Draft for Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR NuScale SMR DESIGN

5.3.2 PRESSURE-TEMPERATURE LIMITS, UPPER-SHELF ENERGY, AND PRESSURIZED THERMAL SHOCK

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of component integrity issues related to reactor vessels

Secondary - None

I. AREAS OF REVIEW

Integral pressurized-small modular reactors (SMRs) generally incorporate the reactor core, and the pressurizer inside the reactor vessel. One or more steam generators (and/or reactor coolant pumps) may be inside the reactor vessel or directly connected to the reactor vessel. For the purpose of this review, the applicant should provide an accurate definition of the reactor coolant pressure boundary, the reactor vessel and its construction.

The staff will review the application with respect to the regulations concerning, (1) pressure-temperature (P-T) limits on maintaining the reactor coolant pressure boundary (RCPB), (2) reactor vessel beltline Charpy upper-shelf energy (USE), and (3) assessment of potential pressurized thermal shock (PTS) (pressurized-water reactor (PWR) only).

The specific areas of review are as follows:

1. Pressure-Temperature Limits/Upper-Shelf Energy/Pressurized Thermal Shock

   A. Pressure-Temperature Limits. The staff will review the P-T limits imposed on the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests, under this section of the Design-Specific Review Standard (DSRS) to ensure adequate safety margins of structural integrity for the ferritic components of the RCPB.

   The regulations in Title 10 of the *Code of Federal Regulations* (CFR), Section 50.60 and associated Appendix G to 10 CFR Part 50, describe the conditions that require P-T limits and provide the general basis for these limits.

   B. Upper-Shelf Energy. The staff will review reactor vessel beltline materials, which must have Charpy USE values of no less than 102 joules (J) (75 foot-pounds (ft-lb)) initially and must maintain USE values throughout the life of the vessel of no less than 68 J (50 ft-lb). If a material's USE values are projected to be less than 68 J, a safety analysis must be performed that will provide margins of safety against fracture equivalent to those described by Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel

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Code (hereinafter referred to as ASME Section XI). Reactor vessel beltline Charpy USE drop may be estimated using Regulatory Guide (RG) 1.99 when surveillance data are not available or are not applicable.

C. Pressurized Thermal Shock. The staff will evaluate iPWR reactor vessel beltline materials to ensure adequate resistance to failure during PTS events. The staff will consider the reference temperature (RT_{PTS}) calculations and screening criterion and, if the RT_{PTS} value is projected to exceed the PTS screening criterion before the expiration date of the license, any associated safety analyses performed to support reactor operation. Projected values of RT_{PTS} for iPWR reactor vessel beltline materials are determined in accordance with 10 CFR 50.61.

2. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this DSRS section in accordance with Standard Review Plan (SRP) Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria," and DSRS Section 14.3.4, "Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this DSRS section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3 and DSRS Section 14.3.4.

3. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other SRP and DSRS sections interface with this section as follows:

1. Review of the material characteristics of the RCPB and the reactor vessel, including fracture toughness properties and the material surveillance program, is performed under SRP Section 5.2.3 and DSRS Section 5.3.1.

2. Review of the overpressure protection system for consistency with the P-T limits in Appendix G to 10 CFR Part 50 is performed under DSRS Section 5.2.2.

3. Review of the peak reactor vessel wall fluence for the design life of the plant is performed under DSRS Section 4.3.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 50.55a, as it relates to quality standards for the design, fabrication, erection, and testing of SSCs that are safety-related or risk-significant;

2. 10 CFR 50.60, as it relates to compliance with the requirements of Appendix G to 10 CFR Part 50;

3. 10 CFR 50.61, as it relates to fracture toughness criteria for iPWRs relevant to PTS events;

4. General Design Criterion (GDC) 1, found in Appendix A to 10 CFR Part 50, as it relates to quality standards for design, fabrication, erection, and testing;

