model (finite element model, lumped-mass stick model, hybrid model, etc.) used for seismic Category I structures. The description should include the criteria and procedures used for modeling in the seismic system analyses and indicate how foundation torsion, rocking, and translation are modeled for the seismic system analyses. The applicant should include criteria and bases used to determine whether a component or structure should be analyzed as part of a system analysis or independently as a subsystem.

Modified Section/Title: C.I.3.7.2.4; Soil-Structure Interaction

Original Section/Title: C.I.3.7.2.4; Soil-Structure Interaction

As applicable, the applicant should provide definition and location of the control motion and modeling methods of SSI analysis used in the seismic system analysis, as well as their bases. This section should include information on (1) extent of embedment, (2) depth of soil over bedrock, (3) layering of soil strata, and (4) strain-dependent shear modulus (reduction curves and hysteretic damping ratio relations) appropriate for each layer of the site soil column. If applicable, the applicant should specify the procedures by which strain-dependent soil properties (e.g., hysteretic damping, shear modulus, and pore pressure) and layering are incorporated into the site response analyses used to generate free-field ground motions, as well as how these soil properties are used when the SSI analysis incorporates the variations of soil properties. The applicant should show how the upper and lower bound iterated soil properties used in the SSI analyses are consistent with those generated from the free-field analyses (if necessary, by referencing the information in FSAR Section 3.7.1.3). The discussion should specify the type of soil foundation model (lumped soil spring model, finite element model, etc.). If using the finite element model, the applicant should specify the criteria for determining the location of the bottom and side boundaries of the analysis model as applicable. The applicant should also specify procedures used to account for effects of adjacent structures (through soil structure-to-structure interaction), if any, on structural response in the SSI analysis.

If it is necessary to apply a forcing function at boundaries of the soil foundation model to simulate earthquake motion for performing a dynamic analysis for the soil-structure system, the applicant should discuss the theories and procedures used to generate the forcing function system such that response motion of the soil media in the free field at the site is identical to the design ground motion, and such that these boundary effects do not influence the SSI analyses.

This section should describe the procedures for incorporating strain dependent soil properties, embedded effects, layering, and variation of soil properties into the analysis. If lumped spring-dashpot methods are used, the discussion should explain the theories and methods for calculating the soil springs and the suitability of such methods for the particular site conditions and the parameters used in the SSI analysis. Also, the applicant should show how the analysis accounts for frequency-dependent soil properties of the lumped spring-dashpot models for different modes of response.

The discussion should include any other methods used for SSI analysis or the basis for not using SSI analysis.

Modified Section/Title: C.I.3.7.2.5; Development of Floor Response Spectra

Original Section/Title: C.I.3.7.2.5; Development of Floor Response Spectra

The applicant should describe the procedures, basis, and justification for developing floor response spectra considering the three components of earthquake motion, two horizontal and one vertical, specified in RG 1.122, "Development of Floor Design Response Spectra Seismic Design of Floor-Supported Equipment or Components." If using a single artificial time history analysis method to develop floor response spectra, the applicant should demonstrate that (1) provisions of RG 1.122, including peak broadening requirements, apply, (2) response spectra of the artificial time history to be employed in the free field envelop the free-field design response spectra for all damping values actually used in the response spectra, and (3) the PSD generated from the time history envelops the target PSD. If the applicant applies multiple time histories to generate floor response spectra, it should provide the basis for the methods used to account for uncertainties in parameters. If the applicant uses a modal response spectrum analysis method to develop floor response spectra, it should provide the basis for the method's conservatism and equivalence to a time history method.

For COL plants, the discussion of PSHA in Section 3.7.1.1.1 of this guide applies.

Modified Section/Title: C.I.3.7.2.6; Three Components of Earthquake Motion

Original Section/Title: C.I.3.7.2.6; Three Components of Earthquake Motion

The applicant should indicate the extent to which procedures for considering the three components of earthquake motion in determining seismic response of SSCs conform to RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," and provide suitable justifications for any exceptions to this guidance.

Modified Section/Title: C.I.3.7.2.7; Combination of Modal Responses

Original Section/Title: C.I.3.7.2.7; Combination of Modal Responses

When the applicant uses a modal time history analysis method and/or a response spectrum analysis method to calculate seismic response of SSCs, it should describe the procedure for combining modal responses (i.e., shears, moments, stresses, deflections, and accelerations), including that for modes with closely spaced frequencies. Also, the description should indicate the extent to which the applicant has followed the recommendations of RG 1.92, including those applicable to adequate consideration of high-frequency modes to combine modal responses.

Modified Section/Title: C.I.3.7.2.8; Interaction of Non-Seismic Category I Structures with Seismic Category I Structures

Original Section/Title: C.I.3.7.2.8; Interaction of Non-Seismic Category I Structures with Seismic Category I Structures

This section should describe the location of all plant structures (seismic Category I, seismic Category II, and nonseismic structures), including the distance between structures and the height of each structure. The description should provide the design criteria used to account for seismic motion of non-seismic Category I (seismic Category II and non-seismic) structures, or portions thereof, in the seismic design of seismic Category I structures or parts thereof. The applicant should describe the seismic design of non-seismic

Category I structures whose continued function is not required, but whose failure could adversely affect the safety function of SSCs or result in incapacitating injury to control room occupants. The description should include the design criteria that will be applied to ensure protection of seismic Category I structures from structural failure of non-Category I structures as a result of seismic effects.

Modified Section/Title: C.I.3.7.2.9; Effects of Parameter Variations on Floor Response Spectra

Original Section/Title: C.I.3.7.2.9; Effects of Parameter Variations on Floor Response Spectra

This section should describe the procedures that the applicant will use to consider effects of expected variations of structural properties, damping values, soil properties, and uncertainties attributable to modeling of soil structure systems on floor response spectra and time histories.

Modified Section/Title: C.I.3.7.2.10; Use of Constant Vertical Static Factors

Original Section/Title: C.I.3.7.2.10; Use of Constant Vertical Static Factors

Where applicable, the applicant should identify and justify the application of equivalent static factors as vertical response loads for the seismic design of seismic Category I SSCs in lieu of using the response loads generated from a vertical seismic system dynamic analysis method.

Modified Section/Title: C.I.3.7.2.11; Method Used To Account for Torsional Effects

Original Section/Title: C.I.3.7.2.11; Method Used To Account for Torsional Effects

The applicant should describe the method used to consider torsional effects in the seismic analysis of seismic Category I structures, including evaluation and justification of static factors or any other approximate methods used (in lieu of a combined vertical, horizontal, and torsional system dynamic analysis) to account for torsional accelerations in seismic design of seismic Category I structures. Also, the applicant should describe the method used to consider the torsional effects attributable to accidental eccentricities for each seismic Category I structure.

Modified Section/Title: C.I.3.7.2.12; Comparison of Responses

Original Section/Title: C.I.3.7.2.12; Comparison of Responses

Where the applicant uses both response spectrum analysis and time history analysis methods, the applicant should provide the responses obtained from both methods at selected points in major seismic Category I structures, together with a discussion comparing the responses.

Modified Section/Title: C.I.3.7.2.13; Methods for Seismic Analysis of Dams

Original Section/Title: C.I.3.7.2.13; Methods for Seismic Analysis of Dams

The applicant should describe the analytical methods and procedures to be used for seismic analysis of seismic Category I concrete dams, including assumptions made, models developed, boundary conditions used, analysis methods used, hydrodynamic effects considered, and procedures by which the analysis incorporates strain dependent material properties of foundations.

Modified Section/Title: C.I.3.7.2.14; Determination of Dynamic Stability of Seismic Category I Structures

Original Section/Title: C.I.3.7.2.14; Determination of Dynamic Stability of Seismic Category I Structures

The applicant should describe the dynamic methods and procedures used to determine dynamic stability (overturning, sliding, and floatation) of seismic Category I structures.

Modified Section/Title: C.I.3.7.2.15; Analysis Procedure for Damping

Original Section/Title: C.I.3.7.2.15; Analysis Procedure for Damping

The applicant should describe the procedure used to account for damping in various elements of a soil-structure system model.

Modified Section/Title: C.I.3.7.3; Seismic Subsystem Analysis
Original Section/Title: C.I.3.7.3; Seismic Subsystem Analysis
This section of the FSAR covers civil structure-related subsystems such as platforms, trusses, buried piping, conduit, tunnels, dams, dikes, above-ground tanks, and the like. Section 3.9.2 of this guide covers the seismic analysis of mechanical subsystems (such as piping, mechanical components, and heat transport system).

Modified Section/Title: C.I.3.7.3.1; Seismic Analysis Methods
Original Section/Title: C.I.3.7.3.1; Seismic Analysis Methods
This section should describe analysis methods to be used for seismic analysis of seismic Category I subsystems. The applicant should provide the information requested in Section 3.7.2 of this guide, but apply it to seismic Category I subsystems. The description should include the basis for using the equivalent static load method of analysis, if applicable, and the procedures for determining equivalent static loads.

Modified Section/Title: C.I.3.7.3.2; Procedures Used for Analytical Modeling
Original Section/Title: C.I.3.7.3.2; Procedures Used for Analytical Modeling
This section should provide the criteria and procedures used for modeling seismic subsystems. The applicant should confirm the use of criteria and bases described in Section 3.7.2 of this guide to determine whether a component or structure should be independently analyzed as a subsystem.

Modified Section/Title: C.I.3.7.3.3; Analysis Procedure for Damping
Original Section/Title: C.I.3.7.3.3; Analysis Procedure for Damping
This section should provide the information requested in Section 3.7.2 of this guide but as it pertains to seismic Category I subsystems.

Modified Section/Title: C.I.3.7.3.4; Three Components of Earthquake Motion
Original Section/Title: C.I.3.7.3.4; Three Components of Earthquake Motion
The applicant should provide the information requested in Section 3.7.2 of this guide but as it pertains to seismic Category I subsystems.

Modified Section/Title: C.I.3.7.3.5; Combination of Modal Responses
Original Section/Title: C.I.3.7.3.5; Combination of Modal Responses
The applicant should provide the information requested in Section 3.7.2 of this guide but as it pertains to seismic Category I subsystems.

Modified Section/Title: C.I.3.7.3.6; Use of Constant Vertical Static Factors
Original Section/Title: C.I.3.7.3.6; Use of Constant Vertical Static Factors
The applicant should provide the information requested in Section 3.7.2 of this guide but as it pertains to seismic Category I subsystems.

Modified Section/Title: C.I.3.7.3.7; Buried Seismic Category I Piping, Conduits, and Tunnels
Original Section/Title: C.I.3.7.3.7; Buried Seismic Category I Piping, Conduits, and Tunnels
This section should describe seismic criteria and methods for considering effects of earthquakes on buried piping, conduits, tunnels, and auxiliary systems. These criteria include compliance characteristics of soil media; dynamic pressures; seismic wave passage; and settlement attributable to earthquake and differential movements at support points, penetrations, and entry points into other structures provided with anchors.

Modified Section/Title: C.I.3.7.3.8; Methods for Seismic Analysis of Category I Concrete Dams

Original Section/Title: C.I.3.7.3.8; Methods for Seismic Analysis of Category I Concrete Dams

The applicant should describe the analytical methods and procedures to be used for seismic analysis of seismic Category I concrete dams, including assumptions made, models developed, boundary conditions used, analysis methods used, hydrodynamic effects considered, and procedures by which the analysis incorporates strain dependent material properties of foundations.

Modified Section/Title: C.I.3.7.3.9; Methods for Seismic Analysis of Aboveground Tanks

Original Section/Title: C.I.3.7.3.9; Methods for Seismic Analysis of Aboveground Tanks

The applicant should provide seismic criteria and analysis methods that consider hydrodynamic forces, tank flexibility, SSI, and other pertinent parameters for seismic analysis of seismic Category I above-ground tanks.

Modified Section/Title: C.I.3.7.4; Seismic Instrumentation

Original Section/Title: C.I.3.7.4; Seismic Instrumentation

This section should discuss the proposed instrumentation system for measuring the effects of an earthquake.

Modified Section/Title: C.I.3.7.4.1; Comparison with Regulatory Guide 1.12

Original Section/Title: C.I.3.7.4.1; Comparison with Regulatory Guide 1.12

The applicant should discuss the proposed seismic instrumentation program and compare it with the seismic instrumentation guidelines of RG 1.12, "Instrumentation for Earthquakes." The applicant should provide the bases for elements of the proposed seismic instrumentation program that differ from those recommended in that regulatory guide.

Modified Section/Title: C.I.3.7.4.2; Location and Description of Instrumentation

Original Section/Title: C.I.3.7.4.2; Location and Description of Instrumentation

This section should describe locations of seismic instrumentation such as triaxial peak accelerographs, triaxial time history accelerographs, and triaxial response spectrum recorders that will be installed in selected seismic Category I structures and components. The description should specify the bases for selection of the seismic instrumentation and installation locations and discuss the extent to which the instrumentation will be used to verify seismic analyses following an earthquake.

Modified Section/Title: C.I.3.7.4.3; Control Room Operator Notification

Original Section/Title: C.I.3.7.4.3; Control Room Operator Notification

This section should describe the procedures to be followed to inform the control room operator of the peak acceleration level, cumulative absolute velocity, and input response spectra values shortly after occurrence of an earthquake. It should include the bases for establishing predetermined values for activating the readout of the seismic instrumentation to the control room operator.

Modified Section/Title: C.I.3.7.4.4; Comparison with Regulatory Guide 1.166

Original Section/Title: C.I.3.7.4.4; Comparison with Regulatory Guide 1.166

The applicant should discuss the response procedure immediately after an earthquake and compare it with RG 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions." The discussion should include the bases for elements of the response procedure that differ from those of the guidelines in RG 1.166. Provide suitable justification for any exceptions/deviations from guidance in RG 1.166.

Modified Section/Title: C.I.3.7.4.5; Instrument Surveillance
Original Section/Title: C.I.3.7.4.5; Instrument Surveillance
The applicant should discuss requirements for instrument surveillance testing and calibration pertaining to instrument operability and reliability.

Modified Section/Title: C.I.3.7.4.6; Program Implementation
Original Section/Title: C.I.3.7.4.6; Program Implementation
If the details of the seismic instrumentation implementation plan are not available at the time the application is prepared, the applicant should provide sufficient detail for the staff to be able to assess the adequacy of the program implementation.

Modified Section/Title: C.I.3.8; Design of Category I Structures
Original Section/Title: C.I.3.8; Design of Category I Structures

Modified Section/Title: C.I.3.8.1; Concrete Reactor Building
Original Section/Title: C.I.3.8.1; Concrete Containment
This section should provide the following information on concrete reactor buildings and on concrete portions of steel/concrete reactor buildings:

1. Physical description
2. Applicable design codes, standards, and specifications
3. Loading criteria, including loads and load combinations
4. Design and analysis procedures
5. Structural acceptance criteria
6. Materials, quality control programs, and special construction techniques
7. Testing and ISI programs, including milestones.

Modified Section/Title: C.I.3.8.1.1; Description of the Reactor Building
Original Section/Title: C.I.3.8.1.1; Description of the Containment
The applicant should define the primary structural aspects and elements relied on to perform the reactor building function by providing a physical description of the concrete reactor building, including plan and section views. The description should include the geometry of the concrete reactor building or concrete portions of steel/concrete reactor buildings, including plan views at various elevations and sections in at least two orthogonal directions. The applicant should describe the arrangement of the reactor building and the relationship and interaction of the reactor building structure with its surrounding structures and with its interior compartments. The discussion should also explain the effect these structures have on the design boundary conditions and expected structural behavior of the reactor building when subjected to design loads.

Modified Section/Title: C.I.3.8.1.2; Applicable Codes, Standards, and Specifications
Original Section/Title: C.I.3.8.1.2; Applicable Codes, Standards, and Specifications
The applicant should provide design codes, standards, specifications, regulations, GDC [PDC], Regulatory guides, and other industry standards applicable to HTGR technology used in the design,

fabrication, construction, testing, and ISI of the reactor building. For each document, the applicant should identify the specific edition, date, or addenda.

Modified Section/Title: C.I.3.8.1.3; Loads and Load Combinations

Original Section/Title: C.I.3.8.1.3; Loads and Load Combinations

The applicant should discuss loads and load combinations utilized in the design of the reactor building structure, and/or the specific edition, date, or addenda of design codes, standards, specifications, regulations, GDC [PDC], regulatory guides, and other industry standards applicable to HTGR technology. The applicant should discuss various combinations of loads that are normally postulated, such as normal operating loads with severe environmental and abnormal loads.

The discussion should include any other site-related or plant-related loads and load combinations applicable to the reactor building. Examples of such loads include those induced by floods, potential aircraft crashes, explosive hazards in proximity to the site, and missiles generated from activities of nearby military installations or turbine failures.

Modified Section/Title: C.I.3.8.1.4; Design and Analysis Procedures

Original Section/Title: C.I.3.8.1.4; Design and Analysis Procedures

The applicant should describe the design and analysis method used for the reactor building, including key assumptions and the basis for selection of structural models and boundary conditions, with emphasis on the extent of compliance with requirements of the ASME Code applicable to HTGR technology, and/or specific edition, date, or addenda of design codes, standards, specifications, regulations, PDC, regulatory guides, and other industry standards applicable to HTGR technology. It should reference all computer programs used to permit identification with available published programs and describe proprietary computer programs in sufficient detail to establish their applicability and the method for validating them. The applicant should discuss the effects of seismic tangential (membrane) shears and provide analysis results of the effects of expected variation in assumptions and material properties.

Modified Section/Title: C.I.3.8.1.5; Structural Acceptance Criteria

Original Section/Title: C.I.3.8.1.5; Structural Acceptance Criteria

The applicant should specify the acceptance criteria relating to stresses, strains, gross deformations, and other parameters that quantitatively identify margins of safety, with emphasis on the extent of compliance with the ASME Code applicable to HTGR technology, and/or to the specific edition, date, or addenda of design codes, standards, specifications, regulations, PDC, regulatory guides, and other industry standards applicable to HTGR technology.

Modified Section/Title: C.I.3.8.1.6; Materials, Quality Control, and Special Construction Techniques

Original Section/Title: C.I.3.8.1.6; Materials, Quality Control, and Special Construction Techniques

The applicant should identify materials used in the construction of the reactor building, with emphasis on the extent of compliance with Articles of the ASME Code applicable to HTGR technology and/or to the specific edition, date, or addenda of design codes, standards, specifications, regulations, PDC, regulatory guides, and other industry standards applicable to HTGR technology. This section should also include a summary of the engineering properties of the materials of construction.

The applicant should describe the quality control program for reactor building fabrication and construction, with emphasis on the extent of compliance with the ASME Code applicable to HTGR technology and/or the specific edition, date, or addenda of design codes, standards, specifications, regulations, PDC, regulatory guides, and other industry standards applicable to HTGR technology. The description should include the extent to which the quality control program covers the examination of materials, including tests to determine the physical properties of material and the combination of materials used for construction. It should also describe the extent to which the quality control program covers the examination of placement of material, erection tolerances, reinforcement, and the prestressing system, as applicable.

The applicant should also identify and describe special, new, or unique construction techniques and the effects that those techniques may have on the structural integrity of the completed reactor building.

Modified Section/Title: C.I.3.8.1.7; Testing and Inservice Inspection Requirements
Original Section/Title: C.I.3.8.1.7; Testing and Inservice Inspection Requirements
The applicant should describe the testing and ISI, including milestones, for the reactor building, with emphasis on the extent of compliance with the ASME Code applicable to HTGR technology and/or the specific edition, date, or addenda of design codes, standards, specifications, regulations, PDC, regulatory guides, and other industry standards applicable to HTGR technology. Additionally, the applicant should define the objectives of the tests, as well as the acceptance criteria for the results, and discuss the extent of additional testing and ISI, including milestones, if it is using new or previously untried design approaches.

Modified Section/Title: C.I.3.8.2; N/A
Original Section/Title: C.I.3.8.2; Steel Containment
Not applicable

Modified Section/Title: C.I.3.8.2.1; N/A
Original Section/Title: C.I.3.8.2.1; Description of the Containment
Not Applicable

Modified Section/Title: C.I.3.8.2.2; N/A
Original Section/Title: C.I.3.8.2.2; Applicable Codes, Standards, and Specifications
Not Applicable

Modified Section/Title: C.I.3.8.2.3; N/A
Original Section/Title: C.I.3.8.2.3; Loads and Load Combinations
Not Applicable

Modified Section/Title: C.I.3.8.2.4; N/A
Original Section/Title: C.I.3.8.2.4; Design and Analysis Procedures
Not Applicable

Modified Section/Title: C.I.3.8.2.5; N/A
Original Section/Title: C.I.3.8.2.5; Structural Acceptance Criteria
Not Applicable

Modified Section/Title: C.I.3.8.2.6; N/A
Original Section/Title: C.I.3.8.2.6; Materials, Quality Control, and Special Construction Techniques
Not Applicable

Modified Section/Title: C.I.3.8.2.7; N/A
Original Section/Title: C.I.3.8.2.7; Testing and Inservice Inspection Requirements
Not Applicable

Modified Section/Title: C.I.3.8.3; Concrete and Steel Internal Structures of Steel or Concrete Reactor Building

Original Section/Title: C.I.3.8.3; Concrete and Steel Internal Structures of Steel or Concrete Containment

The applicant should provide information similar to that requested in Section 3.8.1 of this guide, but for internal structures of the reactor building. The reactor building internal structures are those concrete and steel structures that are inside (not part of) the reactor building and support SSCs that are important to safety. Subsections 3.8.3.1 through 3.8.3.7 describe the recommended information.

Modified Section/Title: C.I.3.8.3.1; Description of the Internal Structures

Original Section/Title: C.I.3.8.3.1; Description of the Internal Structures

The applicant should define the primary structural aspects and elements relied on to perform the safety related functions by including a physical description of the internal structures, including plan and section views. This description should contain general arrangement diagrams and principal features of major internal structures. The major structures to be described include the following:

1. For HTGR
   a. Reactor support system
   b. Steam generator and/or intermediate heat exchanger support system
   c. Reactor cavity
   d. Other major internal structures, such as supports, the refueling cavity walls, the operating floor, intermediate floors, and various platforms.

Modified Section/Title: C.I.3.8.3.2; Applicable Codes, Standards, and Specifications

Original Section/Title: C.I.3.8.3.2; Applicable Codes, Standards, and Specifications

The applicant should provide information similar to that requested for concrete reactor building in Section 3.8.1.2 of this guide and RG 1.142, but the information should pertain to the reactor building internal structures listed in FSAR Section 3.8.3.1.

Modified Section/Title: C.I.3.8.3.3; Loads and Load Combinations

Original Section/Title: C.I.3.8.3.3; Loads and Load Combinations

The applicant should discuss and specify the loads used in the design of the reactor building internal structures listed in FSAR Section 3.8.3.1.

The applicant should discuss the various combinations of the above loads that are usually postulated, such as normal operating loads, normal operating loads with severe environmental loads, and normal operating loads with extreme environmental loads and abnormal loads.

The applicant should provide specific information, emphasizing the following considerations:

1. The extent to which the criteria comply industry structural codes and standards applicable to HTGR technology and applicable regulations, PDC, and regulatory guides;

2. For steel linear supports, the extent to which the applicant’s criteria comply with ASME Code requirements applicable to HTGR technology, augmented by RG 1.57 and/or the specific edition, date, or addenda of design codes, standards, specifications, regulations, PDC, regulatory guides, and other industry standards applicable to HTGR technology as endorsed by the NRC.

Modified Section/Title: C.I.3.8.3.4; Design and Analysis Procedures

Original Section/Title: C.I.3.8.3.4; Design and Analysis Procedures

The applicant should describe the design and analysis method and assumptions and identify the boundary conditions of those internal structures listed in FSAR Section 3.8.3.1. The description should include the expected behavior under load and the mechanisms for load transfer to these structures and then transfer to an anchorage or foundational structure of the RB. The applicant should reference the computer programs utilized to permit identification with available published programs and describe proprietary computer programs in sufficient detail to establish their applicability and the method for validating them.

This section should specify the extent to which the design and analysis procedures comply with industry codes and standards applicable to HTGR technology for concrete and steel structures, respectively, and/or the specific edition, date, or addenda of design codes, standards, specifications, regulations, PDC, and regulatory guides.

The description should include the design and analysis method used with the assumptions regarding boundary conditions, for HPB linear supports. It should also specify and identify the type of analysis (elastic or plastic) and the methods of load transfer, particularly seismic and accident loads. The applicant should specify the extent of compliance with design and analysis procedures delineated in the ASME Code applicable to HTGR technology and/or the specific edition, date, or addenda of design codes, standards, specifications, regulations, PDC, regulatory guides, and other industry standards applicable to HTGR technology.

The applicant should describe the design and analysis method utilized. The description should include the method and assumptions, with particular emphasis on modeling techniques, boundary conditions, and force time functions where elastoplastic behavior is assumed and the ductility of the walls is relied on to absorb the energy associated with jet impingement and missile loads.

Modified Section/Title: C.I.3.8.3.5; Structural Acceptance Criteria

Original Section/Title: C.I.3.8.3.5; Structural Acceptance Criteria

The applicant should provide information similar to that requested for a concrete reactor building in Section 3.8.1.5 of this guide, but the information should pertain to the various reactor building internal structures listed in FSAR Section 3.8.3.1.

Modified Section/Title: C.I.3.8.3.6; Materials, Quality Control, and Special Construction Techniques

Original Section/Title: C.I.3.8.3.6; Materials, Quality Control, and Special Construction Techniques

The applicant should identify and describe the materials, quality control programs, and any special construction techniques. The description should include the major materials of construction, such as the concrete ingredients, reinforcing bars and splices, and the structural steel and various supports and anchors.

The applicant should also describe the quality control program proposed for the fabrication and construction of the reactor building interior structures commensurate with the designed safety functions and include NDE of the materials to determine physical properties, placement of concrete, and erection tolerances. This section of the application should also identify and describe special, new, or unique construction techniques to determine their effects on the structural integrity of the completed interior structure.

The applicant should provide the following information:

1. The extent to which the material and quality control requirements comply with industry concrete codes and specifications applicable to HTGR technology for steel

2. For quality control in general, the extent of compliance with applicable provisions of Sections 3.8 and 17.5 of this guide

3. For welding of reinforcing bars, the extent to which the design complies with ASME Code requirements applicable to HTGR technology (with identification and justification of any exceptions).

Modified Section/Title: C.I.3.8.3.7; Testing and Inservice Inspection Requirements
Original Section/Title: C.I.3.8.3.7; Testing and Inservice Inspection Requirements
The applicant should describe the testing and ISI programs, including milestones, for the internal structures commensurate with the safety functions performed by the structure. The description should specify test requirements for internal structures related directly and critically to the functioning of the reactor building, as well as the ISI requirements. As requested in Section 3.8.3.6 of this guide, the applicant should identify the extent of compliance with the specific edition, date, or addenda of design codes, standards, specifications, regulations, PDC, regulatory guides, and other industry standards applicable to HTGR technology.

Modified Section/Title: C.I.3.8.4; Other Seismic Category I Structures
Original Section/Title: C.I.3.8.4; Other Seismic Category I Structures
The applicant should provide information for all seismic Category I structures not covered by Sections 3.8.1, 3.8.3, or 3.8.5 of this guide. The information provided should be similar to that requested in Section 3.8.1 of this guide.

Modified Section/Title: C.I.3.8.4.1; Description of the Structures
Original Section/Title: C.I.3.8.4.1; Description of the Structures
This section should contain descriptive information, including plan and section views of each structure, to define the primary structural aspects and elements relied on for the structure to perform its safety related function. The applicant should describe the relationship between adjacent structures, including any separation or structural ties, and describe the plant's seismic Category I structures. Items which should be considered include the following:

1. Reactor enclosure buildings
2. Auxiliary buildings
3. Fuel storage buildings
4. Control buildings
5. Other seismic Category I structures, such as pipe and electrical conduit tunnels, waste storage facilities, stacks, intake structures, pumping stations, water wells, cooling towers, and concrete dams, embankments, and tunnels.

The applicant should also describe structures that are safety-related but, because of other design provisions, are not classified as seismic Category I.

Modified Section/Title: C.I.3.8.4.2; Applicable Codes, Standards, and Specifications
Original Section/Title: C.I.3.8.4.2; Applicable Codes, Standards, and Specifications
The applicant should provide information similar to that requested in Section 3.8.1.2 of this guide for concrete reactor buildings, but the information should pertain to all other seismic Category I structures.

Modified Section/Title: C.I.3.8.4.3; Loads and Load Combinations

Original Section/Title: C.I.3.8.4.3; Loads and Load Combinations

The applicant should specify and identify the loads used in the design of all other seismic Category I structures including the following:

1. Loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads attributable to operating temperature, and hydrostatic loads such as those in spent fuel pools

2. Loads sustained in the event of severe environmental conditions, including those induced by the OBE and the design wind specified for the plant site

3. Loads sustained in the event of extreme environmental conditions, including those induced by the SSE and the design-basis tornado specified for the plant site

4. Loads sustained during abnormal plant conditions, such as rupture of high energy piping with associated elevated temperatures and pressures within or across compartments and possibly jet impingement and impact forces. The discussion should cover the various combinations of the above loads that are usually postulated, such as normal operating loads, normal operating loads with severe environmental loads, normal operating loads with extreme environmental loads, normal operating loads with abnormal loads, normal operating loads with severe environmental loads and abnormal loads, and normal operating loads with extreme environmental loads and abnormal loads.

The loads and load combinations described above are generally applicable to most structures. The discussion should include other site related design loads, such as those induced by floods, potential non-deliberate aircraft crashes, explosive hazards in proximity to the site, and projectiles and missiles generated from activities of nearby military installations.

Modified Section/Title: C.I.3.8.4.4; Design and Analysis Procedures

Original Section/Title: C.I.3.8.4.4; Design and Analysis Procedures

The applicant should describe the design and analysis method, with assumptions regarding boundary conditions and emphasis on the extent of compliance with industry codes, standards, and specifications applicable to HTGR technology for concrete and steel structures, respectively. The description should include the expected behavior under load and the mechanisms of load transfer to the foundations. The applicant should reference computer programs to permit identification with available published programs and describe proprietary computer programs to the maximum extent practical to establish the applicability of the programs and the method used to validate them.

Modified Section/Title: C.I.3.8.4.5; Structural Acceptance Criteria

Original Section/Title: C.I.3.8.4.5; Structural Acceptance Criteria

The applicant should specify the design criteria related to stresses, strains, gross deformations, factors of safety, and other parameters that quantitatively identify the margins of safety as provided in design codes, standards, specifications, regulations, PDC, and regulatory guides applicable to HTGR technology.

Modified Section/Title: C.I.3.8.4.6; Materials, Quality Control, and Special Construction Techniques

Original Section/Title: C.I.3.8.4.6; Materials, Quality Control, and Special Construction Techniques

This section should address the materials and quality control programs and identify any new or special construction techniques, as outlined in Section 3.8.3.6 of this guide.

Modified Section/Title: C.I.3.8.4.7; Testing and Inservice Inspection Requirements

Original Section/Title: C.I.3.8.4.7; Testing and Inservice Inspection Requirements

This section should specify any testing and ISI requirements.

Modified Section/Title: C.I.3.8.5; Foundations

Original Section/Title: C.I.3.8.5; Foundations

The applicant should provide information similar to that requested in Section 3.8.1 of this guide for concrete reactor buildings, but the information should pertain to the foundations of all seismic Category I structures. As appropriate, this section, as well as FSAR Section 3.8.1, should discuss concrete foundations of concrete reactor buildings.

The applicant should provide information for foundations for all seismic Category I structures constructed of materials other than soil for the purpose of transferring loads and forces to the basic supporting media.

Modified Section/Title: C.I.3.8.5.1; Description of the Foundations

Original Section/Title: C.I.3.8.5.1; Description of the Foundations

The applicant should provide descriptive information, including plan and section views of each foundation, to define the primary structural aspects and elements relied on to perform the foundation function. The description should include the relationship between adjacent foundations, including any separation and the reasons for such separation. In particular, the applicant should discuss the type of foundation and its structural characteristics and provide the general arrangement of each foundation, with emphasis on the methods of transferring horizontal shears, such as those that are seismically induced, to the foundation media. If the applicant uses shear keys for such purposes, it should include the general arrangement of the keys. If using waterproofing membranes, the applicant should discuss their effect on the capability of the foundation to transfer shears.

This section should include information to adequately describe other types of foundation structures, such as pile foundations, caisson foundations, retaining walls, abutments, and rock and soil anchorage systems.

Modified Section/Title: C.I.3.8.5.2; Applicable Codes, Standards, and Specifications

Original Section/Title: C.I.3.8.5.2; Applicable Codes, Standards, and Specifications

This section should provide information similar to that requested in Section 3.8.1.2 of this guide, but as applicable to the foundations of all seismic Category I structures.

Modified Section/Title: C.I.3.8.5.3; Loads and Load Combinations

Original Section/Title: C.I.3.8.5.3; Loads and Load Combinations

This section should provide information similar to that requested in Section 3.8.4.3 of this guide, but as applicable to the foundations of all seismic Category I structures.

Modified Section/Title: C.I.3.8.5.4; Design and Analysis Procedures

Original Section/Title: C.I.3.8.5.4; Design and Analysis Procedures

This section should provide information similar to that requested in Section 3.8.4.4 of this guide, but as applicable to the foundations of all seismic Category I structures.

The applicant should discuss the assumptions regarding boundary conditions, as well as the methods by which lateral loads and forces and overturning moments are transmitted from the structure to the foundation media. The discussion should address the methods for considering the effects of settlement.

Modified Section/Title: C.I.3.8.5.5; Structural Acceptance Criteria

Original Section/Title: C.I.3.8.5.5; Structural Acceptance Criteria
The applicant should provide information similar to that requested in Section 3.8.4.5 of this guide, but as it pertains to the foundations of all seismic Category I structures.

The applicant should describe, and indicate the design limits imposed on, the various parameters that define the structural stability of each structure and its foundations, including differential settlements and factors of safety against overturning and sliding.

Modified Section/Title: C.I.3.8.5.6; Materials, Quality Control, and Special Construction Techniques

Original Section/Title: C.I.3.8.5.6; Materials, Quality Control, and Special Construction Techniques
This section should provide information similar to that requested in Section 3.8.4.6 of this guide for the foundations of all seismic Category I structures.

Modified Section/Title: C.I.3.8.5.7; Testing and Inservice Inspection Requirements

Original Section/Title: C.I.3.8.5.7; Testing and Inservice Inspection Requirements
The applicant should discuss information similar to that requested in Section 3.8.4.7 of this guide for the foundations of all seismic Category I structures.

If programs for continued surveillance and monitoring of foundations are required, the applicant should define the various aspects of the program, including implementation milestones.

Modified Section/Title: C.I.3.9; Mechanical Systems and Components

Original Section/Title: C.I.3.9; Mechanical Systems and Components

Modified Section/Title: C.I.3.9.1; Special Topics for Mechanical Components

Original Section/Title: C.I.3.9.1; Special Topics for Mechanical Components
The applicant should provide information concerning the design transients and resulting loads and load combinations with appropriate specified design and service limits for seismic Category I components and supports, including those designated as ASME Code Class A or B (or core support) and those not covered by the ASME Code.

Modified Section/Title: C.I.3.9.1.1; Design Transients

Original Section/Title: C.I.3.9.1.1; Design Transients
The applicant should provide a complete list of transients used in the design and fatigue analysis of all ASME Code Class A and CS components, component supports, and reactor internals. The list should include the number of events for each transient, as well as the number of load and stress cycles per event and for events in combination. The applicant should provide the number of transients assumed for the design life of the plant and describe the environmental conditions to which equipment important to safety will be exposed over the life of the plant (e.g., chemical attack). The applicant should classify all transients (or combinations of transients) with respect to the plant and system operating condition categories identified as “normal,” “upset,” “emergency,” “faulted,” or “testing.”

Modified Section/Title: C.I.3.9.1.2; Computer Programs Used in Analyses

Original Section/Title: C.I.3.9.1.2; Computer Programs Used in Analyses
The applicant should list the computer programs used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I ASME Code and non-ASME Code items and provide the following information:

1. Author, source, dated version, and facility

2. Description and the extent and limitations of the code's applications

3. Demonstration of the computer code's solutions to a series of test problems and the source of the test problems.

Modified Section/Title: C.I.3.9.1.3; Experimental Stress Analysis
Original Section/Title: C.I.3.9.1.3; Experimental Stress Analysis
If the applicant uses experimental stress analysis methods in lieu of analytical methods for seismic Category I ASME Code and non ASME Code items, it should provide sufficient information to show the validity of the design.

Modified Section/Title: C.I.3.9.1.4; Considerations for the Evaluation of the Faulted Condition
Original Section/Title: C.I.3.9.1.4; Considerations for the Evaluation of the Faulted Condition
The applicant should describe the analytical methods (e.g., elastic or elastic plastic) used to evaluate stresses for seismic Category I ASME Code and non ASME Code components and component support and discuss their compatibility with the type of dynamic system analysis used. The applicant should show that the stress strain relationship and ultimate strength value used in the analysis for each component is valid. If the applicant invokes the use of elastic or elastic plastic component analysis concurrently with elastic or elastic plastic system analysis, it should show that the calculated component or component support deformations and displacements do not violate the corresponding limits and assumptions on which the method used for the system analysis is based. When elastic-plastic stress or deformation design limits are specified for ASME Code and non ASME Code components, the applicant should provide the methods of analysis used to calculate the stresses and/or deformations resulting from the faulted condition loadings. The applicant should also describe the procedure for developing the loading function for each component.

Modified Section/Title: C.I.3.9.2; Dynamic Testing and Analysis of Systems, Components, and Equipment
Original Section/Title: C.I.3.9.2; Dynamic Testing and Analysis of Systems, Components, and Equipment
The applicant should provide the criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports (including supports for conduit and cable trays and ventilation ducts) under vibratory loadings, including those attributable to flow-induced vibration (FIV), acoustic resonance, postulated pipe breaks, and seismic events.

Modified Section/Title: C.I.3.9.2.1; Piping Vibration, Thermal Expansion, and Dynamic Effects
Original Section/Title: C.I.3.9.2.1; Piping Vibration, Thermal Expansion, and Dynamic Effects
The applicant should provide information concerning the piping vibration, thermal expansion, and dynamic effects testing that it will conduct during startup functional testing on ASME Code Class A and B systems; other high energy piping systems inside seismic Category I structures; high energy portions of systems for which failure could reduce the functioning of any seismic Category I plant feature to an unacceptable level; and seismic Category I portions of moderate energy piping systems located outside reactor building. The applicant should show that these tests will demonstrate that the piping systems, restraints, components, and supports have been designed to (1) withstand the flow induced dynamic loadings under operational transient and steady state conditions anticipated during service and (2) not restrain normal thermal motion.

The applicant should include the following information concerning the piping vibration, thermal expansion, and dynamic effects testing:

1. List of the systems that will be monitored

2. List of the different flow modes of operation and transients such as pump trips and valve closures to which the components will be subjected during the test. Additional guidance provided in RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants"
3. List of the locations selected for visual inspections and measurements in the piping system during the tests. For each of these selected locations, the applicant should include the deflection (peak-to-peak) or other appropriate criteria to show that the stress and fatigue limits are within the design levels. The applicant should also provide the rationale and bases for the acceptance criteria and selection of locations to monitor pipe motions. If the as-built specifics are not available at the time of the application, representative and bounding conditions may be used in the analysis to be submitted with the application for staff review. The applicant should in the application propose an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

4. List of the snubbers on systems that experience sufficient thermal movement to measure snubber travel from cold to hot position. The applicant should in the application propose an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

5. Description of the thermal motion monitoring program to ensure that clearances are adequate to allow unrestrained normal thermal movement of systems, components, and supports

6. Description of the corrective actions that the applicant will take if vibration exceeds acceptable levels, piping system restraints are determined to be inadequate or are damaged, or no snubber piston travel is measured

7. If the piping vibration, thermal expansion, and dynamic effects testing is incomplete at the time the application is filed, the applicant should specify whether they are part of the initial test program (ITP) and should describe the implementation program, including milestones.

Modified Section/Title: C.I.3.9.2.2; Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

Original Section/Title: C.I.3.9.2.2; Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

The applicant should describe the seismic system analysis and qualification of Category I systems, components, equipment, and their supports (including supports for conduit and cable trays and ventilation ducts) performed to ensure functional integrity and operability during and after a postulated seismic occurrence.

Modified Section/Title: C.I.3.9.2.2.1; Seismic Qualification Testing

Original Section/Title: C.I.3.9.2.2.1; Seismic Qualification Testing

The applicant should describe the methods and criteria for seismic qualification testing of seismic Category I mechanical equipment.

Modified Section/Title: C.I.3.9.2.2.2; Seismic System Analysis Methods
Original Section/Title: C.I.3.9.2.2.2; Seismic System Analysis Methods
The applicant should describe the seismic analysis methods (e.g., response spectra, time history, equivalent static load) and include the following information in the description:

1. Manner in which the dynamic system analysis is performed

2. Method chosen for selection of significant modes and an adequate number of masses or degrees of freedom

3. Manner in which the seismic dynamic analysis considers maximum relative displacements between supports

4. Other significant effects accounted for in the seismic dynamic analysis, such as piping interactions, externally applied structural restraints, dynamic effects (both mass and stiffness effects), and nonlinear response.

If the applicant uses a static load method in lieu of a dynamic analysis, it should provide justification that a simple model can realistically represent the system and that the method produces conservative results.

