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Proposed Alternate Methodology for Preparation of a Documented Safety Analysis for the Transformational Challenge Reactor

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MANAGED BY UT-BATTELLE FOR THE US DEPARTMENT OF ENERGY

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Transformational Challenge Reactor Program

Proposed Alternate Methodology for Preparation of a Documented Safety Analysis for the Transformational Challenge Reactor

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Introduction

10 CFR Part 830, Subpart B, Appendix A, Table 2 lists acceptable methodologies for development of a Documented Safety Analysis (DSA) for various types of nuclear facilities. The acceptable methodology listed for a DOE reactor is U.S. Nuclear Regulatory Commission Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*, or successor document. This document was developed for large commercial light-water power reactors. The difference between the Transformational Challenge Reactor (TCR) and commercial reactors in power output, radiological source term, fuel type and composition, and operational life span makes application of the Regulatory Guide to the TCR overly burdensome relative to the value added.

DOE-STD-1083-2009, *Processing Exemptions to Nuclear Safety Rules and Approval of Alternative Methods for Documented Safety Analyses*, Section 4.2, provides a structured procedure for requesting approval of an alternate methodology to develop a DSA other than the methodologies explicitly included in Table 2. Below is a request for approval of an alternate methodology for developing a DSA for the TCR.

The proposed alternate methodology

- NUREG 1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors* will form the basis for the methodology.
- Two chapters will be modified in the TCR DSA to address on-site activities associated with the reactor core including assembling the reactor core in preparation for operation, disassembling the core upon completion of operation, and evaluation/analysis of the core while on the TCR site.
- An analysis pursuant to the requirements of DOE Standard 1027, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports* will be included in chapter three of the DSA.

All aspects of 830.204(b), *Documented Safety Analysis*, will be addressed. Appendix A provides a crosswalk between requirements of 10 CFR 830.204(b) and NUREG 1537. A description of the contents of each chapter of NUREG 1537 is also provided in Appendix A. Appendix B provides a comparison of the requirements of Regulatory Guide 1.70 with that of NUREG 1537.

The design criteria for the facility and a description of proposed integration of safety into the design process will be addressed in a separate Safety Design Strategy document.

Identification of the nuclear facility for which the proposed alternate methodology would be used to develop the DSA

The TCR program will manufacture and test a nuclear core using additive manufacturing combined with machine learning, materials science and data science technologies to develop the information, processes and tools to facilitate expansion of additive manufacturing into advanced nuclear energy systems and other applications requiring a high level of quality assurance for highly demanding applications with large consequence. To accomplish this objective, the program will:

- Evaluate multiple designs, materials and additive manufacturing systems for use in nuclear applications to build previously unachievable designs and understand manufacturing options that could be applied for existing designs.

- Develop a digital platform and associated processes to couple data analytics with design and manufacturing data for use in rapid prototyping and quality evaluations of manufactured products.

- Develop information and processes to integrate sensors and control systems meeting nuclear standards during manufacturing to provide real-time information on performance and enable predictive maintenance and autonomous operations.

- Perform inspection and evaluation of manufactured items to validate the performance predicted by the digital platform, including a short operational test of the manufactured core within a simple reactor system.

- Engage with industry groups, standards development organizations and regulatory bodies to facilitate the regulatory framework for future commercial deployments.

- Publish the information, processes and tools gleaned in the TCR program to assist industry in future commercial deployments.

This request supports the conduct of the short operational test of the additively manufactured core within a simple reactor system.

Supporting Information

This section provides supporting information to assist the reviewer in making the determination that the proposed methodology provides necessary and sufficient detail to conclude that the resulting DSA would meet the requirements of 10 CFR 830, as well as any other applicable requirements, and to conclude that the methodology will provide a DSA with sufficient detail to establish the appropriate hazard controls.

10 CFR 830.7 requires the use of a graded approach, where appropriate, to implement safety basis requirements. 10 CFR 830.3 defines graded approach as the process of ensuring that the level of analysis, documentation, and actions used to comply with a requirement in this part are commensurate with the items listed below. How these factors will be addressed in the TCR DSA is provided under each item:

(1) The relative importance to safety, safeguards, and security

Safety, safeguards, and security is of paramount importance to the TCR and will be fully addressed in the DSA.

