University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology

TOPICAL REPORT

Revision 1

Issued by

Ultra Safe Nuclear Corporation

to

The University of Illinois Urbana-Champaign

under

USNRC Project No. 99902094

February 28, 2024

Approved: March 7, 2024

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 7, 2024

Dr. Caleb S. Brooks, Associate Professor
Department of Nuclear, Plasma,
and Radiological Engineering
University of Illinois at Urbana-Champaign
Talbot Laboratory, Room 111C, MC-234
104 South Wright St.
Urbana, IL 61801

SUBJECT: UNIVERSITY OF ILLINOIS AT URBANA-CHAMPAIGN – SAFETY EVALUATION FOR TOPICAL REPORT RELATED TO EVENT SEQUENCE IDENTIFICATION AND SAFETY CLASSIFICATION METHODOLOGY (EPID NO. L-2023-NFN-0011)

Dear Dr. Brooks:

By letter dated September 7, 2023, the University of Illinois at Urbana-Champaign (UIUC) submitted the topical report (TR) "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology," Revision 0 (Agencywide Documents Access and Management System Accession No. ML23250A318) for U.S. Nuclear Regulatory Commission (NRC) staff review. As part of its review of the TR, the NRC staff sent requests for additional information (RAIs) to UIUC by email dated December 19, 2023 (ML23354A009). By letter dated January 17, 2024 (ML24017A307), UIUC submitted responses to the RAIs, including marked-up pages with proposed edits to the TR. UIUC submitted Revision 1 of the TR, incorporating the proposed edits, by letter dated February 20, 2024 (ML24053A336).

The NRC staff's safety evaluation (SE) for the TR, Revision 1, is enclosed. The enclosed SE will be made publicly available.

The NRC staff requests that UIUC publish an accepted version of the TR within 3 months of receipt of this letter. The accepted version should incorporate this letter and the enclosed SE after the title page. The accepted version should include an "-A" (designating "accepted") following the TR identifier.

If you have any questions, please contact Edward Helvenston at (301) 415-4067, or by email at Edward.Helvenston@nrc.gov.

Sincerely,

Signed by Cruz, Holly on 03/07/24

Holly D. Cruz, Acting Chief
Non-Power Production and Utilization Facility Licensing Branch
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Project No.: 99902094

Enclosure:
As stated

cc: GovDelivery Subscribers

SUBJECT: UNIVERSITY OF ILLINOIS AT URBANA-CHAMPAIGN – SAFETY EVALUATION FOR TOPICAL REPORT RELATED TO EVENT SEQUENCE IDENTIFICATION AND SAFETY CLASSIFICATION METHODOLOGY (EPID NO. L-2023-NFN-0011) DATED: MARCH 7, 2024

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ADAMS Accession No.: ML24039A164 **NRR-106**

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UNIVERSITY OF ILLINOIS AT URBANA-CHAMPAIGN – SAFETY EVALUATION OF REVSION 1 OF TOPICAL REPORT "UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN HIGH TEMPERATURE GAS-COOLED RESEARCH REACTOR: EVENT SEQUENCE IDENTIFICATION AND SSC SAFETY CLASSIFICATION METHODOLOGY" (EPID NO. L-2023-NFN-0011)

APPLICANT INFORMATION

Applicant: University of Illinois at Urbana-Champaign

Applicant Address: 104 South Wright St.
Urbana, Illinois 61801

Docket /Project No(s).: 99902094

APPLICATION INFORMATION

Submittal Date: September 7, 2023

Submittal Agencywide Documents Access and Management System (ADAMS) Accession No.: ML23250A318 (Reference 1)

Supplement and Request for Additional Information (RAI) response ADAMS Accession Nos.: ML24017A307 (Reference 2); ML24053A336 (Reference 3)

Brief Description of the Topical Report: The topical report (TR) "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Classification Methodology," Revision 1 (Reference 3), describes the event sequence identification methodology and safety classification methodology to be applied to the University of Illinois at Urbana-Champaign (UIUC) Micro Modular Reactor (MMR) research reactor. UIUC would utilize these two methodologies when providing details of design aspects of the MMR, designed by Ultra Safe Nuclear Corporation, in a preliminary safety analysis report and final safety analysis report for the proposed UIUC MMR.

As discussed in TR section 2.1, UIUC's stated purpose of the event sequence identification methodology in the TR is to identify event sequences for the UIUC MMR safety analysis that will be intended to meet Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(4). As discussed in TR section 2.3, the elements of the event sequence identification methodology are: identifying postulated initiating events (PIEs), screening PIEs on a deterministic basis, defining event sequences, and grouping event sequences into accident categories, guided by NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" (Reference 4), that have been adapted for use in the high-temperature gas-cooled MMR. TR section 2.3.1 describes PIE identification as a three-phase process: an initial phase that provides a list defined through historical information, regulatory documentation,

Enclosure

and engineering judgement; a top-down phase that determines causes by analyzing design function trees created from master logic diagrams (MLDs); and a bottom-up phase to verify that a more developed design can result in the identified PIE.

As discussed in TR section 3.1, in preparation for the submission of construction permit (CP) and operating license (OL) applications, UIUC has identified requirements in 10 CFR 50.34(a)(4) and 10 CFR 50.34(b)(4) for analyses and evaluation of structures, systems, and components (SSCs). UIUC also details in TR sections 3.2 and 3.3 its intended method for identification of SSCs. SSCs will be categorized as either safety-related (SR) or non-safety-related (NSR). SSCs that have been connected to the reliable operation of safety functions will be defined as SR SSCs. The methodology for safety classification of SSCs will be performed by identifying limiting event sequences connected to safety classification, listing the required SSCs to fulfill a safety function during the developed event sequence, and assigning these identified SSCs as SR with all others as NSR.