Modified Section/Title: C.I.3.9.2.2.3; Determination of Number of Earthquake Cycles
Original Section/Title: C.I.3.9.2.2.3; Determination of Number of Earthquake Cycles
The applicant should describe the number of earthquake cycles assumed during one seismic event, the maximum number of cycles for which systems and components are designed, and the criteria used to establish these parameters.

Modified Section/Title: C.I.3.9.2.2.4; Basis for Selection of Frequencies
Original Section/Title: C.I.3.9.2.2.4; Basis for Selection of Frequencies
The applicant should provide the criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure.

Modified Section/Title: C.I.3.9.2.2.5; Three Components of Earthquake Motion
Original Section/Title: C.I.3.9.2.2.5; Three Components of Earthquake Motion
This section should describe how the three components of earthquake motion are considered in determining the seismic response of systems and components.

Modified Section/Title: C.I.3.9.2.2.6; Combination of Modal Responses
Original Section/Title: C.I.3.9.2.2.6; Combination of Modal Responses
When the applicant uses a response spectra method, it should describe how modal responses (e.g., shears, moments, stresses, deflections, and accelerations) were combined, including those for modes with closely spaced frequencies. Additional guidance is provided in RG 1.92 .

Modified Section/Title: C.I.3.9.2.2.7; Analytical Procedures for Piping
Original Section/Title: C.I.3.9.2.2.7; Analytical Procedures for Piping
The applicant should describe the analytical methods (e.g., response spectra, time history, equivalent static load) used for the seismic analysis of piping systems, including the methods used to consider differential piping support movements at different support points located within a structure and between structures.

Modified Section/Title: C.I.3.9.2.2.8; Multiple-Supported Equipment Components with Distinct Inputs

Original Section/Title: C.I.3.9.2.2.8; Multiple-Supported Equipment Components with Distinct Inputs

This section should describe the analytical methods used for the seismic analysis of equipment and components supported at different elevations within a building and between buildings.

Modified Section/Title: C.I.3.9.2.2.9; Use of Constant Vertical Static Factors

Original Section/Title: C.I.3.9.2.2.9; Use of Constant Vertical Static Factors

Where applicable, the applicant should justify the use of constant static forces instead of vertical seismic system dynamic analysis to compute the vertical response loads for the design of affected systems, components, equipment, and their supports.

Modified Section/Title: C.I.3.9.2.2.10; Torsional Effects of Eccentric Masses

Original Section/Title: C.I.3.9.2.2.10; Torsional Effects of Eccentric Masses

This section should describe the methods used to consider the torsional effects of eccentric masses (e.g., valve operators) in seismic system analyses.

Modified Section/Title: C.I.3.9.2.2.11; Buried Seismic Category I Piping Conduits, and Tunnels

Original Section/Title: C.I.3.9.2.2.11; Buried Seismic Category I Piping Conduits, and Tunnels

The applicant should describe the seismic criteria and methods used to analyze buried piping, conduits and tunnels, including the procedures used to consider the inertia effects of soil media and the differential displacements at structural penetrations.

Modified Section/Title: C.I.3.9.2.2.12; Interaction of Other Piping with Seismic Category I Piping

Original Section/Title: C.I.3.9.2.2.12; Interaction of Other Piping with Seismic Category I Piping

This section should describe the seismic analysis methods used to account for the seismic motion of non-seismic Category I piping systems in the seismic design of seismic Category I piping.

Modified Section/Title: C.I.3.9.2.2.13; Analysis Procedure for Damping

Original Section/Title: C.I.3.9.2.2.13; Analysis Procedure for Damping

This section should describe the criteria used to account for damping in systems, components, equipment, and their supports. Additional guidance is provided in RG 1.61.

Modified Section/Title: C.I.3.9.2.2.14; Test and Analysis Results

Original Section/Title: C.I.3.9.2.2.14; Test and Analysis Results

The applicant should supply the results of tests and analyses to demonstrate adequate seismic qualification. If the seismic qualification testing is incomplete at the time the application is filed, the applicant should describe the implementation program, including milestones.

Modified Section/Title: C.I.3.9.2.3; Dynamic Response Analysis of Reactor Internals under Operational Flow Transients

Original Section/Title: C.I.3.9.2.3; Dynamic Response Analysis of Reactor Internals under Operational Flow Transients

For a demonstration (first of a design) reactor, the applicant should describe the dynamic system analysis and response of the structural components within the reactor vessel caused by operational flow transients and steady state conditions. The applicant should demonstrate the acceptability of the reactor internals design for normal operating conditions and provide the predicted input forcing functions and the vibratory response of the reactor internals.

For a non-demonstration reactor, the applicant should provide references to the reactor that is the reactor design demonstration plant included in the application, along with a brief summary of test and analysis results.

Modified Section/Title: C.I.3.9.2.4; Preoperational Flow-Induced Vibration Testing of Reactor Internals

Original Section/Title: C.I.3.9.2.4; Preoperational Flow-Induced Vibration Testing of Reactor Internals
The applicant should describe the preoperational and startup test program for FIV testing of reactor internals and demonstrate that FIV experienced during normal operation will not cause structural failure or degradation.

For a demonstration reactor, the applicant should describe flow modes, vibration monitoring sensor types and locations, procedures and methods to be used to process and interpret the measured data, planned visual inspections, planned comparisons of test results with analytical predictions, and possible supplementary tests (e.g., component vibration tests, flow tests, scaled model tests).

For a nonprototype reactor, the applicant should provide references to the reactor that is prototypical of the reactor (design included in the application), along with a brief summary of test and analysis results.

The applicant should identify and justify any deviation from the guidance provided in RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperation and Initial Startup Testing." RG 1.20 provides requirements that are specific to LWR features/elements, however, it is functionally applicable to HTGRs.

Modified Section/Title: C.I.3.9.2.5; Dynamic System Analysis of the Reactor Internals under Faulted Condition

Original Section/Title: C.I.3.9.2.5; Dynamic System Analysis of the Reactor Internals under Faulted Condition
The applicant should discuss the dynamic system analysis methods used to confirm the adequacy of the structural design of the reactor internals and the unbroken loop of the reactor vessel system, as it relates to withstanding dynamic effects with no loss of function under the simultaneous occurrence of the most severe postulated accident or steam line break and SSE.

The applicant should include the following information concerning the dynamic system analysis:

1. Typical diagrams of the dynamic system mathematical modeling of piping, pipe supports, and reactor internals, along with fuel compacts and control rod assemblies and drives, used in the analysis and a discussion of the bases for any structural partitioning and directional decoupling of components

2. Methods used to obtain the forcing functions and a description of the forcing functions used for the dynamic analysis of the most severe postulated accident or steam line break and SSE event (including system pressure differentials, direction, rise time, magnitude, duration, initial conditions, spatial distribution, and loading combinations)

3. Methods used to compute the total dynamic structural responses, including the buckling response, of those structures in compression

4. Results of the dynamic analysis.

Modified Section/Title: C.I.3.9.2.6; Correlations of Reactor Internals Vibration Tests with the Analytical Results

Original Section/Title: C.I.3.9.2.6; Correlations of Reactor Internals Vibration Tests with the Analytical Results

The applicant should describe the method used to correlate the results of the reactor internals pre-operational vibration test with the analytical results derived from dynamic analyses of reactor internals under operational flow transients and steady state conditions. The description should include the method used to verify the mathematical model used in the faulted condition (most severe postulated accident, steam line break, and SSE) by comparing certain dynamic characteristics such as natural frequencies.

Modified Section/Title: C.I.3.9.3; ASME Code Class A and B Components, Component Supports, and Core Support

Original Section/Title: C.I.3.9.3; ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support

The applicant should discuss the structural integrity of pressure-retaining components, component supports, and core support structures designed and constructed in accordance with the rules of the ASME Code, Section III, Division 5, as well as with GDC 1, "Quality standards and records," GDC 2 **[PDC 2]**, "Design bases for protection against natural phenomena," GDC 4 **[PDC 4]**, "Environmental and dynamic effects design bases," GDC 14 **[PDC 14]**, "Reactor coolant pressure boundary," and GDC 15 **[PDC 15]**, "Reactor coolant system design." This discussion should also incorporate design information related to component design for steam generators (as requested in Section 5.4.2 of this guide), if applicable, including field run piping and internal parts of components. If request as-built information is not available at the time of the application, the applicant should provide current design information, representative, or bounding information. The applicant should in the application propose an appropriate method (e.g., ITAAC, license condition, FSAR) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

Modified Section/Title: C.I.3.9.3.1; Loading Combinations, System Operating Transients, and Stress Limits

Original Section/Title: C.I.3.9.3.1; Loading Combinations, System Operating Transients, and Stress Limits

The applicant should provide the design and service load combinations (e.g., design and service loads, including system operating transients, in combination with loads resulting from postulated seismic and other transient initiating events) specified for components constructed in accordance with the ASME Code and designated as ASME Code Class A or B. This should include Class A and B component support structures and core support structures to determine that appropriate design and service limits have been designated for all loading combinations. The applicant should describe how actual design and service stress limits and deformation criteria comply with applicable limits specified in the ASME Code. The applicant should provide information on service stress limits that allow inelastic deformation of ASME Code Class A and B components, component supports, and core support structures and provide justification for proposed design procedures. The discussion should include information on field run piping and internal parts of components (e.g., valve discs and seats and pump shafting) subjected to dynamic loading during operation of the component.

The applicant should include the following information for ASME Code Class A components, core support structures, and ASME Code Class A component supports:

1. Summary description of mathematical or test models used

2. Methods of calculations or tests, including simplifying assumptions, identification of method of system and component analysis used, and demonstration of their compatibility (see Section 3.9.1 of this guide) in the case of components and supports that are designed to faulted limits

3. Summary of the maximum total stress, deformation, and cumulative usage factor values for each of the component operating conditions for all ASME Code Class A components. The applicant should identify those values that differ from the allowable limits by less than 10 percent and provide the contribution of each of the loading categories, (e.g., seismic, dead weight, pressure, and thermal) to the total stress for each maximum stress value identified in this range.

The applicant should include the following information for all other classes of components and their supports:

1. Summary description of any test models used (see Section 3.9.1 of this guide)

2. Summary description of mathematical or test models used to evaluate faulted conditions, as appropriate, for components and supports (see Section 3.9.1 of this guide)

3. For all ASME Code Class B components required to shut down the reactor or mitigate consequences of a postulated piping failure without offsite power, a summary of the maximum total stress and deformation values for each of the component operating conditions (with identification of those values that differ from the allowable limits by less than 10 percent).

The discussion should include a list of transients appropriate to ASME Code Class A and B components, core support structures, and component supports categorized on the basis of plant operating condition. In addition, for ASME Code Class A components, core support structures, and component supports, the applicant should include the number of cycles to be used in the fatigue analysis appropriate to each transient (see Section 3.9.1 of this guide).

Modified Section/Title: C.I.3.9.3.2; Design and Installation of Pressure-Relief Devices

Original Section/Title: C.I.3.9.3.2; Design and Installation of Pressure-Relief Devices

The applicant should describe the design and installation criteria applicable to the mounting of pressure relief devices (i.e., safety and relief valves) for overpressure protection of ASME Class A and B components and include information to permit evaluation of applicable load combinations and stress criteria. This section should provide information to allow the design review to consider plans for accommodating the rapidly applied reaction force that occurs when a safety or relief valve opens and the transient fluid-induced loads applied to piping downstream from a safety or relief valve in a closed discharge piping system. The applicant should describe the design of safety and relief valve systems with respect to load combinations postulated for the valves, upstream piping or header, downstream or vent piping, and system supports, if applicable.

For load combinations, the applicant should identify the most severe combination of applicable loads attributable to internal fluid weight, momentum, and pressure; dead weight of valves and piping; thermal load under heat up; steady state and transient valve operation; reaction forces when valves are discharging (i.e., thrust, bending, torsion); and seismic forces (i.e., SSE); if applicable.

The discussion should include the method of analysis and magnitude of any dynamic load factors used. The applicant should discuss and include in the analysis a description of the structural response of the piping and support system, with particular attention to the dynamic or time history analyses employed in evaluating the appropriate support and restraint stiffness effects under dynamic loadings when valves are discharging. The applicant should present the results of this analysis.

If the applicant proposes to use hydraulic snubbers, it should describe snubber performance characteristics to ensure that their effects have been considered in analyses under steady-state valve operation and repetitive load applications caused by cyclic valve opening and closing during the course of a pressure transient.

Modified Section/Title: C.I.3.9.3.3; Component Operability Assurance

Original Section/Title: C.I.3.9.3.3; Pump and Valve Operability Assurance

The applicant should identify all active ASME Class A and B pumps and valves. This section should present criteria to be employed in a test program, or a program consisting of tests and analysis, to ensure operability of pumps required to function and valves required to open or close to perform a safety function during or following the specified plant event. The applicant should discuss features of the program, including conditions of test, scale effects (if appropriate), loadings for specified plant event, transient loads (including seismic component, dynamic coupling to other systems, stress limits, deformation limits), and other information pertinent to assurance of operability. The applicant should include the design stress limits established in FSAR Section 3.9.3.1.

The section should also include program results, summarizing stress and deformation levels and environmental qualification, as well as maximum test envelope conditions for which each component qualifies, including end connection loads and operability results.

Modified Section/Title: C.I.3.9.3.4; Component Supports

Original Section/Title: C.I.3.9.3.4; Component Supports

This section should provide load combinations, system operating transients, stress limits, and deformation limits for component supports, discussed in Section 3.9.3 of this guide.

The applicant should furnish information to enable evaluation of supports for ASME Code Class A and B components, including assessment of design and structural integrity of plate and shell, linear, and component standard types of supports for active components. The applicant should analyze and/or test the component supports as discussed in Section 3.9.3 of this guide, and include their effects on operability in the discussion provided in that section. The applicant should present the criteria used for the analysis or test program, as well as the results of the analysis and/or test programs discussed in Sections 3.9.3 of this guide. The combination of loadings considered for each component support within a system, including the designation of the appropriate service stress for each loading combination should be consistent with the criteria in Appendix A, RG 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Support" and RG 1.130, "Service Limits and Loading Combinations for Class I Plate-and-Shell-Type Component Supports."

Modified Section/Title: C.I.3.9.4; Control Rod Drive Systems

Original Section/Title: C.I.3.9.4; Control Rod Drive Systems

This section should provide information on the control rod drive system (CRDS). For electrical systems, the applicant should include the CRDM up to the coupling interface with neutron control elements. The applicant should treat the CRDM housing as part of the HPB. FSAR Section 4.5.1 should include information on CRDS materials.

If other types of CRDSs are proposed, or if new features that are not specifically mentioned here are incorporated in current types of CRDSs, the applicant should provide information regarding the new systems or new features.

Modified Section/Title: C.I.3.9.4.1; Descriptive Information of CRDS

Original Section/Title: C.I.3.9.4.1; Descriptive Information of CRDS

The applicant should provide an evaluation of the system's adequacy to properly perform its design function. This evaluation should include design criteria, testing programs, drawings, and a summary of the method of operation of the control rod drives.

Modified Section/Title: C.I.3.9.4.2; Applicable CRDS Design Specifications

Original Section/Title: C.I.3.9.4.2; Applicable CRDS Design Specifications

The applicant should indicate the design codes, standards, specifications, and standard practices, as well as PDC, regulatory guides, and positions, applied in the design, fabrication, construction, and operation of the CRDS. The list of the various criteria along with the names of the apparatuses to which they apply should include the following:

1. List of the pressurized parts of the system in FSAR Section 3.2.2:
    a. For those portions that are part of the HPB, the extent of compliance with the Class A requirements in Section III of the ASME Code, Division 5, should be indicated
    b. For those portions that are not part of the HPB, the extent of compliance with other specified parts of Section III or other sections of the ASME Code, should be indicated.

2. An evaluation of the nonpressurized portions of the CRDS, which demonstrates the acceptability of design margins for allowable values of stress, deformation, and fatigue. If the applicant uses an experimental testing program in lieu of analysis, it should discuss how the program adequately covers stress, deformation, and fatigue in the CRDS. If this experimental testing program is incomplete at the time the application is filed, the applicant should describe the implementation program, including milestones, completion dates and expected conclusions.

Modified Section/Title: C.I.3.9.4.3; Design Loads, Stress Limits, and Allowable Deformations

Original Section/Title: C.I.3.9.4.3; Design Loads, Stress Limits, and Allowable Deformations

This section should present information that pertains to the applicable design loads and their appropriate combinations, the corresponding design stress limits, and the corresponding allowable deformations. The deformations of interest are those where a failure of movement could occur and such movement is necessary for a safety related function. The applicant should provide the following information:

1. If experimental testing is used in lieu of establishing a set of stress and deformation allowable, a description of the testing program, including the load combinations, design stress limits, and allowable deformation criteria. If the experimental testing is incomplete at the time the application is filed, a description of the implementation program, including milestones, completion dates and expected conclusions.

2. For components that are not designed to the ASME Code, the design limits and safety margins

3. For components that are designed to the ASME Code, information similar to that requested in Section 3.9.3 of this guide

4. Comparison of the actual design with the design criteria and limits to demonstrate that the criteria and limits have not been exceeded.

Modified Section/Title: C.I.3.9.4.4; CRDS Operability Assurance Program
Original Section/Title: C.I.3.9.4.4; CRDS Operability Assurance Program
The applicant should provide plans for conducting an operability assurance program or references to previous test programs or standard industry procedures for similar apparatuses. This section should show how the operability assurance program includes a life cycle test program.

The applicant should describe the plan for implementing the operability assurance program, including milestones.

Modified Section/Title: C.I.3.9.5; Reactor Pressure Vessel Internals
Original Section/Title: C.I.3.9.5; Reactor Pressure Vessel Internals
The applicant should discuss the specific design codes, load combinations, allowable stress and deformation limits, and other criteria used in designing the reactor internals, for both core support structures designed to the ASME Code and internals designed to other standards. The applicant should ensure the structural and functional integrity of the reactor internals.

Modified Section/Title: C.I.3.9.5.1; Design Arrangements
Original Section/Title: C.I.3.9.5.1; Design Arrangements
The applicant should present the physical or design arrangements of all reactor internals SSC, and assemblies, including the manner of positioning and securing such items within the RPV, the manner of providing for axial and lateral retention and support of the internals components and assemblies, and the manner of accommodating dimensional changes attributable to thermal and other effects. The description should include the functional requirements for each component. The applicant should verify that any significant changes in design from that used in previously licensed plants of similar design do not affect the acoustic and FIV test results requested in Section 3.9.2 of this guide.

Modified Section/Title: C.I.3.9.5.2; Loading Conditions
Original Section/Title: C.I.3.9.5.2; Loading Conditions
This section should specify the plant and system operating conditions and DBEs that provide the basis for the design of the reactor internals to sustain normal operation, vibratory FIV and acoustic loading, anticipated events (AEs), postulated accidents, and seismic events in accordance with the information requested in Section 3.9.1 of this guide.

The applicant should identify the design codes, code cases, and acceptance criteria applicable to the design, analysis, fabrication, and NDE of the internals components. The discussion should identify internal components that are designated as core support structures and internal structures and discuss the implications of this designation on applicable design criteria. The applicant should also indicate the extent to which the design and construction of the core support structures are in accordance with appropriate subsection(s) of the ASME Code and the extent to which the design of other reactor internals will be consistent with appropriate article(s) of the ASME Code. RG 1.20 provides further details on the determination of loading conditions caused by adverse flow effects. (RG 1.20 provides requirements specific to features/elements of LWRs. However, the principles apply to HTGR technology.)

Modified Section/Title: C.I.3.9.5.3; Design Bases
Original Section/Title: C.I.3.9.5.3; Design Bases
The applicant should list all combinations of design and service loadings accounted for in the design of the reactor internals (e.g., acoustic and FIV, operating differential pressure and thermal loads, thermal stratification, seismic loads, FIV loads, acoustic loads, transient pressure loads associated with postulated depressurization events, and asymmetric blowdown pressurization and loading resulting from pipe ruptures at postulated locations that are not excluded based on LBB analyses). This section should define

these loads and describe the method of combining loads for normal, upset, emergency, and faulted service conditions. For each specific load combination, the applicant should provide the allowable design or service limits to be applied to the reactor internals. Considering the effects of component service environments, the applicant should provide the deflection, cycling, and fatigue limits. The applicant should also verify that the allowable deflections will not interfere with the functioning of all related components (e.g., control rod guide tubes and standby cooling systems). The applicant should provide a summary of the maximum calculated total stress, deformation, and cumulative usage factor for each designated design or service limit. FSAR Section 3.9.2 should present the details of the dynamic analyses.

Modified Section/Title: C.I.3.9.5.4; N/A
Original Section/Title: C.I.3.9.5.4; BWR Reactor Pressure Vessel Internals Including Steam Dryer

Modified Section/Title: C.I.3.9.6; Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints
Original Section/Title: C.I.3.9.6; Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves,
This section should describe the functional design and qualification provisions and inservice testing (IST) programs for certain safety-related valves and dynamic restraints (snubbers) (i.e., those safety-related pumps, valves, and snubbers designated as Class A or B under Section III of the ASME Code, plus those pumps, valves, and snubbers not categorized as Class A or B but considered to be safety-related) to ensure that they will be in a state of operational readiness to perform their safety functions throughout the life of the plant.

Modified Section/Title: C.I.3.9.6.1; Functional Design and Qualification of HPB Components and Dynamic Restraints
Original Section/Title: C.I.3.9.6.1; Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints
In this section, the applicant should do the following:

1. Describe the provisions in the design of safety-related components of the HPB that allow testing at the maximum flow rates specified in the plant accident analyses

2. Describe the provisions in the functional design and qualification of each safety-related HPB component that demonstrate the capacity of the components to perform their intended functions for a full range of system differential pressures and flows, ambient temperatures, and available voltage (as applicable) from normal operating to design-basis conditions

3. Verify that the qualification program for safety-related components that are part of the HPB includes testing and analyses to demonstrate that these components will not experience any leakage, or increase in leakage, from their loading

4. Describe the provisions in the functional design and qualification of dynamic restraints in safety-related systems and access for performing IST program activities that comply with the requirements in the latest edition and addenda of the ASME Operations and Maintenance Code (OM Code) incorporated by reference in 10 CFR 50.55a on the date 12 months before the date for initial fuel load

5. Give particular attention to flow-induced loading in functional design and qualification to incorporate degraded flow conditions such as those that might result from the presence of debris, impurities, and contaminants in the fluid system.

Modified Section/Title: C.I.3.9.6.2; Inservice Testing Program for Pumps
Original Section/Title: C.I.3.9.6.2; Inservice Testing Program for Pumps
Not Applicable

Modified Section/Title: C.I.3.9.6.3; Inservice Testing Program for Valves
Original Section/Title: C.I.3.9.6.3; Inservice Testing Program for Valves
See section 3.9.6.3.4.

Modified Section/Title: C.I.3.9.6.3.1; Inservice Testing Program for Motor-Operated Valves
Original Section/Title: C.I.3.9.6.3.1; Inservice Testing Program for Motor-Operated Valves
Not Applicable

Modified Section/Title: C.I.3.9.6.3.2; Inservice Testing Program for Power-Operated Valves Other Than MOVs
Original Section/Title: C.I.3.9.6.3.2; Inservice Testing Program for Power-Operated Valves Other Than MOVs
Not Applicable

Modified Section/Title: C.I.3.9.6.3.3; Inservice Testing Program for Check Valves
Original Section/Title: C.I.3.9.6.3.3; Inservice Testing Program for Check Valves
Not Applicable

Modified Section/Title: C.I.3.9.6.3.4; Pressure Isolation Valve Leak Testing
Original Section/Title: C.I.3.9.6.3.4; Pressure Isolation Valve Leak Testing
The applicant should list the pressure isolation valves, including the classification, allowable leak rate, and test interval for each valve.

Modified Section/Title: C.I.3.9.6.3.5; Containment Isolation Valve Leak Testing
Original Section/Title: C.I.3.9.6.3.5; Containment Isolation Valve Leak Testing
Not Applicable

Modified Section/Title: C.I.3.9.6.3.6; Inservice Testing Program for Safety and Relief Valves
Original Section/Title: C.I.3.9.6.3.6; Inservice Testing Program for Safety and Relief Valves
The applicant should provide a list of valves that are to be included in the IST program, including their type, code class, valve category, valve functions, test parameters, and test frequency.

Modified Section/Title: C.I.3.9.6.3.7; Inservice Testing Program for Manually Operated Valves
Original Section/Title: C.I.3.9.6.3.7; Inservice Testing Program for Manually Operated Valves
The applicant should list the manually operated valves, including their safety-related function.

Modified Section/Title: C.I.3.9.6.3.8; Inservice Testing Program for Explosively Activated Valves
Original Section/Title: C.I.3.9.6.3.8; Inservice Testing Program for Explosively Activated Valves
The applicant should list explosively actuated valves, including a test plan and corrective actions. (NOTE: This section may be N/A if HTGR design does not include squib valves. However, the application should state this if it is the case.)

Modified Section/Title: C.I.3.9.6.4; Inservice Testing Program for Dynamic Restraints

Original Section/Title: C.I.3.9.6.4; Inservice Testing Program for Dynamic Restraints

To describe the IST program for dynamic restraints, the applicant should do the following:

1. Provide a table listing all safety-related components that use snubbers in their support systems, including:
   a. Identification of the systems and components that use snubbers
   b. Indication of the number of snubbers used in each system and on the components in that system
   c. Identification of the type(s) of snubber (hydraulic or mechanical)
   d. Specification of the standards to which the snubbers comply
   e. A statement of whether the snubber is used as a shock, vibration, or dual-purpose snubber
   f. If a snubber is identified as either a dual-purpose or vibration arrester type, indicate whether the snubber and/or component were evaluated for fatigue strength, should be indicated.

2. Describe the IST program (including test frequency and duration and examination methods) related to visual inspections (e.g., checking for degradation, cracked fluid reservoirs, missing parts, and leakage) and functional testing of dynamic restraints; describe and state the basis for dynamic restraint testing

3. Describe the steps to be taken to ensure that all snubbers are properly installed before preoperational piping and plant startup tests

4. Confirm the accessibility provisions for maintenance, ISI and testing, and possible repair or replacement of snubbers

5. Describe the implementation program, including milestones, for the snubber IST programs that comply with the requirements in the latest edition and addenda of the OM Code incorporated by reference in 10 CFR 50.55a on the date 12 months before the date for initial fuel load.

Modified Section/Title: C.I.3.9.6.5; Relief Requests and Alternative Authorizations to ASME OM Code

Original Section/Title: C.I.3.9.6.5; Relief Requests and Alternative Authorizations to ASME OM Code

The applicant should provide information regarding components for which the applicant is requesting relief from (or proposing an alternative to) the ASME OM Code requirements. The information should include the following:

1. Identification of the component by name, number, functions, class under Section III of the ASME Code, valve category (as defined in ISTC-1033 of the ASME OM Code), and pump group (as defined in ISTB-2000 of the ASME OM Code)

2. Identification of the ASME OM Code requirement(s) from which the applicant is requesting relief or to which the applicants is requesting an alternative

3. For a relief request pursuant to 10 CFR 50.55a(f)(6)(I) or (g)(6)(I), the basis for requesting the relief and an explanation of why compliance with the ASME OM Code is impractical or should otherwise not be required

4. For an alternative request pursuant to 10 CFR 50.55a(a)(3), details regarding the proposed alternative(s) demonstrating that (1) the proposed IST will provide an acceptable level of quality and

safety, or (2) compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety

5. Description of the plan, including milestones, for implementing the proposed IST program.

Modified Section/Title: C.I.3.9.7; [Reserved]

Original Section/Title: C.I.3.9.7; [Reserved]
N/A

Modified Section/Title: C.I.3.9.8; [Reserved]

Original Section/Title: C.I.3.9.8; [Reserved]
N/A

Modified Section/Title: C.I.3.10; Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Original Section/Title: C.I.3.10; Seismic and Dynamic Qualification of Mechanical and Electrical Equipment
The applicant should identify all instrumentation, electrical equipment, and mechanical components (other than pipes), including their supports, that should be designed to withstand the effects of earthquakes and the full range of normal and accident loadings. This equipment should include (1) equipment associated with systems that are essential to emergency reactor shutdown and reactor core cooling, (2) equipment essential to preventing significant release of radioactive material to the environment, and (3) instrumentation needed to assess plant and environs conditions during and after an accident as described in RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." The applicant should identify equipment (1) that performs the above functions automatically, (2) that operators use to perform the above functions manually, and (3) for which failure can prevent satisfactory accomplishment of one or more of the above safety functions. This includes equipment in the RPS, ESF Class 1E equipment, the emergency power system, and all auxiliary safety related systems and supports. Examples of mechanical equipment include pumps, valves, fans, valve operators, snubbers, battery and instrument racks, control consoles, cabinets, and panels; examples of electrical equipment include valve operator motors, solenoid valves, relays, pressure switches, level transmitters, electrical penetrations, and pump and fan motors.

Modified Section/Title: C.I.3.10.1; Seismic Qualification Criteria

Original Section/Title: C.I.3.10.1; Seismic Qualification Criteria
The applicant should provide the criteria used for seismic qualification, including the decision criteria for selecting a particular test or method of analysis, the considerations defining the seismic and other relevant dynamic load input motion, and the process to demonstrate the adequacy of the seismic qualification program. The applicant should indicate the extent to which the seismic qualification criteria use the guidance in RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" and provide suitable justifications for any exceptions to this guidance.

Modified Section/Title: C.I.3.10.2; Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation

Original Section/Title: C.I.3.10.2; Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation
This section should describe the methods and procedures, including test and/or analysis results, used to ensure the structural integrity and functionality of mechanical and electrical equipment for operation in the event of an SSE. If the applicant is required to postulate an OBE, the applicant should address five occurrences of the OBE followed by a full SSE event or a number of fractional peak cycles equivalent to

the maximum peak cycle for five OBE events followed by one full SSE, in combination with other relevant design-basis loads.

Modified Section/Title: C.I.3.10.3; Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation
Original Section/Title: C.I.3.10.3; Methods and Procedures of Analysis or Testing of Supports of Mechanical and Instrumentation
In this section, the applicant should describe the methods and procedures, including results, used to analyze or test the supports for mechanical and electrical equipment, as well as the verification procedures used to account for possible amplification of vibratory motion (amplitude and frequency content) under seismic and dynamic conditions. The description should include supports for such items as battery racks and instrument racks, pumps, valves, valve operators, fans, control consoles, cabinets, panels, and cable trays.

Modified Section/Title: C.I.3.10.4; Test and Analyses Results and Experience Database
Original Section/Title: C.I.3.10.4; Test and Analyses Results and Experience Database
The applicant should provide the results of tests and analyses that demonstrate adequate seismic qualification. If the seismic and dynamic qualification testing is incomplete at the time of the application, the applicant should include an implementation program, including implementation program, including milestones and completion dates with appropriate information submitted for staff review and approval prior to installation and equipment. If qualification by experience is proposed, the applicant should submit for staff review and approval the methods and procedures, including details of the experience database, to ensure the structural integrity and functionality of the in-scope mechanical and electrical equipment as described in Section C.I.3.10.2 of this guide.

Modified Section/Title: C.I.3.11; Environmental Qualification of Mechanical and Electrical Equipment
Original Section/Title: C.I.3.11; Environmental Qualification of Mechanical and Electrical Equipment
The applicant should identify the electrical equipment (including instrumentation and control and certain accident monitoring equipment specified in RG 1.97) that is within the scope of 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants." This equipment must perform its safety functions under all normal environmental conditions, AE, and accident and post-accident environmental conditions. Mechanical and electrical equipment associated with systems essential to emergency reactor shutdown, reactor core cooling, and reactor heat removal, should be included. The applicant should also include equipment for which postulated failure might affect the safety function of safety-related equipment or mislead an operator, as well as equipment that is otherwise essential to prevent significant releases of radioactive material to the environment.

Modified Section/Title: C.I.3.11.1; Equipment Location and Environmental Conditions
Original Section/Title: C.I.3.11.1; Equipment Location and Environmental Conditions
This section should specify the location of each piece of equipment, both inside and outside reactor building. For equipment inside the reactor building, the applicant should specify whether the location is inside or outside of the missile shield design features or specify the equipment locations in accordance with harsh environment zones.

The applicant should specify both the normal and accident environmental conditions for each item of equipment, including temperature, pressure, humidity, radiation, chemicals, submergence, and vibration (nonseismic) at the location where the equipment must perform. For the normal environment, the applicant should provide specific values, including those attributable to loss of environmental control systems. For the accident environment, the applicant should identify the cause of the postulated environment (e.g., HPB break, steam line break, or other), specify the environmental conditions as a function of time, and identify the length of time that each item of equipment is required to operate in the accident environment.

Modified Section/Title: C.I.3.11.2; Qualification Tests and Analyses

Original Section/Title: C.I.3.11.2; Qualification Tests and Analyses

The applicant should demonstrate that (1) the equipment is capable of maintaining functional operability under all service conditions postulated to occur during the equipment's installed life for the time it is required to operate, and (2) failure of the equipment after performance of its safety function will not be detrimental to plant safety or mislead an operator. The applicant should consider all environmental conditions that may result from any normal mode of plant operation, AOOs, DBEs, and post DBEs. The applicant should describe the qualification tests and analyses performed on each item of equipment to ensure that it will perform under the specified normal and accident environmental conditions.

In this section, the applicant should document how the design will meet the requirements of 10 CFR 50.49; GDCs [PDCs] 1, 2, 4, and 23, "Protection system failure modes," of Appendix A to 10 CFR Part 50; and Criteria III, XI, and XVII of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. The applicant should indicate the extent to which it will use the guidance contained in applicable regulatory guides (some of which are listed below), or document and justify the use of alternative approaches:

1. RG 1.40, "Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water Cooled Nuclear Power Plants"

2. RG 1.63, "Electric Penetration Assemblies in Containment Structures for Light Water Cooled Nuclear Power Plants"

3. RG 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants"

4. RG 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants"

5. RG 1.97, "Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants"

6. RG 1.156, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants"

7. RG 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants"

8. (8) RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems"

9. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors"

10. RG 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants"

Modified Section/Title: C.I.3.11.3; Qualification Test Results

Original Section/Title: C.I.3.11.3; Qualification Test Results

The applicant should document the qualification test results and qualification status for each type of equipment. Because the Environmental Qualification program is an operational program, as discussed in SECY-05-0197, the program and its implementation milestones should be fully described and reference any applicable standards. "Fully described" should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow for a reasonable assurance finding of acceptability. This statement applies to all of Subsection 3.11.

Modified Section/Title: C.I.3.11.4; Loss of Ventilation

Original Section/Title: C.I.3.11.4; Loss of Ventilation

The applicant should describe the bases that ensure that loss of environmental control systems (e.g., heat tracing, ventilation, heating, air conditioning) will not adversely affect the operability of each item of equipment, including electric control and instrumentation equipment and instrument sensing lines that rely on heat tracing for freeze protection. The description should include the analyses performed to identify the "worst case" environment (e.g., temperature, humidity), including identification and determination of the limiting condition with regard to temperature that would require reactor shutdown. The applicant should describe any testing (factory or on site) performed to confirm satisfactory operability of control and electrical equipment under extreme environmental conditions and document the successful completion of qualification tests and qualification status for each type of equipment.

Modified Section/Title: C.I.3.11.5; Estimated Chemical and Radiation Environment

Original Section/Title: C.I.3.11.5; Estimated Chemical and Radiation Environment

The applicant should identify the chemical environment for both normal operation and the DBA inside the reactor building.

The applicant should identify the radiation dose and dose rate used to determine the radiation environment and indicate the extent to which estimates of radiation exposures are based on a radiation source term that is consistent with NRC staff-approved source terms and methodology (refer to NGNP white paper "Mechanistic Source Terms" INL/EXT-10-17997). For exposure of organic components on ESF systems, the applicant should tabulate beta and gamma exposures separately for each item of equipment and list the average energy of each type of radiation. For ESF systems outside the reactor building, the applicant should indicate whether the radiation estimates account for factors affecting the source term such hold-up/plate out inside the reactor building, meteorological dispersion (if appropriate), and operation of other ESF systems. The applicant should list all assumptions used in the calculation.

This section should document successful completion of qualification tests and qualification status for each type of equipment.

Modified Section/Title: C.I.3.11.6; Qualification of Mechanical Equipment

Original Section/Title: C.I.3.11.6; Qualification of Mechanical Equipment

The applicant should define the process for determining the suitability of environmentally sensitive mechanical equipment (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms) needed for safety-related functions and for verifying that the design of such materials, parts, and equipment is adequate. The applicant should identify the following:

1. Safety-related mechanical equipment located in harsh environmental areas

2. Non-metallic subcomponents of such equipment

3. The environmental conditions and process parameters for which this equipment must be qualified

4. The non-metallic material capabilities

5. The environmental effects on the non-metallic components of the equipment.

The applicant should document successful completion of qualification tests and/or analysis and qualification status for each type of equipment.

Modified Section/Title: C.I.3.12; Piping Design Review
Original Section/Title: C.I.3.12; Piping Design Review

Modified Section/Title: C.I.3.12.1; Introduction
Original Section/Title: C.I.3.12.1; Introduction
This section covers the design of the piping system and piping support for seismic Category I, Category II, and nonsafety systems. It also discusses the adequacy of the structural integrity, as well as the functional capability, of the safety-related piping system, piping components, and their associated supports. The design of piping systems should ensure that they perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. This includes pressure-retaining piping components and their supports, buried piping, instrumentation lines, and the interaction of non-seismic Category I piping and associated supports with seismic Category I piping and associated supports. This section covers the design transients and resulting loads and load combinations with appropriate specified design and service limits for seismic Category I piping and piping support, including those designated as ASME Code Class A and B, and those not covered by the ASME Code.

Modified Section/Title: C.I.3.12.2; Codes and Standards
Original Section/Title: C.I.3.12.2; Codes and Standards
The applicant should provide a table showing compliance with the NRC's regulations in 10 CFR 50.55a to the extent applicable to HTGR technology. This table should identify the piping system and associated supports.

The applicant should discuss the design and analyses of the piping system, including piping components and associated supports in accordance with Section III of the ASME Code. The discussion should cover requirements and procedures used in preparing the design specification of the piping system, including loading combinations, design data, and other design inputs. It should also identify design codes, standards, specifications, regulations, PDC, regulatory guides, and other industry standards used in the design or that will be used in the fabrication, construction, testing, and ISI of the piping system. The applicant should identify the specific edition, date, or addenda of each document.

The ASME Code cases that may be used for the design of the ASME Code Class l, 2,and 3 piping system are those recommended in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," some of which may be adaptable to Code Class A and B components for HTGR technology. The design reports for ASME Code Class A and B piping system and piping support should be available for NRC audit.

Modified Section/Title: C.I.3.12.3; Piping Analysis Methods
Original Section/Title: C.I.3.12.3; Piping Analysis Methods
The applicant should identify and describe the design consistent with seismic subsystem analysis related to seismic analysis methods (e.g., response spectrum analysis, modal time history analysis, direct

integration time history analysis, frequency domain time history analysis, equivalent static load analysis) used for seismic Category I and non-seismic Category I (seismic Category II and nonseismic) piping system and piping support.

The applicant should explain the manner in which the seismic dynamic analysis considers maximum relative displacement among supports and indicate other significant effects accounted for in the analysis, such as hydrodynamic effects and nonlinear response.

This section should describe the procedure used for analytical modeling, number of earthquake cycles, selection of frequencies, damping criteria (consistent with RG 1.61), combination of modal responses, equivalent static factors, the analysis for small bore piping, and interaction of Category I systems with other systems. Since there are numerous technical issues related to piping design and piping support other than seismic and those criteria discussed in this guide, the applicant should also discuss any acceptable methods that are common industry practices and/or practical engineering considerations proven through extensive experience.

Modified Section/Title: C.I.3.12.3.1; Experimental Stress Analyses

Original Section/Title: C.I.3.12.3.1; Experimental Stress Analyses
If the applicant uses experimental stress analysis methods in lieu of analytical methods for seismic Category I ASME Code and non ASME Code piping system design, it should provide sufficient information to show the validity of the design. It is recommended that, before using the experimental stress analysis method, the applicant submit the details of the method, as well as the scope and extent of its application, for NRC approval. The experimental stress analysis method should comply with ASME Code requirements applicable to HTGR technology as endorsed by the NRC.

Modified Section/Title: C.I.3.12.3.2; Modal Response Spectrum Method

Original Section/Title: C.I.3.12.3.2; Modal Response Spectrum Method
Modal response spectrum and time history methods form the basis for the analyses of all major seismic Category I piping systems and components. The applicant should describe the procedures for considering the three components of earthquake motion in determining seismic response of piping system and piping support and the procedure for combining modal responses (i.e., shears, moments, stresses, deflections, and accelerations), including that for modes with closely spaced frequencies. Also, the applicant should indicate the extent to which it has followed the recommendations of RG 1.92, including those applicable to adequate consideration of high-frequency modes, to combine modal responses.

If the applicant uses any alternative seismic analysis method, it should provide the basis for its conservatism and equivalence in safety to the applicable regulatory position.

Modified Section/Title: C.I.3.12.3.3; Response Spectra Method (or Independent Support Motion Method)

Original Section/Title: C.I.3.12.3.3; Response Spectra Method (or Independent Support Motion Method)
As an alternative to the enveloped response spectra method, the applicant may use independent support motion seismic analyses where there is more than one supporting structure for the piping system. This means that all supports are located on the same floor or portions of the floor of a structure. A support group is defined by supports that have the same time history input. The analysis combines the responses from motions of supports in two or more different groups by the square root sum of the squares method. For this procedure, the criteria for damping values should be consistent with those in RG 1.61.