(2) The magnitude of any hazard involved

The magnitude of the radiological hazard of the TCR is much less than that of a typical reactor. This is due to the unusually short operational life of the reactor (measured in hours as opposed to years for a typical reactor) for a total fuel burnup less than one megawatt-day. This will result in a very small buildup of noble gases and other fission products and thus a very small source term. Preliminary analysis indicates that the primary constituent of dose at the site boundary is iodine which can be easily mitigated with filtration.

The brief operational lifecycle of the TCR also reduces the frequency of many events routinely considered for other DOE reactors. This reduction in frequency is several orders of magnitude, and therefore, justifies a more qualitative approach in addressing postulated operational and accident scenarios. In addition, analysis of natural phenomena hazard events can be greatly reduced by adopting an operational plan that takes local weather conditions into account for assembling, operating, and disassembling the reactor core. These aspects of regulatory compliance will be addressed further in the program Safety Design Strategy document.

Use and storage of hazardous chemicals will be minimal.

(3) The life cycle stage of a facility

The total time of on-site activities associated with the reactor core (assembling the reactor core in preparation for operation, operating the reactor, disassembling the core upon completion of operation, and evaluation/analysis) is planned to be less than one year so the life cycle stage could be considered very near end-of-life.

(4) The programmatic mission of a facility

The TCR program will revitalize the Nation's capabilities in delivering nuclear power systems by facilitating the industry's adoption of key technologies that will enable a new approach to nuclear energy deployment.

(5) The particular characteristics of a facility

- Maximum thermal power level – 1 Megawatt (MW)
- Fuel – high assay low enriched uranium (< 20 % enriched)
- Helium cooled
- Operational life – less than 24 hours

(6) The relative importance of radiological and non-radiological hazards

Although the magnitude of the radiological hazard of the TCR is much less than that of a typical reactor, control of radiological hazards is of paramount importance and will be addressed as such. Non-radiological hazards are also of high importance to the construction and operation of the TCR. Safety of the facility worker shall be considered essential to the program.

(7) Any other relevant factor - None.

Reason for requesting use of an alternate methodology

This section includes a description of the likely outcome and consequences of simply complying with the methodology in Table 2 to Appendix A of 10 CFR 830, Subpart B.

10 CFR 830, Appendix B lists U.S. Nuclear Regulatory Commission Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*, or successor document as the acceptable methodology for development of a DSA for a DOE reactor. The latest revision of this document was published in 1978 and was developed for large commercial light-water power reactors. Commercial reactors, for which this Regulatory Guide was developed, operate in the thousands of megawatts range compared to the less than 1 MW for TCR. This Regulatory Guide was written for light-water-cooled reactors with low (less than 5%) enriched uranium fuel operating in the thermal neutron spectrum, whereas, the TCR will be helium-cooled with high-assay

low enriched (<20%) uranium fuel. Applying the Regulatory Guide to TCR would be unnecessarily burdensome, technically unfeasible, and fiscally irresponsible.

Furthermore, DOE Standard 1083, *Processing Exemptions to Nuclear Safety Rules and Approval of Alternative Methods for Documented Safety Analyses*, recognizes that Regulatory Guide 1.70 may not be appropriate for smaller reactors:

> "The method listed for reactors in Table 2 of Appendix A of 10 C.F.R. Part 830, Subpart B, is Nuclear Regulatory Commission Regulatory Guide 1.70 'Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.' That method was developed primarily for power reactors and other methods may be appropriate for certain types of small research reactors."

An analysis was performed of the available methodologies for promulgating a DSA for the TCR. The methodologies examined include:

- DOE STANDARD 3009, Preparation of Nonreactor Nuclear Facility Documented Safety Analysis
- DOE-STD-3011-2016, Preparation of Documented Safety Analysis for Interim Operations at DOE Nuclear Facilities. (This standard was evaluated because of the extremely short operational life of the reactor)
- NUREG 1537, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors
- A combination of NUREG 1537, DOE-STD-3009, and advanced reactor (non-light water reactor) requirements

NUREG 1537 was chosen as the best fit for the TCR. The NUREG:

- was developed by the Nuclear Regulatory Commission (NRC) specifically for small research reactors. Approximately thirty reactors have been or are being licensed by the NRC based on a Safety Analysis Report (synonymous with the DOE DSA) using NUREG 1537. If the TCR was licensed today by the NRC, this NUREG would be the governing document both for preparation of the Safety Analysis Report by the licensee and review by the NRC.
- accommodates use of a graded approach and is adaptable to non-light water reactor designs.
- utilizes several American Nuclear Society standards in various categories.