5. GDC 14, as it relates to ensuring an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture of the RCPB;

6. GDC 31, as it relates to ensuring that the RCPB will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized;

7. GDC 32, as it relates to the reactor vessel materials surveillance program;

8. Appendix G to 10 CFR Part 50, as it relates to material testing and fracture toughness;

9. Appendix H to 10 CFR Part 50, as it relates to material testing and fracture toughness;

10. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission’s (NRC's) regulations; and

10\. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the COL, the provisions of the AEA and the NRC's regulations.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC’s regulations identified above are set forth below. The DSRS is not a substitute for the NRC’s regulations, and compliance with it is not required. As an alternative, and as described in more

detail below, an applicant may identify the differences between a DSRS section and the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and discuss how the proposed alternative provides an acceptable method of complying with the NRC regulations that underlie the DSRS acceptance criteria.

1. Pressure-Temperature Limits

A. Applicable Regulations, Codes, and Basis Documents. The regulations in 10 CFR 50.60 and associated Appendix G to 10 CFR Part 50 describe the conditions that require P-T limits and provide the general basis for these limits. Appendix G specifically requires that P-T limits must be at least as conservative as limits obtained by following Appendix G to ASME Section XI during heatup, cooldown, and test conditions. Appendix G to 10 CFR Part 50 also requires additional safety margins when the reactor core is critical.

Although Appendix G to ASME Section III is usually referenced with regard to facility design and construction, the reviewer should instead apply the provisions of Appendix G to ASME Section XI when using this DSRS. The following provide the rationale for using Appendix G to ASME Section XI instead of Appendix G to Section III of the ASME Code:

i. Appendix G to 10 CFR Part 50 specifically references Appendix G to Section XI to the ASME Code, and Appendix G to Section III to the ASME Code contains similar provisions.

ii. The differences between Appendix G to ASME Section XI and Appendix G to ASME Section III have resulted from a series of ASME code cases, including N-588, N-640, and N-641. Appendix G to ASME Section III has not been updated since those code cases were developed. However, the staff expects that Appendix G of ASME Section III will be updated to be consistent with Appendix G to ASME Section XI.

B. Pressure-Temperature Requirements. Appendix G to 10 CFR Part 50 requires that the P-T limits defined in that appendix be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of ASME Section XI, as stated below:

i. Pressure-Temperature Limits for Preservice Hydrostatic Tests

During preservice hydrostatic tests (if fuel is not in the vessel), a material's lower bound static crack initiation fracture toughness, K_{Ic}, must be greater than the K_I caused by pressure stresses acting on a defined, conservative hypothetical flaw, as shown in the following expression:

$$K_{applied} = K_I(\text{pressure}) < K_{Ic}$$

ii. Pressure-Temperature Limits for Inservice Leak and Hydrostatic Tests

During performance of inservice leak and hydrostatic tests, a material's K_{Ic} must be greater than 1.5 times the K_I caused by pressure, as shown in the following expression:

$$K_{applied} = 1.5\ K_I(\text{pressure}) < K_{Ic}$$

iii. Pressure-Temperature Limits for Heatup and Cooldown Operations

At all times during heatup and cooldown operations, a material's K_{Ic} must be greater than the sum of 2 times the K_I caused by pressure and the K_I caused by thermal gradients, as shown in the following expression:

$$K_{applied} = 2K_I(\text{pressure}) + K_I(\text{thermal}) < K_{Ic}$$

iv. Pressure-Temperature Limits for Core Critical Operation

At all times that the reactor core is critical (except for low-power physics tests), the temperature must be higher than that required for inservice hydrostatic testing. In addition, the P-T relationship must provide at least a 22 °C (40 °F) margin over that required for heatup and cooldown operations.

2. Upper-Shelf Energy

A. Applicable Regulations, Codes, and Basis Documents. Appendix G to 10 CFR Part 50 requires that reactor vessel beltline materials have a Charpy USE value in the transverse direction for base material and along the weld for weld material according to the ASME Code of no less than 102 J (75 ft-lb) initially and must maintain a Charpy USE value throughout the life of the vessel of no less than 68 J (50 ft-lb), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation (NRR), that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of ASME Section XI.