Modified Section/Title: C.I.3.12.3.4; Time History Method

Original Section/Title: C.I.3.12.3.4; Time History Method

The applicant may perform a time history analysis using either the modal superposition method or the direct integration method. The applicant should include the following in its description of the seismic analysis method used:

1. Manner in which the dynamic system analysis is performed

2. Method chosen for selection of significant modes and an adequate number of masses or degrees of freedom

3. Manner in which the seismic dynamic analysis considers maximum relative displacements between supports

4. Other significant effects accounted for in the dynamic seismic analysis, such as piping interactions, externally applied structural restraints, hydrodynamic effects (both mass and stiffness effects), types of loading and condition, damping criteria, and nonlinear response.

If the applicant uses a static load method instead of a dynamic analysis, it should demonstrate that a simple model can realistically represent the system and that the method produces conservative results.

Modified Section/Title: C.I.3.12.3.5; Inelastic Analyses Method

Original Section/Title: C.I.3.12.3.5; Inelastic Analyses Method

The applicant should describe in detail the methodology, the specific system, and the acceptance criteria if it uses the inelastic method for piping design analyses. The acceptance criteria used should be consistent with those contained in Section 3.9.1 of this guide. Before using the inelastic method for analyses, the applicant should submit it for review and approval by the NRC.

Modified Section/Title: C.I.3.12.3.6; Small-Bore Piping Method

Original Section/Title: C.I.3.12.3.6; Small-Bore Piping Method

The response spectrum method is an acceptable seismic analysis methodology for evaluating both small- and large-bore piping. The applicant should describe in detail the method used for seismic analysis, including analyses procedure and criteria for small- and large-bore piping. If the applicant proposes an equivalent static load method, the method should be consistent with the recommendations of Section 3.9.2.II.2.a(2)(C) of this guide. The applicant should explain the basis for the method's conservatism and equivalence in safety to the applicable regulatory position.

Modified Section/Title: C.I.3.12.3.7; Nonseismic/Seismic Interaction (II/I)

Original Section/Title: C.I.3.12.3.7; Nonseismic/Seismic Interaction (II/I)

The applicant should describe the location of all piping systems (seismic Category I, seismic Category II, and non-seismic structures), including the distance between various piping systems. The applicant should provide the design criteria used to account for seismic motion of non- seismic Category I (seismic Category II and non-seismic) piping or portions thereof in the seismic design of seismic Category I structures or portions thereof. The description should include the seismic design of non-seismic Category I piping systems whose continued function is not required, but whose failure could adversely impact the safety function of SSCs. The applicant should describe the design criteria that it will apply to ensure functionality of seismic Category I systems despite impacts from the failure of non-seismic Category I piping because of seismic effects.

Modified Section/Title: C.I.3.12.3.8; Seismic Category I Buried Piping

Original Section/Title: C.I.3.12.3.8; Seismic Category I Buried Piping

The applicant should describe seismic criteria and methods for considering the effects of earthquakes on buried piping, conduits, tunnels, and auxiliary systems. These criteria should include compliance characteristics of soil media; dynamic pressures; seismic wave passage; and settlement resulting from earthquake and differential movements at support points, penetrations, and entry points into other structures provided with anchors.

Modified Section/Title: C.I.3.12.4; Piping Modeling Technique

Original Section/Title: C.I.3.12.4; Piping Modeling Technique

The applicant should provide criteria and procedures used for modeling that are applicable to seismic Category I ASME Code and non-ASME Code piping systems. The applicant should include criteria and bases used to determine whether the piping system and piping support are being analyzed as part of a system analysis or independently as a subsystem. The applicant should describe the types of model (finite element model, lumped-mass stick model, hybrid model, etc.) used for the seismic Category I piping system. Using methods recommended in Section 3.9.1 of this guide, the applicant should describe and provide verification of all computer programs used for analyses of seismic Category I piping designated as ASME Code Class A and B and non-ASME Code items. The applicant should describe the computer codes used for the design of the piping systems and supports and verify that these computer codes are in accordance with those used in the NRC benchmark problems appropriate for these piping analyses methods.

Modified Section/Title: C.I.3.12.4.1; Computer Codes

Original Section/Title: C.I.3.12.4.1; Computer Codes

The applicant should provide a list of computer programs used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I ASME Code and non-ASME Code piping systems, consistent with Section 3.9.1.2 of this guide.

Modified Section/Title: C.I.3.12.4.2; Dynamic Piping Model

Original Section/Title: C.I.3.12.4.2; Dynamic Piping Model

The applicant should describe the types of model (finite element, hybrid model, etc.) used for seismic Category I piping and piping support and provide the criteria and procedures used for modeling in the seismic system analyses. The applicant should indicate how the dynamic piping model for the seismic system analyses accounts for the effects of torsion (including eccentric masses), bending, shear, and axial deformations, and effects resulting from the changes in stiffness values of curved members. The applicant should also include the criteria and bases used to determine whether a piping system is analyzed as part of a larger structural system analysis or independently as a subsystem.

Modified Section/Title: C.I.3.12.4.3; Piping Benchmark Program

Original Section/Title: C.I.3.12.4.3; Piping Benchmark Program

The applicant should provide a list of computer programs used in dynamic and static analyses to determine the structural and functional integrity of the seismic Category I piping system design and the non-ASME Code piping system design. The applicant should also verify that the computer programs used for the analysis are in accordance with the appropriate NRC benchmark problems for the analyses methods used for design.

The applicant should provide the mathematical models for a series of selected piping systems and the associated analyses using the computer programs identified above. This section should compare the results of the analyses of each model to modal frequencies, maximum pipe moments, maximum support loads, maximum equipment nozzle loads, and maximum deflections. For values obtained using the computer program, the applicant should justify any deviations from values obtained using the approved dynamic analyses method.

Modified Section/Title: C.I.3.12.4.4; Decoupling Criteria

Original Section/Title: C.I.3.12.4.4; Decoupling Criteria

The applicant should provide the criteria used to decouple smaller piping systems from larger piping systems. When piping is supported by larger piping, the applicant should use either a coupled dynamic model of the supported piping and supporting piping or the amplified response spectra at the connection point to the supporting piping, with a decoupled model of the supported piping.

Modified Section/Title: C.I.3.12.5; Piping Stress Analysis Criteria

Original Section/Title: C.I.3.12.5; Piping Stress Analysis Criteria

Modified Section/Title: C.I.3.12.5.1; Seismic Input Envelope vs. Site-Specific Spectra

Original Section/Title: C.I.3.12.5.1; Seismic Input Envelope vs. Site-Specific Spectra

The applicant should provide design GMRS spectra for the SSE. If the ground response spectra differ from the generic ground response spectra, such as the response criteria provided in RG 1.60, the applicant should provide the procedure to calculate response spectra and its basis for each damping ratio used.

The applicant should describe the procedures, basis, and justification for developing floor response spectra as specified in RG 1.122. If the applicant uses a single artificial time history analysis method to develop floor response spectra, it should demonstrate that (1) provisions of RG 1.122, including peak broadening requirements, apply, and (2) the response spectra of the artificial time history to be employed in the free field envelop the free-field design response spectra for all damping values actually used in the response spectra. If the applicant applies multiple time histories to generate floor response spectra, it should provide the basis for the methods used to account for uncertainties in parameters.

Modified Section/Title: C.I.3.12.5.2; Design Transients

Original Section/Title: C.I.3.12.5.2; Design Transients

The applicant should provide a complete list of transients used in the design and fatigue analysis of all ASME Code Class A piping system and support components consistent with Section 3.9.1.1 of this guide.

Modified Section/Title: C.I.3.12.5.3; Loadings and Load Combination

Original Section/Title: C.I.3.12.5.3; Loadings and Load Combination

This section should provide the design and service loading combinations for piping system and pipe support, consistent with Section 3.9.3.1 of this guide.

Modified Section/Title: C.I.3.12.5.4; Damping Values

Original Section/Title: C.I.3.12.5.4; Damping Values

The applicant should provide the specific percentage of critical damping values used for seismic Category I piping system and piping support (e.g., damping values for the type of construction or fabrication). Also, the applicant should compare the damping values assigned to the piping system and piping support with the acceptable damping values provided in RG 1.61. The applicant should explain the basis for any proposed damping values that differ from those recommended in RG 1.61 and the rationale for the proposed variation.

Modified Section/Title: C.I.3.12.5.5; Combination of Modal Responses

Original Section/Title: C.I.3.12.5.5; Combination of Modal Responses

When using the response spectrum analysis method to evaluate seismic response of piping system and piping support, the applicant should describe the procedure for combining modal responses (i.e., shears, moments, stresses, deflections, and accelerations), including that for modes with closely spaced frequencies. Also, the applicant should indicate the extent to which it is following the recommendations for combining modal responses given in RG 1.92, including those applicable to adequate consideration of high-frequency modes.

Modified Section/Title: C.I.3.12.5.6; High-Frequency Modes
Original Section/Title: C.I.3.12.5.6; High-Frequency Modes
The applicant should describe the method used to account for selection of high-frequency modes in seismic response spectrum analysis of the piping system and piping support. The method proposed should be consistent with Section 3.7.2 of this guide. If the applicant proposes an alternative in lieu of these methods, it should provide the basis for the alternative's conservatism and equivalence in safety to the applicable regulatory position.

Modified Section/Title: C.I.3.12.5.7; Fatigue Evaluation of ASME Code Class A Piping
Original Section/Title: C.I.3.12.5.7; Fatigue Evaluation of ASME Code Class 1 Piping
The applicant should describe the method used to account for effects of the environment on the fatigue design of the piping system.

Modified Section/Title: C.I.3.12.5.8; Fatigue Evaluation of ASME Code Class B Piping
Original Section/Title: C.I.3.12.5.8; Fatigue Evaluation of ASME Code Class 2 and 3 Piping
This section should describe the method used to account for effects of the environment on the fatigue design of the Class B piping system and associated support.

Modified Section/Title: C.I.3.12.5.9; Thermal Oscillations in Piping Connected to the Helium Pressure Boundary
Original Section/Title: C.I.3.12.5.9; Thermal Oscillations in Piping Connected to the Reactor Coolant System
The applicant should describe the piping stress analysis methodology developed for the design of the piping system connected to the HPB for identification and evaluation of piping systems susceptible to thermal stresses from unanalyzed temperature oscillation. The applicant should describe a program to ensure continued integrity of the piping system consistent with NRC Bulletin Letter 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," issued in June 1988. If the applicant proposes an alternative in lieu of these methods to ensure the integrity of the piping system, it should provide the basis for the alternative's conservatism and equivalence in safety to the applicable regulatory position.

Modified Section/Title: C.I.3.12.5.10; Thermal Stratification
Original Section/Title: C.I.3.12.5.10; Thermal Stratification
Not Applicable

Modified Section/Title: C.I.3.12.5.11; Safety Relief Valve Design, Installation, and Testing
Original Section/Title: C.I.3.12.5.11; Safety Relief Valve Design, Installation, and Testing
The applicant should describe the design and installation criteria applicable to the piping system and piping support when connected to pressure relief devices (i.e., safety and relief valves) for overpressure protection of ASME Class A and B components meeting the criteria specified in Section 3.9.3.2 of this guide.

Modified Section/Title: C.I.3.12.5.12; Functional Capability
Original Section/Title: C.I.3.12.5.12; Functional Capability
The applicant should identify and describe the design of all ASME Code Class A and B piping systems whose functionality is essential for safe shutdown for all Service Level D loading conditions. The design

should be consistent with recommendations in NUREG-1367, "Functional Capability of Piping Systems," and PDC 2.

Modified Section/Title: C.I.3.12.5.13; Combination of Inertial and Seismic Anchor Motion Effects

Original Section/Title: C.I.3.12.5.13; Combination of Inertial and Seismic Anchor Motion Effects

If piping is supported at multiple locations within a single structure or is attached to two separate structures, the applicant should describe the methods and analyses of the piping system relative to building movements at supports and anchors (seismic anchor motion), as well as with respect to the effects of seismic inertial loads. The applicant should also evaluate the effects of relative displacements at support points by imposing the maximum support displacements in the most unfavorable combination consistent with Section 3.9.2 of this guide.

Modified Section/Title: C.I.3.12.5.14; Operating-Basis Earthquake as a Design Load

Original Section/Title: C.I.3.12.5.14; Operating-Basis Earthquake as a Design Load

Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50 allows the use of single-earthquake design by providing the option to use an OBE value of one third the maximum vibratory ground acceleration of the SSE and to eliminate the requirement to perform explicit response analyses for the OBE.

For applications that use this option, the applicant should provide an evaluation to determine the effects of displacement-limited seismic anchor motions on ASME Code components and supports to ensure their functionality during and following an SSE. For piping systems, the evaluation should combine the effects of seismic anchor motions from an SSE with the effects of other normal operational loadings that might occur concurrently. NUREG-1503, "Final Safety Evaluation Report (SER) Related to Certification of the Advanced BWR Design," issued in 1994, states the conditions for these criteria.

Modified Section/Title: C.I.3.12.5.15; Welded Attachments

Original Section/Title: C.I.3.12.5.15; Welded Attachments

The applicant should describe and explain the design of support members, connections, or attachments welded to piping. These should be designed such that their failure under unanticipated loads does not cause failure in the pipe pressure boundary. Any code cases used as the basis for design of welded attachments should be consistent with those in RG 1.84 or otherwise approved by the NRC.

Modified Section/Title: C.I.3.12.5.16; Modal Damping for Composite Structures

Original Section/Title: C.I.3.12.5.16; Modal Damping for Composite Structures

The applicant should describe the procedure used to determine the composite modal damping value for the piping system. Composite modal damping for coupled building and piping systems may be used for piping systems that are coupled to concrete building structures.

Composite modal damping may also be used for piping systems coupled to flexible equipment or flexible valves. The composite modal damping approach should be consistent with the acceptance criteria given in Section 3.7.2 of this guide.

Modified Section/Title: C.I.3.12.5.17; Minimum Temperature for Thermal Analyses

Original Section/Title: C.I.3.12.5.17; Minimum Temperature for Thermal Analyses

This section should provide the thermal expansion analyses criteria for the piping design to evaluate the stresses and loadings above the stress-free reference temperature.

Modified Section/Title: C.I.3.12.5.18; Intersystem Accident
Original Section/Title: C.I.3.12.5.18; Intersystem Loss-of-Coolant Accident
This section should describe and evaluate the various design features of the low-pressure piping systems that interface with the HPB. The design of the low-pressure piping systems should be such that it can withstand full heat transport system pressure without compromising its functionality.

Modified Section/Title: C.I.3.12.5.19; Effects of Environment on Fatigue Design
Original Section/Title: C.I.3.12.5.19; Effects of Environment on Fatigue Design
The applicant should describe the method and procedures used to account for the effects of the environment on the fatigue design of piping system and associated support connected to HPB components. The method proposed should be consistent with the recommendations of the RG 1.76.

Modified Section/Title: C.I.3.12.6; Piping Support Design Criteria
Original Section/Title: C.I.3.12.6; Piping Support Design Criteria
This section should describe the method used in the design of ASME Code Class A and B pipe supports.

Modified Section/Title: C.I.3.12.6.1; Applicable Codes
Original Section/Title: C.I.3.12.6.1; Applicable Codes
The applicant should provide design codes, standards, specifications, regulations, PDC, regulatory guides, and other industry standards that are used in the design or that will be used in the fabrication, construction, testing, and ISI of the piping support. The application should identify the specific edition, date, or addenda of each document.

Modified Section/Title: C.I.3.12.6.2; Jurisdictional Boundaries
Original Section/Title: C.I.3.12.6.2; Jurisdictional Boundaries
This section should describe the jurisdictional boundaries between pipe supports and interface attachment points. The jurisdictional boundaries should be in accordance with Subsection NF of Section III of the ASME Code and AISC N690 (1994) including Supplement 2 (2004).

Modified Section/Title: C.I.3.12.6.3; Loads and Load Combinations
Original Section/Title: C.I.3.12.6.3; Loads and Load Combinations
The applicant should provide loads, loading combinations (including system operating transients), and stress criteria for piping supports, including margins of safety. The stress limits for pipe support designs should meet the criteria of ASME Code Section III, Subsection NF.

Modified Section/Title: C.I.3.12.6.4; Pipe Support Baseplate and Anchor Bolt Design
Original Section/Title: C.I.3.12.6.4; Pipe Support Baseplate and Anchor Bolt Design
The applicant should describe the design of pipe support baseplate and anchor bolts. The design of the pipe support baseplate and anchor bolts should be consistent with NRC Bulletin Letter 79 02, Revision 2, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," issued in March 1979. If any other design is used, provide the basis for the design's conservatism and equivalence in safety to the applicable regulatory position.

Modified Section/Title: C.I.3.12.6.5; Use of Energy Absorbers and Limit Stops
Original Section/Title: C.I.3.12.6.5; Use of Energy Absorbers and Limit Stops
The applicant should provide the design and analyses of the special engineered supports (rigid gapped supports) used in the piping system. The recommended analyses consist of an iterative response spectra analysis of the piping and support system. The iterations establish calculated piping displacements that are compatible with the stiffness and gap of the rigid gapped supports.

Modified Section/Title: C.I.3.12.6.6; Use of Snubbers
Original Section/Title: C.I.3.12.6.6; Use of Snubbers
If the applicant proposes to use hydraulic snubbers for piping support, the design and analyses should be consistent with Section 3.9.3.2 of this guide.

Modified Section/Title: C.I.3.12.6.7; Pipe Support Stiffnesses
Original Section/Title: C.I.3.12.6.7; Pipe Support Stiffnesses
The applicant should discuss and describe pipe support stiffness values and support deflection limits used in the piping analyses and support designs.

Modified Section/Title: C.I.3.12.6.8; Seismic Self-Weight Excitation
Original Section/Title: C.I.3.12.6.8; Seismic Self-Weight Excitation
This section should describe the design and analyses with consideration of the service loading combination resulting from postulated events and the designation of appropriate service limits for pipe support seismic loads.

Modified Section/Title: C.I.3.12.6.9; Design of Supplementary Steel
Original Section/Title: C.I.3.12.6.9; Design of Supplementary Steel
The applicant should describe the design and analysis of structural steel used as pipe supports. The design of pipe support from structural steel should be in accordance with Subsection NF of Section III of the ASME Code and AISC N690 (1994) including Supplement 2 (2004).

Modified Section/Title: C.I.3.12.6.10; Consideration of Friction Forces
Original Section/Title: C.I.3.12.6.10; Consideration of Friction Forces
For sliding type of supports, the applicant should describe and analyze the friction loads induced by the pipe on the support.

Modified Section/Title: C.I.3.12.6.11; Pipe Support Gaps and Clearances
Original Section/Title: C.I.3.12.6.11; Pipe Support Gaps and Clearances
This section should provide information on pipe support gaps and clearances to be used between the pipe and the frame type of support.

Modified Section/Title: C.I.3.12.6.12; Instrumentation Line Support Criteria
Original Section/Title: C.I.3.12.6.12; Instrumentation Line Support Criteria
The applicant should provide the design criteria for instrumentation line supports. The design loads and load combinations for safety-related instrumentation supports are similar to those for pipe supports. The design for instrumentation line support should be in accordance with criteria described in ASME Code Section III, Subsection NF.

Modified Section/Title: C.I.3.12.6.13; Pipe Deflection Limits
Original Section/Title: C.I.3.12.6.13; Pipe Deflection Limits
The applicant should provide and describe the pipe deflection limits for standard component pipe supports. The standard component pipe support movement should remain within the manufacturer's recommended design limits. This criterion applies to limit stops, snubbers, rods, hangers, and sway struts.

Modified Section/Title: C.I.3.13; Threaded Fasteners (ASME Code Class A and B)
Original Section/Title: C.I.3.13; Threaded Fasteners (ASME Code Class 1, 2, and 3)
The applicant should provide the criteria used to select materials to fabricate threaded fasteners (e.g., threaded bolts, studs) in ASME Code Class A or B systems, as well as the criteria to fabricate, design, test, and inspect the threaded fasteners in these systems, both before initial service and during service.

Modified Section/Title: C.I.3.13.1; Design Considerations
Original Section/Title: C.I.3.13.1; Design Considerations
See subsections 3.13.1.1 through 3.13.1.5.

Modified Section/Title: C.I.3.13.1.1; Materials Selection
Original Section/Title: C.I.3.13.1.1; Materials Selection
The applicant should provide information pertaining to the selection of materials and material testing of threaded fasteners and indicate the level of conformance with applicable codes or standards. For threaded fasteners made from ferritic steels (i.e., low-alloy steel or carbon grades), the applicant should discuss the material testing used to establish the fracture toughness of the materials.

Modified Section/Title: C.I.3.13.1.2; Special Materials Fabrication Processes and Special Controls
Original Section/Title: C.I.3.13.1.2; Special Materials Fabrication Processes and Special Controls
The applicant should provide information pertaining to the fabrication of threaded fasteners. It should identify particular fabrication practices or special processes used to mitigate the occurrence of stress-corrosion cracking or other forms of material degradation in the fasteners during service. The discussion should include any environmental considerations related to the selection of materials used to fabricate threaded fasteners. The applicant should discuss the use of lubricants and/or surface treatments in mechanical connections secured by threaded fasteners.

Modified Section/Title: C.I.3.13.1.3; Fracture Toughness Requirements for Threaded Fasteners Made of Ferriti Materials
Original Section/Title: C.I.3.13.1.3; Fracture Toughness Requirements for Threaded Fasteners Made of Ferriti Materials
For threaded fasteners in ASME Code Class A systems that are fabricated from ferritic steels, the applicant should discuss the fracture toughness tests performed on the threaded fasteners and demonstrate compliance with applicable acceptance criteria set forth in Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50.

Modified Section/Title: C.I.3.13.1.4; [Reserved]
Original Section/Title: C.I.3.13.1.4; [Reserved]

Modified Section/Title: C.I.3.13.1.5; Certified Material Test Reports
Original Section/Title: C.I.3.13.1.5; Certified Material Test Reports
The applicant should summarize the material fabrication results and material property test results in the certified material test reports, pursuant to ASME Code requirements as endorsed by the NRC.

Modified Section/Title: C.I.3.13.2; Inservice Inspection Requirements
Original Section/Title: C.I.3.13.2; Inservice Inspection Requirements
The applicant should demonstrate compliance with the ISI requirements of 10 CFR 50.55a and Section XI of the ASME Code, Division 2.

If the preservice inspections, fracture toughness testing, or certified material test reports are incomplete at the time the application is filed, the applicant should describe the implementation program, including milestones, completion dates and expected conclusions.

Appendix D

Chapter 4. Reactor System

Appendix D
Chapter 4. Reactor System

Modified Section/Title: C.I.4; Reactor System

Original Section/Title: C.I.4; Reactor

Chapter 4 of the FSAR should provide an evaluation and supporting information to establish the capability of the reactor system to perform its safety functions throughout its design lifetime under all normal operational modes, including transient, steady-state, and accident conditions. This chapter should also include information to support the accident safety and risk analyses provided in Chapters 15 and 19.

The information submitted in this chapter should reference typical or bounding fuel and reactor design information. The applicant must later submit Cycle 1-related core fuel design, control rod design, core loading pattern, and related core parameters (related to Sections C.I.4.3 and C.I.4.4) for approval.

Modified Section/Title: C.I.4.1; Summary Description

Original Section/Title: C.I.4.1; Summary Description

In this section, the applicant should provide a summary description of the mechanical, nuclear, and thermal-hydraulic designs of the various reactor system components which includes the reactor core, reactor internals, and neutron control components. (Information on the reactor vessel is provided in Section 5.3.) This summary description should indicate the independent and interrelated performance and safety functions of each component. (Information on control rod drive mechanism and reactor internals provided in Sections 3.9.4 and 3.9.5 of the FSAR may be incorporated by reference, or the information on these components provided in this section may be incorporated in Sections 3.9.4 and 3.9.5 by reference.)

This summary description should also describe the major elements of Section 1.3 that are applicable to the design of the reactor system. The description should include functional requirements, the role of the reactor system in the overall safety design, and principal design criteria. Interfaces with other systems should be identified.

In addition, this description should include a summary table of the important design and performance characteristics as well as a tabulation of analysis techniques used and load conditions considered (including computer code names).

Modified Section/Title: C.I.4.2; Reactor System Design

Original Section/Title: C.I.4.2; Fuel System Design

The reactor system is defined as consisting of the following components: reactor core, reactor internals, and neutron control components. The reactor core is defined as consisting of the standard fuel elements and reserve shutdown control fuel elements (including, e.g., fuel particles, fuel compacts, prismatic graphite fuel element blocks, fuel hole plugs, lumped burnable poison, dowels and sockets, and the fuel handling hole), the standard and neutron control replaceable side and center graphite reflector elements, upper and lower replaceable graphite reflector elements, and the startup neutron sources. The reactor internals are defined as consisting of the permanent graphite side reflector, the metallic upper core restraint elements, the upper plenum shroud, the lower core graphite and metallic support structures, the metallic core barrel and coolant riser channels on the exterior of the core barrel, and seismic keys. Neutron control components are defined as consisting of the control rods and the reserve shutdown control material. For the control rods, this section should include the reactivity control elements that extend from the coupling interface of the control rod drive mechanism described in Chapter 3. For the reserve shutdown material, this section should include the reactivity control material that is released from the reserve shutdown equipment above the core described in Chapter 3. In addition, applicants should present the design bases for the mechanical, chemical, and thermal-hydraulic designs of the reactor system, which can affect or limit the safe, reliable operation of the plant.

The description of the reactor system design should include, as a minimum, the following aspects for normal operating, AOO, and postulated accident conditions:

1. Reactor Core
    a. Fast neutron fluence limits
    b. Peak allowable graphite stress
    c. Maximum metallic upper core restraint element temperature
    d. Maximum fuel and graphite element temperature and time at temperature as a function of burnup and operating conditions
    e. Fuel particle coating integrity criteria, both as-manufactured and during reactor operation.

2. Reactor Internals
    a. Core barrel temperature during conduction cooldown DBEs and DBAs
    b. Upper plenum shroud temperature during conduction cooldown DBEs and DBAs.

3. Neutron Control Components
    a. Maximum control rod cladding temperature.

The description should include a listing of material properties and the considerations that were taken into account in materials selection. The description should address coolant chemistry, including consideration of all possible reactor system/coolant impurity interactions.

Modified Section/Title: C.I.4.2.1; Design Bases

Original Section/Title: C.I.4.2.1; Design Bases

Applicants should explain and substantiate the selection of design bases from the perspective of safety considerations. Where the limits selected are consistent with proven practice, a referenced statement to that effect will suffice; however, where the limits exceed present practice, this section should provide an evaluation and explanation based on developmental work or analysis. These design bases may be expressed as either explicit numbers or general conditions. In addition, the discussion of design bases should include a description of the functional characteristics in terms of desired performance under stated conditions. This should relate systems, components, and materials performance under normal operating, AOO, and postulated accident conditions. The discussion should consider the following aspects with respect to performance:

1. Fuel particle kernel and coatings
    a. Mechanical properties of the coatings (e.g., Young's modulus, Poisson's ratio, design dimensions, strength, and failure modes and limits) and effects of design temperature and irradiation on those properties
    b. Coating stress-strain limits
    c. Chemical properties of the fuel kernel and coatings
    d. Fission product retention characteristics.

2. Fuel compacts
    a. Thermal-physical properties of the fuel compacts (e.g., thermal conductivity, density, and specific heat) and effects of design temperature and irradiation on those properties
    b. Effects of irradiation-induced dimensional change
    c. Chemical properties of the fuel compacts
    d. Fission product retention characteristics.

3. Lumped burnable poison
   a. Thermal-physical properties of the absorber material
   b. Compatibility of the absorber and fuel element graphite
   c. Irradiation behavior of absorber material.

4. Fuel and replaceable reflector elements
   a. Thermal-physical properties of the graphite (e.g., thermal conductivity, density, and specific heat) and effects of design temperature and irradiation on those properties
   b. Effects of irradiation-induced dimensional change
   c. Chemical properties of the graphite
   d. Fission product retention characteristics.

5. Fuel performance/radionuclide retention
   a. Analytical models and conservatism in the input data
   b. Ability of the models to predict experimental or operating characteristics
   c. Standard deviation or statistical uncertainty associated with the correlations or analytical models.

6. Reactor internals
   a. Structural design
   a. Thermal-hydraulic design
   b. Compatibility with coolant impurities.

7. Control rods
   a. Thermal-physical properties of the absorber material
   b. Compatibility with coolant impurities
   c. Compatibility of the absorber and cladding materials
   d. Cladding stress-strain limits
   e. Irradiation behavior of absorber material.

8. Reserve shutdown material
   a. Thermal-physical properties of the absorber material
   b. Compatibility with coolant impurities.

Modified Section/Title: C.I.4.2.2; Description and Design Drawings

Original Section/Title: C.I.4.2.2; Description and Design Drawings

Applicants should provide a description and final design drawings of the reactor system (reactor core, reactor internals, and neutron control components) showing arrangements, dimensions, critical tolerances, and handling features. In addition, this section should include a discussion of design features that prevent improper orientation or placement of fuel within the core.

Applicants should at a minimum also provide the following design drawings:

- fuel particle schematic
- fuel compact schematic
- fuel element cross-section
- fuel element outline
- fuel element schematic
- replaceable reflector element cross-section
- replaceable reflector element outline
- replaceable reflector schematic
- control rod cross-section

- control rod outline
- control rod schematic
- reserve shutdown material schematic
- metallic upper core restraint element schematic
- lumped burnable poison rod schematic
- reactor core cross-section
- reactor core outline
- reactor internals cross-section
- reactor internals outline
- graphite core support outline
- metallic core support outline
- core barrel cross section with coolant riser channels.

Modified Section/Title: C.I.4.2.3; Design Evaluation

Original Section/Title: C.I.4.2.3; Design Evaluation

Applicants should provide an evaluation of the reactor system design for the physically feasible combinations of chemical, thermal, irradiation, mechanical/structural, and hydraulic interactions, as well as fuel and radionuclide retention performance. The evaluation of these interactions should include the effects of normal reactor operations, AOOs, and postulated accidents. A discussion of potential failure modes and effects should be provided for reactor system components.

When conclusive operating experience is not available, applicants should discuss any prototype testing associated with the fuel and reactor system design, including in-reactor testing of design features and lead elements of a new design.

Modified Section/Title: C.I.4.2.4; Testing and Inspection Plan

Original Section/Title: C.I.4.2.4; Testing and Inspection Plan

This section should describe the testing and inspections to be performed to verify the design characteristics of the reactor core components, including fuel particle coating integrity; fuel element dimensions; fuel enrichment; burnable poison concentration; and characteristics of the fuel, control rod compacts, and reserve shutdown material. This section should also include descriptions of the inspection program for new fuel elements and new control rods. Testing and inspections for reactor internals, including the core support structure should also be described. Where testing and inspection programs are essentially the same for plants previously licensed (or designs previously certified) under 10 CFR Part 50 or 10 CFR Part 52, applicants should provide a statement to that effect, along with an identification of the fabricator and a table summarizing the important design and performance characteristics.

This section should describe the online fuel performance monitoring methods and the postirradiation fuel and replaceable reflector surveillance package as well as surveillance of control rods.

Modified Section/Title: C.I.4.3; Nuclear Design

Original Section/Title: C.I.4.3; Nuclear Design

Modified Section/Title: C.I.4.3.1; Design Bases

Original Section/Title: C.I.4.3.1; Design Bases

This section should provide and discuss the design bases for the nuclear design of the reactor core and reactivity control systems, including initial and equilibrium core fuel loadings and mass flow rates, nuclear and reactivity control limits such as excess reactivity, fuel burnup, negative reactivity feedback, core design lifetime, fuel replacement program, reactivity coefficients, stability criteria, maximum controlled reactivity insertion rates, control of power distribution, shutdown margins, stuck rod criteria, rod speeds, burnable poison requirements, and reserve shutdown provisions. Information should be provided regarding overall reactivity control requirements, including maximum operating excess reactivity, hot to cold temperature effect, xenon and other short term radionuclide decay, and shutdown margin. Information regarding time dependent decay heat following reactor shutdown should also be provided.

Modified Section/Title: C.I.4.3.2; Description
Original Section/Title: C.I.4.3.2; Description
Applicants should describe the nuclear characteristics of the design, including the information indicated in the following subsections.

Modified Section/Title: C.I.4.3.2.1; Nuclear Design Description
Original Section/Title: C.I.4.3.2.1; Nuclear Design Description
Applicants should list, describe, or illustrate features of the nuclear design that are not discussed in specific subsections for appropriate times in the fuel cycle. The description should include such areas as fuel enrichment distributions, burnable poison distributions, other physical features of the fuel elements or active core relevant to nuclear design parameters, delayed neutron fraction and neutron lifetimes, core lifetime and burnup, plutonium buildup, and the relationship to cooldown, xenon burnout, or other transient requirements.

Modified Section/Title: C.I.4.3.2.2; Power Distribution
Original Section/Title: C.I.4.3.2.2; Power Distribution
This section should provide full quantitative information on calculated "normal" power distributions, including distributions within fuel elements, axial distributions, gross radial distributions, and nonseparable aspects of radial and axial distributions. Information regarding the resulting fast neutron fluence and fuel burnup distributions should also be provided and compared with the design limits for these parameters. This should include a full range of both representative and limiting power distributions related to representative and limiting conditions of such relevant parameters as power, flow, flow distribution, rod patterns, time in cycle (burnup and possible burnup distributions), cycle, burnable poison, and xenon. The information should cover these patterns in sufficient detail to ensure that normally anticipated distributions are fully described and the effects of all parameters important in affecting distributions are displayed. This should include details of transient power shapes and magnitudes accompanying normal transients, such as load following, xenon buildup, decay or redistribution, and xenon oscillation control. Applicants should describe the radial power distribution within a fuel element and its variation with fluence and/or burnup if this is used in thermal calculations.

This section should discuss and assign specific magnitudes to errors or uncertainties that may be associated with these calculated distributions and provide the experimental data, including results from both critical experiments and previous or current operating reactors that support the analysis, likely distribution limits, and assigned uncertainty magnitudes. It should also discuss experimental checks to be performed on this reactor as well as the criteria for satisfactory results.

Applicants should provide detailed descriptions of the design power distributions (shapes and magnitudes) and design peaking factors to be used in steady-state limit statements and transient analysis initial conditions. The description should include all relevant components and such variables as maximum allowable peaking factors versus axial position or changes over the fuel cycle. Applicants should justify the selections by discussing the relationships of these design assumptions to the previously provided expected and limiting distributions and uncertainty analysis.

This section should describe the relationship of these distributions to the monitoring instrumentation, discussing in detail the adequacy of the number of instruments and their spatial deployment (including allowed failures); required correlations, if any, between readings and peaking factors, calibrations and errors, and operational procedures and specific operational limits; axial and azimuthal asymmetry limits; limits for alarms, and rod blocks, scrams, and other items to demonstrate that sufficient information is available to determine, monitor, and limit distributions associated with normal operation to within proper limits. Applicants should describe in detail all calculations, computer codes, and computers used in the course of operations that are involved in translating power distribution-related measurements into calculated power distribution information. This section should provide the frequency with which the calculations are normally performed and execution times of the calculations. It should also describe the input data required for the codes. In addition, applicants should provide a full quantitative analysis of the uncertainties associated with the sources and processing of information used to produce operational power distribution results. This should include consideration of allowed instrumentation failures.

Modified Section/Title: C.I.4.3.2.3; Reactivity Coefficients
Original Section/Title: C.I.4.3.2.3; Reactivity Coefficients
This section should provide full quantitative information on calculated reactivity coefficients, including the fuel Doppler coefficient, moderator coefficient, and power coefficient. It should state the precise definitions or assumptions related to parameters involved (e.g., effective fuel temperature for Doppler, parameters held constant in the power coefficient, spatial variation of parameters, and flux weighting used). The information should primarily take the form of curves covering the full applicable range of parameters (temperature and power) from cold startup through limiting values used in accident analyses. It should include quantitative discussions of both spatially uniform parameter changes and those nonuniform parameter and flux weighting changes appropriate to operational and accident analyses as well as the methods used to treat nonuniform changes in transient analyses.

Applicants should provide sufficient information to illustrate the normal and limiting values of parameters appropriate to operational and accident states, considering factors such as cycle, time in cycle, control rod insertions, burnable poisons, and power distribution. This section should discuss potential uncertainties in the calculations and experimental results that support the analysis and assigned uncertainty magnitudes and experimental checks to be made in this reactor. Where limits on coefficients are especially important (e.g., positive moderator coefficients in the power range), applicants should fully detail the experimental checks on these limits.

This section should provide the coefficients actually used in transient analyses and show (by reference to previous discussions and uncertainty analyses) that suitably conservative values are used (1) for both beginning-of-life (BOL) and end-of-life (EOL) analyses, (2) where most negative or most positive (or least negative) coefficients are appropriate, and (3) where spatially nonuniform changes are involved.

Modified Section/Title: C.I.4.3.2.4; Control Requirements
Original Section/Title: C.I.4.3.2.4; Control Requirements
This section should provide tables and discussions related to core reactivity balances for BOL, EOL, and (where appropriate) intermediate conditions. The discussions should consider such reactivity influences as control rod bank and reserve shutdown material requirements and expected and minimum worths, burnable poison worths, stuck rod allowances, moderator and fuel temperature defects, burnup and fission products, xenon and samarium poisoning and decay of other short-lived radionuclides, permitted rod insertions at power, and error allowances. Applicants should also provide and discuss the required and expected shutdown margin as a function of time in cycle, along with uncertainties in the shutdown margin and experimental confirmations from previous or current operating reactors.

Applicants should fully describe all methods, paths, and limits for normal operational control involving such areas as control rod motion and use of the reserve shutdown material. Descriptions should consider cold, hot, and peak xenon startup, load following and xenon reactivity control, power shaping (e.g., xenon redistribution or oscillation control), and burnup.

Modified Section/Title: C.I.4.3.2.5; Control Rod Patterns and Reactivity Worths

Original Section/Title: C.I.4.3.2.5; Control Rod Patterns and Reactivity Worths

This section should provide full information on control rod patterns expected to be used throughout a fuel cycle. It should include details concerning separation into groups or banks if applicable; order and extent of withdrawal of individual rods or banks; limits (with justification) to be imposed on rod or bank positions as a function of power level and/or time in cycle or for any other reason; and expected positions of rods or banks for cold critical, hot standby critical, and full power for both BOL and EOL.

Applicants should describe allowable deviations from these patterns for misaligned or stuck rods or for any other reason (such as spatial power shaping). For allowable patterns (including allowable deviations), applicants should indicate for various power, EOL, and BOL conditions the maximum worth of rods that might be postulated to be removed from the core in rod withdrawal accidents. Applicants should also give the worths of these rods as a function of position, describe any experimental confirmations of these worths, and provide maximum reactivity increase rates associated with these withdrawals.

This section should describe fully and give the methods for calculating the scram reactivity as a function of time after scram signal, including consideration of scram time, stuck rods, power level and shape, time in cycle, and any other parameters important for control rod reactivity worth and axial reactivity shape functions.

Modified Section/Title: C.I.4.3.2.6; Criticality of Reactor During Refueling

Original Section/Title: C.I.4.3.2.6; Criticality of Reactor During Refueling

This section should state the maximum value of Keff for the reactor during refueling and describe the basis for assuming that this maximum value will not be exceeded.

Modified Section/Title: C.I.4.3.2.7; Stability

Original Section/Title: C.I.4.3.2.7; Stability

This section should define the degree of predicted stability with regard to xenon oscillations in both the axial direction and the horizontal plane. If any form of xenon instability is predicted, it should include evaluations of higher-mode oscillations. Applicants should describe in detail the analytical and experimental bases for the predictions and include an assessment of potential error in the predictions. Applicants should also show how unexpected oscillations would be detectable before safety limits are exceeded.

This section should provide unambiguous positions regarding stability or lack thereof. That is, where stability is claimed, it should provide corroborating data from sufficiently similar power plants or provide commitments to demonstrate stability. Applicants should indicate criteria for determining whether the reactor will be stable. Where instability or marginal stability is predicted, applicants should provide details regarding the detection and control of oscillations as well as provisions for protection against exceeding safety limits. In cases in which the applicant does not provide a means for detecting and suppressing instabilities, the application should include a methodology for predicting margins to instability, and show that the reactor meets adequate acceptance criteria in this regard. A stability analysis should be performed on a cycle-specific basis to determine the limits of operation where stability is assured. Applicants should incorporate the commitment to perform the analysis with an approved methodology through reporting requirements (Section 5.6.5) in the Technical Specifications.

In addition, applicants should provide analyses of overall reactor stability against power oscillations (other than xenon).

Modified Section/Title: C.I.4.3.2.8; Vessel Irradiation

Original Section/Title: C.I.4.3.2.8; Vessel Irradiation

This section should provide the neutron flux distribution and spectrum in the core, at core boundaries, and at the pressure vessel wall for appropriate times in the reactor life for nil ductility temperature determinations. It should clearly state the assumptions used in the calculations, including power level, use factor, type of fuel cycle, and vessel design life. Applicants should also discuss the computer codes used in the analysis database for fast neutron cross-sections, geometric modeling of the reactor, core barrel, and pressure vessel, as well as the calculation uncertainties.