A statement on the limitation of the uses of the alternate methodology

Some aspects of NUREG 1537 do not apply to DOE reactors. For example, Chapter 15, *Financial Qualifications* clearly does not apply to DOE reactors. This and any other area of NUREG 1537 that should not be applied to the creation of the TCR DSA will be addressed in the Safety Design Strategy document for the facility.

Appendix A

Crosswalk of 10 CFR 830.204(b) Requirements and NUREG 1537

10 CFR 830.204(b) Requirement	NUREG 1537 Chapter	
A facility description (including the design of safety structures, systems and components) and the work to be performed.	Chapter 1 Chapter 2 Chapter 3 Chapter 4 Chapter 5 Chapter 6 Chapter 7 Chapter 8 Chapter 9	The Facility Site Characteristics Design of Structures, Systems, and Components Reactor Description Reactor Coolant Systems Engineered Safety Features Instrumentation and Controls Systems Electrical Power Systems Auxiliary Systems
A systematic identification of both natural and man-made hazards associated with the facility.	Chapter 2 Chapter 11 Chapter 13	Site Characteristics (geologic and seismic hazards) Radiation Protection Program and Waste Management (radiological hazards) Accident Analysis (radiological hazards)
Evaluation of normal, abnormal, and accident conditions, including consideration of natural and man-made external events, identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents which may be beyond the design basis of the facility.	Chapter 13 Normal Conditions are addressed in: Chapter 4 Chapter 5 Chapter 6 Chapter 7 Chapter 8 Chapter 9	Accident Analysis (This chapter will be supplemented with an analysis of beyond-design-basis events) Reactor Description Reactor Coolant Systems Engineered Safety Features Instrumentation and Controls Systems Electrical Power Systems Auxiliary Systems
Derivation of the hazard controls necessary to ensure adequate protection of workers, the public, and the environment; demonstration of the adequacy of these controls to eliminate, limit, or mitigate identified hazards; and definition of the process for maintaining the hazard controls current at all times and controlling their use.	Chapter 11 Chapter 13 Chapter 14	Radiation Protection Program and Waste Management (radiological hazards) Accident Analysis (radiological hazards) Technical Specifications (A Technical Safety Requirements document will be generated instead of this chapter)
Definition of the characteristics of the safety management programs necessary to ensure the safe operation of the facility, including (where applicable) quality assurance, procedures, maintenance, personnel training, conduct of operations, emergency	Chapter 11 Chapter 12 Chapter 14	Radiation Protection Program and Waste Management (radiological hazards) Conduct of Operations Technical Specifications (A Technical Safety Requirements document will be generated instead of this chapter)

10 CFR 830.204(b) Requirement	NUREG 1537 Chapter
preparedness, fire protection, waste management, and radiation protection.	

NUREG 1537 Chapter Summaries

Chapter 1 summarizes the principal design bases and considerations, general descriptions of the reactor facility that illustrate the anticipated operations, and the design safety considerations, including the limiting potential accidents. This chapter summarizes the detailed information found in subsequent chapters of the DSA.

Chapter 2 describes the bases for the site selection and describes the applicable site characteristics, including geography, demography, meteorology, hydrology, geology, seismology, and interaction with nearby installations and facilities.

Chapter 3 describes the design bases and facility structures, systems, and components, and the responses to environmental factors on the reactor site (e.g., floods).

Chapter 4 describes the design bases and the functional characteristics of the reactor core and its components. In this chapter, the safety considerations and features of the reactor are discussed.

Chapter 5 lists the design bases and describes the functions of the reactor coolant and associated systems at the facility, including the primary and secondary systems as applicable, and coolant makeup and purification systems. The chapter also describes provisions for adequate heat removal while the reactor is operating and while it is shut down. Note: TCR will have no coolant makeup and purification systems.

Chapter 6 lists the design bases and describes the functions of engineered safety features (ESFs) that may be required to mitigate consequences of postulated accidents at the facility. This includes design-basis accidents and a maximum hypothetical accident (MHA). The MHA, which assumes an incredible failure that can lead to fuel cladding or to a fueled experiment containment breach, is used to bound credible accidents in the accident analysis.

Chapter 7 lists the design bases and describes the functions of the instrumentation and control systems and subsystems at the facility, placing emphasis on safety related systems and safe reactor shutdown.

Chapter 8 lists the design bases and describes the functions of the normal and emergency (if applicable) electrical power systems at the facility.