B. Upper-Shelf Energy Requirements. Appendix G to 10 CFR Part 50 contains the following USE requirements:

i. Initially, the USE value in the transverse direction for base material and along the weld must not be less than 102 J (75 ft-lb).

ii. Charpy USE throughout the life of the vessel must be maintained at no less than 68 J (50 ft-lb), unless it is demonstrated in a manner approved by the Director, NRR, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of ASME Section XI.

3. Pressurized Thermal Shock

A. Applicable Regulations, Codes, and Basis Documents. Projected values of RT_{PTS} must be determined for iPWR reactor vessel beltline materials in accordance with 10 CFR 50.61. For RT_{PTS} values projected to exceed the screening criteria, safety analyses must be provided that include proposed flux reduction programs or other corrective actions to prevent potential PTS-related

failure of the reactor vessel if continued plant operation beyond the screening criterion is allowed.

B. Pressurized Thermal Shock Requirements. In accordance with 10 CFR 50.61, values of RT_{PTS} projected using the methods of 10 CFR 50.61 for the time of the initial application submittal and for the projected expiration date of the operating license must not exceed the screening criteria of 132 °C (270 °F) for plates, forgings, and axial weld materials, and 149 °C (300 °F) for circumferential weld materials, throughout the facility's licensed operating permit. This assessment must be updated whenever projected values of RT_{PTS} change significantly, or upon request for a change in the expiration date for operation of the facility. For RT_{PTS} values projected to exceed the screening criteria, safety analyses must be provided that include proposed flux reduction programs or other corrective actions to prevent potential PTS-related failure of the reactor vessel if continued plant operation beyond the screening criterion is allowed.

4. 10 CFR 52.47(b)(1) requires that a DC application contain proposed ITAAC necessary and sufficient to assure the plant is built and will operate in accordance with the DC. 10 CFR 52.80(a) requires that the COL identify the ITAAC necessary and sufficient to assure that the facility has been constructed and will be operated in conformity with the license. SRP Section 14.3 provides guidance for reviewing the ITAAC. The requirements of 10 CFR 52.47(b)(1) and 10 CFR 52.80(a) will be met, in part, by identifying ITAAC of the top-level design features with respect to the regulations concerning, (1) P-T limits on maintaining the reactor coolant pressure boundary (RCPB), (2) reactor vessel beltline Charpy USE, and (3) assessment of potential PTS in the DC and COL applications, respectively.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this DSRS section is discussed in the following paragraphs:

1. GDC 1 and 10 CFR 50.55a establish quality assurance requirements for the design, fabrication, erection, and testing of SSCs important to safety. GDC 1 establishes that the quality standards to be applied to SSCs shall be commensurate with the importance of the safety functions to be performed. 10 CFR 50.55a establishes, in relevant part, those provisions of the ASME Code that must be complied with to ensure that SSCs are designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. The RCPB's primary safety functions include preventing a loss of reactor coolant through leakage or gross failure of RCPB piping or components, and acting as a containment barrier to the release of fission products in the event of an accident resulting in fuel damage. In accordance with Appendix G to ASME Section III , P-T limits are established for the RCPB to ensure the satisfaction of the RCPB material fracture toughness requirements. Compliance with GDC 1 and 10 CFR 50.55a provides assurance that the RCPB meets the appropriate quality standards of the ASME Code, and thus that the probability of RCPB material failure and the subsequent effects on reactor core cooling and confinement are minimized.

2. GDC 14 establishes that the RCPB must be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating

failure, and of gross rupture. The RCPB provides for the confinement of reactor coolant and acts as a barrier to the release of fission products in the event of an accident resulting in fuel failure. The P-T limits established for the RCPB ensure that the material fracture toughness requirements for the RCPB piping and components are met and that the RCPB will act in a nonbrittle manner under operating, maintenance, testing, and postulated accident conditions. Application of GDC 14 to the RCPB, with regard to the P-T limits, provides assurance that the RCPB meets the material fracture toughness requirements and will act in a nonbrittle manner, thereby providing a low probability of significant degradation or of gross failure of the RCPB that could cause a loss of reactor coolant inventory and a reduction in the capability to confine fission products.