Modified Section/Title: C.I.4.3.3; Analytical Methods

Original Section/Title: C.I.4.3.3; Analytical Methods

This section should describe in detail the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, and burnup effects. This detailed description should include the computer codes used, including the code name and type, how it is used, its validity (based on critical experiments or confirmed predictions of operating plants), and methods of obtaining nuclear parameters (such as neutron cross-sections). In addition, the detailed descriptions of analytical methods should include estimates of the accuracy of each method.

Modified Section/Title: C.I.4.3.4; Changes

Original Section/Title: C.I.4.3.4; Changes

This section should list any changes in reactor core design features, calculational methods, data, or information relevant to determining important nuclear design parameters that depart from prior reactor design practices and identify the parameters affected by each change. Details regarding the nature and effects of these changes should be treated in appropriate subsections.

Modified Section/Title: C.I.4.4; Thermal-Hydraulic Design

Original Section/Title: C.I.4.4; Thermal-Hydraulic Design

Modified Section/Title: C.I.4.4.1; Design Bases

Original Section/Title: C.I.4.4.1; Design Bases

This section should provide the design bases for the thermal-hydraulic design of the reactor core and reactor internals. It should include such items as maximum fuel temperatures, fuel and graphite time at temperature spatial distributions, and fuel compact to graphite fuel element gap characteristics as a function of burnup and/or fast neutron fluence, flow velocities and flow distribution control, hydraulic stability, transient limits, fuel particle coating integrity criteria, and fuel element integrity criteria.

Modified Section/Title: C.I.4.4.2; Description of Thermal-Hydraulic Design of the Reactor Core and Reactor Internals

Original Section/Title: C.I.4.4.2; Description of Thermal-Hydraulic Design of the Reactor Core

This section should describe the thermal-hydraulic characteristics of the reactor core and internals design and include the information indicated in the following subsections.

Modified Section/Title: C.I.4.4.2.1; Summary Comparison

Original Section/Title: C.I.4.4.2.1; Summary Comparison

Applicants should provide a summary comparison of the reactor core and internals thermal-hydraulic design parameters with previously approved reactors of similar design. This should include, for example, core geometry, helium coolant temperatures, fuel and graphite temperature distributions, coolant velocities, surface heat fluxes, core pressure drop, and bypass flow characteristics.

Modified Section/Title: C.I.4.4.2.2; N/A
Original Section/Title: C.I.4.4.2.2; Critical Heat Flux Ratios

Modified Section/Title: C.I.4.4.2.3; N/A
Original Section/Title: C.I.4.4.2.3; Linear Heat Generation Rate

Modified Section/Title: C.I.4.4.2.4; N/A
Original Section/Title: C.I.4.4.2.4; Void Fraction Distribution

Modified Section/Title: C.I.4.4.2.5; Core Coolant Flow Distribution
Original Section/Title: C.I.4.4.2.5; Core Coolant Flow Distribution
Applicants should describe and discuss the coolant flow distribution. The distribution of helium flow among the coolant channels and various bypass flow paths should be described, and the manner in which the effects of bypass flow on fuel temperatures are taken into account should be provided. Core pressure drop characteristics, including total pressure drop and axial variations in pressure drop for different flow paths and the resulting cross flow between columns, should be described. The effects of coolant channel flow blockage on fuel temperature should be discussed.

The core hydraulics evaluation should include a discussion of the results of flow model tests (with respect to pressure drop for the various flowpaths through the reactor and flow distributions at the core inlet).

Modified Section/Title: C.I.4.4.2.6; Core Pressure Drops and Hydraulic Loads
Original Section/Title: C.I.4.4.2.6; Core Pressure Drops and Hydraulic Loads
Applicants should identify core pressure drops and hydraulic loads during normal and accident conditions that Chapter 15 of the FSAR does not address.

Modified Section/Title: C.I.4.4.2.7; Correlations and Physical Data
Original Section/Title: C.I.4.4.2.7; Correlations and Physical Data
This section should discuss the correlations and physical data employed in determining important characteristics such as heat transfer coefficients and pressure drop.

Modified Section/Title: C.I.4.4.2.8; Thermal Effects of Operational Transients
Original Section/Title: C.I.4.4.2.8; Thermal Effects of Operational Transients
This section should evaluate the capability of the core to withstand thermal effects resulting from anticipated operational transients.

Modified Section/Title: C.I.4.4.2.9; Uncertainties in Estimates
Original Section/Title: C.I.4.4.2.9; Uncertainties in Estimates
Applicants should discuss the uncertainties associated with estimating the peak or limiting conditions for thermal-hydraulic analysis (e.g., fuel temperature and time at temperature, graphite temperature, and pressure drops).

Modified Section/Title: C.I.4.4.2.10; Flux Tilt Considerations
Original Section/Title: C.I.4.4.2.10; Flux Tilt Considerations
This section should describe the approaches used to avoid flux tilts that could result in unacceptable power distributions in the core.

Modified Section/Title: C.I.4.4.3; Description of the Thermal and Hydraulic Design of the Reactor and Heat Transport System

Original Section/Title: C.I.4.4.3; Description of the Thermal and Hydraulic Design of the Reactor Coolant System

This section should describe the thermal-hydraulic design of the Reactor and Heat Transport Systems. The description should include the information indicated in the following subsections. If Chapter 5 of the FSAR provides the applicable information for the Heat Transport System, this section may incorporate it by reference.

Modified Section/Title: C.I.4.4.3.1; Plant Configuration Data

Original Section/Title: C.I.4.4.3.1; Plant Configuration Data

This section should provide the following summary information on plant configuration and operation with details provided in Chapter 5.

- Description of the Heat Transport System (HTS), including isometric drawings that show the configuration and approximate dimensions of the HTS components,
- Listing of all valves and fittings (e.g., elbows, tees) in the HTS
- Total coolant flow through each flowpath (e.g., total loop flow, core flow, bypass flow)
- Elevation of the bottom of each volume with respect to some reference elevation
- Minimum flow areas of each component
- Steady-state pressure and temperature distribution throughout the system.

Modified Section/Title: C.I.4.4.3.2; Operating Restrictions on Helium Circulators

Original Section/Title: C.I.4.4.3.2; Operating Restrictions on Pumps

This section should state the operating restrictions that will be imposed on the helium circulators to meet speed, load, vibration, and temperature limitations, including motor cooling requirements.

Modified Section/Title: C.I.4.4.3.3; N/A

Original Section/Title: C.I.4.4.3.3; Power-Flow Operating Map (BWR)

Modified Section/Title: C.I.4.4.3.4; N/A

Original Section/Title: C.I.4.4.3.4; Temperature-Power Operating Map (PWR)

Modified Section/Title: C.I.4.4.3.5; Load-Following Characteristics

Original Section/Title: C.I.4.4.3.5; Load-Following Characteristics

Applicants should describe the load-following characteristics of the Reactor and Heat Transport Systems as well as the techniques employed to provide this capability, if any.

Modified Section/Title: C.I.4.4.3.6; Thermal and Hydraulic Characteristics Summary Table

Original Section/Title: C.I.4.4.3.6; Thermal and Hydraulic Characteristics Summary Table

Applicants should provide a table summarizing the thermal-hydraulic characteristics of the Reactor and Heat Transport Systems.

Modified Section/Title: C.I.4.4.4; Evaluation of the Thermal-Hydraulic Design of the Reactor and Heat Transport Systems

Original Section/Title: C.I.4.4.4; Evaluation

This section should provide an evaluation of the thermal-hydraulic design of the Reactor and Heat Transport Systems. This evaluation should include the information indicated in the following subsections.

Modified Section/Title: C.I.4.4.4.1; N/A
Original Section/Title: C.I.4.4.4.1; Critical Heat Flux

Modified Section/Title: C.I.4.4.4.2; N/A
Original Section/Title: C.I.4.4.4.2; Core Hydraulics
(NOTE: Only item 1 of original text applies to HTGR; that was moved to Section 4.4.2.5)

Modified Section/Title: C.I.4.4.4.3; Influence of Power Distribution
Original Section/Title: C.I.4.4.4.3; Influence of Power Distribution
This section should discuss the influence of axial and radial power distributions on the thermal‑ and hydraulic design. It should include an analysis to determine which fuel columns control the thermal limits of the reactor.

Modified Section/Title: C.I.4.4.4.4; Core Thermal Response
Original Section/Title: C.I.4.4.4.4; Core Thermal Response
Applicants should evaluate the thermal response of the core at rated power, at design overpower, and during expected transient conditions.

Modified Section/Title: C.I.4.4.4.5; Analytical Methods
Original Section/Title: C.I.4.4.4.5; Analytical Methods
This section should describe the analytical methods and data used to determine the Reactor and Heat Transport System flow rate. This should include classical fluid mechanics relationships and empirical correlations. In addition, this description should provide estimates of the uncertainties in the calculations as well as the resultant uncertainty in Reactor and Heat Transport System flow rate.

This section should provide a comprehensive discussion of the analytical techniques used in evaluating the core thermal-hydraulics, including estimates of uncertainties. This discussion should include such items as hydraulic instability and application of power peaking factors. Applicants may describe computer codes by referencing documents available to the NRC.

Modified Section/Title: C.I.4.4.5; Testing and Verification
Original Section/Title: C.I.4.4.5; Testing and Verification
This section should discuss the testing and verification techniques used to ensure that the planned thermal-hydraulic design characteristics of the Reactor System have been provided and will remain within required limits throughout the core lifetime. This discussion should address the applicable portions of RG 1.68, (Rev. 3), with appropriate adjustments to accommodate the operating characteristics of the modular HTGR. References to the appropriate portions of Chapter 14 of the FSAR are acceptable.

Modified Section/Title: C.I.4.4.6; Instrumentation Requirements
Original Section/Title: C.I.4.4.6; Instrumentation Requirements
This section should discuss the functional requirements for instrumentation to be employed in monitoring and measuring those thermal-hydraulic parameters that are important to safety. For example, this discussion should include the requirements for instrumentation to confirm predicted power distributions or core thermal-hydraulic behavior. Chapter 7 of the FSAR should provide details of the instrumentation design and logic.

Modified Section/Title: C.I.4.5; Reactor Materials
Original Section/Title: C.I.4.5; Reactor Materials

Modified Section/Title: C.I.4.5.1; Control Rod Drive Structural Materials

Original Section/Title: C.I.4.5.1; Control Rod Drive System Structural Materials

For the purpose of this section, the control rod drive includes the Control Rod Drive Mechanism (CRDM) and extends to the coupling interface with the reactivity control (poison) elements in the reactor vessel. It does not include the electrical systems necessary to actuate the CRDMs. This section should provide the information described in the following subsections.

Modified Section/Title: C.I.4.5.1.1; Materials Specifications

Original Section/Title: C.I.4.5.1.1; Materials Specifications

This section should provide a list of the materials, including weld materials, and their specifications for each CRDM component. Applicants should furnish information regarding the mechanical properties of any material not included in either Appendix I to Section III of the ASME Code or in RG 1.84, (Rev. 35), to the extent that its code cases have been accepted by the NRC for the HTGR, and justify the use of such materials.

Applicants should state whether the CRDM design uses any materials that have yield strength greater than 90,000 psi, such as cold-worked austenitic stainless steels, precipitation hardenable stainless steels, or hardenable martensitic stainless steels. If such materials are used, applicants should identify their usage and provide evidence that stress-corrosion cracking will not occur during service life in components fabricated from the materials.

Modified Section/Title: C.I.4.5.1.2; Austenitic Stainless Steel Components

Original Section/Title: C.I.4.5.1.2; Austenitic Stainless Steel Components

This section should describe the processes, inspections, and tests used to ensure that austenitic stainless steel CRDM components, to the extent that any are used in the HTGR control rod drives, are free from increased susceptibility to intergranular stress corrosion cracking (ISCC) caused by sensitization. If special processing or fabrication methods subject the materials to temperatures between 800–1500 °F (427–816 °C) or involve slow cooling from temperatures over 1500 °F (816 °C), applicants should describe the processing or fabrication methods and provide justification to show that such treatment will not cause susceptibility to ISCCG.

(NOTE: that these temperature values were developed from LWR experience; thus, the COL applicant should provide a basis if alternate values are used for the HTGR.)

Applicants should indicate the degree of conformance to the recommendations of RG 1.44, “Control of the Use of Sensitized Stainless Steel,” (Rev. 1) as well as Regulatory Position C.5 of RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” (Rev. 1) as it relates to controls for abrasive steel surfaces. Applicants should justify any deviations from these recommendations.

Modified Section/Title: C.I.4.5.1.3; Other Materials

Original Section/Title: C.I.4.5.1.3; Other Materials

This section should describe the tempering and aging temperatures for martensitic precipitation-hardening stainless steels to prevent their deterioration by stress corrosion during plant operation. It should also describe the processing and treatment of other special purpose materials, such as cobalt-base alloys (Stellites), nickel-based alloys (Inconel), titanium, Colmonoy-type surfacing materials, and GRAPHITAR-type mechanical carbon materials. Identify all metallic and non-metallic materials used in the CRDM that are not included in Section III, Appendix I, Division 5 of the ASME Code, Section II, “Materials,” Parts A, B, C, and D; and Section III, “Rules for Construction of Nuclear Plant Components,” Division 1, including Appendix I.

Modified Section/Title: C.I.4.5.1.4; Cleaning and Cleanliness Control
Original Section/Title: C.I.4.5.1.4; Cleaning and Cleanliness Control
This section should provide details regarding the steps that will be taken to protect austenitic stainless steel materials and parts of these systems during fabrication, shipping, and onsite storage to ensure that all cleaning solutions, processing compounds, degreasing agents, and detrimental contaminants are completely removed and all parts are dried and properly protected following any flushing treatment with water. It should indicate the degree of conformance to the recommendations of RG 1.37 (Rev. 1) and justify any deviations from these recommendations.

Modified Section/Title: C.I.4.5.2; Reactor Internals and Core Support Materials
Original Section/Title: C.I.4.5.2; Reactor Internals and Core Support Materials
This section should discuss the materials used for reactor internals and core support materials and include the information described in the following subsections.

Modified Section/Title: C.I.4.5.2.1; Materials Specifications
Original Section/Title: C.I.4.5.2.1; Materials Specifications
This section should list the materials, including weld materials, and their specifications for components of the reactor internals and core support structures. It should include materials treated to enhance corrosion resistance, strength, and hardness. Applicants should furnish information regarding the mechanical properties of any material not included in Part D of Section II of the ASME Code and justify the use of such materials.

Modified Section/Title: C.I.4.5.2.2; Controls on Welding
Original Section/Title: C.I.4.5.2.2; Controls on Welding
This section should indicate the methods and controls that will be used when welding reactor internals components and core support structures and provide assurance that such welds will meet the acceptance criteria of Article HG-5000 in Section III, Division 5 of the ASME Code.

Modified Section/Title: C.I.4.5.2.3; Nondestructive Examination
Original Section/Title: C.I.4.5.2.3; Nondestructive Examination
This section should indicate that the NDE procedures used to examine tubular products, if any, conform to the requirements of the ASME Code. Applicants should justify any deviations from these requirements.

Modified Section/Title: C.I.4.5.2.4; Fabrication and Processing of Austenitic Stainless Steel Components
Original Section/Title: C.I.4.5.2.4; Fabrication and Processing of Austenitic Stainless Steel Components
This section should indicate the degree of conformance to the recommendations of RG 1.44 (Rev. 1), and RG 1.37 (Rev.1), to the extent that the materials addresses in these regulatory guides may be used in the HTGR. If alternative measures are used, applicants should show that they will provide the same assurance of component integrity as would be achieved by following the recommendations of the listed regulatory guides. Applicants should indicate the maximum yield strength of all cold-worked stainless steels used in the reactor internals.

Modified Section/Title: C.I.4.5.2.5; Other Materials
Original Section/Title: C.I.4.5.2.5; Other Materials
Applicants should submit information on the mechanical properties, corrosion resistance, and fabrication of any materials other than austenitic stainless steels. In particular, applicants should discuss the tempering temperature of hardenable martensitic stainless steels and the aging temperature and aging time of precipitation-hardening stainless steels. This section should also discuss the processing and treatment of other special purpose materials, such as cobalt-base alloys (Stellites), nickel-based alloys (Inconel),

titanium, and Colmonoy-type surfacing materials. This section should also discuss the use of insulating material, such as silica brick, in reactor internals. This section should also discuss the use of graphite in the reactor internals, including information on graphite properties, including behavior in the presence of oxidizing impurities in the helium coolant at concentrations expected during normal operation and under accident conditions, and component fabrication.

Modified Section/Title: C.I.4.6; Functional Design of Reactivity Control Systems
Original Section/Title: C.I.4.6; Functional Design of Reactivity Control Systems
This section should provide information to establish that the control rod drives and the reserve shutdown mechanisms, which include the essential ancillary equipment, are designed to provide the required functional performance and are properly isolated from other equipment. It should also provide information to establish the bases for assessing the combined functional performance of all the reactivity control systems to mitigate the consequences of anticipated transients and postulated accidents.

Modified Section/Title: C.I.4.6.1; Information for Control Rod Drives and Reserve Shutdown Mechanisms
Original Section/Title: C.I.4.6.1; Information for CRDS
Information submitted should include drawings of the control rod drives and reserve shutdown mechanisms, process flow diagrams, piping and instrumentation diagrams, component descriptions and characteristics, and a description of the functions of all related ancillary equipment. This should also include the control rod drive cooling system for plants that have this system. Applicants may provide this information in conjunction with the information requested for Section 3.9.4 of the FSAR.

Modified Section/Title: C.I.4.6.2; Evaluations of the Control Rods Drives and Reserve Shutdown Mechanisms
Original Section/Title: C.I.4.6.2; Evaluations of the CRDS
Applicants should provide failure mode and effects analyses of the control rod drives and reserve shutdown mechanisms in tabular form, with supporting discussion to delineate the logic employed. The failure analysis should demonstrate that the control rod drives and reserve shutdown mechanisms, which for purposes of these evaluations include all essential ancillary equipment, can perform the intended safety functions with sufficient reliability for design basis events.

These evaluations and assessments should establish that all essential elements of the control rod drives and reserve shutdown mechanisms are identified and provisions are made for isolation from nonessential elements. In addition, this discussion should establish that all essential equipment is amply protected from common-mode failures (such as failure of high pressure lines, including their dynamic and environmental effects and postulated generated missiles).

Modified Section/Title: C.I.4.6.3; Testing and Verification of the Control Rod Drives and Reserve Shutdown Mechanisms
Original Section/Title: C.I.4.6.3; Testing and Verification of the CRDS
This section should describe the functional testing program. This should include control rod and reserve shutdown material insertion and withdrawal tests, thermal and fluid dynamic tests simulating postulated operating and accident conditions, and test verification of the control rod drives and reserve shutdown mechanisms with imposed single failures, as appropriate.

Applicants should provide preoperational and initial startup test programs. Program descriptions should include the test objectives, methods, and acceptance criteria. If Chapter 14 of the FSAR provides the applicable information, applicants may incorporate it in this section by reference.

Modified Section/Title: C.I.4.6.4; Information for Combined Performance of Reactivity Systems

Original Section/Title: C.I.4.6.4; Information for Combined Performance of Reactivity Systems

Other sections of the FSAR (e.g., Sections TBD) provide piping and instrumentation diagrams, layout drawings, process diagrams, failure analyses, descriptive material, and performance evaluations related to specific evaluations of the control rods and the reserve shutdown system. This section should include sufficient plan and elevation layout drawings to provide bases for establishing that these systems are not vulnerable to common-mode failures.

Chapter 15 of the FSAR provides evaluations pertaining to the plant's response to postulated process disturbances and equipment malfunctions or failures. This section should list all postulated accidents evaluated in Chapter 15, if any, that take credit for both reactivity control systems to prevent or mitigate each accident.

Modified Section/Title: C.I.4.6.5; Evaluations of Combined Performance

Original Section/Title: C.I.4.6.5; Evaluations of Combined Performance

This section should evaluate the combined functional performance for accidents where both reactivity systems are used, if any. The neutronic, instrumentation, controls, time sequencing, and other process-parameter-related features primarily appear in Chapters 4, 7, and 15 of the FSAR. This section should include failure analyses to demonstrate that the reactivity control systems perform with sufficient reliability during design basis events. These failure analyses should consider failures originating within each reactivity control system as well as those originating from plant equipment other than reactivity systems and should be provided in tabular form with supporting discussion and logic.

Modified Section/Title: C.I.4.7; Reactor System Structural Performance

Original Section/Title: NEW; NEW

This section should provide information on the structural performance of the reactor system components.

Modified Section/Title: C.I.4.7.1; Graphite Stress Phenomena

Original Section/Title: NEW; NEW

This section should provide a general description of the phenomena that affect stress in the graphite components of the reactor system. The role of spatial variations in fast neutron fluence and the temperature field on graphite stress should be discussed for both operating and shutdown conditions. The role of irradiation-induced creep in stress relief should be described. The effects of mechanical loads on graphite components (gravity, helium flow, and seismic), including effects of fatigue, should be discussed. The effects of oxidation of graphite on component strength and other physical properties should be presented. A general discussion of graphite failure criteria should provided, with particular attention to the role of limited localized cracking as a stress relief mechanism relative to large scale component failure. Relevant industry codes and standards for mechanical performance of graphite components in nuclear systems should be cited.

Modified Section/Title: C.I.4.7.2; Graphite Component Structural Performance

Original Section/Title: NEW; NEW

This section should provide the results of structural performance analyses for the reactor system graphite components. The results should be presented to demonstrate compliance with the relevant industry codes and standards cited in Section 4.7.1.

A listing of the analysis codes and models used to evaluate graphite component structural performance should be provided, including a general description of the input parameters used in the codes and the sources of the input data. Information should be provided on the validation of the codes used in the analyses.

Maximum stress/strength ratios expected in fuel elements and other graphite reactor system components under normal operation, AOOs, and postulated accidents should be presented and compared with maximum allowable values. Assumptions regarding the level of impurities in the helium coolant and the resulting amount of graphite oxidation under normal operation should be described and justified. The effects of fatigue on graphite components and a comparison with fatigue limits should be presented.

Modified Section/Title: C.I.4.7.3; Graphite Abrasion and Dust Generation
Original Section/Title: NEW; NEW
This information should provide information on the amount of dust generation expected to occur in the reactor system as a result of graphite abrasion and other sources. The results should be taken into account in the radionuclide retention analyses presented in Section 4.8.

Modified Section/Title: C.I.4.7.4; Metallic Component Structural Performance
Original Section/Title: NEW; NEW
This section should provide the results of structural performance analyses for the reactor system metallic components. Relevant industry codes and standards for mechanical performance of metallic components in nuclear systems should be cited. The results should be presented to demonstrate compliance with these relevant industry codes and standards.

A listing of the analysis codes and models used to evaluate metallic component structural performance should be provided, including a general description of the input parameters used in the codes and the sources of the input data. Information should be provided on the validation of the codes used in the analyses.

Maximum stress/strength ratios expected in metallic reactor system components under normal operation, AOOs, and postulated accidents should be presented and compared with maximum allowable values. The effects of fatigue on metallic components and a comparison with fatigue limits should be presented.

Modified Section/Title: C.I.4.8; Fuel Performance and Radionuclide Retention
Original Section/Title: NEW; NEW
This section should provide information on fuel performance and radionuclide retention in the reactor system. Information should be provided regarding the requirements for radionuclide retention in the fuel, analyses of fuel performance during normal operation, and analyses of fission product transport and release in the reactor system.

Modified Section/Title: C.I.4.8.1; Radionuclide Control Design Requirements
Original Section/Title: NEW; NEW
This section should provide information on radionuclide control design requirements to ensure compliance with regulatory requirements for worker exposure and off site doses from normal operations, AOOs, and postulated accidents. In addition information should be provided regarding radionuclide retention requirements for meeting design goals for EPA Protective Action Guidelines.

At a minimum, the following items should be addressed:

1. Reactor system functional requirements related to radionuclide control

2. 50% and 95% confidence limits on circulating and plateout radionuclide inventories in the primary circuit during normal operation for key radionuclides, with a discussion of how each of these limits are used in assessing compliance with regulatory requirements and design goals

3. Basis for the limits on circulating and plateout inventories

4. Limits on radionuclide release from the plant during postulated accidents for key radionuclides

5. A brief summary of the approach to calculation of mechanistic source terms, with reference as needed to topical reports on this subject.

Modified Section/Title: C.I.4.8.2; Fuel Performance Analysis - Normal Operation
Original Section/Title: NEW; NEW
This section should provide information on calculated fuel performance during normal operation. At a minimum, information to be provided should include the following:

1. Models and assumptions used to determine fuel operating conditions

2. Fuel particle performance (failure) models and their bases and validation status, with reference as needed to topical reports or white papers on this subject

3. A listing of codes used in fuel performance analyses

4. Results of fuel performance analyses, including fuel temperature and burnup distributions and coated particle fuel failure rates.

Modified Section/Title: C.I.4.8.3; Fission Product Transport and Release Analysis - Normal Operation
Original Section/Title: NEW; NEW
This section should provide information on the calculation of fission product generation, transport, and release from the reactor system using the fuel performance analyses discussed in Section 4.8.2 as input. At a minimum, information to be provided should include the following:

1. Models and assumptions used to determine fission product transport and release and their bases and validation status, with reference as needed to topical reports or white papers on this subject

2. A listing of codes used in fission product transport and release

3. Results of fission product analyses and comparison with the radionuclide design control requirements of Section 4.8.1.

Appendix E

Chapter 5. Helium Pressure Boundary and Connecting Systems

Appendix E
Chapter 5. Helium Pressure Boundary and Connecting Systems

Modified Section/Title: C.I.5; Helium Pressure Boundary and Connecting Systems
Original Section/Title: C.I.5; Reactor Coolant and Connecting Systems
Chapter 5 of the FSAR should provide information regarding the Helium Pressure Boundary (HPB) and systems to which it connects. Special consideration should be given to the HPB and pressure-containing appendages out to and including isolation valving. This section should provide an overview describing the interfaces between the various HPB systems (including the vessel system) and connecting systems such as heat transport system, shutdown cooling system, and helium purification system.

This section should include evaluations, together with the necessary supporting material, to show that the HPB is adequate to accomplish its intended objective and to maintain core geometry, ensure the reactor vessel maintains its emissivity characteristics, and transfer of core heat from the reactor vessel to the reactor cavity cooling system, as necessary to mitigate postulated events, including both normal and accident conditions. The information should permit an independent determination of the adequacy of the evaluations; that is, assurance that the evaluations included are correct and complete, and all necessary evaluations have been performed. Applicants should reference evaluations included in other chapters that have a bearing on the HPB.

Modified Section/Title: C.I.5.1; Summary Description
Original Section/Title: C.I.5.1; Summary Description
This section of the FSAR should provide a summary description of the HPB (including the vessel system) and its various connecting systems and components such as the heat transport system, shutdown cooling system, and helium cleanup system. This description should indicate the independent and interrelated performance and safety functions of each system and component and should include a tabulation of important design and performance characteristics.

Modified Section/Title: C.I.5.1.1; Schematic Flow Diagram
Original Section/Title: C.I.5.1.1; Schematic Flow Diagram
This section should provide a schematic flow diagram of the HPB denoting all major components, principal pressures, temperatures, flow rates, and total helium mass under normal steady-state full-power operating conditions.

Modified Section/Title: C.I.5.1.2; Piping and Instrumentation Diagram
Original Section/Title: C.I.5.1.2; Piping and Instrumentation Diagram
Applicants should provide a simplified piping and instrumentation diagram of the HPB and connected systems delineating the following three items, as applicable:

- Extent of the systems located within the reactor building
- Points of separation between the HPB (heat transport) and the secondary (heat utilization or removal) system
- Isolation capability of the HPB as provided by the use of isolation valves between the radioactive and nonradioactive sections of the system, isolation valves between the HPB and connected systems, and passive barriers between the HPB and other systems.

Modified Section/Title: C.I.5.1.3; Elevation Drawing
Original Section/Title: C.I.5.1.3; Elevation Drawing
Applicants should provide an elevation drawing showing principal dimensions of the HPB in relation to the supporting or surrounding concrete structures from which a measure of the protection afforded by the arrangement and the safety considerations incorporated in the layout can be gained.

Modified Section/Title: C.I.5.2; Integrity of the Helium Pressure Boundary
Original Section/Title: C.I.5.2; Integrity of the Reactor Coolant Pressure Boundary
This section of the FSAR should discuss the intended level of integrity and the measures employed that assure the intended level of integrity appropriate to each HPB component is maintained throughout the design lifetime of that component.

Modified Section/Title: C.I.5.2.1; Compliance with Codes and Code Cases
Original Section/Title: C.I.5.2.1; Compliance with Codes and Code Cases

Modified Section/Title: C.I.5.2.1.1; Compliance with 10 CFR 50.55a
Original Section/Title: C.I.5.2.1.1; Compliance with 10 CFR 50.55a
This section should provide a table showing how compliance is maintained with applicable sections of the NRC regulations in 10 CFR 50.55a. This table should identify vessel system components, piping, circulators, compressors, and valves. The applicable component code and code edition and addenda of each Class A component that is part of the HPB and that is covered by Section III of the ASME Code, may be identified by reference to the table of SSCs in Section 3.2 of the FSAR or included in this section of the FSAR.

Certain paragraphs of 10 CFR 50.55a may not apply to HTGRs as currently written. The application of any and all 10 CFR 50.55a requirements to an HTGR must be reviewed and endorsed by NRC. Paragraph (a)(3) of 10CFR 50.55a allows applicants to propose alternative approaches to address relevant requirements. If conformance to the regulations of 10 CFR 50.55a would result in hardships or unusual difficulties without a compensating increase in the level of safety and quality, applicants should provide a complete description of the circumstances resulting in such cases and the basis for proposed alternative requirements. The description should cover how the proposed alternative requirements will provide an equivalent and acceptable level of safety and quality.

Modified Section/Title: C.I.5.2.1.2; Compliance with Applicable Code Cases
Original Section/Title: C.I.5.2.1.2; Compliance with Applicable Code Cases
Applicants should provide a list of ASME Code Cases that will be applied to components comprising the HPB. The list should identify each component to which a code case has been applied by code case number, revision, and title. Applicants are advised to ensure that the applicable revision of the code case is identified for each component application. ASME Code cases related specifically to High Temperature Gas-cooled Reactors may be necessary and should consider Section III, Division 5, "Rules for Construction of Nuclear Facility Components." RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III" lists those ASME Code cases that are acceptable to the NRC staff for design, fabrication, and materials used in an LWR. Applicants should indicate the extent of conformance of their design with these code cases and justify why they are appropriate for use. If applicants use code cases other than those listed (and endorsed by NRC), they should show that their use will result in as acceptable a level of quality and safety for the component as would be achieved by following code cases endorsed by the NRC staff. All code cases are subject to endorsement by NRC prior to their use.

Modified Section/Title: C.I.5.2.2; Overpressure Protection

Original Section/Title: C.I.5.2.2; Overpressure Protection

This section should provide information, as set forth in the following subsections, to accommodate an evaluation of the systems that protect various RCS components from overpressure and thereby preserve the HPB from breach or degradation. The sources of overpressure, including those attributable to fluid ingress from high pressure energy transfer components like steam generators or intermediate heat exchangers, should be described. Such systems include pressure-relieving devices (safety and relief valves) and steam generator isolation and dump systems that are relied upon to limit available moisture ingress inventory. Discuss the affect overpressure systems may have on the function of plant safety systems.

Modified Section/Title: C.I.5.2.2.1; Design Bases

Original Section/Title: C.I.5.2.2.1; Design Bases

This section should provide the design bases on which the functional design of overpressure protection was established. It should address overpressure protection for RCS components that comprise the HPB during modular reactor power operation. Applicants should describe how the guidance of GDC 15, Appendix A to 10 CFR Part 50 [PDC 15], will be met as it relates to not exceeding design conditions during any condition of normal operation or AOO. Applicants should also describe how the underlying intent of GDC 31 [PDC 31] will be met as it relates to designing RCS components with sufficient margin to ensure that they behave in a nonbrittle manner and minimize the probability of rapidly propagating fracture.

Modified Section/Title: C.I.5.2.2.2; Design Evaluation

Original Section/Title: C.I.5.2.2.2; Design Evaluation

This section should provide an evaluation of the functional design of the overpressure protection system. This evaluation should include an analysis of the system's capability to perform its function, describe the analytical model used in the analysis, and discuss the bases for its validity. Applicants should also discuss and justify the assumptions used in the analysis, including the plant initial conditions and system parameters. They should list the systems and equipment assumed to operate and describe their performance characteristics. This section should provide studies that show the sensitivity of the system's performance to variations in these conditions, parameters, and characteristics.

Applicants should provide analysis that demonstrates how overpressure protection is achieved for components that comprise the HPB. This section should identify all overpressure events and, as a subset, identify the events that can be prevented by means such as preventive interlocks or locking-out power. Applicants should describe how the overpressure protection system is enabled, the alarms and indications associated with the system, and the power source for the system. They should discuss whether any credit is taken for active components to mitigate an overpressure event and the additional analysis performed that considers inadvertent system initiation or actuation. If this system uses pressure relief from a low-pressure system, this section should discuss how the low-pressure interlocks will not interfere with the operation of this system.

Modified Section/Title: C.I.5.2.2.3; Piping and Instrumentation Diagrams

Original Section/Title: C.I.5.2.2.3; Piping and Instrumentation Diagrams

This section should provide piping and instrumentation diagrams for the modular HTGR overpressure protection system that affect HPB integrity and show the number, type, and location of all components, including valves, piping, instrumentation, and controls. Applicants should identify the connections and interfaces with other systems.

Modified Section/Title: C.I.5.2.2.4; Equipment and Component Description
Original Section/Title: C.I.5.2.2.4; Equipment and Component Description
This section should describe the equipment and components of the overpressure protection system that affect HPB integrity. Include schematic drawings of safety and relief valves and discuss how the valves operate. It should identify significant design parameters for each component that may impact plant safety and include the design, throat area, capacity, set points of the valves, and the diameter, length, and routing of piping. Applicants should list the design parameters (e.g., pressure and temperature) for these components and specify the number and type of operating cycles as well as the environmental conditions (e.g., temperature and pressure) for which the component is designed.

Modified Section/Title: C.I.5.2.2.5; Mounting of Pressure-Relief Devices
Original Section/Title: C.I.5.2.2.5; Mounting of Pressure-Relief Devices
This section should describe the design and installation details concerning the mounting of pressure-relief devices that are part of or may otherwise affect HPB integrity. Include devices on the secondary side of the heat transport system (e.g., intermediate heat exchanger or steam generator) that may directly or indirectly affect the overall ability of HPB components to perform their intended containment function. For devices that may affect HPB integrity, applicants should specify the design bases for assumed loads (i.e., thrust, bending, and torsion) imposed on the valves, nozzles, and connected piping in the event that all valves discharge. Describe how these loads can be accommodated and include a listing of loads and resulting stresses. Applicants may cross-reference material contained in Section 3.9.3.3 of the FSAR.

Modified Section/Title: C.I.5.2.2.6; Applicable Codes and Classification
Original Section/Title: C.I.5.2.2.6; Applicable Codes and Classification
This section should identify the applicable industry codes and classifications applied to the components and systems that comprise the HPB.

Modified Section/Title: C.I.5.2.2.7; Material Specification
Original Section/Title: C.I.5.2.2.7; Material Specification
Applicants should identify the material specifications for each component.

Modified Section/Title: C.I.5.2.2.8; Process Instrumentation
Original Section/Title: C.I.5.2.2.8; Process Instrumentation
Applicants should identify all process instrumentation.

Modified Section/Title: C.I.5.2.2.9; System Reliability
Original Section/Title: C.I.5.2.2.9; System Reliability
Applicants should discuss system reliability and the consequences of equipment/component failures.

Modified Section/Title: C.I.5.2.2.10; Testing and Inspection
Original Section/Title: C.I.5.2.2.10; Testing and Inspection
This section should identify the tests and inspections to be performed (1) before operation and during startup that demonstrate functional performance and (2) inservice surveillance to ensure continued reliability. Applicants should describe specific testing suitable for all modes of normal operation.

Modified Section/Title: C.I.5.2.3; Helium Pressure Boundary Component Materials
Original Section/Title: C.I.5.2.3; Reactor Coolant Pressure Boundary Materials

Modified Section/Title: C.I.5.2.3.1; Material Specifications

Original Section/Title: C.I.5.2.3.1; Material Specifications

This section should provide a list of specifications for the principal ferritic materials, austenitic stainless steels, and nonferrous metals (including bolting and weld materials) to be used in fabricating and assembling each component (e.g., vessels, piping, compressors, and valves) that comprises the HPB (excluding vessels, which are described in Chapter 5.3). It should identify the grade or type and final metallurgical condition of the material placed in service. "Metallurgical condition" is a technical term used to describe the microstructure of the materials. Based on its phase diagram, the microstructure of a material can vary in accordance with the heat treatments applied to the materials. Different microstructures of a material will possess different mechanical properties. One example is the heat treatment of the austenitic stainless steel in a certain temperature range will create a sensitized microstructure, which is characterized by chromium depletion along the grain boundary. Austenitic stainless steel with sensitized microstructure is susceptible to inter-granular stress corrosion cracking (IGSCC) in LWRs. Materials engineers with metallurgy background should be able to provide the requested information.

If the as-procured, as-built grade, type and final metallurgical condition of the materials are unavailable at the time of the COL application, representative or bounding data/information may be submitted for review as part of the COL application. The COL applicant should submit the as-procured, as-built grade, type and final metallurgical condition of the materials to the staff at a pre-determined time agreed upon by the both parties. The applicant may need to work with the NRC staff during the review to arrive at an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

Modified Section/Title: C.I.5.2.3.2; Compatibility with Reactor Coolant Impurities

Original Section/Title: C.I.5.2.3.2; Compatibility with Reactor Coolant

Applicants should provide the following information relative to the compatibility of the materials of construction and insulation of HPB components with reactor coolant impurities:

[OPEN ITEM: Need To Insert Requirements For HPB Chemistry Control When Available.]

Regarding coolant chemistry, applicants should provide sufficient information about the location and performance of coolant chemistry monitoring and other details of the coolant chemistry program to assure the facility can maintain proper coolant chemistry balance at the level established by the applicant.

Regarding the compatibility of construction materials with reactor coolant impurities, applicants should provide a list of the materials of construction exposed to reactor coolant and a description of material compatibility with the impurities to which the materials may be exposed. Nonmetallic materials exposed to reactor coolant should include a description of the compatibility of these materials with the coolant and its impurities.

Regarding the compatibility of construction materials with materials like insulation and reactor coolant impurities, applicants should provide a list of the materials of construction of HPB components and a description of their compatibility with insulation and the environment. Applicants should provide sufficient information about the selection, procurement, testing, storage, and installation of any nonmetallic thermal insulation for austenitic stainless steel to indicate whether the concentrations of chloride, fluoride, sodium, and silicate in thermal insulation will be within the ranges recommended in RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel" as they are relevant and applicable to HPB components. They should provide information on the leachable contaminants in insulation on nonaustenitic piping.

Modified Section/Title: C.I.5.2.3.3; Fabrication and Processing of Ferritic Materials
Original Section/Title: C.I.5.2.3.3; Fabrication and Processing of Ferritic Materials
Applicants should provide the following types of information relative to fabrication and processing of ferritic materials used for components of the HPB:

Regarding fracture toughness of the ferritic materials, including bolting materials for components (e.g., vessels, piping, compressors, and valves that are not otherwise covered in section 5.3) that comprise the HPB, applicants should indicate how compliance with applicable test and acceptance guidelines is achieved. Appendix G to 10 CFR Part 50 and Division 1 to Section III of the ASME Code exemplify fracture toughness acceptance criteria for LWRs; similar requirements must be defined and addressed for HTGRs. ASME Section III, Division 5 provides testing and acceptance guidance for High Temperature Gas-Cooled Reactors; additional guidance may be required (see INL/EXT-09-17187, "NGNP High Temperature Materials White Paper" for further information on possible approaches). Applicants should submit fracture toughness data in tabular form and include information regarding the calibration of instruments and equipment (FSAR).

If the actual, as procured fracture toughness data is unavailable at the time of the COL application, representative or bounding data and information may be submitted for staff review as part of the COL application.

The COL applicant should submit the actual, as procured fracture toughness data to the staff at a pre-determined time agreed upon by the both parties. The applicant may need to work with the NRC staff during the review to arrive at an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as built plant is consistent with the design reviewed during the licensing process.

Applicants should provide the following information relative to the control of welding of ferritic materials used for components of the HPB:

Sufficient information regarding the avoidance of cold cracking during welding of low-alloy steel components to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Applicants should provide details on proposed minimum preheat temperature and maximum interpass temperature during procedure qualification and production welding. They should provide information on the moisture control for low-hydrogen, covered-arc-welding electrodes.

Sufficient information for electroslag welds in the low-alloy steel components to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of RG 1.34, "Control of Electroslag Weld Properties." Use the appropriate paragraphs of the Code, Section III, Division 5, Subsection HC, as they apply to the HTGR design. For Position C.3.a of RG 1.34 (Rev 1), use the appropriate article 2000 of the Code, Subsection III, Division 5, Subsection HB. They should provide details on the control of welding variables and the metallurgical tests required during procedure qualification and production welding.

In regard to welding and weld repair during fabrication and assembly of ferritic steel components, applicants should provide sufficient details on welder qualification for weld areas of limited accessibility, requalification, and monitoring of production welding for adherence to welding qualification requirements to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of RG 1.71, "Welder Qualification for Areas of Limited Accessibility."