Chapter 9 lists the design bases and describes the functions of such auxiliary systems at the facility as heating, ventilation, air exhaust, air conditioning, service water, compressed air, and fuel handling and storage.

Chapter 10 lists the design bases and describes the functions of experimental facilities. Non-power reactors are designed with irradiation capabilities for research, education, and technological development. This chapter discusses the characteristics of experiment and irradiation facilities based on the proposed experimental programs. This chapter is not applicable to the TCR since the facility does not contain experimental or irradiation facilities.

NUREG 1537 Chapter Summaries

Chapter 11 lists the design bases and describes the functions of the radiation protection and the radioactive waste management programs at the facility. This chapter also describes the control of byproduct materials produced in the reactor and utilized under the 10 CFR Part 50 reactor operating license. The description of the radiation protection program should include health physics procedures, monitoring programs for personnel exposures and effluent releases, and assessment and control of radiation doses, both to workers and the public. The program to maintain radiation exposures and releases as low as is reasonably achievable (ALARA) includes the control and disposal of radiological waste from reactor operations and from experimental programs. Note: TCR will not produce any by product materials as a result of its operation.

Chapter 12 lists the bases and describes the functions of plans and procedures for the conduct of facility operations. These include discussions of the management structure, personnel training and evaluation, provisions for safety review and auditing of operations by the safety committees, and other required functions, such as reporting, security planning, emergency planning, and planning for reactor startup.

Chapter 13 lists the bases, scenarios, and analyses of accidents at the reactor facility, and describes an MHA, which may include a fission product release, and radiological consequences to the operational staff reactor users, the public, and the environment. The function of ESFs is discussed in the accident analysis, as applicable. Note: TCR will not host a user facility therefore only operational staff and public/environmental consequences will be addressed.

Chapter 14 presents the technical specifications, which state the operating limits and conditions and other requirements for the facility to acceptably ensure protection of the health and safety of the public. Note: A separate Technical Safety Requirements document will be generated in compliance with 10 CFR 830.205, Subpart B, *Safety Basis Requirements*.

Chapter 15 concerns financial qualifications of the non-power reactor applicant for initial construction, continuing operations, and decommissioning. Note: The TCR is government-funded therefore financial considerations are not applicable.

Chapter 16 discusses assembling the reactor core on-site in preparation for operation.

Chapter 17 addresses on-site activities associated with disassembling the core upon completion of operation and evaluation/analysis of the core while on the TCR site. This chapter also gives guidance on decommissioning.

Chapter 18 discusses the conversion of the reactor from highly enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel, includes topics covered in Chapters 1 to 17 as related to HEU to LEU conversions. Note: TCR will utilize high-assay LEU fuel so this chapter is not applicable.

Appendix B

Comparison of NUREG 1537 to Regulatory Guide 1.70

Regulatory Guide 1.70	NUREG 1537
Chapter 1 Introduction and General Description of Plant	Chapter 1 The Facility
Chapter 2 Site Characteristics	Chapter 2 Site Characteristics
Chapter 3 Design of Structures, Components, Equipment, and Systems	Chapter 3 Design of Structures, Systems, and Components
Chapter 4 Reactor	Chapter 4 Reactor Description
Chapter 5 Reactor Coolant System and Connected Systems	Chapter 5 Reactor Coolant Systems
Chapter 6 Engineered Safety Features	Chapter 6 Engineered Safety Features
Chapter 7 Instrumentation and Controls	Chapter 7 Instrumentation and Controls Systems
Chapter 8 Electric Power	Chapter 8 Electrical Power Systems
Chapter 9 Auxiliary Systems	Chapter 9 Auxiliary Systems
Chapter 10 Steam and Power Conversion System	Not applicable to TCR since there is no power conversion capability
Chapter 11 Radioactive Waste Management	Chapter 11, Section 11.2, Waste Management
Chapter 12 Radiation Protection	Chapter 11, Section 11.1, Radiation Protection Program
Chapter 13 Conduct of Operations	Chapter 12 Conduct of Operations
Chapter 14 Initial Test Program	Chapter 12, Section 12.11, "Startup Plan"
Chapter 15 Accident Analyses	Chapter 13 Accident Analysis
Chapter 16 Technical Specifications	Chapter 14 Technical Specifications
Chapter 17 Quality Assurance	Chapter 12, Section 12.9, Quality Assurance (10 CFR 830, Subpart A will be addressed instead of 10 CFR 50.34)