3. GDC 31 establishes that the RCPB must be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. The design must reflect consideration of service temperatures and other conditions of the boundary material and the uncertainties in determining material properties; the effects of irradiation on material properties; residual, steady-state, and transient stresses; and the size of flaws. The RCPB provides a fission product barrier, confinement of reactor coolant, and flowpaths to facilitate core cooling. RG 1.99 provides methods for predicting irradiation effects on fracture toughness properties that are applicable to compliance with the requirements of GDC 31. Application of GDC 31 ensures that the P-T limits for the RCPB are appropriately determined and provide sufficient margin to account for uncertainties associated with flaws and the effects of service and operating conditions, and thereby provide a minimum probability of brittle material behavior leading to rapidly propagating failure. The probability of substantial reduction in the capability to contain reactor coolant inventory, reduction in the capability to confine fission products, and interference with core cooling is thereby minimized.

4. 10 CFR 50.60 requires that all light-water nuclear power reactors meet the fracture toughness requirements, including P-T limits, as set forth in Appendix G to 10 CFR Part 50. Compliance with the requirements of this rule and Appendix G provides assurance regarding the structural integrity of the RCPB and, specifically, the reactor vessel. The next item discusses the technical rationale for this rule.

5. Appendix G to 10 CFR Part 50 establishes that the pressure-retaining components of the RCPB that are made of ferritic materials must meet requirements of the ASME Code, supplemented by the additional requirements set forth in Appendix G to 10 CFR Part 50 for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. Fracture toughness properties of ferritic materials increase significantly above the point referred to as the nil-ductility transition temperature. This temperature is established for the RCPB material in accordance with ASME Section XI, as supplemented by the requirements of Appendix G to 10 CFR Part 50. The P-T limits established in accordance with the ASME Code and Appendix G to 10 CFR Part 50 are used to establish operating parameters that provide assurance that the RCPB will act in a nonbrittle manner when subjected to stresses associated with normal operations, maintenance, testing, and anticipated operational occurrences. The P-T limits must be adjusted to account for the effects of radiation embrittlement of the RCPB materials over the life of the plant. Compliance with the requirements of Appendix G provides a method of satisfying the requirements of GDCs 14 and 31 with regard to ensuring that the RCPB

acts in a nonbrittle manner and that the probability of rapidly propagating failure and gross rupture of the RCPB is extremely low.

6. 10 CFR 50.61 establishes fracture toughness requirements for protection against PTS events, which involve transients in iPWRs that cause severe overcooling in conjunction with overpressurization. The thermal stresses in combination with the pressure stresses increase the potential for brittle fracture in the presence of an initiating flaw in material with low toughness. This material may be present in the reactor vessel beltline where neutron radiation gradually embrittles the material over the plant lifetime. The PTS rule provides calculational methods and acceptance criteria for determining the effect of embrittlement on the reactor vessel materials and for establishing the material reference temperature limits beyond which corrective actions and plant-specific safety analyses must justify continued operation of the plant. Establishing, monitoring, and maintaining the structural integrity of the reactor vessel materials are essential in protecting against a failure of the RCPB and the subsequent loss of core cooling and fission product containment. Compliance with the requirements of 10 CFR 50.61 provides assurance that the reactor vessel materials will not be subject to failure from PTS during the life of the reactor.