Describe the controls to limit the occurrence of underclad cracking in low-alloy steel components clad with stainless steel. Provide sufficient information about the program for NDE of ferritic steel tubular products (pipe, tubing, flanges, and fittings) for components to indicate whether detection of unacceptable defects (regardless of defect shape, orientation, or location in the product) is consistent with the ASME Code.

If data, test results, or other information is unavailable at the time of the COL application, representative or bounding data and information may be submitted for staff review as part of the COL application. The COL applicant should submit the data, test results, or other information that was not available at the time of COL application to the staff at a pre determined time agreed upon by the both parties. The applicant may need to work with the NRC staff during the review to arrive at an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as built plant is consistent with the design reviewed during the licensing process.

Modified Section/Title: C.I.5.2.3.4; Fabrication and Processing of Austenitic Stainless Steels

Original Section/Title: C.I.5.2.3.4; Fabrication and Processing of Austenitic Stainless Steels

Applicants should provide the following types of information relative to fabrication and processing of austenitic stainless steels for components that comprise the HPB and have a potential impact to safety:

Applicants should provide information relative to avoidance of stress-corrosion cracking of austenitic stainless steels for components of the HPB during all stages of component manufacture and reactor construction.

Applicants should include sufficient details about the avoidance of sensitization during fabrication and assembly of austenitic stainless steel components of the HPB to indicate whether the degree of freedom from sensitization will be sufficient. Comparable recommendations for LWR power plants are provided in RG 1.44, “Control of the Use of Sensitized Stainless Steel.” Additional supporting information must be provided concerning modular HTGR high temperature applications.

Applicants should provide sufficient details about the process controls to minimize exposure to contaminants capable of causing stress-corrosion cracking of austenitic stainless steel components of the HPB to show whether process controls will provide, during all stages of component manufacture and reactor construction, the appropriate degree of surface cleanliness. Applicants should describe the controls for abrasive work on austenitic stainless steel surfaces. Applicants should identify any pickling used in processing austenitic stainless steel components and describe the restrictions placed on pickling of sensitized materials. Applicants should also identify the upper yield strength limit of the austenitic stainless steel materials used.

If the actual, as-procured yield strength of the austenitic stainless steel materials is unavailable at the time of the COL application, representative or bounding data and information may be submitted for review as part of the COL application. The COL applicant should submit the actual, as-procured yield strength of the austenitic stainless steel materials to the staff at a pre-determined time agreed upon by the both parties. The applicant may need to work with the NRC staff during the review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as built plant is consistent with the design reviewed during the licensing process.

Applicants should provide assurance that cold-worked austenitic stainless steels will not be susceptible to stress-corrosion cracking in HPB component applications. The RCPBs of LWR power plans have a maximum 0.2-percent offset yield strength of 620 megapascal (90,000 psi) to reduce the probability of stress-corrosion cracking; justification concerning the use of this or any other alternative standard must be provided when used in a modular HTGR. Applicants should identify augmented ISI to ensure the structural integrity of the components during service. In general, cold-worked austenitic stainless steels should not be used for HPB applications but they may be used when no proven alternative is available. If such materials are used, applicants should describe the service experience and laboratory testing in the simulated environment to which the components will be exposed. They should describe the controlled, measured, and documented fabrication process for cold-worked components.

Applicants should provide information relative to the control of welding of austenitic stainless steels for components of the HPB. Applicants should provide sufficient information on electroslag welds in austenitic stainless steel components of the HPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of RG 1.34, "Control of Electroslag Weld Properties." The information should include control of welding variables and metallurgical tests required during procedure qualification and production welding.

In regard to welding and weld repair during fabrication and assembly of austenitic stainless steel components of the HPB, applicants should provide sufficient details about welder qualification for areas of limited accessibility, requalification, and monitoring of production welding for adherence to welding qualification requirements to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of RG 1.71, "Welder Qualification for Areas of Limited Accessibility."

Applicants should provide sufficient information about the program for NDE of austenitic stainless steel tubular products (pipe, tubing, flanges, and fittings) for components of the HPB to indicate whether detection of unacceptable defects (regardless of defect shape, orientation, or location in the product) is consistent with NRC-endorsed ASME Code.

Modified Section/Title: C.I.5.2.3.5; Not Applicable

Original Section/Title: C.I.5.2.3.5; Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Base Alloys (PWRs only)

Modified Section/Title: C.I.5.2.3.6; Threaded Fasteners

Original Section/Title: C.I.5.2.3.6; Threaded Fasteners

This section should provide a summary description of the program for ensuring the integrity of bolting and threaded fasteners and their adequacy. Applicants should reference FSAR Section 3.13, as appropriate.

Modified Section/Title: C.I.5.2.4; Inservice Inspection and Testing of Class A HPB Components

Original Section/Title: C.I.5.2.4; Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

Modified Section/Title: C.I.5.2.4.1; Inservice Inspection and Testing Program

Original Section/Title: C.I.5.2.4.1; Inservice Inspection and Testing Program

This section should discuss the ISI and testing program for Class A components of the HPB (ASME Code, Section III) that must comply with the applicable requirements of 10 CFR 50.55a. The applicant should provide sufficient detail to show that the ISI/IST program meets the requirements of Section XI of the ASME Code when they are endorsed by the NRC. Because the ISI/IST is an operational program, applicants should describe the program and its implementation with sufficient scope and level of detail to enable the staff to make a reasonable assurance finding regarding acceptability. Therefore, applicants should provide descriptive information on the system boundary subject to inspection. In particular, they should discuss components (other than steam generator tubes and intermediate heat exchanger circuits) and associated supports to include all pressure vessels, piping, compressors, valves, and bolting covering the following areas:

1. Regarding accessibility, applicants should describe provisions for access to components and identify any remote access equipment needed to perform inspections

2. Regarding examination categories and methods, applicants should discuss the methods, techniques, and procedures used to meet NRC-endorsed ASME Code requirements

3. Regarding inspection intervals, applicants should discuss program scheduling in comparison with the ASME Code

4. Applicants should discuss provisions for evaluating examination results, including evaluation methods for detected flaws and repair procedures for components that reveal defects

5. Applicants should provide descriptive information on system pressure tests and correlated TS requirements

6. Applicants should identify components that are exempted from the ASME Code Section XI examination requirements

7. Applicants should discuss any requests for relief from ASME Code requirements that are impractical as a result of limitations of component design, geometry, or materials of construction

8. Applicants should identify all ASME Code cases that are invoked.

Because ISI/IST programs are operational programs, the programs and their implementation milestones should be fully described and reference any applicable standards. Fully described should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow for a reasonable assurance finding of acceptability.

Modified Section/Title: C.I.5.2.4.2; Preservice Inspection and Testing Program

Original Section/Title: C.I.5.2.4.2; Preservice Inspection and Testing Program

This section should describe the preservice examination program that meets the applicable guidelines of Section III of the ASME Code. Because the preservice inspection and preservice testing programs are operational programs, the programs and their implementation milestones should be fully described and reference any applicable standards. Fully described should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow for a reasonable assurance finding of acceptability.

Modified Section/Title: C.I.5.2.5; Helium Pressure Boundary Leakage Detection

Original Section/Title: C.I.5.2.5; Reactor Coolant Pressure Boundary Leakage Detection

Applicants should describe the HPB leakage detection system. The description should provide sufficient detail to provide a thorough understanding of the purpose, functionality, and safety implications of the HPB leakage detection system. Although written for water rather than helium, the LWR-specific recommendations of RG 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” may offer insight into factors relevant to the design and operation of a HPB leakage detection system; RG 1.29, “Seismic Design Classification," also offers supplemental insight. These RGs are cited as examples and describe methods that may be inappropriate to modular HTGRs unless first adapted.

The applicant should provide information that will permit an evaluation of system adequacy by giving a detailed description of the systems employed, their sensitivity and response time, and the reliance placed on their proper functioning. This section should also identify the limiting leakage conditions that will be included in the TS if they are established.

Applicants should describe the system used for alarm as an indirect indication of leakage and provide the design criteria. They should describe how signals from the various leakage detection systems are correlated to provide information to plant operators regarding leakage location and quantitative leakage flow rate.

Applicants should demonstrate adequate monitoring capability to ensure that the limits of intersystem leakage assumed in the accident analyses are not exceeded. For radioactivity monitoring leakage detection, they should describe the primary coolant radioactivity concentration assumption being used to analyze the sensitivity of the leak detection systems.

This section should describe the provisions to test and calibrate all leakage detection systems and provide and justify the frequency of testing and calibration. The applicant should describe the periodic testing of the collection system(s) that will ensure operability.

Modified Section/Title: C.I.5.3; Vessel System

Original Section/Title: C.I.5.3; Reactor Vessels

Modified Section/Title: C.I.5.3.1; Vessel System Materials

Original Section/Title: C.I.5.3.1; Reactor Vessel Materials

This section of the FSAR should contain pertinent data in sufficient detail to provide assurance that the materials (including weld materials), fabrication methods, and inspection techniques used for the vessel system (i.e., reactor vessel, cross vessel, steam generator and/or IHX vessel) and applicable attachments and appurtenances conform to all applicable regulations. The FSAR should also describe the specifications and criteria to be applied and should demonstrate that the requirements have been met.

Modified Section/Title: C.I.5.3.1.1; Material Specifications

Original Section/Title: C.I.5.3.1.1; Material Specifications

This section should list all materials in the vessel system, applicable attachments, and appurtenances and provide the material specifications, making appropriate references to the applicable sections in Chapter 3 of the FSAR. If any materials other than those listed in Part D to Section II of the ASME Code are used, applicants should provide the data identified in Appendix 5 to Part D of Section II of the ASME Code for approval of the new material. This section should reference information regarding material specifications provided in other documents or sections of the FSAR. It should address mechanical and physical properties of these materials and describe the effects of radiation on these materials, where applicable.

Modified Section/Title: C.I.5.3.1.2; Special Processes Used for Manufacturing and Fabrication

Original Section/Title: C.I.5.3.1.2; Special Processes Used for Manufacturing and Fabrication

This section should describe the manufacture of the product forms and methods used to fabricate the vessel system or any of its applicable attachments and appurtenances. Applicants should discuss any special or unusual processes used and show that they will not compromise the integrity of the vessel system.

Modified Section/Title: C.I.5.3.1.3; Special Methods for Nondestructive Examination

Original Section/Title: C.I.5.3.1.3; Special Methods for Nondestructive Examination

This section should describe in detail all special procedures for detecting surface and internal discontinuities, with emphasis on procedures that differ from those in Section III of the ASME Code. Applicants should pay particular attention to calibration methods, instrumentation, method of application, sensitivity, reliability, reproducibility, and acceptance standards.

Modified Section/Title: C.I.5.3.1.4; Special Controls for Ferritic and Austenitic Stainless Steels

Original Section/Title: C.I.5.3.1.4; Special Controls for Ferritic and Austenitic Stainless Steels

Making appropriate references to the applicable sections in Chapter 3 of the FSAR, applicants should describe controls on welding, composition, heat treatments, and similar processes covered by regulatory guides to verify that these recommendations or equivalent controls are employed. The description should include controls for abrasive work (e.g., grinding) on austenitic stainless steel. Applicants should address the following guidance:

- RG 1.34, "Control of Electroslag Weld Properties" except for Position C.3.a (in Rev 1), use the appropriate paragraph in Article 2000 of the Code, Section III, Division 5, Subsection HB.
- RG 1.44, "Control of the Use of Sensitized Stainless Steel" contains NRC staff positions related to unstabilized austenitic stainless steel of the AISI Type 3XX series. High temperature austenitic stainless steel applications in vessel system components must meet similar stabilization criteria for the specific temperature application.
- RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel"
- RG 1.71, "Welder Qualification for Areas of Limited Accessibility"
- RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials" - for the reactor vessel only
- RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" for the reactor vessel only - methods in this RG should be adjusted based on the HTGR design.

Modified Section/Title: C.I.5.3.1.5; Fracture Toughness

Original Section/Title: C.I.5.3.1.5; Fracture Toughness

This section should describe the fracture testing and acceptance criteria specified for materials of the reactor vessel and appurtenances thereto. In particular, it should describe how the toughness requirements of Appendix G to 10 CFR Part 50 will be met. This section should include a discussion regarding how high temperature material aging are addressed in vessel system components – refer to **NGNP White Paper INL/EXT-09-17187.**

Applicants should specify the maximum nil ductility reference temperature (RTNDT) to which the ferritic materials of the vessel system will be fabricated. They should identify ITAAC that will be completed to verify that these ferritic vessel system materials meet these specifications.

Modified Section/Title: C.I.5.3.1.6; Reactor Vessel Material Surveillance

Original Section/Title: C.I.5.3.1.6; Material Surveillance

This section should describe the material surveillance program for the reactor vessel in sufficient detail to provide assurance that the program meets the requirements of Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50. It should describe the method for calculating neutron fluence for the reactor vessel beltline and the surveillance capsules. Because the material surveillance program is an operational program, as discussed in SECY-05-0197, the program and its implementation milestones should be fully described and reference any applicable standards. Fully described should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow for a reasonable assurance finding of acceptability. In particular, applicants should address the following six topics:

- Basis for selection of material in the program
- Number and type of specimens in each capsule
- Number of capsules and proposed withdrawal schedule comparable with American Society for Testing and Materials (ASTM) Standard E 185, "Surveillance Tests on Structural Materials in Nuclear Reactors," as referenced in 10 CFR Part 50, Appendix H
- Neutron flux and fluence calculations for vessel wall and surveillance specimens and conformance with the guidance of RG 1.190
- Expected effects of radiation on vessel wall materials and basis for estimation
- Location of capsules, method of attachment, and provisions to ensure that capsules will be retained in position throughout the reactor vessel lifetime.

Modified Section/Title: C.I.5.3.1.7; Vessel System Fasteners
Original Section/Title: C.I.5.3.1.7; Reactor Vessel Fasteners
This section should describe the materials and design for the stud bolts, washers, nuts, and other fasteners for the reactor vessel, cross vessel, and steam generator vessel closure. It should include sufficient detail regarding materials property requirements, nondestructive evaluation techniques, lubricants or surface treatments, and protection provisions to show the specifications of Appendix I to Section III of the ASME Code, providing the data called for under Appendix IV to Section III of the ASME Code and the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," or equivalent measures, are followed. The FSAR should describe the mechanical property and fracture toughness tests that will be performed to demonstrate that the materials from which these fasteners are fabricated conform to these recommendations or their equivalent. Applicants should identify any ITAAC that will be completed to verify that the materials from which these fasteners are constructed met these specifications.

Modified Section/Title: C.I.5.3.2; Pressure-Temperature Limits and Charpy Upper-Shelf Energy Data and Analyses
Original Section/Title: C.I.5.3.2; Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses
This section of the FSAR should describe the bases for setting operational limits on pressure and temperature for the vessel system during any condition of normal operation, including AOO, and pressure tests. In addition, this discussion should provide detailed assurance that Appendices G and H to 10 CFR Part 50 will be complied with throughout the life of the plant.

Modified Section/Title: C.I.5.3.2.1; Limit Curves
Original Section/Title: C.I.5.3.2.1; Limit Curves
This section should describe how the applicant will develop pressure-temperature limit curves for (1) preservice system hydrostatic tests, (2) inservice leak and pressure tests, (3) normal operation, including heatup and cooldown, and (4) reactor core operation.

If procedures or criteria other than those recommended in the ASME Code are used, applicants should show that equivalent safety margins are provided. This section should describe the bases used to determine these limits and provide typical curves with temperatures relative to the RTNDT of the limiting material (as defined in the appropriate paragraph 2300 of the Code, Section III, Division 5, Subsection HB).

Based on material properties, such as initial RTNDT and material chemical composition, to which vessel system ferritic materials will be procured, applicants should demonstrate how pressure-temperature limits that meet the requirements of Appendix G to 10 CFR Part 50 can be met for the licensed life of the facility. Applicants should describe the bases used for the prediction and indicate the extent to which the recommendations of RG 1.99 are followed.

This section should describe procedures that will be used to update these limits while in service and address radiation effects and the extent to which the recommendations of RG 1.190 are followed.

Modified Section/Title: C.I.5.3.2.2; Operating Procedures
Original Section/Title: C.I.5.3.2.2; Operating Procedures
This section should describe how the applicant will develop operating procedures that will ensure that the pressure-temperature limits in Section 5.3.2.1 of the FSAR will not be exceeded during any condition of normal operation, including AOO, and system hydrostatic tests. The FSAR should include a commitment that plant operating procedures will ensure that the pressure-temperature limits identified in Section 5.3.2.1 of the FSAR will not be exceeded during any foreseeable upset condition.

This section should also describe the provisions to ensure that the emissivity of the reactor vessel continues to meet the requirements for assumed heat removal.

Modified Section/Title: C.I.5.3.2.3; N/A
Original Section/Title: C.I.5.3.2.3; Pressurized Thermal Shock (PWRs only)

Modified Section/Title: C.I.5.3.2.4; Vessel System Upper-Shelf Energy
Original Section/Title: C.I.5.3.2.4; Upper-Shelf Energy
Applicants should specify minimum Charpy upper-shelf energy values to which ferritic materials of the vessel system will be procured. Applicants should provide projected Charpy upper-shelf energy values at the expiration date of the operating license based on the methodology in RG 1.99 and demonstrate that beltline materials will satisfy the requirement of Appendix G (paragraph IV.A.1.a) to 10 CFR Part 50 for the reactor vessel.

Modified Section/Title: C.I.5.3.3; Vessel System Integrity
Original Section/Title: C.I.5.3.3; Reactor Vessel Integrity
This section of the FSAR should provide a summary of all information related to the integrity of the vessel system, including the major considerations in achieving vessel system safety and a description of the factors contributing to the vessel system's integrity. The COL applicant may identify a specific manufacturer, if one has been chosen, and provide a description of its experience.

Modified Section/Title: C.I.5.3.3.1; Design
Original Section/Title: C.I.5.3.3.1; Design
This section should briefly describe the vessel system design, preferably with a schematic, including materials, construction features, fabrication methods, and inspections. Applicants should summarize applicable design codes and bases and reference other sections of the FSAR as appropriate.

This section should describe how the design of the reactor internals, vessel system, vessel system supports ensure that their integrity is maintained during postulated accidents to (1) provide a geometry conducive to removal of residual heat from the reactor core to the ultimate heat sink, and (2) permit sufficient insertion of neutron absorbers to effect reactor shutdown to maintain the reactor within specified acceptable radionuclide release limits.

Modified Section/Title: C.I.5.3.3.2; Materials of Construction
Original Section/Title: C.I.5.3.3.2; Materials of Construction
Applicants should identify the vessel system materials, including weld materials, and describe any special requirements. They should emphasize the reasons for selection and provide assurance of suitability.

Modified Section/Title: C.I.5.3.3.3; Fabrication Methods
Original Section/Title: C.I.5.3.3.3; Fabrication Methods
This section should identify the vessel system fabrication methods, including forming, welding, and machining. Applicants should describe the service history of vessels constructed using these methods and the vessel system supplier's experience with the procedures.

Modified Section/Title: C.I.5.3.3.4; Inspection Requirements
Original Section/Title: C.I.5.3.3.4; Inspection Requirements
This section should summarize the inspection test methods and requirements, paying particular attention to the level of initial integrity. Applicants should describe any methods that are in addition to the guidelines established in Section III of the ASME Code.

Modified Section/Title: C.I.5.3.3.5; Shipment and Installation
Original Section/Title: C.I.5.3.3.5; Shipment and Installation
This section should summarize the means used to protect the vessel system so that it's as-manufactured integrity will be maintained during shipment and site installation. Applicants should reference other FSAR sections as appropriate.

Modified Section/Title: C.I.5.3.3.6; Operating Conditions
Original Section/Title: C.I.5.3.3.6; Operating Conditions
This section should summarize the operational limits that will be specified to ensure vessel system safety. Applicants should provide a basis for concluding that vessel system integrity will be maintained during the most severe postulated transients, referencing other FSAR sections as appropriate.

This section should also describe the lower emissivity limit that applies to the reactor vessel for heat removal.

Modified Section/Title: C.I.5.3.3.7; Reactor Vessel Inservice Surveillance
Original Section/Title: C.I.5.3.3.7; Inservice Surveillance
This section should summarize the ISI and material surveillance programs and explain their adequacy relative to the guidelines of Appendix H to 10 CFR Part 50 and Section XI of the ASME Code. Applicants should reference Section 5.3.1.6 as appropriate.

Modified Section/Title: C.I.5.3.3.8; Threaded Fasteners
Original Section/Title: C.I.5.3.3.8; Threaded Fasteners
This section should summarize the programs for ensuring the integrity of bolting and threaded fasteners and their adequacy. Applicants should reference FSAR Section 3.13 as appropriate.

Modified Section/Title: C.I.5.4; Subsystems and Components Connected to the Vessel System
Original Section/Title: C.I.5.4; Reactor Coolant System Component and Subsystem Design
This section of the FSAR should provide information regarding performance requirements and design features to ensure the various systems, subsystems, and components within or allied with the HPB accomplish their safety functions. Because these components and subsystems differ for various types and designs of reactors, the components and subsystems are not assigned specific subsection numbers.

The FSAR should contain a separate subsection (numbered 5.4.1 through 5.4.x) for each principal component or subsystem. Each subsection should present the design bases, description, evaluation, and necessary tests and inspections for the component or subsystem, including radiological considerations from the viewpoints of how radiation affects operation and how radiation levels affect the operators and capabilities of operation and maintenance. Applicants should describe the appropriate details regarding the mechanical design in FSAR Sections 3.7, 3.9, and 5.2.

The following subsections identify components and subsystems that should be discussed and the corresponding information that should be provided. As appropriate to the specific reactor type and design, certain subsections are not applicable, and additional subsections are necessary to address other components and subsystems (e.g., the heat transport system that include the hot duct assembly, steam generator, intermediate heat exchanger, and helium circulator, and the shutdown cooling system which includes its heat exchanger and circulator, and those portions within the isolation valves of the helium purification system).

Modified Section/Title: C.I.5.4.1; Cross Vessel
Original Section/Title: C.I.5.4.1; Reactor Coolant Pumps
N/A - Relocated to section 5.3

Modified Section/Title: C.I.5.4.1.1; N/A
Original Section/Title: C.I.5.4.1.1; Pump Flywheel Integrity (PWRs only)

Modified Section/Title: C.I.5.4.2; Heat Transport System
Original Section/Title: C.I.5.4.2; Steam Generators (PWRs only)

(NOTE: Discussion based on RG 1.206, C.I.5.3.3, Rx Vessel Integrity)

This section of the FSAR should provide a summary of all information related to the required safety functions if any of the HTS and include the integrity of those portions of the heat transport system that are part of the HPB. The COL applicant may identify specific types of components, for example an axial compressor with magnetic bearings or a helical coil steam generator, and a specific manufacturer, if one has been chosen, and provide a description of its experience.

Design
This section should briefly describe the HTS design, preferably with a schematic, including materials, construction features, fabrication methods, and inspections. Applicants should summarize applicable design codes and bases and reference other sections of the FSAR as appropriate.

Materials of Construction
Applicants should identify the HTS materials, including weld materials, and describe any special requirements. They should emphasize the reasons for selection and provide assurance of suitability.

Fabrication Methods
This section should identify the HTS fabrication methods, including forming, welding, cladding, and machining. Applicants should describe the service history of vessels constructed using these methods and the vessel supplier's experience with the procedures.

Inspection Requirements
This section should summarize the inspection test methods and requirements, paying particular attention to the level of initial integrity. Applicants should describe any methods that are in addition to the guidelines established in Section III, Division 5, of the ASME Code.

Shipment and Installation
This section should summarize the means used to protect the HTS so that it's as-manufactured integrity will be maintained during shipment and site installation. Applicants should reference other FSAR sections as appropriate.

Operating Conditions
This section should summarize the operational limits that will be specified to ensure the HTS required safety functions over the range of normal operation, AOOs, and postulated accidents. Applicants should provide a basis for concluding that heat transport system will perform its required safety functions over the spectrum of normal and off-normal design conditions, referencing other FSAR sections as appropriate.

Inservice Surveillance
This section should summarize the ISI and material surveillance programs, if any, and explain their adequacy relative to the guidelines of Appendix H to 10 CFR Part 50 and Section XI of the ASME Code. Applicants should reference Section 5.3.1.6 as appropriate.

Threaded Fasteners
This section should summarize the programs for ensuring the integrity of bolting and threaded fasteners and their adequacy as appropriate. Applicants should reference FSAR Section 3.13 as appropriate.

Modified Section/Title: *C.I.5.4.2.1; Hot Duct Assembly*
Original Section/Title: C.I.5.4.2.1; Steam Generator Materials

(NOTE: Discussion based on HTS SDD, RG 1.206, C.I.5.1 and C.I.5.2)

This section of the FSAR should provide a summary description of the hot duct assembly. This description should indicate the safety functions of the hot duct assembly, if any, including a tabulation of important design and performance characteristics.

Applicants should provide a plan and elevation drawing showing principal dimensions of the hot duct assembly in relation to the surrounding concrete structures from which a measure of the protection is afforded by the arrangement and the safety considerations incorporated in the layout can be gained.

This section of the FSAR should discuss the measures to be employed to ensure and maintain the integrity of the hot duct assembly throughout the plant design lifetime.

This section should provide a table showing compliance with the NRC regulations in 10 CFR 50.55a. 10 CFR 50.55a specifies minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of systems, structures, and components commensurate with the importance of the safety function to be performed by requiring compliance with appropriate editions of published industry codes and standards. The requirements in 10 CFR 50.55a, paragraphs (a) through (g) for meeting ASME codes are not endorsed by the NRC for an HTGR. Hence, the applicant should address 10 CFR 50.55a paragraphs (a) through (g) as guidance, referencing HTGR applicable ASME codes instead of those referenced in 10 CFR 50.55a paragraph (b) where appropriate. The applicant should address 10 CFR 50.55a, paragraph (h) as a requirement when addressing standards for protection and safety systems.

If conformance to the regulations of 10 CFR 50.55a would result in hardships or unusual difficulties without a compensating increase in the level of safety and quality, applicants should provide a complete description of the circumstances resulting in such cases and the basis for proposed alternative requirements. The description should cover how the proposed alternative requirements will provide an equivalent and acceptable level of safety and quality.

This section should identify the applicable industry codes and classifications applied to the hot duct assembly.

Applicants should provide a list of NRC approved ASME Code cases that will be applied to the hot duct assembly. If applicants use code cases other than those approved, they should show that their use will result in as acceptable a level of quality and safety for the component.

Applicants should identify the material specifications for the hot duct assembly.
Applicants should identify all process instrumentation.

This section should identify the tests and inspections to be performed (1) before operation and during startup that demonstrate the functional performance and (2) as inservice surveillance to ensure continued reliability. Applicants should describe specific testing of the low-temperature overpressure protection system, particularly operability testing, exclusive of relief valves, before each shutdown.

Structural Requirements (NOTE: Discussion based on HTS SDD)
The hot duct assembly shall be designed for the mechanical and thermal loads resulting from specified design transients. Failure the hot duct assembly shall not cause failure of any “safety-related” SSC during a Safe Shutdown Earthquake (SSE). The hot duct assembly shall be designed to resume operations after an Operating Basis Earthquake (OBE). The hot duct assembly shall be designed for the maximum acoustic pressure levels resulting from HTS or SCS operation. The hot duct assembly shall be designed for the maximum flow-induced forces resulting from HTS or SCS operation.

Environmental Requirements
For portions of the hot duct assembly exposed to the primary coolant, describe the chemical impurities to which the hot duct assembly will be exposed and the measures to prevent degradation of the assembly due to such exposure.

Maintenance Requirements
Describe the capability to access the hot duct assembly to perform maintenance and inspection, including shielding and decontamination requirements and required instrumentation.

Quality Assurance Requirements
The hot duct assembly is not safety-related. Non-safety related items shall come under a Quality Assurance Program based on their safety significance as determined by the applicant.

Modified Section/Title: *C.I.5.4.2.2; Steam Generator (excluding vessel)*
Original Section/Title: C.I.5.4.2.2; Steam Generator Program

(NOTE: This section based on RG 1.206, section 5.4.2 and to a lesser extent the LANL Report, section 5.4.2, which follows RG 1.206 fairly closely)

Steam Generators
Applicants should provide the steam generator design criteria to prevent unacceptable damage to the heat exchange medium from flow-induced vibration (FIV) and cavitation, referencing the information provided in FSAR Section 3.9.3 and including the following two specific pieces of information:

1. Design conditions and transients that will be specified in the design of the steam generator heat exchange medium and the operating condition category selected that defines the allowable stress intensity limits to be used and the justification for this selection

2. Extent of heat exchange medium wall thinning that could be tolerated without exceeding the allowable stress intensity limits defined above under the postulated condition of a design-basis pipe break in the HPB or a break in the secondary piping during reactor operation.

Modified Section/Title: C.I.5.4.2.2.1; Steam Generator Materials
Original Section/Title: NEW; NEW

(NOTE: Based on RG 1.206, section 5.4.2.1, Steam Generator Materials)

Applicants should discuss the design of the steam generator, including (1) the selection, processing, testing, and inspection (during fabrication/processing) of the materials used to fabricate the steam generator, (2) design provisions for limiting the susceptibility of the steam generator to degradation and/or corrosion, (3) fracture toughness of the ferritic materials used in the steam generator, (4) fabrication and processing of austenitic stainless materials (if used in pressure boundary applications), (5) compatibility of materials with the primary (reactor) and secondary coolant, and (6) provisions for accessing the secondary side of the steam generator for maintenance and cleaning.

Applicants should address the following considerations:

1. Making appropriate references to FSAR Section 5.2.3, applicants should provide information on the selection and fabrication of materials for components of the steam generators.

2. Applicants should provide information on the fracture toughness properties of ferritic materials, making appropriate references to FSAR Section 5.2.3.

3. Applicants should provide information on those aspects of design that may affect the performance of steam generator materials.

4. Applicants should provide information on the fabrication and processing of austenitic stainless steel materials (if used in pressure boundary applications), as discussed in Section 5.2.3.4 of this guide.

5. Applicants should provide information on the compatibility of steam generator heat exchange medium with both the primary and secondary coolant. They should describe the methods used in monitoring and maintaining the chemistry of the primary and secondary coolant within the specified ranges.

6. Applicants should describe the design provisions for removing surface deposits, sludge, loose parts (foreign objects), and excessive corrosion products from the secondary side of the steam generator. Describe onsite cleaning and cleanliness control provisions and show that they produce results equivalent to those obtained by following the recommendations of Regulatory Guide 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled

Nuclear Power Plants," and ANSI Standard N45.21-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants," or their functional equivalents.

Modified Section/Title: C.I.5.4.2.2.2; Steam Generator Program

Original Section/Title: NEW; NEW

(NOTE: Based on RG 1.206, section C.I.5.4.2.2, Steam Generator Program)

Applicants should describe provisions in the design of the primary and secondary side of the steam generator that permit implementation of a steam generator integrity program. They should describe the elements of the steam generator integrity program, addressing the following three considerations:

1. Regarding steam generator design, applicants should describe the design provisions for permitting access to both the primary and secondary side of the steam generator. They should discuss the extent to which accessibility is afforded for periodic inspection, testing, and repair using currently available methods and techniques (which are capable of finding the forms of degradation that may affect service life). The application should describe design provisions for inspecting and removing loose parts (foreign objects) from the steam generator as well as for limiting the introduction of loose parts into the steam generator.

2. Applicants should discuss the method for determining repair criteria and describe the scope and extent of the preservice inspection.

3. Applicants should describe the steam generator inspection and reporting requirements to be adopted into the TS (including the limiting conditions for operation (LCO), surveillance requirements, and primary-to-secondary leakage limits). See NGNP white paper INL/EXT-09-17187, ***NGNP High Temperature Materials*** for further information on materials).

Modified Section/Title: C.I.5.4.3; N/A

Original Section/Title: C.I.5.4.3; Reactor Coolant Piping

Not applicable.

Modified Section/Title: C.I.5.4.4; Reserved

Original Section/Title: C.I.5.4.4; [Reserved]

Not applicable.

Modified Section/Title: C.I.5.4.4.1; Intermediate Heat Exchanger

Original Section/Title: NEW; NEW

(NOTE: This section based on RG 1.206, section 5.4.2 and to a lesser extent the LANL Report, section 5.4.2, which follows RG 1.206 fairly closely)

Intermediate Heat Exchanger

Applicants should provide the IHX design criteria to prevent unacceptable damage to the heat exchange medium from flow-induced vibration (FIV), referencing the information provided in FSAR Section 3.9.3 and including the following two specific pieces of information:

1. Design conditions and transients that will be specified in the design of the IHX heat exchange medium and the operating condition category selected that defines the allowable stress intensity limits to be used and the justification for this selection

2. Extent of heat exchange medium wall thinning that could be tolerated without exceeding the allowable stress intensity limits defined above under the postulated condition of a design-basis pipe break in the HPB or a break in the secondary piping during reactor operation.

Modified Section/Title: C.I.5.4.4.2; Helium Circulator

Original Section/Title: NEW; NEW

(NOTE: This is based on LANL Report, Section 5.4.1)

Provide specific information to demonstrate that the main (reactor coolant) circulator provides the pressure head that is necessary for forced circulation of the reactor coolant through the heat transport system (HTS) to perform the core cooling function over the entire range of operation required. Demonstrate that the analyses account for uncertainties associated with reactor coolant flow resistance during passage through the core and HTS, heat transfer coefficients, flow and bypass flow distribution, and reactor coolant heat losses to arrive at the required sizing, speed (rpm), and flow rate characteristics of the circulator. Provide operational envelopes to show that, considering the above uncertainties, the circulator is correctly sized. Address the necessity to provide a full scale mock-up of the HTS and to provide for cold-and hot-flow testing. Show that the design allows for extended periods of operation with minimum wear and deterioration, provides for expeditious removal and decontamination, and addresses overspeed considerations. Discuss the design and materials selection of the main circulator to ensure integrity, durability, compatibility with the service environment, and serviceability.

Show that the main-loop shutoff valve, which is a part of the main circulator assembly, closes within the required time when the main circulator is shutdown and that the projected backflow through the valve is conservative. Show that the design allows for monitoring the position of the valve and that any override mechanism is feasible. Demonstrate that the closing surfaces of the valve will not deteriorate in the reactor coolant atmosphere. Discuss the necessity of bench testing the valve mechanism, and show provisions for in situ testing.

Show that the main circulator motor cavity has adequate provisions for motor cooling and prevents inadvertent leakage of the reactor coolant at all reactor pressures. Address fluid ingress to the reactor coolant, from either the circulator bearings or the motor-cooling heat exchangers.

Provide information that the vibrational characteristics of the main circulator combined bearings and rotor system should preclude encountering resonant frequencies over the entire range of operation. Show that the design of the circulator is such that no loss of reactor vessel or reactor coolant pressure boundary integrity will result from circulator-generated vibration. Discuss the potential for circulator-generated missiles from rotor failure to cause consequential damage to the reactor coolant pressure boundary, to in-vessel cooling systems that could lead to fluid ingress, and to other structures, systems, and components important to safety. Show provisions to address the consequences of any of the above occurrences.

Provide information that fasteners, bolts, shrouds, etc., associated with the circulator assembly are not subject to vibrational or stress corrosion failure. Include enough detail regarding materials property requirements, nondestructive evaluation procedures, lubricants or surface treatments, and protection provisions to show that the recommendations of Regulatory Guide 1.65, “Materials and Inspections of Reactor Vessel Closure Studs,” Rev. 1, or its functional equivalent, are followed.

Modified Section/Title: C.I.5.4.5; N/A

Original Section/Title: C.I.5.4.5; [Reserved]

Not applicable.

Modified Section/Title: C.I.5.4.6; N/A
Original Section/Title: C.I.5.4.6; Reactor Core Isolation Cooling System (BWRs only)
Not applicable.

Modified Section/Title: C.I.5.4.6.1; N/A
Original Section/Title: C.I.5.4.6.1; Design Bases
Not applicable.

Modified Section/Title: C.I.5.4.6.2; N/A
Original Section/Title: C.I.5.4.6.2; System Design
Not applicable.

Modified Section/Title: C.I.5.4.6.3; N/A
Original Section/Title: C.I.5.4.6.3; Performance Evaluation
Not applicable.

Modified Section/Title: C.I.5.4.7; Shutdown Cooling System
Original Section/Title: C.I.5.4.7; Residual Heat Removal System

Modified Section/Title: C.I.5.4.7.1; Design Bases
Original Section/Title: C.I.5.4.7.1; Design Bases

(NOTE: Based on HTGR PSID 5.4.1 and RG 1.206, 5.4.7.1)

Applicants should provide a summary description of the Shutdown Cooling System (SCS), including design features to provide system reliability.

Applicants should discuss design bases of the SCS, including its heat rejection capability and peak cooling capacity.

Applicants should discuss functional design bases, including the time required to reduce the RCS temperature to approximately 100 °C (212°F) [applicant to provide alternative values] and to a temperature that would permit refueling. They should present the design-basis times for the case where the entire SCS is operable as well as the case with the most limiting single failure in the SCS.

Applicants should discuss the design bases for the isolation of the SCS from the HPB. The discussion should cover the isolation design bases, including any interlocks that are provided, and the design bases regarding prevention of SCS damage in the event of closure of the isolation valves.

Applicants should discuss the design bases of the SCS for the prevention of an interfacing system accident.

Applicants should discuss the design bases for the pressure relief capacity of the SCS. The discussion should include the design bases and considerations for limiting transients, equipment malfunctions, and possible operator errors during plant startup and cooldown when the SCS is not isolated from the HPB.

Applicants should discuss the design bases for reliability and operability requirements. They should describe the design bases regarding the manual actions required to operate the system, emphasizing any operations that cannot be performed from the control room in the event of a single failure. The description should cover protection against single failure in terms of piping arrangement and layout, selection of valve types and locations, redundancy of various system components, redundancy of power supplies, and redundancy of instrumentation. It should also include protection against valve motor flooding and spurious single failures.

Applicants should discuss the design bases established to protect the SCS from physical damage. The discussion should cover the design bases for the SCS support structure and for protection against incidents and accidents that could render redundant components inoperative (e.g., fires, pipe whip, internally generated missiles, accident loads, seismic events).

Applicants should discuss the design bases of the SCS for shutdown operations.

Applicants should discuss the design bases of the SCS relief valves.

Since HTGR designs might include active SCS components designated as non-safety related systems for defense in depth functions, applicants should provide an evaluation in accordance with the process of non-safety related with special treatment (NSRST) to determine necessary regulatory oversight for the active SCS.

Modified Section/Title: C.I.5.4.7.1.1; Shutdown Cooling HX and Circulator

Original Section/Title: NEW; NEW
See section 5.4.7.1 for requirements.

Modified Section/Title: C.I.5.4.7.1.2; Shutdown Cooling Water System

Original Section/Title: NEW; NEW
See section 9.2.1 for requirements.

Modified Section/Title: C.I.5.4.7.2; System Design

Original Section/Title: C.I.5.4.7.2; System Design

(NOTE: Based on RG 1.206, section C.I.5.4.7.2)

Applicants should provide the following six pieces of information about SCS design:

1. Applicants should provide a description of the shutdown cooling system, including schematic piping and instrumentation diagrams showing all components, piping, points where connecting systems and subsystems tie together, and I&C associated with subsystem and component actuation. The description should cover component interlocks. Applicants should provide a mode diagram showing temperatures, pressures, and flow rates for each mode of SCS operation.

2. Applicants should provide equipment and component descriptions that cover each component of the system. The descriptions should identify the significant design parameters for each component, state the design pressure and temperature of components for various portions of the system, and explain the bases for their selection. Applicants should provide circulator characteristic curves and circulator power requirements. Applicants should describe heat exchanger characteristics, including design flow rates, inlet and outlet temperatures for the cooling fluid and for the fluid being cooled, the overall heat transfer coefficient, and the heat transfer area. They should identify each component of the SCS that is also a portion of some other system.

3. Regarding control, applicants should state the SCS relief valve capacity and settings and the method of collecting fluids discharged through the relief valve. Applicants should describe provisions with respect to the control circuits for motor-operated isolation valves in the SCS, including consideration of inadvertent actuation. The description should include discussions of the controls and interlocks for these values (e.g., intent of Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations"), considerations for automatic valve closure (e.g., RCS pressure exceeds design pressure of SCS), valve position indications, and valve interlocks and alarms.

4. Applicants should identify the applicable industry codes and classifications for the system design.

5. Applicants should discuss system reliability considerations, including provisions incorporated in the design to ensure that the system will operate when needed and will deliver the required flow rates (e.g., redundancy and separation of components and power sources).

6. Applicants should discuss all manual actions that an operator must take for the SCS to operate properly with all components assumed to be operable. The discussion should identify any actions that must be taken from outside the control room. Applicants should repeat this discussion for the most limiting single failure in the SCS.

Modified Section/Title: C.I.5.4.7.2.1; Shutdown Cooling HX and Circulator
Original Section/Title: NEW; NEW
See section 5.4.7.1 for requirements.

Modified Section/Title: C.I.5.4.7.2.2; Shutdown Cooling Water System
Original Section/Title: NEW; NEW
See section 5.4.7.1 for requirements.

Modified Section/Title: C.I.5.4.7.3; Performance Evaluation
Original Section/Title: C.I.5.4.7.3; Performance Evaluation

(NOTE: Based on RG 1.206, section C.I.5.4.7.3)

Applicants should provide an evaluation of the ability of the SCS to reduce the temperature of reactor coolant at a rate consistent with the design basis (see Section C.I.5.4.7.1 of this guide).

Applicants should describe the analytical methods used and clearly state all assumptions. They should provide curves showing the reactor coolant temperature as a function of time for the two following cases:

1. All SCS components are operable

2. The most limiting single failure has occurred in the SCS.

Modified Section/Title: C.I.5.4.7.3.1; Shutdown Cooling HX and Circulator
Original Section/Title: NEW; NEW
See section 5.4.7.3 for requirements.

Modified Section/Title: C.I.5.4.7.3.2; Shutdown Cooling Water System
Original Section/Title: NEW; NEW
See section 5.4.7.3 for requirements.

Modified Section/Title: C.I.5.4.8; Helium Purification System

Original Section/Title: C.I.5.4.8; Reactor Water Cleanup System (BWRs only)
See sections 5.4.8.1, 5.4.8.2, and 5.4.8.3 for requirements.

Modified Section/Title: C.I.5.4.8.1; Design Bases

Original Section/Title: C.I.5.4.8.1; Design Bases

(NOTE: Based on LANL Report, section 5.4.10)

Describe the helium purification system. The design basis for the helium purification system should include consideration of the capability for the control of reactor coolant purity, capability for maintaining the required reactor coolant system inventory, code design requirements, and system design to detect and control the release of radioactive system effluents to the environment.

Modified Section/Title: C.I.5.4.8.2; System Description

Original Section/Title: C.I.5.4.8.2; System Description

(NOTE: Based on LANL Report, Section 5.4.10)

Describe each piece of equipment as to size and or capacity, flow rates, and storage capabilities. Indicate the design pressure and temperature of each piece of equipment, and cite pertinent previous experience with such equipment.

Modified Section/Title: C.I.5.4.8.3; Performance Evaluation

Original Section/Title: C.I.5.4.8.3; Performance Evaluation

(NOTE: Based on LANL Report, section 5.4.1)

Provide an evaluation of the helium purification system that includes an analysis of the effects of component malfunctions, an analysis of the capability to control the concentrations of chemical and radioactive impurities in the primary reactor coolant within acceptable limits, an analysis of the availability and reliability of the system, and an analysis of the capability to isolate the system in the event of a malfunction or rupture that would release radioactivity. The radiological evaluation for normal operation should be presented in Chapters 11 and 12.

Demonstrate the adequacy of the isolation valve(s) which separates the helium purification system from the reactor coolant system. Show that test provisions are available to demonstrate this adequacy. Provide information showing justification for any shared components or subsystems.

Modified Section/Title: C.I.5.4.9; Reserved

Original Section/Title: C.I.5.4.9; [Reserved]

Modified Section/Title: C.I.5.4.10; N/A

Original Section/Title: C.I.5.4.10; [Reserved]
This section is not applicable.

Modified Section/Title: C.I.5.4.11; N/A

Original Section/Title: C.I.5.4.11; Pressurizer Relief Tank (PWRs only)
This section is not applicable.

Modified Section/Title: C.I.5.4.11.1; N/A
Original Section/Title: C.I.5.4.11.1; Design Bases
This section is not applicable.

Modified Section/Title: C.I.5.4.11.2; N/A
Original Section/Title: C.I.5.4.11.2; System Description
This section is not applicable.

Modified Section/Title: C.I.5.4.11.3; N/A
Original Section/Title: C.I.5.4.11.3; Performance Evaluation
This section is not applicable.

Modified Section/Title: C.I.5.4.11.4; N/A
Original Section/Title: C.I.5.4.11.4; Instrumentation
This section is not applicable.

Modified Section/Title: C.I.5.4.12; Overpressure Protection
Original Section/Title: C.I.5.4.12; Reactor Coolant System High Point Vents
N/A - This topic is addressed in section 5.2.2

Modified Section/Title: C.I.5.4.12.1; Design Bases
Original Section/Title: C.I.5.4.12.1; Design Bases

(NOTE: For LWRs, this section is meant to address high point vents. HTGR Overpressure Protection is presented in section 5.2.2)

Modified Section/Title: C.I.5.4.12.2; System Design
Original Section/Title: C.I.5.4.12.2; System Design

(NOTE: For LWRs, this section is meant to address high point vents. HTGR overpressure protection is presented in section 5.2.2)

Modified Section/Title: C.I.5.4.12.3; Performance Evaluation
Original Section/Title: C.I.5.4.12.3; Performance Evaluation

(NOTE: For LWRs, this section addresses high point vents. For HTGRs, overpressure protection is addressed in section 5.2.2)

Modified Section/Title: C.I.5.4.13; [Reserved]
Original Section/Title: C.I.5.4.13; [Reserved]
This section is not applicable.

Modified Section/Title: C.I.5.4.14; [Reserved]
Original Section/Title: C.I.5.4.14; [Reserved]
This section is not applicable.

Appendix F
Chapter 6. Engineered Safety Features

Appendix F
Chapter 6. Engineered Safety Features

Section/Title: C.I.6; Engineered Safety Features
Original Section/Title: C.I.6; Engineered Safety Features
Chapter 6 of the FSAR should provide a discussion of how the design of ESF meets the applicable regulatory requirements and available regulatory guidance.

Describe the major elements of Section 1.3 that are applicable to the design of the engineered safety features. The description should include functional requirements, the role of the ESF in the overall safety design, and principle design criteria.

ESFs are provided to mitigate the consequences of postulated accidents in the unlikely event that an accident occurs. Together with 10 CFR 50.55a the following GDC, as set forth in Appendix A to 10 CFR Part 50 require that certain systems must be provided to serve as ESF systems:

1. GDC 1, "Quality standards and records" **[PDC 1]**

2. GDC 4, "Environmental and dynamic effects design bases" **[PDC 4]**.

Functional Containment and Reactor Cavity Cooling System (RCCS) are typical of the systems that are required to be provided as ESFs. The application should include information on the plant's ESF systems in sufficient detail to permit an adequate evaluation of the performance capability of these features.

The ESF systems provided in plant designs may vary. The ESF systems explicitly discussed in this chapter are those that are commonly used to limit the consequences of postulated accidents in high-temperature gas-cooled power reactors, and should be treated as illustrative of the ESF systems and of the kind of informative material that is needed. This section of the FSAR should list and discuss each system that is considered to be part of the ESF systems. The discussions on ESF designs should identify functional requirements, demonstrate how the functional requirements comply with regulatory requirements, and demonstrate how the ESF design meets or exceeds the functional requirements.

Modified Section/Title: C.I.6.1; Engineered Safety Feature Materials
Original Section/Title: C.I.6.1; Engineered Safety Feature Materials
The applicant should discuss the materials used in ESF systems/components (RCCS and the reactor building). For RCCS designs, the applicant should also discuss the material interactions with RCCS fluids that could potentially impair operation of ESF systems.

The intent of the information included in this section of the FSAR is to ensure compatibility of the materials with the environmental conditions to which the materials are subjected. The application should include adequate and sufficient information to ensure compliance with the applicable Commission regulations in 10 CFR Part 50 (including applicable GDC **[PDC]**), the positions of applicable regulatory guides and branch technical positions, and the applicable provisions of the ASME Boiler and Pressure Vessel Code.

Modified Section/Title: C.I.6.1.1; Metallic Materials
Original Section/Title: C.I.6.1.1; Metallic Materials

Modified Section/Title: C.I.6.1.1.1; Materials Selection and Fabrication

Original Section/Title: C.I.6.1.1.1; Materials Selection and Fabrication

The applicant should provide information on the selection and fabrication of the materials in the plant's ESF systems, such as the reactor building and the RCCS. This information should include materials treated, as well as the treatment processes used, to enhance corrosion resistance, strength, and hardness, or other properties that would be applicable to the selection of materials for HTGR systems. The application should:

1. List the material specifications for all pressure-retaining ferritic materials, austenitic stainless steels, and nonferrous metals, including bolting and welding materials, in each component (e.g., vessels, piping, pumps, valves) that are part of the ESF systems. The applicant should identify the grade or type and final metallurgical conditions of the materials placed in service.

2. List the ESF construction materials that would be exposed to the HPB discharge in the event of a break in the HPB, or discharge of any other pressurized system. Describe test data and service experience to show that the construction materials used are compatible with any impurities that may be in the subject pressurized fluid.

3. For topics below that apply to HTGR systems, provide the following information to demonstrate that the integrity of safety-related components of the ESF systems is maintained during all stages of component manufacture and reactor construction.
    a. Provide sufficient details regarding the means used to avoid significant sensitization during fabrication and assembly of austenitic stainless steel components of the ESF systems. In so doing, demonstrate that the degree of freedom from sensitization is comparable to that obtainable by following the recommendations of RG 1.44 or functional equivalent that applies to HTGR ESF systems (NOTE: that this RG contains NRC staff positions related to unstabilized austenitic stainless steel of the AISI Type 3XX series. High temperature austenitic stainless steel applications must meet similar stabilization criteria for the specific temperature application). This RG describes acceptable criteria for preventing intergranular corrosion and IGSCC of stainless steel components of the ESF systems. The application should discuss the measures in place to prevent furnace-sensitized material from being used in the ESF systems, and how methods described in this guide are followed in testing the materials prior to fabrication to ensure that no deleterious sensitization occurs during welding.
    b. If stress corrosion cracking is a concern related to HTGR ESF systems, provide sufficient details on process controls used to limit the exposure of austenitic stainless steel ESF components to contaminants that are capable of causing stress-corrosion cracking. Show that the degree of surface cleanliness during all stages of component manufacture and reactor construction is comparable to that obtainable by following the recommendations of RGs 1.44 and 1.37. New methods beyond these RGs may need to be proposed, justified, and endorsed for use in certain HTGR high-temperature applications.
    c. If nonmetallic thermal insulation is used by any systems located in proximity to ESF, provide sufficient information on the selection, procurement, testing, storage, and installation of nonmetallic thermal insulation to demonstrate that the leachable concentrations of chloride, fluoride, sodium, and silicate are comparable to those recommended in RG 1.36 if stress corrosion cracking is a concern related to HTGR ESF systems.
    d. Provide sufficient information to show the degree of agreement that the fracture toughness properties of the ferritic materials with the guidelines of the ASME Code.
    e. Describe the controls imposed on abrasive work performed on austenitic stainless steel surfaces to minimize cold working of surfaces and introduction of contaminants that promote stress-corrosion cracking of the materials.

4. Provide sufficient information to determine that the corrosion allowances specified for ESF materials that are exposed to process fluids are supported by adequate technical bases, and that the specified corrosion allowances are adequate for the proposed design life of affected components and piping.

5. Provide sufficient information to show that the preheat temperatures for welding low alloy steel comply with RG 1.50; for welding carbon steel materials, the preheat temperatures should comply with Section III, Division 5 of the ASME Code.

6. Provide sufficient information to ensure that moisture control on low-hydrogen welding materials comply with the guidelines in Section III, Division 5 of the ASME Code, unless alternative procedures are justified.

7. Provide sufficient information to show that the methods for qualifying welders for making welds at locations where access is limited, and the methods for monitoring and certifying such welds, are in accordance with RG 1.71.

8. Provide sufficient information to show that the applicable guidance pertaining to material selection and fabrication provided in FSAR Chapters 5 and 10 is met (to be provided later).

Modified Section/Title: C.I.6.1.1.2; Composition and Compatibility of ESF Coolants
Original Section/Title: C.I.6.1.1.2; Composition and Compatibility of Core Cooling Coolants and Containment Sprays
The applicant should provide the following information described below regarding the composition and compatibility of the RCCS coolant and other processing fluids (i.e., fluids used during fabrication and cleaning), as they relate to the materials of the ESF systems. The applicant should provide the following information.

1. Provide information to verify the compatibility of materials used in manufacturing ESF components with the ESF fluids.

2. Provide information concerning the proposed approach to control the chemistry of the cooling water if water is used by the RCCS design.

Modified Section/Title: C.I.6.1.2; Organic Materials
Original Section/Title: C.I.6.1.2; Organic Materials
Section 6.1.2 addresses the concern that organic materials and coatings may fail (delaminate from the substrate) and become a debris source that could prevent decay heat removal systems from performing their safety function by blocking the containment sump suction. Given that HTGRs will not use these types of decay heat removal systems, this subsection has been deleted.

Modified Section/Title: C.I.6.2; Functional Containment
Original Section/Title: C.I.6.2; Containment Systems

Modified Section/Title: C.I.6.2.1; Functional Containment Design
Original Section/Title: C.I.6.2.1; Containment Functional Design
The description of the overall approach to the functional containment and the three fuel barriers is in Sections 1.3 and 4.8. The description of the helium pressure boundary barrier is in Chapter 5. This section should describe the reactor building analyses, considering shutdown conditions, when appropriate, to provide a basis for procedures, instrumentation, operator response, equipment interactions, and equipment response, if any, during normal operations and postulated accidents.

The applicant should describe how the reactor building functions within the functional containment to meet the intent of GDC 4 [PDC 4], GDC 16 [PDC 16], and GDC 50 [PDC 50] in Appendix A to 10 CFR Part 50 and 10 CFR 50.46 [as modified for HTGRs]. GDC 4 [PDC 4] provides the basic environmental and dynamic effects design requirements for all SSC commensurate with the importance of the safety functions to be performed. GDC 16 [PDC 16] establishes the fundamental requirement to design the functional containment that is a barrier against the release of radioactivity to the environment. For the HTGR, PDC 16 does not require that the reactor building be an essentially leaktight barrier since the other four barriers of the functional containment contribute to the requirement of retaining radionuclides. GDC 50 [PDC 50] requires that the reactor building structurally protects the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink to maintain the specified acceptable core radionuclide release design limits.

Modified Section/Title: C.I.6.2.1.1; Reactor Building

Original Section/Title: C.I.6.2.1.1; Containment Structure

1. Functions
   Provide a description of the safety functions of the reactor building during normal operations and Chapter 15 events.

2. Interfaces
   Provide a description of the key interfaces between the reactor building and safety-related and non-safety related SSCs.

3. Design Bases
   The applicant should discuss the design bases for the reactor building to withstand a spectrum of postulated accidents that includes helium pressure boundary breaks, water ingress events, external events that could impact vessel geometry and main steamline break accidents. In particular, this discussion should include the following information.
   a. Discuss the postulated accident conditions and the extent of simultaneous occurrences (accidents plus associated equipment failures) that determine the reactor building accident pressure (including both internal and external design pressure requirements). Applicants should credit only seismically qualified equipment for accident mitigation in reactor building safety analyses. The maximum calculated accident pressure and temperature should be stated. The applicant should describe the flow path from the HPB to the reactor building vent for the spectrum of events analyzed.
   b. Discuss the postulated accident conditions and the extent of simultaneous occurrences (accidents plus associated equipment failures) that determine the accident pressure and temperature requirements for the internal structures of reactor buildings. Applicants should credit only seismically qualified equipment for accident mitigation in reactor building safety analyses.
   c. Discuss the sources and amounts of mass and energy that might be released into the reactor building with reference to the design evaluations provided in FSAR Sections 6.2.1.3 and 6.2.1.4.
   d. Discuss the capability for energy removal from the reactor building under various scenarios assumed in the Chapter 15 analyses.
   e. Discuss the bases for the analysis of the reactor building conditions used in the RCCS performance studies and explain how this bases supports the assumptions used in the analysis of offsite radiological consequences of the accident.

4. Design Features
   In this section of the FSAR, the applicant should discuss the loads experienced in the reactor building, describe the design features of the reactor building and internal structures, and include appropriate general arrangement drawings. The applicant should provide the following information:
   a. Discuss the qualification tests proposed to demonstrate the functional capability of the SSCs in the reactor building that are assumed to perform safety functions and a discussion of the status of any incomplete developmental test programs together with a schedule for test program completion and subsequent submittal of supplemental application information, as necessary. (FSAR Section 1.5 should also identify any incomplete developmental test programs.)
   b. Discuss the design provisions to protect the integrity of the reactor building structure under external pressure-loading conditions. Specify the design values of the external design pressure of the reactor building and the lowest expected internal pressure.

5. Design Evaluation
   The applicant should provide evaluations of the functional capability of the reactor building design. The applicant should:
   a. Provide analyses of the reactor building pressure response to a spectrum of postulated events analyzed in Chapter 15; specify the break size and location of each postulated HPB break analyzed; describe the flow paths from the HPB to the reactor building vents for each event analyzed.
   b. Identify the reactor building computer codes used to determine the pressure and temperature response; discuss and justify the inherent conservatisms in the assumptions made in the analyses regarding initial reactor building conditions (e.g., pressure, temperature, free volume, humidity), reactor building heat removal (if applicable), and RCCS operability.
   c. Provide the results of a failure modes and effects analysis of the RCCS (or refer to Section 6.3).
   d. Provide analyses of the temperature and pressure response of the reactor building to postulated secondary-system pipe ruptures (e.g., steam and feedwater line breaks). The break size and location of each postulated break analyzed should be specified, the method of analysis described and the computer codes used (provide detailed mass and energy release analyses in Section 6.2.1.4 of the FSAR) described. The assumptions made regarding the operating conditions of the reactor, closure times of secondary-system isolation valves, and operation of other ESFs (if applicable) should be discussed and justified. The results of each accident analyzed in Chapter 15 should be tabulated.
   e. With respect to modeling any heat sinks for heat transfer calculations, the applicant should provide and justify the computer mesh spacing used for concrete, steel, and steel-lined concrete heat sinks. It should justify the steel-concrete interface resistance used for steel-lined concrete heat sinks, as well as the heat transfer correlations used in heat transfer calculations. The condensing heat transfer coefficient as a function of time for the most severe steam or feedwater line pipe breaks should be graphically illustrated.
   f. For the limiting HPB break, secondary line break and water ingress event; indicating the time of occurrence (in seconds after the break occurs) of events, such as the following:
      - any key RCCS operations that are time dependent
      - peak reactor building pressure during the blowdown phase
      - end of the blowdown phase
      - peak reactor building pressure subsequent to the end of the blowdown phase
      - end of steam generator energy release for secondary system breaks
   g. For the most severe HPB breaks, secondary line breaks and water ingress events provide energy inventories that show the distribution of energy prior to the accident, at the time of peak pressure, and at the end of the blowdown phase.
   h. Describe the model for determining the distribution of mass and energy from the postulated break in the reactor building atmosphere.

i. Provide a summary description of the instrumentation provided to monitor and record reactor building during the course of an accident within the reactor building with appropriate reference to Chapter 7 of the FSAR. The range, accuracy, and response of the instrumentation, as well as the tests conducted to qualify the instruments for use in the post-accident reactor building environment (or reference Chapter 7 of the FSAR) should be provided.

Modified Section/Title: C.I.6.2.1.2; Reactor Building Subcompartments
Original Section/Title: C.I.6.2.1.2; Containment Subcompartments

1. Design Bases
   The applicant should discuss the design bases for the reactor building subcompartments and include the following information:
   a. Synopsis of the how the subcompartments work following a HPB depressurization event or a steam generator line break.

2. Design Features
   The applicant should describe each subcompartment analyzed, and provide plan and elevation drawings showing component and equipment locations, routing of high-energy lines, and vent locations and configurations. The applicant should tabulate the subcompartment free volumes and vent areas and identify the vent areas that become available only after the occurrence of a postulated HPB depressurization (e.g., as a result of insulation collapsing or blowing out, blowout panels being blown out, or hinged doors swinging open), and describe the manner in which they are treated. The availability of these vent areas should be justified. Dynamic analyses of the available vent area as a function of time should be provided and supported with appropriate test data.

3. Design Evaluation
   The applicant should identify the computer program(s) used, and/or provide or reference a detailed description of the analytical model, for subcompartment pressure response analyses. It should provide the results of the analyses, and include the following information:
   a. Describe the computer program used to calculate the mass and energy releases from a postulated break. The applicant should discuss the conservatism of the blowdown model with respect to the pressure response of the subcompartment.
   b. Specify the assumed initial operating conditions of the plant, such as reactor power level and subcompartment pressure, temperature, and humidity.
   c. Identify the piping system within a subcompartment that is assumed to rupture, the location of the break within the subcompartment, and the break size. The inside diameter of the ruptured line, as well as the locations and sizes of any flow restrictions within the line that is postulated to fail should be provided.
   d. Provide a graph showing the pressure response within a subcompartment as a function of time to permit evaluation of the effects on structures and component supports.
   e. Provide mass and energy release data for the postulated pipe breaks in tabular form, with time in seconds, mass release rate in lbm/s, enthalpy of mass released in Btu/lbm, and energy release rate in Btu/s.
   f. Provide for all vent flowpaths, specification of the flow conditions (subsonic or sonic) up to the time of peak pressure.
   g. Describe, in detail, the method used to determine vent loss coefficients and a table showing the vent paths and loss coefficients for each subcompartment.

Modified Section/Title: C.I.6.2.1.3; Mass and Energy Release Analyses for Postulated Helium Pressure Boundary Breaks

Original Section/Title: NEW; NEW

The applicant should identify the computer codes used, and provide or reference detailed descriptions of the analytical models employed to calculate the mass and energy released following a postulated HPB breaks and water ingress events. It should discuss the analyses performed on various HPB break locations and a spectrum of pipe break sizes at each location to identify the limiting pipe break locations and sizes. The discussion should be divided into the accident phases in which different physical processes occur, as follows:

1. Blowdown phase (i.e., when the helium coolant is being rapidly injected into the reactor building)

2. Long-term cooling phase (i.e., when core decay heat and remaining stored energy in the primary and secondary systems are being added to the reactor building).

The following information should be included:

1. Mass and Energy Release Data
   Describe the mass and energy release data for the spectrum of limiting break sizes and locations during the period when there is significant energy release. Using the tabular form, the applicant should provide this information with time in seconds, mass release rate in lbm/s, and enthalpy of mass released in Btu/lbm.

2. Energy Sources
   Identify the sources of generated and stored energy in the HPB and secondary steam system considered in analyses of HPB breaks and water ingress events, and a description of the methods used and assumptions made in calculations of the energy available for release from these sources. The conservatism in the calculation of the available energy for each source should be addressed. The stored energy sources and the amounts of stored energy should be tabulated. For each source of generated energy, curves showing the energy release rate and integrated energy released should be provided.

3. Description of the Blowdown Model
   Describe the procedure used to calculate the mass and energy released from the HPB during the blowdown phase of a accident (or reference as appropriate). All significant equations and correlations used in the analysis should be included. Conservatism in the mass and energy release calculations from the standpoint of predicting the highest reactor building pressure response, and justify any assumptions should be discussed. For example, the calculations used to determine the energy transferred to the helium coolant from heated surfaces, as well as the release of helium coolant to the reactor building during blowdown should be described. In addition, the heat transfer correlations used, and justify their application should be provided and justified.

4. N/A

5. Description of the Long-Term Cooling Model
   Describe the calculations used to determine the mass and energy released to the reactor building during the long-term cooling phase of the accident (or reference as appropriate) including (or referencing) all significant equations and correlations used in the analysis. The conservatism in the mass and energy release calculations, from the standpoint of predicting the highest reactor building pressure response should be discussed and justified. For example, discuss and justify the methods

used to calculate (a) core inlet and exit flow rates and (b) removal of all sensible heat from primary system metal surfaces and the steam generators. The heat transfer correlations used should be described, and their application justified.

6. Failure Mode and Effects Analysis
   Provide a failure modes and effects analysis of the RCCS to determine the failures that maximizes the energy release to the reactor building following a HPB break or water ingress event. Analyses for each postulated break location should be provided.

7. N/A

8. N/A

9. Additional Information Required for Confirmatory Analysis
   To enable confirmatory analyses to be performed, provide a tabulation of the elevations, flow areas, and friction coefficients within the primary system that are used for the reactor building analyses. Representative values with justification for empirical correlations that are significant to the analysis should be provided.

Modified Section/Title: C.I.6.2.1.4; Mass and Energy Release Analysis for Postulated Secondary-System Pipe Ruptures Inside the Reactor Building

Original Section/Title: C.I.6.2.1.4; Mass and Energy Release Analysis for Postulated Secondary-System Pipe Ruptures Inside Containment (PWR)

The applicant should identify the computer code used, and provide (or reference) a detailed description of the analytical model used to calculate the mass and energy released following a secondary-system steam and feedwater line break inside the reactor building. A spectrum of break sizes and various reactor operating conditions should be analyses to ensure that the most severe secondary-system pipe rupture has been identified. The following information should be included:

1. Mass and Energy Release Data
   Provide mass and energy release data for the most severe secondary-system pipe rupture with regard to break size and location and operating power level of the reactor, in tabular form with time in seconds, mass flow rate in lbm/s, and corresponding enthalpy in Btu/lbm. Separate tables for the mass and energy released from each side of a double-ended break should be provided.

2. Failure Modes and Effects Analysis
   Provide a failure modes and effects analysis to determine the most severe break location, for the purpose of maximizing the mass and energy released to the reactor building and the reactor building pressure response. This analysis should consider, for example, the failure of a steam or feedwater line isolation valve, or the feedwater pump to trip.

3. Initial Conditions
   Describe the analysis, including assumptions, to determine the fluid mass available for release into the reactor building. In general, the analysis should be performed in a manner that is conservative from a reactor building response standpoint (i.e., maximizes the fluid mass available for release).

4. Description of Blowdown Model
   Identify the computer code used should be identified, and the procedure used for calculations including all significant equations (or reference the appropriate report) should be described. Calculations of the energy transferred from the primary system to the secondary system, stored

energy removed from the secondary system metal, break flow, and steam-water separation should be conservative for reactor building analysis. This conservatism should be discussed and justified. The correlations used to calculate the heat transferred from the steam generator tubes and shell should be provided and justified.

5. Energy Inventories
   Provide for the most severe secondary-system pipe rupture, the inventories of the energy transferred from the primary (if applicable) and secondary systems to the reactor building.

6. Additional Information Required for Confirmatory Analyses
   To permit confirmatory analyses to be performed, provide a tabulation of the elevations, flow areas, and friction coefficients within the secondary system, as well as the feedwater flow rate as a function of time. Representative values with justification for empirical correlations (such as those used to predict heat transfer and liquid entrainment) that are significant to the analysis should be provided.

Modified Section/Title: C.I.6.2.1.5; N/A

Original Section/Title: C.I.6.2.1.5; Minimum Containment Pressure Analysis for Performance Capability Studies of the Emergency Core Cooling System (PWR)
N/A

Modified Section/Title: C.I.6.2.1.6; Testing and Inspection - Reactor Building

Original Section/Title: C.I.6.2.1.6; Testing and Inspection
The applicant should provide information on the reactor building testing and inspection with regard to preoperational testing and periodic inservice surveillance to ensure the functional capability of the reactor building and associated SSCs. The applicant should emphasize those tests and inspections that are considered essential to determine that performance objectives have been achieved, and performance capability is maintained above pre-established limits throughout the plant's lifetime. The applicant should include information on the following:

1. Planned tests and inspections, including the need and purpose of each test and inspection

2. Selected frequency for performing each test and inspection, including justification

3. The manner in which tests and inspections are conducted

4. Requirements and bases for acceptability

5. Action to be taken in the event that acceptability requirements are not met.

The applicant should emphasize those surveillance-type tests that are of such importance to safety that they may become part of the TS of an operating license, and discuss the bases for such surveillance requirements.

Modified Section/Title: C.I.6.2.1.7; Instrumentation Requirements - Reactor Building

Original Section/Title: C.I.6.2.1.7; Instrumentation Requirements
The applicant should discuss the instrumentation proposed to be installed to monitor conditions inside the reactor building and to actuate safety functions when abnormal conditions are sensed. The appropriate FSAR section of the application that discusses the design details and logic of the instrumentation should be referenced.

Modified Section/Title: C.I.6.2.2; N/A
Original Section/Title: C.I.6.2.2; Containment Heat Removal Systems
N/A

Modified Section/Title: C.I.6.2.2.1; N/A
Original Section/Title: C.I.6.2.2.1; Design Bases
N/A

Modified Section/Title: C.I.6.2.2.2; N/A
Original Section/Title: C.I.6.2.2.2; System Design
N/A

Modified Section/Title: C.I.6.2.2.3; N/A
Original Section/Title: C.I.6.2.2.3; Design Evaluation
N/A

Modified Section/Title: C.I.6.2.2.4; N/A
Original Section/Title: C.I.6.2.2.4; Tests and Inspections
N/A

Modified Section/Title: C.I.6.2.2.5; N/A
Original Section/Title: C.I.6.2.2.5; Instrumentation Requirements
N/A

Modified Section/Title: C.I.6.2.3; N/A
Original Section/Title: C.I.6.2.3; Secondary Containment Functional Design
N/A

Modified Section/Title: C.I.6.2.3.1; N/A
Original Section/Title: C.I.6.2.3.1; Design Bases
N/A

Modified Section/Title: C.I.6.2.3.2; N/A
Original Section/Title: C.I.6.2.3.2; System Design
N/A

Modified Section/Title: C.I.6.2.3.3; N/A
Original Section/Title: C.I.6.2.3.3; Design Evaluation
N/A

Modified Section/Title: C.I.6.2.3.4; N/A
Original Section/Title: C.I.6.2.3.4; Tests and Inspections
N/A

Modified Section/Title: C.I.6.2.3.5; N/A
Original Section/Title: C.I.6.2.3.5; Instrumentation Requirements
N/A

Modified Section/Title: C.I.6.2.4; Reactor Building Venting
Original Section/Title: C.I.6.2.4; Containment Isolation System

This section of the FSAR should provide the design and functional capability of the reactor building vent system.

Modified Section/Title: C.I.6.2.4.1; Design Bases

Original Section/Title: C.I.6.2.4.1; Design Bases
The applicant should discuss the design bases for the reactor building venting system, including the following:

1. Governing conditions under which reactor building venting is necessary

2. Design requirements for reactor building vents.

Modified Section/Title: C.I.6.2.4.2; System Design

Original Section/Title: C.I.6.2.4.2; System Design
The applicant should provide a table of design information regarding the reactor building venting system. This table should include the following information:

1. Reference to a figure in the application showing arrangement of reactor building vents

2. Vent equipment tag number

3. Location of valve

4. Leakage criteria

5. Type of operator

6. Primary mode of vent actuation

7. Normal vent position

8. Shutdown vent position

9. Postaccident vent position

10. Power failure vent position

11. Automatic closure signals

12. Closure time

13. Power source.

The applicant should specify the signals that initiate closure of the reactor building vents, or refer to the FSAR section of the application that provides this information. The applicant should discuss the bases for the reactor building vent closure times.

The applicant should discuss the design requirements for reactor building vents, including the following:

1. The quality standards and seismic design classification

2. Assurance of protection against loss of function from missiles, jet forces, pipe whip, and earthquakes and the provisions taken to ensure that closure of the vent is not prevented by debris that could become entrained in the escaping gas

3. Assurance of the operability of valves and valve operators in the reactor building atmosphere under normal plant operating conditions and postulated accident conditions

4. Mechanical and electrical redundancy to preclude common-mode failures.

The applicant should discuss the design provisions to test the operability and the leakage rate of the vents. The applicant should describe the environmental qualification tests that have been (or will be) performed on mechanical and electrical components that may be exposed to the accident environment inside the reactor building. The applicant should show the expected environmental conditions as functions of time, or refer to the section of the FSAR where this information can be found. The applicant should identify the codes, standards, and regulatory guides applied in the design of the system and its components.

Modified Section/Title: C.I.6.2.4.3; Design Evaluation

Original Section/Title: C.I.6.2.4.3; Design Evaluation
The applicant should provide an evaluation of the functional capability of the reactor building vent system, in conjunction with a failure modes and effects analysis of the system.

Modified Section/Title: C.I.6.2.4.4; Tests and Inspections

Original Section/Title: C.I.6.2.4.4; Tests and Inspections
The applicant should describe the program for initial functional testing and subsequent periodic operability testing of the reactor building vents system. The applicant should discuss the scope and limitations of the tests and describe the inspection program for the isolation system and its components.

Modified Section/Title: C.I.6.2.5; N/A

Original Section/Title: C.I.6.2.5; Combustible Gas Control in Containment
N/A

Modified Section/Title: C.I.6.2.5.1; N/A

Original Section/Title: C.I.6.2.5.1; Design Bases
N/A

Modified Section/Title: C.I.6.2.5.2; N/A

Original Section/Title: C.I.6.2.5.2; System Design
N/A

Modified Section/Title: C.I.6.2.5.3; N/A

Original Section/Title: C.I.6.2.5.3; Design Evaluation
N/A

Modified Section/Title: C.I.6.2.5.4; N/A

Original Section/Title: C.I.6.2.5.4; Tests and Inspections
N/A

Modified Section/Title: C.I.6.2.5.5; N/A
Original Section/Title: C.I.6.2.5.5; Instrumentation Requirements
N/A

Modified Section/Title: C.I.6.2.6; N/A
Original Section/Title: C.I.6.2.6; Containment Leakage Testing
N/A

Modified Section/Title: C.I.6.2.6.1; N/A
Original Section/Title: C.I.6.2.6.1; Containment Integrated Leakage Rate Test
N/A

Modified Section/Title: C.I.6.2.6.2; N/A
Original Section/Title: C.I.6.2.6.2; Containment Penetration Leakage Rate Test
N/A

Modified Section/Title: C.I.6.2.6.3; N/A
Original Section/Title: C.I.6.2.6.3; Containment Isolation Valve Leakage Rate Test
N/A

Modified Section/Title: C.I.6.2.6.4; N/A
Original Section/Title: C.I.6.2.6.4; Scheduling and Reporting of Periodic Tests
N/A

Modified Section/Title: C.I.6.2.6.5; N/A
Original Section/Title: C.I.6.2.6.5; Special Testing Requirements
N/A

Modified Section/Title: C.I.6.2.7; N/A
Original Section/Title: C.I.6.2.7; Fracture Prevention of Containment Pressure Vessel
N/A

Modified Section/Title: C.I.6.3; Reactor Cavity Cooling System
Original Section/Title: C.I.6.3; Emergency Core Cooling System

Modified Section/Title: C.I.6.3.1; Design Bases
Original Section/Title: C.I.6.3.1; Design Bases
The applicant should provide a summary description of the RCCS, and identify all major subsystems of the RCCS, such as air or water cooled systems that are necessary for the RCCS to perform its heat removal function. Applicable reference(s) to nuclear plants or designs that employ the same RCCS design and are operating or have been licensed or certified should be provided. Describe the purpose of the RCCS, including its heat removal function, and identify each accident or transient for which the required protection includes the operation of the RCCS.

The applicant should describe how the RCCS design complies with relevant rules, regulations, and regulatory requirements, including the following:

GDC 2, “Design bases for protection against natural phenomena” **[PDC 2]**
GDC 4, “Environmental and dynamic effects design bases” **[PDC 4]**
GDC 5, “Sharing of structures, systems, and components” **[PDC 5]**

GDC 17, “Electric power systems” **[PDC 17 - In Work]**
GDC 34, "Residual Heat Removal" **[PDC 34]**
GDC 36, "Inspection of **[Passive Residual Heat Removal]** System", as applicable to the RCCS for rejecting heat the Ultimate Heat Sink **[PDC 36]**
GDC 37, "Testing of **[Passive Residual Heat Removal]** System", as applicable to the RCCS for rejecting heat to the Ultimate Heat Sink **[PDC 37]**
GDC 50, "**[Reactor Building]** Design Basis", as it relates to maintaining the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink to maintain the specified acceptable core radionuclide release limits.

The applicant should describe how the RCCS design and analysis incorporate the resolutions of the relevant USIs, and medium and high priority GSIs that are determined to be applicable to HTGRs and are specified in the version of NUREG 0933, that is current 6 months before the application submittal date.

The applicant should describe how the RCCS design incorporates operating experience insights from generic letters and bulletins issued up to 6 months before the application submittal date, to the extent that they are applicable to the HTGR design. It is the COL applicant’s responsibility to identify all relevant items applicable to their reactor designs.

The applicant should specify the design bases for selecting the functional requirements, such as core residual heat removal, maintaining reactor vessel, vessel support, and reactor building integrity, mitigating radionuclide releases, achieving and maintaining the plant in a safe stable state, with long term cooling. The applicant should discuss the bases for selecting such system parameters as operating pressure, operating temperature, flow rate, and hydraulic flow resistance of RCCS channels.

The applicant should specify the design bases concerned with reliability requirements. It should describe the protection against postulated equipment failures in terms of physical arrangement and layout, selection and redundancy of various system components, and redundancy of instrumentation. It should also describe how RCCS operation is protected against postulated equipment failures and flooding.

The applicant should specify the requirements that have been established to protect the RCCS from physical damage. This discussion should include design bases for RCCS support structure design, pipe whip protection, missile protection, and protection against such accident loads as helium depressurization or seismic loads.

The applicant should specify the environmental design bases during RCCS operation.

Modified Section/Title: C.I.6.3.2; System Design

Original Section/Title: C.I.6.3.2; System Design

Modified Section/Title: C.I.6.3.2.1; Schematic Piping and Instrumentation Diagrams

Original Section/Title: C.I.6.3.2.1; Schematic Piping and Instrumentation Diagrams
The applicant should provide piping and instrumentation diagrams showing the location of all components and cooling panels, piping and flow channels, flow channel inlet and discharge points, points where connecting systems and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation for all modes of RCCS operation, along with a complete description of component interlocks.

Modified Section/Title: C.I.6.3.2.2; Equipment and Component Descriptions

Original Section/Title: C.I.6.3.2.2; Equipment and Component Descriptions

The applicant should describe each component of the system, and identify its significant design parameters. The applicant should state the design and operating pressure and temperature of components for various portions of the system, and explain the bases for their selection. The applicant should state the available quantity of coolant (if applicable) for natural convection flow, including water storage capacity for water cooled systems. The applicant should describe the overall RCCS performance factors for each function (e.g., heat removal mechanism for thermal radiation and natural convection flow). The applicant should provide elevations of tanks in the passive water cooled systems, with reference to RCCS cooling loop elevation. Cooling panels and heat exchanger characteristics, including design flow rates, inlet and outlet temperatures for the cooling fluid and the fluid being cooled, the overall heat transfer coefficient, and the heat transfer area should be described. The applicant should describe the RCCS flow paths, including the intake and exhaust structure(s).

The applicant should state the relief valve capacity and settings or venting provisions included in the system, if provided. Specify design requirements for RCCS delivery lag times, if the concept of lag times from event initiation until RCCS cooling is provided is an applicable concept.

Modified Section/Title: C.I.6.3.2.3; Applicable Codes and Classifications

Original Section/Title: C.I.6.3.2.3; Applicable Codes and Classifications

The applicant should identify the applicable industry codes and classifications for the design of the system.

Modified Section/Title: C.I.6.3.2.4; Material Specifications and Compatibility

Original Section/Title: C.I.6.3.2.4; Material Specifications and Compatibility

The applicant should identify the material specifications for the RCCS, and discuss material compatibility and chemical effects of all expected conditions. The applicant should list the materials used in or on the RCCS by their commercial names, quantities (estimate where necessary), and chemical composition and show that the radiolytic or pyrolytic decomposition products, if any, of each material will not interfere with the safe operation of this or any other ESF.

Modified Section/Title: C.I.6.3.2.5; System Reliability

Original Section/Title: C.I.6.3.2.5; System Reliability

The applicant should discuss the reliability considerations incorporated in the design to ensure that the system provides the required cooling functions (e.g., redundancy and separation of components). The applicant should provide a failure modes and effects analysis of the RCCS, identifying the functional consequences of each postulated licensing basis event, including the effects of any postulated equipment or component failure, or operator error that can adversely affect the RCCS. The applicant should discuss how all potential passive and active failures of systems and components were considered for long term cooling. (Refer to NGNP Project white papers for additional guidance on event selection, and system and component failure application).

The applicant should describe how the design considered the adverse impact of air, gas or water accumulation in the RCCS flow channels on the RCCS operability, including water hammer effects, if applicable.

For a passive safety system design that relies exclusively on natural forces to perform design basis safety functions, and includes active systems to provide defense in depth capabilities for reactor cavity cooling and residual heat removal, the applicant should describe how the passive system reliability and the impact of adverse system interactions on the safety functions were considered. The applicant should describe how the regulatory oversight of the active nonsafety systems was considered in using the process of "non-safety related with special treatment" described in the NGNP SSC Classification white paper, if that classification is applied.

Modified Section/Title: C.I.6.3.2.6; Protection Provisions

Original Section/Title: C.I.6.3.2.6; Protection Provisions

The applicant should describe the provisions to protect the system (including connections other systems) against damage that might result from movement (between components within the system and connecting systems), from missiles, thermal stresses, or other causes (e.g., helium depressurization with loss of forced cooling, seismic events).

Modified Section/Title: C.I.6.3.2.7; Provisions for Performance Testing and Inspection

Original Section/Title: C.I.6.3.2.7; Provisions for Performance Testing and Inspection

The applicant should describe the provisions to facilitate performance testing and inspection of components.

Modified Section/Title: C.I.6.3.2.8; Manual Actions

Original Section/Title: C.I.6.3.2.8; Manual Actions

The applicant should identify all manual actions that an operator is required to take in order for the RCCS to operate properly. The applicant should identify all process instrumentation available to the operator in the control room to assist in assessing post-accident conditions. The applicant should discuss the information available to the operator, the time delay during which the operator's failure to act properly has no unsafe consequences, and the consequences if the operator fails to perform the action at all.