III. REVIEW PROCEDURES

These review procedures are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Selected Programs and Guidance - In accordance with the guidance in NUREG-0800, "Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Integral Pressurized Water Reactor Edition" (NUREG-0800 Intro Part 2) as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800 Intro Part 2, the NRC requirements that must be met by an SSC do not change under the SMR framework. Using the graded approach described in NUREG-0800 Intro Part 2, the NRC staff may determine that, for certain structures, systems, and components (SSCs), the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified where applicable as part of completion of the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is described in Figure 1 of NUREG-0800, Introduction - Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

    - 10 CFR Part 50, Appendix A, General Design Criteria (GDC), Overall Requirements, Criteria 1 through 5
    - 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
    - 10 CFR 50.49, Environmental Qualification of Electrical Equipment (EQ) Program
    - 10 CFR 50.55a, Code Design, Inservice Inspection and Inservice Testing (ISI/IST) Programs

- 10 CFR 50.65, Maintenance Rule requirements
- Reliability Assurance Program (RAP)
- 10 CFR 50.36, Technical Specifications
- Availability Controls for SSCs Subject to Regulatory Treatment of Non-Safety Systems (RTNSS)
- Initial Test Program (ITP)
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

This list of examples is not intended to be all-inclusive. It is the responsibility of the technical reviewers to determine whether the information in the application, including the degree to which the applicant seeks to rely on such selected programs and guidance, demonstrates that all acceptance criteria have been met to support the safety finding for a particular SSC.

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), and 10 CFR 52.79(a)(17), (20) and (37), for design certification or combined license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG 0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v) for a DC application, and except paragraphs (f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v) for a COL application. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.

3. <u>Construction Permit/Design Certification Reviews</u>. The staff will review the information in the technical submittal for a commitment that the fracture toughness of the ferritic materials in the RCPB will comply with the requirements of Appendix G to 10 CFR Part 50, as detailed in ASME Section XI, and that the materials in the beltline region of the reactor vessel will comply with the requirements of 10 CFR 50.61 and the guidance of RG 1.99.

4. <u>Operating License/Combined License Reviews</u>. The plant technical specifications or pressure-temperature limits report should show the P-T limits using real temperature. The staff will review these curves and their bases to determine acceptability in the following areas:

   A. The limiting RT_{NDT} has been properly determined and radiation effects are included in an acceptable manner.

   B. Limits are shown for all required conditions and provide all required information.

   C. The limits proposed are consistent with the acceptance criteria described in Section II above.

5. <u>Acceptability Determination Methods</u>

A. Pressure-Temperature Limits. The reviewer will perform an independent evaluation of one or more proposed P-T limit curves. The reviewer will base this evaluation on the methodology for constructing P-T limit curves found in Appendix G to Section XI of the most recent edition and addenda of the ASME Code that 10 CFR 50.55a has endorsed. The reviewer will also apply the additional minimum temperature requirements specified in Appendix G to 10 CFR Part 50.

For checking any P-T limit curve, the following steps describe the general form of the staff's evaluation.

i. Verify what each axis of the P-T limit plot represents. "Temperature," normally given on the horizontal axis, may be either the reactor coolant system fluid temperature (most common) or the metal temperature of the vessel. "Pressure," normally given on the vertical axis, is the system pressure but may be given in absolute or gauge values.

The reviewer should also check to see whether the curves include pressure and/or temperature measurement uncertainties have been included in the curves. If so, these must be removed before the evaluation outlined below will give comparable results.

ii. Determine the reference temperature (RT_{NDT}) at the 1/4 thickness (1/4 T) and 3/4 T locations for each vessel beltline material. For preservice hydrostatic testing curves, this determination shall be based on the initial material properties for each material determined in accordance with ASME Section III, NB-2331 or Branch Technical Position (BTP) 5-3.

For all other curves (inservice leak/hydrostatic testing, heatup, cooldown, core critical operation), a period of applicability should be specified, usually in effective full-power years (EFPYs) of operation. This should be specified, along with the 1/4 T and 3/4 T neutron fluence (E > 1.0 MeV) at the EFPY value for each beltline material, to account for the effects of radiation on the material properties of each beltline material. The RT_{NDT} values at the 1/4 T and 3/4 T locations for each beltline material through the end of the specified period of applicability should be determined in accordance with RG 1.99.

iii. The following fundamental equation should be satisfied at each P-T point along any P-T limit curve:

$$K_{I\ applied} = K_{Ic}$$

where:

$K_{I\ applied}$ = The stress intensity due to pressure (membrane) and thermal gradient (bending) loads at the tip of the 1/4 T defect postulated in Appendix G to ASME Section XI.

K_{Ic} = The lower bound, plane strain, crack initiation fracture toughness for the material as represented in Figure 1.

iv. $K_{I\ applied}$ shall be calculated from an equation of the general form:

$$K_{I\ applied} = SF * M_m * (p * R_i / t) + K_{I\ thermal}$$

where:

SF = A structural factor applied to the pressure loading as specified in Appendix G to ASME Section XI and dependent on which P-T curve is being evaluated.

M_m = An influence coefficient to convert applied stress to crack tip stress intensity. M_m depends on the orientation of the flaw being evaluated (axial flaws for plates, forgings, and axial welds; circumferential flaws for circumferential welds) and the thickness of the material. Appendix G to ASME Section XI specifies values for M_m.

p = The pressure at the specified condition.

R_i = The vessel inside radius.

t = The vessel wall thickness.

$K_{I\ thermal}$ = The stress intensity at the crack tip due to thermal loadings (which are only considered for heatup, cooldown, and core critical operation curves). $K_{I\ thermal}$ may be conservatively calculated from equations given in Appendix G to ASME Section XI, which depend on heatup/cooldown rate and the vessel thickness. $K_{I\ thermal}$ may also be more accurately obtained from ORNL/NRC/LTR-03/03 for a given heatup/cooldown rate and the vessel thickness.

v. K_{Ic} is determined from Figure 1 (taken from Appendix G to ASME Section XI). K_{Ic} is a function of a material's RT_{NDT} value and temperature at the location of interest (i.e., either the 1/4 T or 3/4 T location).

It should be noted that a material's RT_{NDT} value will vary through the wall thickness as the neutron fluence decreases from the vessel inside diameter to the vessel outside diameter. The reviewer should apply the methods of RG 1.99 for determining the appropriate RT_{NDT} values.

It should also be noted that the temperature to be applied in using Figure 1 is the <u>metal</u> temperature at the tip of the postulated flaw from Appendix G to ASME Section XI (i.e., at either the 1/4 T or 3/4 T location). The metal temperature at throughwall locations will depend on the reactor coolant system fluid temperature and the rate of change of the reactor coolant system fluid temperature. Throughwall metal temperatures can

be determined from methods given in Appendix G to ASME Section XI or from ORNL/NRC/LTR-03/03.

vi. Based on the discussion above, it is recommended that the following equation be solved to determine the allowable pressure associated with a specified temperature along a P-T limit curve:

$$\text{allowable pressure} = t * (K_{Ic} - K_{I\ thermal}) / (SF * M_m * R_i)$$

which is an algebraic rearrangement of the equation from (iv). The reviewer should keep in mind; however, that four of the quantities (K_{Ic}, $K_{I\ thermal}$, SF, and M_m) are dependent on other, more basic variables or conditions:

K_{Ic}—Depends on metal temperature and material RT_{NDT}

$K_{I\ thermal}$—Depends on heatup/cooldown rate and vessel wall thickness

SF—Depends on the curve being evaluated and the assumed flaw orientation

M_m—Depends on the vessel wall thickness and the assumed flaw orientation

vii. The reviewer should verify that all minimum temperature requirements specified in Appendix G to 10 CFR Part 50 for the P-T limit curve being verified have been met. These requirements have been imposed to ensure that highly stressed, nonbeltline regions (i.e., the vessel flange region) are protected from brittle failure.

viii. It should also be noted that some applications may provide P-T limit curves for other specific nonbeltline regions (e.g., the nozzle course and/or the bottom head of a boiling-water reactor) to address specific modes of operation. These nonbeltline curves are normally submitted to provide additional operational/testing flexibility and are generally less restrictive than the corresponding beltline curve. The development of these curves necessitates the evaluation of complex geometries where discontinuities are present. No simple review procedure can be specified for the review of such curves. However, the reviewer may use the guidance provided in Appendix G to ASME Section XI and Welding Research Council Bulletin 175 for such situations.