Modified Section/Title: C.I.6.3.3; Performance Evaluation

Original Section/Title: C.I.6.3.3; Performance Evaluation

The applicant should discuss the RCCS performance through the safety analyses of a spectrum of postulated design basis events and design basis accidents. These analyses should be included in FSAR Chapter 15. In this section of the FSAR, the applicant should list the accidents discussed in Chapter 15 that rely on RCCS operation. The applicant should summarize the conclusions of the accident analyses. The applicant should provide the bases for any operational restrictions, such as minimum functional capacity or testing requirements that might be appropriate for inclusion in the TS of the license. The applicant should indicate all existing criteria that are used to judge the adequacy of RCCS performance, including those contained in the (**HTGR variant of 10 CFR 50.46 [to be developed]**). RCCS cooling performance evaluation should include an evaluation of postulated equipment and component failures (consistent with the NGNP white paper regarding failure criterion).

The applicant should provide simplified functional flow diagrams showing the alignment of valves, flow paths in the system, and the capacity of RCCS heat removal for typical accident conditions (e.g., helium depressurization with loss of forced cooling). The applicant should provide typical heat removal rates as a function of time for the various accidents, and discuss the time sequence of RCCS operation for short and long term cooling. Analysis supporting any lag times (e.g., the period between the time an accident has occurred and the time RCCS is discharged) should be included, and if credit is taken for operator action that sequence should be indicated.

The applicant should discuss the extent to which components or portions of the RCCS are required for operation of other systems, and the extent to which components or portions of other systems are required for operation of the RCCS. In the analysis of how these dependent systems would function, the applicant should include system priority (which system takes preference) and conditions under which various

components or portions of one system function as part of another system. Any limitations on operation or maintenance included to ensure minimum capability should be delineated.
The applicant should state the bounds within which principal system parameters need to be maintained in the interest of constant readiness to provide the heat removal function(s) assumed in the accident analysis (e.g., maximum number of inoperable components, maximum allowable time period for which a component can be out of service). The failure modes and effects analysis provided in FSAR Section 6.3.2.5 identifies possible degraded RCCS performances caused by postulated failures. The accident analyses provided in Chapter 15 of the FSAR consider each of the degraded RCCS cases in the selection of the most significant events to be analyzed. The applicant should discuss the conclusions of the various accident analyses to show that the RCCS is adequate to perform its intended function.

Modified Section/Title: C.I.6.3.4; Tests and Inspections

Original Section/Title: C.I.6.3.4; Tests and Inspections

Modified Section/Title: C.I.6.3.4.1; RCCS Performance Tests

Original Section/Title: C.I.6.3.4.1; ECCS Performance Tests
The applicant should provide a description, or reference the description of the preoperational test program performed for the RCCS. The program should provide for testing each train of the RCCS under both ambient and simulated hot operating conditions. The tests should demonstrate that the heat removal capacity provided by each RCCS flowpath is within the design specifications. Any exceptions taken during the performance test should be justified.

Modified Section/Title: C.I.6.3.4.2; Reliability Tests and Inspections

Original Section/Title: C.I.6.3.4.2; Reliability Tests and Inspections
Although the RCCS is a system that is normally operating, the applicant should identify the periodic test and inspection program for verifying the performance and operability requirements of the system (e.g. normal operation, AOOS, postulated accident conditions, post accident operation), and explain the reasons why the planned program is believed to be appropriate.

This discussion should include the following information:

1. Description of planned tests

2. Considerations that led to periodic testing and the selected test frequency

3. Test methods to be used

4. Requirements and bases for acceptability of observed performance

5. Description of the program for ISI, including items to be inspected, accessibility requirements, and the types and frequency of inspection.

The applicant should provide a cross reference if information about planned tests is available anywhere else in the application; repetition is not necessary.

The applicant should emphasize those surveillance type tests that are of such importance to safety that they may become part of the TS of an operating license. The applicant should provide the bases for such surveillance requirements as part of the application.

Modified Section/Title: C.I.6.3.5; Instrumentation Requirements
Original Section/Title: C.I.6.3.5; Instrumentation Requirements
The applicant should discuss the instrumentation provisions for RCCS (equipment designed to IEEE Std 603 standards, if applicable). The discussion of design details and logic of the instrumentation provided in Chapter 7 of the FSAR should be referenced.

Modified Section/Title: C.I.6.4; Habitability Systems (Presumed N/A)
Original Section/Title: C.I.6.4; Habitability Systems
(NOTE: All portions of section C.I.6.4 are currently presumed to be not applicable to HTGR technology and cannot be verified until a final design decision is made. No updates have been incorporated in the writers guide)

Modified Section/Title: C.I.6.4.1; N/A
Original Section/Title: C.I.6.4.1; Design Basis

Modified Section/Title: C.I.6.4.2; N/A
Original Section/Title: C.I.6.4.2; System Design

Modified Section/Title: C.I.6.4.2.1; N/A
Original Section/Title: C.I.6.4.2.1; Definition of Control Room Envelope

Modified Section/Title: C.I.6.4.2.2; N/A
Original Section/Title: C.I.6.4.2.2; Ventilation System Design

Modified Section/Title: C.I.6.4.2.3; N/A
Original Section/Title: C.I.6.4.2.3; Leaktightness

Modified Section/Title: C.I.6.4.2.4; N/A
Original Section/Title: C.I.6.4.2.4; Interaction with Other Zones and Pressure-Containing Equipment

Modified Section/Title: C.I.6.4.2.5; N/A
Original Section/Title: C.I.6.4.2.5; Shielding Design

Modified Section/Title: C.I.6.4.3; N/A
Original Section/Title: C.I.6.4.3; System Operational Procedures

Modified Section/Title: C.I.6.4.4; N/A
Original Section/Title: C.I.6.4.4; Design Evaluations

Modified Section/Title: C.I.6.4.4.1; N/A
Original Section/Title: C.I.6.4.4.1; Radiological Protection

Modified Section/Title: C.I.6.4.4.2; N/A
Original Section/Title: C.I.6.4.4.2; Toxic Gas Protection

Modified Section/Title: C.I.6.4.5; N/A
Original Section/Title: C.I.6.4.5; Testing and Inspection

Modified Section/Title: C.I.6.4.6; N/A
Original Section/Title: C.I.6.4.6; Instrumentation Requirement

Modified Section/Title: C.I.6.5; Fission Product Removal and Control Systems (Presumed N/A)
Original Section/Title: C.I.6.5; Fission Product Removal and Control Systems
(NOTE: All portions of section C.I.6.5 are currently presumed to be not applicable to HTGR technology and cannot be verified until a final design decision is made. No updates have been incorporated in the writer's guide)

Modified Section/Title: C.I.6.5.1; N/A
Original Section/Title: C.I.6.5.1; ESF Filter Systems

Modified Section/Title: C.I.6.5.1.1; N/A
Original Section/Title: C.I.6.5.1.1; Design Bases

Modified Section/Title: C.I.6.5.1.2; N/A
Original Section/Title: C.I.6.5.1.2; System Design

Modified Section/Title: C.I.6.5.1.3; N/A
Original Section/Title: C.I.6.5.1.3; Design Evaluation

Modified Section/Title: C.I.6.5.1.4; N/A
Original Section/Title: C.I.6.5.1.4; Tests and Inspections

Modified Section/Title: C.I.6.5.1.5; N/A
Original Section/Title: C.I.6.5.1.5; Instrumentation Requirements

Modified Section/Title: C.I.6.5.1.6; N/A
Original Section/Title: C.I.6.5.1.6; Materials

Modified Section/Title: C.I.6.5.2; N/A
Original Section/Title: C.I.6.5.2; Containment Spray Systems

Modified Section/Title: C.I.6.5.2.1; N/A
Original Section/Title: C.I.6.5.2.1; Design Bases

Modified Section/Title: C.I.6.5.2.2; N/A
Original Section/Title: C.I.6.5.2.2; System Design (for Fission Product Removal)

Modified Section/Title: C.I.6.5.2.3; N/A
Original Section/Title: C.I.6.5.2.3; Design Evaluation

Modified Section/Title: C.I.6.5.2.4; N/A
Original Section/Title: C.I.6.5.2.4; Tests and Inspections

Modified Section/Title: C.I.6.5.2.5; N/A
Original Section/Title: C.I.6.5.2.5; Instrumentation Requirements

Modified Section/Title: C.I.6.5.2.6; N/A
Original Section/Title: C.I.6.5.2.6; Materials

Modified Section/Title: C.I.6.5.3; N/A
Original Section/Title: C.I.6.5.3; Fission Product Control Systems

Modified Section/Title: C.I.6.5.3.1; N/A
Original Section/Title: C.I.6.5.3.1; Primary Containment

Modified Section/Title: C.I.6.5.3.2; N/A
Original Section/Title: C.I.6.5.3.2; Secondary Containments

Modified Section/Title: C.I.6.5.4; N/A
Original Section/Title: C.I.6.5.4; Ice Condenser as a Fission Product Cleanup System

Modified Section/Title: C.I.6.5.5; N/A
Original Section/Title: C.I.6.5.5; Pressure Suppression Pool as a Fission Product Cleanup System

Modified Section/Title: C.I.6.5.5.1; N/A
Original Section/Title: C.I.6.5.5.1; Design Bases

Modified Section/Title: C.I.6.5.5.2; N/A
Original Section/Title: C.I.6.5.5.2; System Design (for the Fission Product Removal)

Modified Section/Title: C.I.6.5.5.4; N/A
Original Section/Title: C.I.6.5.5.4; Tests and Inspections

Modified Section/Title: C.I.6.6; Inservice Inspection of Class B Components (possibly N/A - new ASME Div 2, Section XI)
Original Section/Title: C.I.6.6; Inservice Inspection of Class 2 and 3 Components
The applicant should discuss the ISI program for Class B components in accordance with ASME XI, Division 2.

Modified Section/Title: C.I.6.6.1; Components Subject to Examination
Original Section/Title: C.I.6.6.1; Components Subject to Examination
The applicant should indicate whether Class A components not forming part of the helium pressure boundary and Class B components, including those included in Section XI, Division 2, of the ASME Code are examined in accordance with ASME Code guidelines.

Modified Section/Title: C.I.6.6.2; Accessibility
Original Section/Title: C.I.6.6.2; Accessibility
The applicant should indicate whether the design and arrangement of Class B components provide adequate clearances to conduct the examinations at the ASME Code-defined inspection interval. The applicant should describe any special design arrangements for those components that are to be examined during normal reactor operation.

Modified Section/Title: C.I.6.6.3; Examination Techniques and Procedures
Original Section/Title: C.I.6.6.3; Examination Techniques and Procedures
The applicant should indicate the extent the examination techniques and procedures described in Section XI, Division 2 of the ASME Code are used. The applicant should describe any special examination techniques and procedures that might be used to meet the ASME Code guidelines.

Modified Section/Title: C.I.6.6.4; Inspection Intervals
Original Section/Title: C.I.6.6.4; Inspection Intervals
The applicant should indicate whether an inspection schedule for Class B components is in accordance with the guidance in Section XI, Division 2, of the ASME Code.

Modified Section/Title: C.I.6.6.5; Examination Categories and Requirements
Original Section/Title: C.I.6.6.5; Examination Categories and Requirements
The applicant should indicate whether the ISI categories and guidelines for Class B components are in agreement with Section XI, Division 2, Appendix V, of the ASME Code.

Modified Section/Title: C.I.6.6.6; Evaluation of Examination Results
Original Section/Title: C.I.6.6.6; Evaluation of Examination Results
The applicant should indicate whether the evaluation of Class B component examination results are in agreement with the guidelines of Section XI, Division 2, of the ASME Code. In addition, the applicant should indicate whether repair procedures for Class B components are in agreement with the guidelines of Section XI, Division 2 of the ASME Code.

Modified Section/Title: C.I.6.6.7; System Pressure Tests
Original Section/Title: C.I.6.6.7; System Pressure Tests
The applicant should indicate whether the program for Class B component pressure testing is in agreement with Section XI, Division 2, of the ASME Code.

Modified Section/Title: C.I.6.6.8; Augmented ISI to Protect against Postulated Piping Failures
Original Section/Title: C.I.6.6.8; Augmented ISI to Protect against Postulated Piping Failures
Not applicable.

Modified Section/Title: C.I.6.6.9; Code Interpretations and Exemptions
Original Section/Title: NEW; NEW
The applicant should identify all interpretations and exemptions from Code examination requirements.

Modified Section/Title: C.I.6.6.10; Relief Requests
Original Section/Title: NEW; NEW
The applicant should identify each provision of the Code for which relief is requested.

Modified Section/Title: C.I.6.6.11; Code Cases
Original Section/Title: NEW; NEW
The applicant should identify exemptions from Code requirements to be invoked that are permitted by approved Code cases.

Modified Section/Title: C.I.6.6.12; Operational Programs
Original Section/Title: NEW; NEW
The applicant should describe the operational program and proposed implementation milestones for the Preservice Inspection and Inservice Inspection and testing programs for Class B components.

Modified Section/Title: C.I.6.7; Feedwater and Main Steam Isolation

Original Section/Title: C.I.6.7; Main Steamline Isolation Valve Leakage Control System (BWRs)

This section applies to modular HTGRs that employ a primary coolant loop steam generator. The function of the feedwater and main steam isolation system is to isolate and limit the ingress of water into the lower steam pressure primary coolant following a steam generator tube leak or rupture. The applicant should describe the design bases and criteria to be applied to this system (if present), the modes of operation, and describe how design criteria are demonstrated to be met.

Modified Section/Title: C.I.6.7.1; Design Bases

Original Section/Title: C.I.6.7.1; Design Bases

The applicant should provide the design bases for all systems and components that support the essential safety function of isolating the steam generator feedwater and main steam and assuring proper operation at the intended time. The design basis should address the following considerations:

1. Safety-related function of the system.

2. System functional performance requirements, including the ability to function following an AOO and postulated accident.

3. Seismic and quality group classification of the system.

4. Requirements for protection from missiles, pipe whip, and jet forces, as well as the system's ability to withstand adverse environments associated with flow induced and seismic excitations.

5. System capability to provide sufficient capability and reliability to perform its safety function.

6. Requirements for system initiation and actuation consistent with the requirements for instrumentation, controls, and interlocks provided for engineered safety systems.

7. Requirements for inspection and testing during and subsequent to power operations.

Modified Section/Title: C.I.6.7.2; System Description

Original Section/Title: C.I.6.7.2; System Description

The applicant should provide a detailed description of all systems and components that support the isolation function of steam generator feedwater and main steam lines. Include interconnected system piping and instrumentation diagrams, system drawings, and locations of necessary support components in the station complex as well as the individual HTGR module. Indicate the physical division between the safety-related and nonessential portions of the systems. The description and drawings should include subsystems, system operation (function), system interactions, components utilized, connection points, and the instrumentation and controls that are utilized to assure proper system function. The applicant should identify inspection and testing requirements for the steam generator feedwater and main steam isolation system and describe provisions to be used that accomplish such inspections and testing.

Modified Section/Title: C.I.6.7.3; System Evaluation

Original Section/Title: C.I.6.7.3; System Evaluation

The applicant should provide an evaluation of the capability of the steam generator feedwater and main steam isolation system to function as intended during and following a postulated event. This evaluation should consider:

1. Ability of the system to maintain its safety function when subjected to missiles, pipe whip, jet forces, adverse environmental conditions, and loss of offsite power coincident with the postulated event,

2. Protection afforded the system from the effects of failure of any nonseismic Category I system or component,

3. Capability of the system to provide effective isolation of components and nonessential systems or equipment,

4. Failure modes and effects analysis to demonstrate that appropriate safety-grade instrumentation, controls, and interlocks will provide safe operating conditions, ensure system actuation under designed conditions, and preclude inadvertent system actuation.

5. Assurance that a system malfunction or inadvertent operation has no adverse effect on other safety-related systems, components, or functions.

Modified Section/Title: C.I.6.7.4; Instrumentation Requirements

Original Section/Title: C.I.6.7.4; Instrumentation Requirements
The applicant should describe the steam generator feedwater and main steam isolation system instrumentation and controls. The description must address detection and signaling of need for initiation of protective actions, verification of protective action performance, and demonstrate adequacy to meet postulated accident conditions.

Modified Section/Title: C.I.6.7.5; N/A

Original Section/Title: C.I.6.7.5; Inspection and Testing
(NOTE: Inspection and testing moved to C.I.6.7.2)

Appendix G

Chapter 9. Auxiliary Systems

Appendix G
Chapter 9. Auxiliary Systems

Modified Section/Title: C.I.9; Auxiliary Systems
Original Section/Title: C.I.9; Auxiliary Systems
Chapter 9 of the FSAR should provide information about the facility's auxiliary systems. In particular, this chapter should identify systems that are essential for safe shutdown of the plant or for protection of the health and safety of the public. For each system, the description should provide the design bases for the system and its critical components, a safety evaluation demonstrating how the system satisfies the design bases, the testing and inspection to be performed to verify system capability and reliability, and the required instrumentation and controls. For systems that have little or no role in protecting the public against exposure to radiation, the description should provide enough information to allow the NRC staff to understand the design and operation and their effect on reactor safety, with emphasis on those aspects of design and operation that might affect the reactor and its safety features or contribute to the control of radioactivity. In addition, the information provided (e.g., a failure analysis) should clearly show the system's capability to function without compromising the safe operation of the plant under both normal operating and transient situations.

The applicant should state seismic design classifications with reference to detailed information provided in Chapter 3 of the FSAR, where appropriate. The applicant should also summarize radiological considerations associated with the operation of each system under normal and accident conditions, where applicable, with reference to detailed information in Chapters 11 or 12, as appropriate.

Modified Section/Title: C.I.9.1; Fuel Storage and Handling
Original Section/Title: C.I.9.1; Fuel Storage and Handling

Modified Section/Title: C.I.9.1.1; Criticality Safety of Fresh and Spent Fuel Storage and Handling
Original Section/Title: C.I.9.1.1; Criticality Safety of Fresh and Spent Fuel Storage and Handling

Modified Section/Title: C.I.9.1.1.1; Design Bases
Original Section/Title: C.I.9.1.1.1; Design Bases
The applicant should provide the design bases for new and spent fuel storage facilities, including such considerations as quantity of fuel to be stored, means for maintaining a subcritical array, the degree of subcriticality provided for the most reactive condition possible together with the methods, approximations and assumptions used in this analysis.

Modified Section/Title: C.I.9.1.1.2; Facilities Description
Original Section/Title: C.I.9.1.1.2; Facilities Description
The applicant should provide a description of the new and spent fuel storage facilities, including drawings, and their locations in the station complex.

Modified Section/Title: C.I.9.1.1.3; Safety Evaluation
Original Section/Title: C.I.9.1.1.3; Safety Evaluation
The applicant should provide an evaluation of the capability of the new and spent fuel storage facilities to reduce the probability of occurrence of unsafe conditions. This evaluation should include consideration of the degree of subcriticality for all normal and credible abnormal conditions that could involve the storage and handling of fresh and spent fuel. The evaluation should include descriptions of the methods used, approximations and assumptions made, and handling of design tolerances and uncertainties. Additional

guidance regarding acceptable design of the spent fuel storage facilities is given in RG 1.13. Guidance in RG 1.13 regarding spent fuel pool water purification is not applicable. Guidance in RG 1.13 regarding the safety related aspects of monitoring instrumentation, water level, water cooling, and makeup water systems are not applicable to the HTGR.

Modified Section/Title: C.I.9.1.2; New and Spent Fuel Storage

Original Section/Title: C.I.9.1.2; New and Spent Fuel Storage

Modified Section/Title: C.I.9.1.2.1; Design Bases

Original Section/Title: C.I.9.1.2.1; Design Bases

This section should provide the design bases for the new and spent fuel storage facilities, including such considerations as quantity of fuel to be stored, the configuration of the storage facilities, and the design of the storage wells. This information should address measures to provide drainage of new and spent fuel storage areas, measures to prevent flooding of dry new and spent fuel storage areas, circulation of coolant through the storage wells, shielding requirements, design loadings to be withstood, and protection against natural phenomena and internal missiles.

Modified Section/Title: C.I.9.1.2.2; Facilities Description

Original Section/Title: C.I.9.1.2.2; Facilities Description

This section should provide a description of the spent fuel storage facilities, including drawings, and their location in the station complex.

Modified Section/Title: C.I.9.1.2.3; Safety Evaluation

Original Section/Title: C.I.9.1.2.3; Safety Evaluation

The applicant should provide an evaluation of the protection of the spent fuel storage facilities against unsafe conditions. This evaluation should include the following considerations:

- Governing codes for design
- Protection against natural phenomena
- Ability to withstand design loads and forces
- Design features to maintain an adequate cooling capability in spent fuel storage areas under normal and accident conditions
- Design features (e.g., drains) to prevent flooding of dry new and spent fuel storage areas
- Effectiveness of coolant circulation
- Configuration of fuel storage and associated handling areas to preclude accidental dropping of heavy objects on spent fuel
- Material compatibility requirements
- Radiological shielding design (present details in FSAR Chapter 12)
- Ability of the fuel storage wells to withstand accident forces associated with fuel handling
- Safety implications related to sharing (for multi-unit facilities).

RG 1.13 gives additional guidance on the acceptable design of the spent fuel storage facilities. Guidance in RG 1.13 regarding spent fuel pool water purification is not applicable to the HTGR. Guidance in RG 1.13 regarding the safety related aspects of monitoring instrumentation, water level, water cooling, and makeup water systems are not applicable to the HTGR.

Modified Section/Title: C.I.9.1.3; Spent Fuel Pool Cooling System

Original Section/Title: C.I.9.1.3; Spent Fuel Pool Cooling and Cleanup System

Modified Section/Title: C.I.9.1.3.1; Design Bases

Original Section/Title: C.I.9.1.3.1; Design Bases

The applicant should provide the design bases for the cooling system for the spent fuel facilities, including the following considerations:

- The heat generation rate of the stored fuel
- The heat removal paths for normal and accident conditions
- Protection of essential components against natural phenomena and internal missiles
- The capability of essential components to withstand design loadings
- Cooling water temperature limits for normal and accident conditions
- Provisions to preclude inadvertent or accidental draining or siphoning of water coolant
- Provisions to collect system leakage and instrumentation to indicate water inventory and temperature
- Radiation levels under normal and anticipated accident conditions.

Modified Section/Title: C.I.9.1.3.2; System Description

Original Section/Title: C.I.9.1.3.2; System Description

The applicant should provide a detailed description and drawings of the cooling system, including the instrumentation and alarms.

Modified Section/Title: C.I.9.1.3.3; Safety Evaluation

Original Section/Title: C.I.9.1.3.3; Safety Evaluation

The applicant should provide an evaluation of the cooling system, including the following considerations:

- Capability to transfer the necessary heat from the surrounding helium gas to the cooling water and then to an UHS under normal and accident conditions without exceeding specified spent fuel temperatures
- Capability to maintain and makeup as necessary the helium gas surrounding the spent fuel
- Capability of the makeup water system to maintain adequate water level for cooling and shielding requirements under normal and accident conditions
- Provision of passive design features to ensure that the cooling water level will not be inadvertently reduced below the minimum level necessary for adequate cooling and shielding
- The ability to maintain occupational exposure as low as reasonably achievable
- Capability to withstand design loads and forces
- Protection of essential components from the effects of natural phenomena
- Provision of features to collect cooling water system leakage
- Safety implications related to sharing (for multi-unit facilities).

Modified Section/Title: C.I.9.1.3.4; Inspection and Testing Requirements

Original Section/Title: C.I.9.1.3.4; Inspection and Testing Requirements

The applicant should describe the inspection and testing requirements for the cooling system.

Modified Section/Title: C.I.9.1.3.5; Instrumentation Requirements

Original Section/Title: C.I.9.1.3.5; Instrumentation Requirements

The applicant should describe system instrumentation, including instrumentation to indicate helium gas inventory, water inventory, temperature, and radiation levels under normal and anticipated accident conditions.

Modified Section/Title: C.I.9.1.4; Light-Load Handling System (Related to Refueling)

Original Section/Title: C.I.9.1.4; Light-Load Handling System (Related to Refueling)

Modified Section/Title: C.I.9.1.4.1; Design Bases
Original Section/Title: C.I.9.1.4.1; Design Bases
The applicant should provide the design bases for the fuel handling system (FHS), including the load handling requirements, handling control features, and provisions to prevent fuel handling accidents.

Modified Section/Title: C.I.9.1.4.2; System Description
Original Section/Title: C.I.9.1.4.2; System Description
The applicant should provide a description of the FHS, including all components for transporting and handling fuel from the time it reaches the plant until it leaves the plant. The applicant should provide an outline of the procedures used in new fuel receipt and storage, reactor refueling operations, and spent fuel storage and shipment. Toward that end, the FSAR should also provide component drawings, building layouts, and illustrations showing important aspects of the fuel handling process. For example, illustrations and component drawings should show the arrangement of equipment for fuel movement within the reactor and the equipment used for fuel transfer. Include detailed descriptions and drawings, and provide the design data, seismic category, and quality class for all principal components. Also identify the design codes and standards used for design, manufacture, testing, operation, maintenance, and seismic design aspects.

Modified Section/Title: C.I.9.1.4.3; Safety Evaluation
Original Section/Title: C.I.9.1.4.3; Safety Evaluation
The applicant should provide an evaluation of the FHS, including the system's capability to preclude unacceptable releases of radiation as a result of mechanical damage to fuel, maintain an adequate degree of subcriticality, and maintain acceptable shielding during fuel handling. This evaluation should consider the design of components and mechanisms to withstand earthquakes and interlocks and design features to ensure that the applicant will perform fuel handling within acceptable limits.

Modified Section/Title: C.I.9.1.4.4; Inspection and Testing Requirements
Original Section/Title: C.I.9.1.4.4; Inspection and Testing Requirements
The applicant should describe the inspection and testing requirements for FHS subsystems and components, including shop tests, preoperational tests, and periodic operational tests.

Modified Section/Title: C.I.9.1.4.5; Instrumentation Requirements
Original Section/Title: C.I.9.1.4.5; Instrumentation Requirements
The applicant should describe the system I&C, alarms, and communication system(s). Include a description of the adequacy of safety-related interlocks.

Modified Section/Title: C.I.9.1.5; Overhead Heavy-Load Handling System
Original Section/Title: C.I.9.1.5; Overhead Heavy-Load Handling System

Modified Section/Title: C.I.9.1.5.1; Design Bases
Original Section/Title: C.I.9.1.5.1; Design Bases
The applicant should provide the design bases for the overhead heavy-load handling system with respect to critical load handling evolutions. Critical load handling evolutions are those handling evolutions with the potential for inadvertent operations or equipment malfunctions to affect the handling system in the following ways:

- Cause a significant release of radioactivity
- Cause a loss of margin to criticality
- Cause a loss of cooling to irradiated fuel in the reactor vessel or spent fuel
- Damage equipment essential to achieve or maintain safe shutdown.

Necessary information includes parameters defining the load that, if dropped, would cause the greatest damage; the areas of the plant where the load would be handled; the design of the overhead heavy-load handling system; and the operating, maintenance, and inspection procedures applied to the load handling system. A heavy load is defined as a load weighing more than one fuel element and its associated handling device.

Modified Section/Title: C.I.9.1.5.2; System Description

Original Section/Title: C.I.9.1.5.2; System Description

The applicant should provide a description of the overhead heavy-load handling system, including component drawings, building layouts, and illustrations of special lifting devices. For all principal components, provide the relevant design data, seismic category, and quality class, and identify the design codes and standards used for design, manufacture, testing, operation, maintenance, and seismic design aspects.

Modified Section/Title: C.I.9.1.5.3; Safety Evaluation

Original Section/Title: C.I.9.1.5.3; Safety Evaluation

The applicant should provide an evaluation of the overhead heavy-load handling system in satisfying the applicable objectives of Section 5.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," including the following capabilities:

- Preclude unacceptable releases of radiation through mechanical damage to fuel
- Prevent damage that could threaten the ability to maintain an adequate degree of subcriticality
- Prevent damage that could result in a loss of cooling capability to the reactor vessel or spent fuel
- Prevent damage that alone could result in a loss of essential safe-shutdown functions.

This evaluation should describe the extent of conformance with the general load handling practices of Section 5.1.1 of NUREG-0612 and describe design features or analyses demonstrating that the design will achieve the objectives of Section 5.1 of NUREG-0612. These design features and analyses may include one or more of the following:

- Mechanical stops or electrical interlocks to preclude load drops in critical areas
- Analyses of potential load drops demonstrating that the system would satisfy the objectives in the event of a load drop
- A highly reliable load handling system to assure a low probability of a load drop in a critical area.

Modified Section/Title: C.I.9.1.5.4; Inspection and Testing Requirements

Original Section/Title: C.I.9.1.5.4; Inspection and Testing Requirements

The applicant should describe the inspection and testing requirements for the overhead heavy-load handling system components, including shop tests, preoperational tests, and periodic operational tests and inspections.

Modified Section/Title: C.I.9.1.5.5; Instrumentation Requirements

Original Section/Title: C.I.9.1.5.5; Instrumentation Requirements

The applicant should describe the system I&C, alarms and communication system(s), and the adequacy of safety-related interlocks.

Modified Section/Title: C.I.9.2; Water Systems

Original Section/Title: C.I.9.2; Water Systems

This section of the FSAR should discuss each of the plant's water systems. The applicant should provide separate subsections (numbered 9.2.1 through 9.2.x) for each of the systems.

Modified Section/Title: C.I.9.2.1; Shutdown Cooling Water and Service Water Systems and Subsystems
Original Section/Title: C.I.9.2.1; Station Service Water System (Open, Raw Water Cooling Systems)

Modified Section/Title: C.I.9.2.1.1; Design Bases
Original Section/Title: C.I.9.2.1.1; Design Bases
The applicant should provide the design bases for the shutdown cooling water and service water system and subsystems, including the following considerations:
- Cooling requirements for normal and accident conditions
- Capability to provide essential cooling for normal and accident conditions
- Capability to provide essential cooling using either offsite power supplies or onsite emergency power supplies
- Capability to isolate nonessential portions of the system
- Protection of essential (i.e., needed for accident mitigation) components against natural phenomena and internal missiles
- Capability of essential components to withstand design loadings
- Provisions for inspection and functional testing of essential components and system segments
- Provisions to detect leakage of radioactive material into the system and control leakage out of the system
- Provisions to protect against adverse environmental, operating, and accident conditions that can occur, such as freezing, thermal overpressurization, and waterhammer
- Capability of the system to function at the minimum UHS design conditions.

Modified Section/Title: C.I.9.2.1.2; System Description
Original Section/Title: C.I.9.2.1.2; System Description
The applicant should provide a detailed description and drawings of the shutdown cooling water system and the service water system, including components cooled by the system, nonessential components that may be isolated from the shutdown cooling water system and the service water system, cross-connection capability between trains and units, and instrumentation and alarms.

Modified Section/Title: C.I.9.2.1.3; Safety Evaluation
Original Section/Title: C.I.9.2.1.3; Safety Evaluation
If either the shutdown cooling water system or the service water system and subsystems perform any safety functions, the applicant should provide an evaluation of the shutdown cooling water system and service water system and subsystems, including the following considerations, as applicable:
- Capability to transfer the necessary heat to an UHS under normal and accident conditions with and without offsite power available
- Capability to isolate nonessential portions of the system
- The protection of essential components against natural phenomena and internal missiles
- Capability of essential components to withstand design loadings and adverse environmental, operating, and accident conditions
- Capability of the system to function during adverse environmental conditions and abnormally high and low water levels
- Measures used to prevent long-term corrosion and organic fouling that may degrade system performance
- Safety implications related to sharing of systems that can be cross-tied (for multi-unit facilities).

Modified Section/Title: C.I.9.2.1.4; Inspection and Testing Requirements
Original Section/Title: C.I.9.2.1.4; Inspection and Testing Requirements
The applicant should describe the inspection and testing requirements for the shutdown cooling water system and service water system and subsystems, including inspection and testing necessary to demonstrate that the applicant will effectively manage fouling and degradation mechanisms applicable to the site to maintain acceptable system performance and integrity, and periodic flow testing though normally isolated safety-related components and infrequently used cross-connections between trains/units.

Modified Section/Title: C.I.9.2.1.5; Instrumentation Requirements
Original Section/Title: C.I.9.2.1.5; Instrumentation Requirements
The applicant should describe the system alarms, instrumentation, and controls. This description should include the adequacy of instrumentation to support required testing, as well as the adequacy of alarms to notify operators of degraded conditions.

Modified Section/Title: C.I.9.2.2; Reactor Plant Cooling Water System
Original Section/Title: C.I.9.2.2; Cooling System for Reactor Auxiliaries (Closed Cooling Water Systems)

Modified Section/Title: C.I.9.2.2.1; Design Bases
Original Section/Title: C.I.9.2.2.1; Design Bases
The applicant should provide the design bases for the reactor plant cooling water system, including the following considerations:

- Cooling requirements for normal and accident operations
- Capability to provide essential cooling for normal and accident conditions
- Capability to provide essential cooling using either offsite power supplies or onsite emergency power supplies
- Capability to isolate nonessential portions of the system
- Protection of essential components against natural phenomena and internal missiles
- Capability of essential components to withstand design loadings
- Provisions to protect against adverse environmental, operating, and accident conditions that can occur, such as thermal overpressurization and waterhammer
- Provisions for inspection and functional testing of essential components and system segments
- Provisions to detect and control leakage of radioactive material into or out of the system
- Provisions to withstand loss of pressure boundary integrity in one train and expected long-term leakage without a loss of system functional capability.

Modified Section/Title: C.I.9.2.2.2; System Description
Original Section/Title: C.I.9.2.2.2; System Description
The applicant should provide a detailed description and drawings of the reactor plant cooling water system, including the components cooled by the system, nonessential components that may be isolated, cross-connection capability between trains and units, and instrumentation and alarms.

Modified Section/Title: C.I.9.2.2.3; Safety Evaluation
Original Section/Title: C.I.9.2.2.3; Safety Evaluation
If the reactor plant cooling water system performs any safety functions, the applicant should provide an evaluation of the reactor plant cooling water system, including the following considerations, as applicable:

- Capability to transfer the necessary heat to an UHS under normal and accident conditions, with and without offsite power available

- Capability to isolate nonessential portions of the system
- Protection of essential components against natural phenomena and internal missiles
- Capability of essential components to withstand design loadings and adverse environmental, operating, and accident conditions
- Prevention of long-term corrosion that may degrade system performance
- Safety implications related to sharing (for multi-unit facilities)
- Capability to withstand loss of pressure boundary integrity in one train and expected long-term leakage without a loss of system functional capability.

Modified Section/Title: C.I.9.2.2.4; Inspection and Testing Requirements
Original Section/Title: C.I.9.2.2.4; Inspection and Testing Requirements
The applicant should describe the inspection and testing requirements for the reactor plant cooling water system.

Modified Section/Title: C.I.9.2.2.5; Instrumentation Requirements
Original Section/Title: C.I.9.2.2.5; Instrumentation Requirements
The applicant should describe the system alarms, instrumentation, and controls. Include a description of the adequacy of instrumentation to support required testing, as well as the adequacy of alarms to notify operators of degraded conditions.

Modified Section/Title: C.I.9.2.3; Reactor Cavity Cooling System Water Cooling Subsystem
Original Section/Title: C.I.9.2.3; [Reserved]

Modified Section/Title: C.I.9.2.3.1; Design Basis
Original Section/Title: NEW; NEW
The applicant should provide the design bases for the Reactor Cavity Cooling System Water Cooling Subsystem, including the following considerations:
- Cooling requirements for normal and accident operations
- Capability to provide essential cooling for normal and accident conditions
- Capability to provide essential cooling using either offsite power supplies or onsite emergency power supplies
- Capability to isolate nonessential portions of the system
- Protection of essential components against natural phenomena and internal missiles
- Capability of essential components to withstand design loadings
- Provisions to protect against adverse environmental, operating, and accident conditions that can occur, such as thermal overpressurization and waterhammer
- Provisions for inspection and functional testing of essential components and system segments
- Provisions to detect and control leakage of radioactive material into or out of the system
- Provisions to withstand loss of pressure boundary integrity in one train and expected long-term leakage without a loss of system functional capability.

Modified Section/Title: C.I.9.2.3.2; System Description
Original Section/Title: NEW; NEW
The applicant should provide a detailed description and drawings of the Reactor Cavity Cooling System Water Cooling Subsystem, including the components cooled by the system, nonessential components that may be isolated, cross-connection capability between trains and units, and instrumentation and alarms.

Modified Section/Title: C.I.9.2.3.3; Safety Evaluation
Original Section/Title: NEW; NEW
If the Reactor Cavity Cooling System Water Cooling Subsystem performs any safety functions, then the applicant should provide an evaluation of the Reactor Cavity Cooling System Water Cooling Subsystem, including the following considerations, as applicable:

- Capability to transfer the necessary heat to an UHS under normal and accident conditions, with and without offsite power available
- Capability to isolate nonessential portions of the system
- Protection of essential components against natural phenomena and internal missiles
- Capability of essential components to withstand design loadings and adverse environmental, operating, and accident conditions
- Prevention of long-term corrosion that may degrade system performance
- Safety implications related to sharing (for multi-unit facilities)
- Capability to withstand loss of pressure boundary integrity in one train and expected long-term leakage without a loss of system functional capability.

Modified Section/Title: C.I.9.2.3.4; Inspection and Testing Requirements
Original Section/Title: NEW; NEW
The applicant should describe the inspection and testing requirements for the Reactor Cavity Cooling System Water Cooling Subsystem

Modified Section/Title: C.I.9.2.3.5; Instrumentation Requirements
Original Section/Title: NEW; NEW
The applicant should describe the system alarms, instrumentation, and controls. Include a description of the adequacy of instrumentation to support required testing, as well as the adequacy of alarms to notify operators of degraded conditions.

Modified Section/Title: C.I.9.2.4; Potable, Sanitary, and Hot Water Systems
Original Section/Title: C.I.9.2.4; Potable and Sanitary Water Systems
The applicant should provide a description of the potable, sanitary, and hot water systems. This description should include system design criteria addressing connections to the nuclear island and provisions for the prevention of connections to systems having the potential to contain radioactive material.

Modified Section/Title: C.I.9.2.5; N/A
Original Section/Title: C.I.9.2.5; Ultimate Heat Sink
N/A
See section 6.3

Modified Section/Title: C.I.9.2.5.1; N/A
Original Section/Title: C.I.9.2.5.1; Design Bases
N/A
See section 6.3

Modified Section/Title: C.I.9.2.5.2; N/A
Original Section/Title: C.I.9.2.5.2; System Description
N/A
See section 6.3

Modified Section/Title: C.I.9.2.5.3; N/A
Original Section/Title: C.I.9.2.5.3; Safety Evaluation
N/A
See section 6.3

Modified Section/Title: C.I.9.2.5.4; N/A
Original Section/Title: C.I.9.2.5.4; Inspection and Testing Requirements
N/A
See section 6.3

Modified Section/Title: C.I.9.2.5.5; N/A
Original Section/Title: C.I.9.2.5.5; Instrumentation Requirements
N/A
See section 6.3

Modified Section/Title: C.I.9.2.6; N/A
Original Section/Title: C.I.9.2.6; Condensate Storage Facilities

Modified Section/Title: C.I.9.2.6.1; N/A
Original Section/Title: C.I.9.2.6.1; Design Bases

Modified Section/Title: C.I.9.2.6.2; N/A
Original Section/Title: C.I.9.2.6.2; System Description

Modified Section/Title: C.I.9.2.6.3; N/A
Original Section/Title: C.I.9.2.6.3; Safety Evaluation

Modified Section/Title: C.I.9.2.6.4; N/A
Original Section/Title: C.I.9.2.6.4; Inspection and Testing Requirements

Modified Section/Title: C.I.9.2.6.5; N/A
Original Section/Title: C.I.9.2.6.5; Instrumentation Requirements

Modified Section/Title: C.I.9.3; Process Auxiliaries
Original Section/Title: C.I.9.3; Process Auxiliaries
This section of the FSAR should discuss each of the auxiliary systems associated with the reactor process system. Because these auxiliary systems vary in number, type, and nomenclature for various plant designs, the standard format does not assign specific subsection numbers to these systems. The applicant should provide separate subsections (numbered 9.3.1 through 9.3.x) for each of the systems. For each system, these subsections should provide the following information:

- Design bases, including the GDC [PDC] to which the system is designed
- System description
- Safety evaluation
- Testing and inspection requirements
- Instrumentation requirements.

The following paragraphs provide examples of systems that the section should discuss, as appropriate to the individual plant, and identify some specific information that the section should provide in addition to the items identified above. These examples are not intended to represent a complete list of systems to be discussed in this section.

Modified Section/Title: C.I.9.3.1; Compressed Air Systems

Original Section/Title: C.I.9.3.1; Compressed Air Systems

The applicant should describe the compressed (instrument and service) air systems that provide station air for service and maintenance uses, and include discussion of provisions for air cleanliness and quality requirements, and environmental design requirements. Include a description of the capabilities to interconnect and/or isolate the instrumentation and control air system from the station service air system if the design provides two such systems that can be interconnected.

The description of the compressed air system should include a failure analysis (including diverse sources of electric power), the maintenance of air cleanliness to ensure system reliability, the capability to isolate the system, if required, and safety implications related to sharing (for multi-unit plants). Include in the failure analyses a description of the system's capability to function in the event of adverse environmental phenomena, abnormal operational, or accident conditions. Address the potential for overpressurization of air-supplied components. The applicant should describe the I&C features to determine and ensure that the system is operating correctly, including the means to detect leakage from radioactive systems to the I&C air system and to preclude releases to the environment. The applicant should describe the provisions for periodic testing of air quality, testing of pressure and leakage, and any necessary periodic functional testing of the safety-related portions of the I&C air system.

Modified Section/Title: C.I.9.3.2; Process and Postaccident Sampling Systems

Original Section/Title: C.I.9.3.2; Process and Postaccident Sampling Systems

The applicant should describe the sampling system for the various plant gases and fluids.