B. Upper-Shelf Energy. The reviewer will evaluate the initial Charpy USE values for the reactor vessel materials in accordance with the acceptance criterion specified in paragraph IV.A.1.a of Appendix G to 10 CFR Part 50. Reactor vessel materials that do not meet the specified initial Charpy USE acceptance criterion shall be evaluated in accordance with the provisions for additional analysis also specified in paragraph IV.A.1.a. In addition to the ASME Code, RG 1.161 provides an acceptable methodology for the performance of analyses intended to meet the provisions for the additional analysis in paragraph IV.A.1.a.

The reviewer will also evaluate the end-of-license (EOL) Charpy USE values for the reactor vessel materials in accordance with the acceptance criterion specified in paragraph IV.A.1.a of Appendix G to 10 CFR Part 50. RG 1.99 provides guidance on evaluating Charpy USE drop and, hence, USE values. Reactor vessel materials that do not meet the projected EOL Charpy USE minimum 68 J (50 ft-lb) criterion shall be evaluated in accordance with the provisions for additional analysis also specified in paragraph IV.A.1.a. In accordance with paragraph IV.A.1.c., this analysis must be submitted to the staff for review and approval on an individual case basis at least 3 years prior the date on which the predicted Charpy USE will no longer satisfy the requirements of paragraph IV.A.1.a, or on a schedule approved by the Director, NRR. In addition to the ASME Code, RG 1.161 provides an acceptable methodology for the performance of analyses intended to meet the provisions for additional analysis specified in paragraph IV.A.1.a.

C. Pressurized Thermal Shock. The reviewer will evaluate the projected values for RT_{PTS}, including the calculational methods and assumptions, and compare the projected values with the screening criteria in 10 CFR 50.61. For each iPWR where the RT_{PTS} value for any material in the beltline is projected to exceed the PTS screening criterion before the expiration date of the operating license, the licensee should submit an analysis and schedule for the implementation of flux reduction programs that are reasonably practical to avoid exceeding the PTS screening criterion. If the analysis indicates that no reasonably practical flux reduction program will prevent the value of RT_{PTS} from exceeding the PTS screening criterion before the expiration date of the operating license, the licensee can choose between the two options in 10 CFR 50.61 to meet PTS requirements. The licensee can submit a safety analysis to determine the modifications necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. The staff will review these safety analyses against the requirements of 10 CFR 50.61. Alternatively, the licensee can perform a thermal-annealing treatment of the reactor vessel pursuant to 10 CFR 50.61(b)(7) to recover fracture toughness. In accordance with 10 CFR 50.61, the licensee must submit for NRC approval details of the approach selected at least 3 years before the reactor vessel is projected to exceed the PTS screening criteria.

For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's technical review and calculations (if applicable) support conclusions of the following type to be included in the staff's SER. The reviewer also states the bases for those conclusions.

The pressure-temperature limits imposed on the reactor coolant system for all operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure are in conformance with the fracture toughness criteria of Appendix G to 10 CFR Part 50 and Appendix G, "Protection Against Nonductile Failure" to Section XI of the ASME Boiler and Pressure Vessel Code. The applicant has adequately addressed the upper-shelf energy criteria in accordance with Appendix G to 10 CFR Part 50 and thermal shock events in accordance with 10 CFR 50.61. The staff concludes that the use of operating limits, based upon the criteria defined in DSRS Section 5.3.2, provides reasonable assurance that nonductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the applicable requirements of 10 CFR 50.55a, 10 CFR 50.60, 10 CFR 50.61 and GDCs 1, 14, and 31 of Appendix A to 10 CFR Part 50.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The regulations in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), and 10 CFR 52.79(a)(41) establish requirements for applications for ESPs, DCs, and COLs, respectively. These regulations require the application to include an evaluation of the site (ESP), standard plant design (DC), or facility (COL) against the Standard Review Plan (SRP) revision in effect six months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed small modular reactor (SMR) designs, however, differ significantly from large light-water nuclear reactor power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405) (SRM). In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated pre-application activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for the evaluation of a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address

the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether an application complies with NRC regulations for NuScale SMR applications, provided that the application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. If the design or siting assumptions in a NuScale application deviate significantly from the design and siting assumptions the staff used in preparing the DSRS, the staff will use the more general guidance in the SRP as specified in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), or 10 CFR 52.79(a)(41), depending on the type of application. Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design or siting assumptions.

VI. REFERENCES

1. 10 CFR 52.47, "Contents of Applications."

2. 10 CFR 50.55a, "Codes and Standards."

3. 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation."

4. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

5. 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records."

6. 10 CFR Part 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary."

7. 10 CFR Part 50, Appendix A, GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

8. 10 CFR Part 50, Appendix A, GDC 32, "Inspection of Reactor Coolant Pressure Boundary."

9. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

10. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

11. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

12. 10 CFR 52.97, "Issuance of Combined Licenses."

13. ASME Boiler and Pressure Vessel Code, Section III, including Appendix G, “Protection Against Nonductile Failure,” American Society of Mechanical Engineers.

14. ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” American Society of Mechanical Engineers.

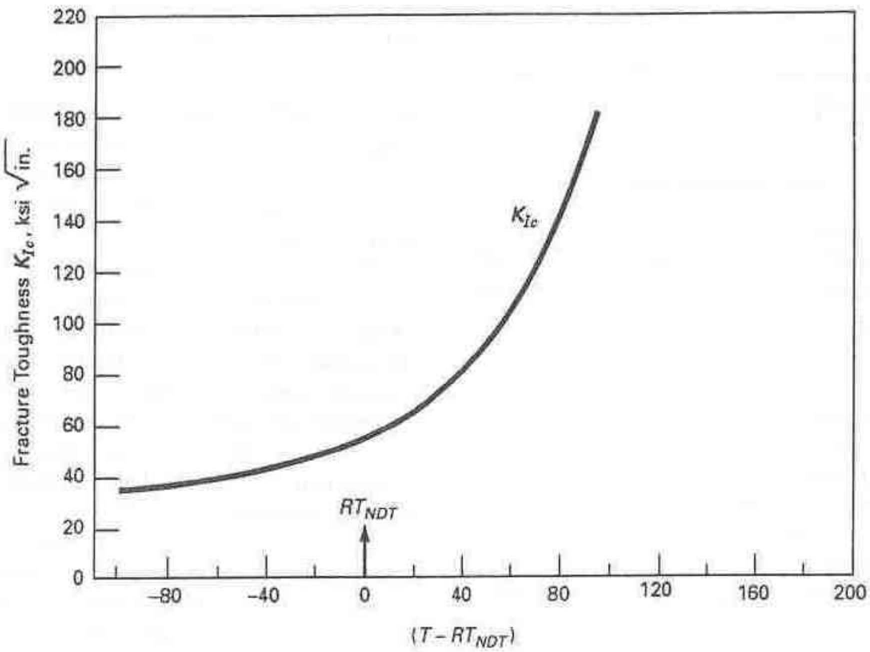
15. BTP 5-3, “Fracture Toughness Requirements.”

16. ORNL/NRC/LTR-03/03, “Tabulation of Thermally-Induced Stress Intensity Factors (K_{IT}) and Crack Tip Temperatures for Generating P-T Curves per ASME Section XI - Appendix G.”

17. RG 1.99, “Radiation Embrittlement of Reactor Vessel Materials.”

18. RG 1.161, “Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb.”

19. Welding Research Council Bulletin 175, “PVRC Recommendation on Fracture Toughness,” Welding Research Council, Pressure Vessel Research Committee Ad Hoc Group on Toughness Requirements, August 1972.



$$K_{Ic}=33.2+20.734 \exp [0.02(T-RT_{NDT})]$$

Temperature Relative to RT_{NDT}, $(T-RT_{NDT})$, Fahrenheit Degrees

FIGURE 1