Include consideration of sample size and handling necessary to ensure that a representative sample is obtained from liquid and gaseous process streams and tanks. The applicant should describe provisions for purging sampling lines and reducing plateout in sample lines (e.g., heat tracing). The applicant should describe provisions to purge and drain sample streams back to the system of origin, or to an appropriate waste treatment system, to minimize personnel exposure.

The applicant should describe provisions for isolating the system and the means to limit reactor coolant losses; requirements to minimize, to the extent practical, hazards to plant personnel; and design of the system, including pressure, temperature, materials of construction, and applicable code requirements. The description should delineate process streams and points where samples will be obtained, along with the parameters to be determined through sampling (e.g., gross beta-gamma concentration). The applicant should describe measures to ensure that samples will be representative samples, and address the effect of sharing on plant safety (for multi-unit facilities).

Having the postaccident sampling system is not mandatory. However, although the process sampling system does not have postaccident sampling capability its design should allow for collection of highly radioactive samples provided the contingency plan exists for their handling, no decrease in the effectiveness of emergency plans occurs, radioactivity including iodines is monitored and the capability for sampling and analyzing the reactor building atmosphere exists.

Modified Section/Title: C.I.9.3.3; Equipment and Floor Drainage System

Original Section/Title: C.I.9.3.3; Equipment and Floor Drainage System

The applicant should describe the drainage systems for collecting the radioactive effluent from high-activity and low-activity liquid drains from various specified equipment and buildings. Include piping and pumps from equipment or floor drains to the sumps, and any additional equipment that may be necessary to route effluents to the drain tanks and then to the radwaste system.

Discuss design considerations for precluding back-flooding of equipment in safety-related compartments, as well as preventing transfer of contaminated fluids to noncontaminated drainage systems. Identify areas where the drainage system is used to detect leakage from safety systems or to identify conditions that are adverse to safety, such as excessive leakage that could compromise the capability of SSC to perform safety functions or could result in an uncontrolled release of radioactive material to the environment. The applicant should describe the performance of interfacing reviews under the sections dealing with protecting drainage systems against flooding, internally and externally generated missiles, and high- or moderate-energy pipe breaks.

The applicant should describe the seismic and safety classifications of the various portions of the system. Identify those portions of the system that are classified as seismic Category I and Quality Group C.

FSAR Chapters 11 and 12 should present an evaluation of radiological considerations for normal operation and postulated spills and accidents, including the effects of sharing (for multi-unit plants).

Modified Section/Title: C.I.9.3.4; Helium Storage and Transfer System

Original Section/Title: C.I.9.3.4; Chemical and Volume Control System (Including Boron Recovery System) (Pressurized-Water Reactors Only)

This section should provide a summary description of the Helium Storage and Transfer System. The description should include identification of system components and any shared systems or components. The description should also include the following:

- System and subsystem arrangement, configuration, and operation including process and instrument drawings and general arrangement drawings
- Materials of construction
- Identification of system interfaces
- Power generation functions
- Radionuclide control functions
- Helium storage requirements and their bases
- Electric power requirements
- Instrumentation and control requirements
- Failure modes effects
- Anticipated operational occurrences and their consequences
- Design basis events performance
- Inspection and testing requirements.

Modified Section/Title: C.I.9.3.4.1; Design Bases

Original Section/Title: C.I.9.3.4.1; Design Bases

The design bases for the helium storage and transfer system should include the capability to supply helium to the primary heat transfer systems.

Modified Section/Title: C.I.9.3.4.2; System Description

Original Section/Title: C.I.9.3.4.2; System Description

The applicant should provide a complete description of the system and components, including any piping and instrumentation diagrams. Include design data, seismic category, and quality class for all components. The applicant should describe the principles of both automatic and manual system operation for steady-state, transient, startup, shutdown, and accident conditions. The applicant should describe controls, design provisions, and automatic features. Outline the operating procedures, including the controls for pumpup and pumpdown modes of operation.

Discuss helium coolant purity/chemistry requirements. The applicant should describe temperature control provisions, if any, including provision for alarm failures. The applicant should provide tables of system design parameters and component design data.

Modified Section/Title: C.I.9.3.4.3; Safety Evaluation

Original Section/Title: C.I.9.3.4.3; Safety Evaluation

The applicant should provide a safety evaluation that addresses, at a minimum, the following considerations:

- Design for safe operation, shutdown, and prevention/mitigation of postulated accidents, including the ability of the system to provide sufficient capacity and capability to support the plant's ability to withstand, or cope with, as applicable, and recover from, a SBO
- Pumping capability of system for reactor coolant makeup, and for small pipe and component failures
- Provisions for a leakage detection and control program in accordance with 10 CFR 50.34(f)(xxvi)
- Design for limitation of radioactive releases to the environment within normal and accident limits
- Justification for the component and piping seismic design category and quality class assigned
- Results of failure modes and effects analyses for prevention/mitigation of postulated accidents
- System provisions to prevent such vacuum conditions that could cause wall inward buckling and failure in tanks
- Compliance with GDC [PDC]
- Extent to which the applicant has followed applicable regulatory guides protection of essential portions of systems from failure of non-seismic Category I equipment and piping.

And also from the following events:

- Flooding
- Adverse environmental occurrences (e.g., hurricanes, tornadoes)
- Abnormal operational conditions, or accident conditions, such as the following:
- Internally and externally generated missiles
- Lose of offsite power
- The effects of high- and moderate-energy line failures.

Modified Section/Title: C.I.9.3.4.4; Inspection and Testing Requirements

Original Section/Title: C.I.9.3.4.4; Inspection and Testing Requirements

The applicant should describe the inspection and testing requirements for the Helium Storage and Transfer System.

Modified Section/Title: C.I.9.3.4.5; Instrumentation Requirements

Original Section/Title: C.I.9.3.4.5; Instrumentation Requirements

The applicant should describe the system I&C, including the adequacy of to fulfill their functions.

Modified Section/Title: C.I.9.3.5; Decontamination Services Subsystem

Original Section/Title: C.I.9.3.5; Standby Liquid Control System (Boiling-Water Reactors)

This section should provide a summary description of the Decontamination Services Subsystem. The description should include identification of system components and any shared systems or components. The description should also include the following:

- System and subsystem arrangement, configuration, and operation including process and instrument drawings and general arrangement drawings

- Identification of equipment receiving decontamination services
- Identification of decontamination processes to be used
- Identification of decontamination chemicals or agents to be used
- Identification of system interfaces, including in particular with radwaste systems
- Anticipated decontamination activities and techniques
- Decontamination controls and processing of waste streams
- Electric power requirements
- Instrumentation and control requirements
- Failure modes effects
- Anticipated operational occurrences and their consequences
- Design basis events performance
- Inspection and testing requirements.

Modified Section/Title: C.I.9.3.5.1; Design Bases
Original Section/Title: C.I.9.3.5.1; Design Bases
See section 9.3.5.

Modified Section/Title: C.I.9.3.5.2; System Description
Original Section/Title: C.I.9.3.5.2; System Description
See section 9.3.5.

Modified Section/Title: C.I.9.3.5.3; Safety Evaluation
Original Section/Title: C.I.9.3.5.3; Safety Evaluation
See section 9.3.5.

Modified Section/Title: C.I.9.3.5.4; Inspection and Testing Requirements
Original Section/Title: C.I.9.3.5.4; Inspection and Testing Requirements
See section 9.3.5.

Modified Section/Title: C.I.9.3.5.5; Instrumentation Requirements
Original Section/Title: C.I.9.3.5.5; Instrumentation Requirements
See section 9.3.5.

Modified Section/Title: C.I.9.3.6; Liquid Nitrogen Subsystem
Original Section/Title: NEW; NEW
This section should provide a summary description of the Liquid Nitrogen Subsystem. The description should include identification of system components and any shared systems or components. The description should also include the following:

- Subsystem arrangement, configuration, normal operation, reactor depressurization operation, and shutdown operation should be described Process and instrument drawings and general arrangement drawings
- Materials of construction
- Identification of system interfaces
- Power generation functions
- Radionuclide control functions
- Nitrogen storage requirements and their bases
- Electric power requirements
- Instrumentation and control requirements
- Failure modes effects

- Anticipated operational occurrences and their consequences
- Design basis events performance
- Inspection and testing requirements.

Modified Section/Title: C.I.9.3.6.1; Design Bases
Original Section/Title: NEW; NEW
See section 9.3.6.

Modified Section/Title: C.I.9.3.6.2; System Description
Original Section/Title: NEW; NEW
See section 9.3.6.

Modified Section/Title: C.I.9.3.6.3; Safety Evaluation
Original Section/Title: NEW; NEW
See section 9.3.6.

Modified Section/Title: C.I.9.3.6.4; Inspection and Testing Requirements
Original Section/Title: NEW; NEW
See section 9.3.6.

Modified Section/Title: C.I.9.3.6.5; Instrumentation Requirements
Original Section/Title: NEW; NEW
See section 9.3.6.

Modified Section/Title: C.I.9.3.7; Helium Purification System
Original Section/Title: NEW; NEW
This section should provide a summary description of the Helium Purification System. The description should include identification of system components and any shared systems or components. The description should also include the following:

- Subsystem arrangement, configuration, normal operation, reactor depressurization operation, and refueling operation should be described Process and instrument drawings and general arrangement drawings
- Materials of construction
- Identification of system interfaces
- Power generation functions
- Radionuclide design requirements and control functions
- Radiation shielding requirements
- Electric power requirements
- Instrumentation and control requirements
- Failure modes effects
- Anticipated operational occurrences and their consequences
- Design basis events performance
- Inspection and testing requirements.

Modified Section/Title: C.I.9.3.7.1; Design Bases
Original Section/Title: NEW; NEW
See section 9.3.7.

Modified Section/Title: C.I.9.3.7.2; System Description
Original Section/Title: NEW; NEW
See section 9.3.7.

Modified Section/Title: C.I.9.3.7.3; Safety Evaluation
Original Section/Title: NEW; NEW
See section 9.3.7.

Modified Section/Title: C.I.9.3.7.4; Inspection and Testing Requirements
Original Section/Title: NEW; NEW
See section 9.3.7.

Modified Section/Title: C.I.9.3.7.5; Instrumentation Requirements
Original Section/Title: NEW; NEW
See section 9.3.7.

Modified Section/Title: C.I.9.4; Air Conditioning, Heating, Cooling, and Ventilation Systems
Original Section/Title: C.I.9.4; Air Conditioning, Heating, Cooling, and Ventilation Systems
The following subsections discuss examples of systems that the applicant should address, as appropriate to the individual plant, and identify some specific information that the applicant should provide. These examples are not intended to represent a complete list of systems to be discussed in this section. For each system, these subsections should provide the following information:

- Design bases, including the GDC [PDC] to which the system is designed
- System description
- Safety evaluation
- Testing and inspection requirements
- Instrumentation requirements.

Modified Section/Title: C.I.9.4.1; Control Room Area Ventilation System
Original Section/Title: C.I.9.4.1; Control Room Area Ventilation System
See sections 9.4.1.1 thru 9.4.1.5.

Modified Section/Title: C.I.9.4.1.1; Design Bases
Original Section/Title: C.I.9.4.1.1; Design Bases
Discuss the design bases for the air handling and treatment system for the control room and other auxiliary rooms (e.g., relay rooms and emergency switchgear rooms) considered to be part of the control room envelope. Include the criteria and/or features that ensure the performance (e.g., flow rates, temperature limits, humidity limits, filtration) and reliability of the system (i.e., single failure, redundancy, seismic design, missile protection, environmental qualification) for all modes of operation, including normal, abnormal, SBO, and toxic gas modes. The design bases should also include requirements for manual or automatic actuation, system isolation, monitoring for radiation and/or toxic gas, and other controls essential to the performance of the system functions.

Modified Section/Title: C.I.9.4.1.2; System Description
Original Section/Title: C.I.9.4.1.2; System Description
The system description should include the system's major components, key parameters, essential controls, and operating modes. This description should also include a process flow diagram or piping and instrument diagram to enhance understanding of system operation and flow paths and tables showing the key parameters and features of major components. In addition, the description should address realignment

of the system as a result of automatic actuation or operator action for all modes of operation, with reference to response to radiation, toxic gas, smoke and/or other actuation signals.

Modified Section/Title: C.I.9.4.1.3; Safety Evaluation
Original Section/Title: C.I.9.4.1.3; Safety Evaluation
Describe the functional performance requirements of the Control Room Area HVAC system to maintain plant performance capability and to maintain ambient environmental conditions within acceptable levels for component operation and protection and for personnel protection. Discuss the manner in which the system achieves each performance requirement. Identify all system/component interfaces between the Control Room Area HVAC and the Nuclear Island and the safety considerations for any such interfaces.

Modified Section/Title: C.I.9.4.1.4; Inspection and Testing Requirements
Original Section/Title: C.I.9.4.1.4; Inspection and Testing Requirements
The applicant should describe the inspection and testing requirements for the control room area ventilation system, including ISI requirements for applicable components. Identify the inspection and testing programs to ensure that the system will meet its functional and plant capability requirements, especially those that will be controlled through TS surveillance, if any.

Modified Section/Title: C.I.9.4.1.5; Instrumentation Requirements
Original Section/Title: C.I.9.4.1.5; Instrumentation Requirements
The applicant should describe the system I&C. Include provisions for operational testing and the I&C features to verify that the system is available to operate in the correct mode.

Modified Section/Title: C.I.9.4.2; Spent Fuel Storage Area Ventilation System
Original Section/Title: C.I.9.4.2; Spent Fuel Pool Area Ventilation System
See sections 9.4.2.1 thru 9.4.2.4.

Modified Section/Title: C.I.9.4.2.1; Design Bases
Original Section/Title: C.I.9.4.2.1; Design Bases
The design bases of the air handling and treatment system for the spent fuel storage area should include the criteria and/or features to ensure the system's functional performance requirements (i.e., flow rates, temperature limits, humidity limits, filtration) and reliability (i.e., single failure, redundancy, seismic design, environmental qualification) for all modes of operation, including normal, abnormal, and SBO modes. The design bases should also include requirements for manual or automatic actuation, system isolation, monitoring for radiation and filtration, and other controls essential to the performance of the system functions.

Modified Section/Title: C.I.9.4.2.2; System Description
Original Section/Title: C.I.9.4.2.2; System Description
The system description should include the system's major components, key parameters, essential controls, and operating modes. This description should also include a process flow diagram or piping and instrument diagram to enhance understanding of system operation and flow paths and include tables showing the key parameters and features of major components. In addition, the description should address realignment of the system as a result of automatic actuation or operator action for all modes of operation with reference to response to radiation or other actuation signals.

Modified Section/Title: C.I.9.4.2.3; Safety Evaluation
Original Section/Title: C.I.9.4.2.3; Safety Evaluation
Identify the functional performance objectives to be achieved by the spent fuel storage area ventilation system confinement, containment, or reduction of contamination by isolation and filtering or alternately

maintenance of acceptable zone temperature and humidity to prevent degradation of important equipment. Discuss the manner in which the system achieves each functional performance objectives actuation signals and subsequent equipment actuation, as well as the capability to reduce contamination by HEPA or carbon filters. Include a discussion of the ability to (1) detect radiation in the area of the spent fuel storage area and (2) filter the contaminants out of the air before exhausting it to the environment or prevent the contaminated air from leaving the spent fuel storage area.

Modified Section/Title: C.I.9.4.2.4; Inspection and Testing Requirements
Original Section/Title: C.I.9.4.2.4; Inspection and Testing Requirements
The applicant should describe the inspection and testing requirements for the spent fuel storage area ventilation system components important to safety. Identify the inspection and testing programs to ensure that the system will meet its functional performance requirements, especially those that will be controlled through TS surveillance, if any, which may include confirmation of filter efficiencies, pressure drops, flow rates, and temperatures through test programs.

Modified Section/Title: C.I.9.4.3; Makeup Water Treatment and Auxiliary Boiler Building HVAC System
Original Section/Title: C.I.9.4.3; Auxiliary and Radwaste Area Ventilation System
See sections 9.4.3.1 thru 9.4.3.4.

Modified Section/Title: C.I.9.4.3.1; Design Bases
Original Section/Title: C.I.9.4.3.1; Design Bases
The design bases for the air handling and treatment system for the Makeup Water Treatment and Auxiliary Boiler Building HVAC System should include the criteria and/or features to ensure the system's performance (i.e., flow rates, temperature limits, humidity limits, filtration) and reliability (i.e., redundancy, seismic design, environmental qualification) for all modes of operation, including normal, abnormal, and SBO. Also describe requirements for manual or automatic actuation, system isolation, monitoring for radiation, and other controls essential to the performance of the system functions, if any. Include, as appropriate, the preferred direction of airflow from areas of low potential radioactivity to areas of high potential radioactivity, any differential pressures to be maintained and measured, and any requirements for the treatment of exhaust air, during normal, abnormal, and accident conditions.

Modified Section/Title: C.I.9.4.3.2; System Description
Original Section/Title: C.I.9.4.3.2; System Description
The system description should include the system's major components, key parameters, essential controls, and operating modes. This description should also include a process flow diagram or piping and instrument diagram to enhance understanding of system operation and flow paths and tables showing the key parameters and features of major components. In addition, the description should address the realignment of the system as a result of automatic actuation or operator action for all modes of operation, with reference to response to radiation or other actuation signals.

Modified Section/Title: C.I.9.4.3.3; Safety Evaluation
Original Section/Title: C.I.9.4.3.3; Safety Evaluation
The applicant should provide an evaluation of the Makeup Water Treatment and Auxiliary Boiler Building HVAC System. Identify the functional performance objectives to be achieved by the system which may include confinement, containment, or reduction of contamination by isolation and filtering or alternately to maintain an acceptable zone temperature and humidity to prevent degradation of important equipment. Discuss the manner in which the system achieves each functional performance objective which may include actuation signals and subsequent equipment actuation, as well as the system's

capability to reduce contamination by HEPA or carbon filters. Chapters 11 and 12 of the FSAR should present the evaluation of radiological considerations for normal operation.

Modified Section/Title: C.I.9.4.3.4; Inspection and Testing Requirements

Original Section/Title: C.I.9.4.3.4; Inspection and Testing Requirements

The applicant should describe the inspection and testing requirements for the Makeup Water Treatment and Auxiliary Boiler Building HVAC System. Identify the inspection and testing programs to ensure that the system will meet its functional performance requirements, especially those that will be controlled through TS surveillance, if any, which may include confirmation of filter efficiencies, pressure drops, flow rates, and temperatures through test programs.

Modified Section/Title: C.I.9.4.4; Turbine Building Area Ventilation System

Original Section/Title: C.I.9.4.4; Turbine Building Area Ventilation System

See sections 9.4.4.1 thru 9.4.4.4.

Modified Section/Title: C.I.9.4.4.1; Design Bases

Original Section/Title: C.I.9.4.4.1; Design Bases

The design bases for the air handling and treatment system for the turbine-generator area in the turbine building should include the criteria and/or features to ensure the system's functional performance (i.e., flow rates, temperature limits, humidity limits, filtration) and reliability for all modes of operation, including normal, abnormal, and SBO conditions. The design bases should also include requirements for manual or automatic actuation, system isolation, and other controls essential to the performance of system functions.

Modified Section/Title: C.I.9.4.4.2; System Description

Original Section/Title: C.I.9.4.4.2; System Description

The system description should include the system's major components, key parameters, essential controls, and operating modes. This description should also include a process flow diagram or piping and instrument diagram to enhance understanding of system operation and flow paths. Tables should be included to show the key parameters and features of major components. In addition, the description should address the realignment of the system as a result of automatic actuation or operator action for all modes of operation with reference to response to radiation or other actuation signals. Identify which, if any, portions of the system are essential (classified as seismic Category I) and how those portions can be isolated from nonessential portions of the system.

Modified Section/Title: C.I.9.4.4.3; Safety Evaluation

Original Section/Title: C.I.9.4.4.3; Safety Evaluation

The applicant should provide an evaluation of the turbine building area ventilation system. This evaluation should include a system failure analysis (including effects of inability to maintain preferred airflow patterns). Identify the functional performance objectives to be achieved by the system which may include confinement, containment, or reduction of contamination by isolation and filtering or alternately to maintain acceptable zone temperature and humidity to prevent degradation of important equipment. Discuss the manner in which the system achieves each functional performance objective which may include actuation signals and subsequent equipment actuation, as well as the capability of the system to reduce contamination by HEPA or carbon filters. FSAR Chapters 11 and 12 should evaluate radiological considerations for normal operation.

Modified Section/Title: C.I.9.4.4.4; Inspection and Testing Requirements

Original Section/Title: C.I.9.4.4.4; Inspection and Testing Requirements

The applicant should describe the inspection and testing requirements for the turbine building area ventilation system. Identify the inspection and testing programs to ensure that the system will meet its functional performance requirements, especially those that will be controlled through TS surveillance, if any, which may include confirmation filter efficiencies, pressure drops, flow rates, and temperatures through test programs.

Modified Section/Title: C.I.9.4.5; Standby Power Building Heating and Ventilation System

Original Section/Title: C.I.9.4.5; Engineered Safety Feature Ventilation System

See sections 9.4.5.1 thru 9.4.5.4.

Modified Section/Title: C.I.9.4.5.1; Design Bases

Original Section/Title: C.I.9.4.5.1; Design Bases

The design bases for the air handling and treatment system for the Standby Power Building Heating and Ventilation System should include the criteria and/or features to ensure the system's functional performance (i.e., flow rates, temperature limits, humidity limits, filtration) and reliability for all modes of operation, including normal, abnormal, and SBO conditions. The design bases should also include requirements for manual or automatic actuation, system isolation, and other controls essential to the performance of system functions.

Modified Section/Title: C.I.9.4.5.2; Systems Description

Original Section/Title: C.I.9.4.5.2; Systems Description

The system description should include the system's major components, key parameters, essential controls, and operating modes. This description should also include a process flow diagram or piping and instrument diagram to enhance understanding of system operation and flow paths. Tables should be included to show the key parameters and features of major components. In addition, the description should address the realignment of the system as a result of automatic actuation or operator action for all modes of operation with reference to response to radiation or other actuation signals. Identify which, if any, portions of the system are essential (classified as seismic Category I) and how those portions can be isolated from non-essential portions of the system.

Modified Section/Title: C.I.9.4.5.3; Safety Evaluation

Original Section/Title: C.I.9.4.5.3; Safety Evaluation

The applicant should provide an evaluation of the Standby Power Building Heating and Ventilation System. This evaluation should include a system failure analysis (including effects of inability to maintain preferred airflow patterns). Identify the functional performance objectives to be achieved by the system which may include confinement, containment, or reduction of contamination by isolation and filtering. Another may be to maintain acceptable zone temperature and humidity to prevent degradation of important equipment. Discuss the manner in which the system achieves each functional performance objective which may include the objective of confinement, containment, or contamination reduction and address the actuation signals and subsequent equipment actuation, as well as the capability of the system to reduce contamination by HEPA or carbon filters.

FSAR Chapters 11 and 12 should evaluate radiological considerations for normal operation.

Modified Section/Title: C.I.9.4.5.4; Inspection and Testing Requirements

Original Section/Title: C.I.9.4.5.4; Inspection and Testing Requirements

The applicant should describe the inspection and testing requirements for the Standby Power Building Heating and Ventilation System. Identify the inspection and testing programs to ensure that the system will meet its functional performance requirements, especially those that will be controlled through TS surveillance, if any, to confirm filter efficiencies, pressure drops, flow rates, and temperatures through test programs.

Modified Section/Title: C.I.9.5; Other Auxiliary Systems
Original Section/Title: C.I.9.5; Other Auxiliary Systems
This section includes examples of other systems important to the safe operation of the facility, such as fire protection systems, lighting systems, communication systems, and backup power generator support systems. The level of information to be provided will reflect the design bases for the system; therefore, the non-safety systems will likely have reduced discussion.

Modified Section/Title: C.I.9.5.1; Nuclear Area and Plant Fire Protection
Original Section/Title: C.I.9.5.1; Fire Protection Program
Because the Fire Protection Program (FPP) is an operational program, as discussed in SECY-05-0197, the program and its implementation milestones should be fully described and reference any applicable standards. Fully described should be understood to mean that the program is clearly and sufficiently described in terms of the scope and level of detail to allow for a reasonable assurance finding of acceptability.

Modified Section/Title: C.I.9.5.1.1; Design Bases
Original Section/Title: C.I.9.5.1.1; Design Bases
The applicant should provide the design bases for the FPP to demonstrate that the FPP satisfies the Commission's fire protection objectives through a defense-in-depth philosophy. SRP Section 9.5.1.1 and RG 1.189, "Fire Protection for Nuclear Power Plants," discuss the design bases for an acceptable FPP. At a minimum, the FSAR should include the following design bases:

- Overall FPP design bases to meet 10 CFR 50.48, "Fire Protection," as well as the criteria for new reactor enhanced fire protection in accordance with Appendix A to SRP Section 9.5.1.1.
- A list of the industry codes, standards, and guidance documents that will be the basis for the design, construction, testing, inspection and maintenance of the FPP, including the applicable edition date (which should be within 6 months of the COL application submittal date for plant specific FPP features, or within 6 months of the design certification application, as applicable). The applicant should identify exceptions to the guidance and/or provisions included in these documents and provide the basis for each exception.
- The assumptions and bases for assumptions applied to analyses of fire-induced multiple spurious actuations that could prevent safe shutdown. This discussion should include the protection provided to ensure that one train of safe-shutdown SSC remains free of fire damage.
- The acceptance criteria for operator manual actions or recovery actions credited to achieve and maintain safe shutdown during and after a fire. The applicant should identify where it has credited operator manual or recovery actions and describe the associated fire scenario for each, as well as the analyses (including the appropriate thermo-hydraulic analysis) to demonstrate that safe shutdown can be achieved and maintained.

Some of this information may not be available or possible to provide at the time the COL application is submitted. In those cases, the applicant should submit the information that is available, justify its inability to provide the unavailable information in the COL application, and furnish details describing implementation plans, milestones, and sequences and/or ITAAC or commitments for developing, completing, and submitting this information during the construction period, prior to fuel receipt on site.

Modified Section/Title: C.I.9.5.1.2; System Description
Original Section/Title: C.I.9.5.1.2; System Description
The applicant should provide a description of the FPP, including the fire protection system piping and instrumentation diagrams. SRP Section 9.5.1.1 describes the scope of the facility FPP and the related NRC-approved acceptance criteria. The applicant should describe each element of the FPP well enough to

permit an independent assessment of the program's capability to satisfy the Commission's fire protection objectives. As a minimum, the system description should include the following:

- Overall FPP provisions, including the fire protection organization; administrative policies; fire prevention controls; applicable administrative, operations, maintenance, and emergency procedures; QA; access to fire areas for fire fighting; and fire brigade and emergency response capability.
- Evaluation of the FPP against RG 1.189 and SRP Section 9.5.1.1. This evaluation should identify and describe all differences between the facility's FPP design features, analytical techniques, and procedural measures, and those given in RG 1.189 and Section 9.5.1.1. Where such differences exist, the evaluation should discuss how the proposed alternative provides an acceptable method of complying with applicable NRC rules or regulations that underlie RG 1.189 and SRP Section 9.5.1.1.
- Provide a plant layout, facility site arrangement, and structural design features, which provide separation or isolation of redundant systems important to safety.
- Selection and design of fire detection, alarm, control, and suppression on the basis of the fire hazards analysis; design, testing, qualification, inspection and maintenance of fire barriers; use of noncombustible materials; design of floor drains, ventilation, emergency lighting, and communication systems to the extent that they impact the FPP.
- Cover fire protection and control provisions (for multi-unit sites) to maintain the integrity and operability of any shared fire protection systems and to ensure that fire hazards associated with one unit will not have an adverse effect on the adjacent unit(s).
- Design features that prevent migration of smoke, hot gases, or fire suppressant material into other fire areas, causing adverse effects on safe-shutdown capabilities, including operator actions.
- Any emergency backup functions performed by the fire protection system to support operation of safe-shutdown systems. This description should include the extent to which the facility relies on this backup function for safe shutdown (e.g., the backup function is required for safe shutdown or is provided only for additional defense in depth and is not essential to achieving or maintaining safe shutdown).
- The facility's design for smoke and heat control during a fire in areas important to safety.
- Contain a description of any portions of the fire protection system that are designed to remain functional following a safe-shutdown earthquake and provisions for isolating those portions from the rest of the system.
- Electrical cable and raceway penetrations in fire barriers and raceway fire barrier systems, including qualification tests and acceptance criteria.
- Provide the schedule and detailed implementation plan for the FPP, to ensure that the program is properly established and implemented in time to provide adequate protection prior to fueling and operation of the nuclear power plant. The description should include the implementation plans to establish, train, and equip the site fire brigade to ensure adequate manual firefighting capability for areas with structures, systems, and components important to safety. As discussed in Section 13.4 of this guide, applicants should provide implementation milestones for operational programs.

Modified Section/Title: C.I.9.5.1.3; Safety Evaluation

Original Section/Title: C.I.9.5.1.3; Safety Evaluation

The applicant should provide a postfire, safe-shutdown analysis to demonstrate that the FPP satisfies the Commission's fire protection objectives, in accordance with the enhanced fire protection criteria for new reactors described in Appendix A to SRP Section 9.5.1.1. This analysis should include the list of systems and components needed to provide postfire safe-shutdown capability; the arrangement of the systems and components within the plant fire areas; the separation between redundant safe-shutdown systems and components; fire protection for safe-shutdown systems and components; and potential interactions between non-safety systems, fire protection systems, and systems important to safety as they relate to potential adverse effects on the safe-shutdown capability. SRP Section 9.5.1.1 and RG 1.189 provide guidance for an acceptable FPP safety evaluation and supporting analyses. To support the safe-shutdown analysis, the applicant should provide a fire hazards analysis evaluating (1) the potential fire hazards for areas containing equipment important to safety throughout the plant, and (2) the effect of postulated fires and explosions relative to maintaining the ability to perform safe-shutdown functions and minimizing radioactive releases to the environment. The fire hazards analysis should specify measures for fire prevention, detection, suppression, and containment, as well as alternative shutdown capability for each fire area containing SSC important to safety in accordance with NRC guidelines and regulations.

Chapter 19 of this Writer's Guide offers guidance for fire PRA.

If the applicant is going to implement a performance-based fire protection program, then the applicant should describe in the FSAR how it will meet the applicable NRC guidance documents.

Modified Section/Title: C.I.9.5.1.4; Inspection and Testing Requirements
Original Section/Title: C.I.9.5.1.4; Inspection and Testing Requirements
The applicant should provide a description of the inspection and testing requirements for the fire protection system for both initial system startup and periodic inspections and tests following startup.

Modified Section/Title: C.I.9.5.2; Communication Systems
Original Section/Title: C.I.9.5.2; Communication Systems
See sections 9.5.2.1 thru 9.5.2.3.

Modified Section/Title: C.I.9.5.2.1; Design Bases
Original Section/Title: C.I.9.5.2.1; Design Bases
This section should provide design bases for the communication systems for intraplant and plant to-offsite communications and should include a discussion of the use of diverse system types. Address the integrated design of the system and related plant features to support effective communication between plant personnel in all vital areas of the plant during normal operation, as well as during accident or incident conditions under maximum potential noise levels or other conditions that could interfere with communication (e.g., electromagnetic interference).

FSAR Section 13.6 should discuss communications associated with security.

Modified Section/Title: C.I.9.5.2.2; System Description
Original Section/Title: C.I.9.5.2.2; System Description
The FSAR should provide a detailed description and evaluation of the communication systems, including drawings. For all vital areas, the FSAR should address the environmental conditions including weather, moisture, noise level, and electromagnetic interference/radiofrequency interference that might interfere with effective communication. Environmental conditions also include fire and radiological events in which personnel must be able to communicate effectively while equipped with respiratory protection.

Chapter 7.9 of this Writer's Guide offers recommendations for data communication systems.

Modified Section/Title: C.I.9.5.2.3; Inspection and Testing Requirements
Original Section/Title: C.I.9.5.2.3; Inspection and Testing Requirements
The applicant should provide inspection and testing requirements and any associated inspection/test procedures for the communication systems.

Modified Section/Title: C.I.9.5.3; Lighting Systems
Original Section/Title: C.I.9.5.3; Lighting Systems
The applicant should provide a description of the plant's normal, emergency, and supplementary lighting systems, including the capability of these systems to provide adequate lighting during all plant operating conditions (e.g., normal operation and anticipated fire, transient, and accident conditions). Discuss the effect of a loss of all alternating current power (i.e., during a SBO event) on emergency lighting systems. In the description of these lighting systems, include the following considerations:

- Design criteria
- Provisions for lighting needed in areas required for firefighting
- Provisions for lighting needed in areas for control and maintenance of safety-related equipment
- Access routes to and from these areas
- A failure analysis.

Modified Section/Title: C.I.9.5.4; Backup Power Generator Support Systems
Original Section/Title: C.I.9.5.4; Diesel Generator Fuel Oil Storage and Transfer System
See sections 9.5.4.1 thru 9.5.4.3.

Modified Section/Title: C.I.9.5.4.1; Design Bases
Original Section/Title: C.I.9.5.4.1; Design Bases
The applicant should provide the design bases for the fuel oil storage and transfer system for the diesel generators, including the requirement for onsite storage capacity, capability to meet code design requirements, capability to detect and control system leakage, and environmental design bases. The applicant should also describe, as applicable, any diesel generator cooling water, air start, and lubricating systems.

Modified Section/Title: C.I.9.5.4.2; System Description
Original Section/Title: C.I.9.5.4.2; System Description
The applicant should provide a description and drawings of the diesel generator fuel oil storage and transfer system and, if applicable, diesel generator cooling water, air start, and lubricating systems.

Modified Section/Title: C.I.9.5.4.3; Safety Evaluation
Original Section/Title: C.I.9.5.4.3; Safety Evaluation
If the fuel oil storage and transfer system performs any safety functions, then the applicant should provide an evaluation of the fuel oil storage and transfer system. This evaluation should include, as applicable to the safety function, the potential for material corrosion and fuel oil contamination, a failure analysis to demonstrate the system's capability to meet design criteria (e.g., seismic requirements, capability to perform its function in the event of SBO, implications of sharing between units at a multi unit site, ability to meet independence and redundancy requirements for onsite electric power supplies), ability to withstand environmental design conditions, external and internal missiles and forces associated with pipe breaks, and the plans for procuring additional fuel oil and recharging storage tanks, if necessary.

If the diesel generator cooling water, air start, or lubricating systems are used and perform any safety functions, then the applicant should provide an evaluation of the system. This evaluation should include, as applicable to the safety function, a failure analysis to demonstrate the system's capability to meet design criteria (e.g., seismic requirements, capability to perform its function in the event of SBO, implications of sharing between units at a multi unit site, ability to meet independence and redundancy requirements for onsite electric power supplies), ability to withstand environmental design conditions, external and internal missiles and forces associated with pipe breaks.

Modified Section/Title: C.I.9.5.4.4; Inspection and Testing Requirements
Original Section/Title: C.I.9.5.4.4; Inspection and Testing Requirements
The applicant should describe the test and inspection procedures for the diesel generator fuel oil storage and transfer system and, if applicable, diesel generator cooling water, air start and lubricating systems.

Modified Section/Title: C.I.9.5.5; N/A
Original Section/Title: C.I.9.5.5; Diesel Generator Cooling Water System
N/A

Modified Section/Title: C.I.9.5.5.1; N/A
Original Section/Title: C.I.9.5.5.1; Design Bases
N/A

Modified Section/Title: C.I.9.5.5.2; N/A
Original Section/Title: C.I.9.5.5.2; System Description
N/A

Modified Section/Title: C.I.9.5.5.3; N/A
Original Section/Title: C.I.9.5.5.3; Safety Evaluation
N/A

Modified Section/Title: C.I.9.5.5.4; N/A
Original Section/Title: C.I.9.5.5.4; Inspection and Testing Requirements
N/A

Modified Section/Title: C.I.9.5.6; N/A
Original Section/Title: C.I.9.5.6; Diesel Generator Starting Air System
N/A

Modified Section/Title: C.I.9.5.6.1; N/A
Original Section/Title: C.I.9.5.6.1; Design Bases
N/A

Modified Section/Title: C.I.9.5.6.2; N/A
Original Section/Title: C.I.9.5.6.2; System Description
N/A

Modified Section/Title: C.I.9.5.6.3; N/A
Original Section/Title: C.I.9.5.6.3; Safety Evaluation
N/A

Modified Section/Title: C.I.9.5.7; N/A
Original Section/Title: C.I.9.5.7; Diesel Generator Lubrication System
N/A

Modified Section/Title: C.I.9.5.7.1; N/A
Original Section/Title: C.I.9.5.7.1; Design Basis
N/A

Modified Section/Title: C.I.9.5.7.2; N/A
Original Section/Title: C.I.9.5.7.2; System Description
N/A

Modified Section/Title: C.I.9.5.7.3; N/A
Original Section/Title: C.I.9.5.7.3; Safety Evaluation
N/A

Modified Section/Title: C.I.9.5.8; N/A
Original Section/Title: C.I.9.5.8; Diesel Generator Combustion Air Intake and Exhaust System
N/A

Modified Section/Title: C.I.9.5.8.1; N/A
Original Section/Title: C.I.9.5.8.1; Design Bases
N/A

Modified Section/Title: C.I.9.5.8.2; N/A
Original Section/Title: C.I.9.5.8.2; System Description
N/A

Modified Section/Title: C.I.9.5.8.3; N/A
Original Section/Title: C.I.9.5.8.3; Safety Evaluation
N/A

Modified Section/Title: C.I.9.5.8.4; N/A
Original Section/Title: C.I.9.5.8.4; Inspection and Testing Requirements
N/A

Appendix H

Chapter 15. Transient and Accident Analysis

Appendix H
Chapter 15. Transient and Accident Analysis

Table of Contents

Mod Subsection	Mod Subject	Orig Subsection	Orig Subject
C.I.15	Transient and Accident Analyses	C.I.15	Transient and Accident Analyses
C.I.15.1	Transient and Accident LBE Classification	C.I.15.1	Transient and Accident Classification
C.I.15.2	Frequency of Occurrence	C.I.15.2	Frequency of Occurrence
C.I.15.3	Plant Characteristics Considered in the Safety Evaluation	C.I.15.3	Plant Characteristics Considered in the Safety Evaluation
C.I.15.4	Expected and Assumed Plant Response	C.I.15.4	Assumed Protection System Actions
C.I.15.5	Evaluation of Individual LBE Sequences (DBEs and DBAs)	C.I.15.5	Evaluation of Individual Initiating Events
C.I.15.6	Event Evaluation (DBEs and DBAs)	C.I.15.6	Event Evaluation
C.I.15.6.1	Identification of Initiating Event Causes and Number of Reactor Modules Affected	C.I.15.6.1	Identification of Causes and Frequency Classification
C.I.15.6.2	Sequence of Events and Plant Response	C.I.15.6.2	Sequence of Events and Systems Operation
C.I.15.6.3	Functional Containment Performance (Fuel, HPB, and RB Barriers)	C.I.15.6.3	Core and System Performance
C.I.15.6.3.1	Evaluation Model	C.I.15.6.3.1	Evaluation Model
C.I.15.6.3.2	Input Parameters and Initial Conditions	C.I.15.6.3.2	Input Parameters and Initial Conditions
C.I.15.6.3.3	Results	C.I.15.6.3.3	Results
C.I.15.6.4	N/A	C.I.15.6.4	Barrier Performance
C.I.15.6.4.1	N/A	C.I.15.6.4.1	Evaluation Model
C.I.15.6.4.2	N/A	C.I.15.6.4.2	Input Parameters and Initial Conditions
C.I.15.6.4.3	N/A	C.I.15.6.4.3	Results
C.I.15.6.5	Radiological Consequences	C.I.15.6.5	Radiological Consequences
C.I.15 - Appendix A	Risk-Informed Event Sequences to be Analyzed	C.I.15 - Appendix A	Representative Initiating Events to be Analyzed
C.I.15 - Appendix B	Typical Input Parameters and Initial Conditions For Transients and Accidents	C.I.15 - Appendix B	Typical Input Parameters and Initial Conditions For Transients and Accidents
C.I.15 - Appendix C	Representative Parameters to be Tabulated for Accident Analyses	C.I.15 - Appendix C	Representative Parameters to be Tabulated for Postulated Accident Analyses

Mod Subsection	Mod Subject	Orig Subsection	Orig Subject
C.I.15 - Appendix D	Applicability TBD Referenced in the Summary Tabulations for Specific Design-Basis Accidents	C.I.15 - Appendix D	Additional Parameters and Information to be Provided or
C.I.15 - Appendix E	Applicability TBD	C.I.15 - Appendix E	Summary of Initial Conditions and Computer Codes Used
C.I.15 - Appendix F	Applicability TBD	C.I.15 - Appendix F	Nominal Values of Pertinent Plant Parameters Used in the Accident Analyses
C.I.15 - Appendix G	Applicability TBD	C.I.15 - Appendix G	Safety Analysis RPS and ESFAS Trip Setpoints and Delay Times
C.I.15 - Appendix H	Applicability TBD	C.I.15 - Appendix H	Single Failures
C.I.15 - Appendix I	Applicability TBD	C.I.15 - Appendix I	Limiting Single Failures Assumed in Transient and Accident Analyses
C.I.15 - Appendix J	Applicability TBD	C.I.15 - Appendix J	Nonsafety-Related System and Equipment Used To Mitigate Transients and Accident