For additional details on the submittal, refer to the documents available at the ADAMS Accession Nos. identified above.

EVALUATION CRITERIA

As discussed in TR sections 1.0 and 1.4, UIUC plans to apply for licensing of the MMR design under the provisions of 10 CFR 50.21(c), as a Class 104(c) research reactor. Specifically, UIUC plans to provide CP and OL applications in the future for U.S. Nuclear Regulatory Commission (NRC) staff review to support licensing the construction and operation of the UIUC MMR, respectively. Because the MMR would be a non-light-water reactor (non-LWR), and the UIUC MMR would also be a Class 104(c) research reactor, the NRC's regulations specifically pertaining to light-water reactors (LWRs) may not be applicable. UIUC has submitted and the NRC staff is currently evaluating a separate TR entitled, "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Applicability of Nuclear Regulatory Commission Regulations" (Reference 5). As of the date of this SE, the NRC staff's review of the regulatory applicability TR is still in progress.

To support the review of this TR, the NRC staff considered the regulations it expected to be applicable to the UIUC MMR based on insights from its review of the TR and from general information provided by UIUC regarding the proposed MMR in UIUC's Regulatory Engagement Plan submitted to the NRC by letter dated June 26, 2023 (Reference 6). The evaluation criteria section in this SE, continued below, describes regulations and guidance that the NRC staff determined to be appropriate to support the review and approval of this TR.

The regulations at 10 CFR 50.34(a)(3) require, in part, that applications for CPs include principal design criteria (PDC), which are discussed later in this SE section. In addition, the regulations at 10 CFR 50.34(a)(4) and 10 CFR 50.34(b)(4) require that CP and OL applications, respectively, include analyses and evaluation of SSCs. The NRC staff anticipates that an eventual UIUC MMR application would reference the contents of this TR and SE and would be subject to the provisions of 10 CFR 50.34. Therefore, the staff considered sections of 10 CFR 50.34, particularly 10 CFR 50.34(a)(3), 10 CFR 50.34(a)(4), and 10 CFR 50.34(b)(4), in reviewing the TR and developing this SE.

NRC Regulatory Guide (RG) 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," Revision 0 (Reference 7), outlines example PDC formulated to be generally applicable to non-LWR designs. Appendix C of RG 1.232, in particular, contains a set of PDC specifically tailored to modular high-temperature gas-cooled reactor (MHTGR) designs, such as the MMR. RG 1.232 states that an applicant may use the RG's guidance to develop all or part of a reactor design's PDC. The proposed PDC for the UIUC MMR design is the subject of a separate TR, "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria" (Reference 8). As of the date of this SE, the NRC staff's review of the PDC TR is still in progress. In its PDC TR, UIUC generally applied the PDC developed for MHTGRs in appendix C of RG 1.232, with minimal adaptation as needed, to the UIUC MMR design. Although the staff has not made a final determination on the UIUC PDC TR, the staff considered the general characteristics of the proposed PDC in the PDC TR as well as in appendix C of RG 1.232 to inform its review of this TR. Two of the proposed UIUC PDC that the staff considered most applicable to the subject matter of this TR review are listed below (reproduced from Reference 8):

- UIUC MMR PDC 2, "Design bases for protection against natural phenomena," which states, in part, that "[SSCs] important to safety shall be designed to withstand the effects of natural phenomena … without loss of capability to perform their safety functions."

- UIUC MMR PDC 10, "Reactor design," which states that "[t]he reactor system and associated heat removal, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits [SARRDLs] are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences [AOOs]."

Additionally, the NRC staff noted that multiple other proposed PDCs in the UIUC PDC TR include specific information regarding the design of SSCs used to ensure that the consequences of postulated accident conditions do not exceed SARRDLs.

NUREG-1537, Part 1, "Format and Content," and Part 2, "Standard Review Plan and Acceptance Criteria," provide guidance for preparing and reviewing, respectively, applications for non-power reactors. The most relevant guidance with respect to this TR is contained in chapter 6, "Engineered Safety Features," and chapter 13, "Accident Analyses," of Parts 1 and 2. In these chapters, NUREG-1537 describes, in part, the engineered safety features (ESFs) approach and how ESFs are identified for a design, if required, and incorporated in the safety analysis report (SAR). The approach includes identification and evaluation of postulated accidents, including a maximum hypothetical accident (MHA) for which potential consequences are shown to exceed and bound all credible accidents. These accidents are evaluated against the applicable regulatory limits regarding radiological consequences (e.g., public dose limits in 10 CFR Part 20). NUREG-1537, Parts 1 and 2, chapter 13, identifies some specific types of accident scenarios (e.g., reactivity insertion, loss of coolant) that should be discussed by the applicant in the SAR.

TECHNICAL EVALUATION

Scope of Review

Section 1.5 of the TR requests NRC review and approval of the methodologies presented in TR sections 2 and 3. Therefore, TR sections 2 and 3 were the focus of the NRC staff's review as summarized in this SE. Section 2 of the TR describes the methodology that UIUC plans to use to identify the event sequences that will be evaluated and described in the UIUC MMR safety analysis that will be provided as part of future licensing submittals. TR sections 1.4 and 2.1 state that the specific method(s) that UIUC will use to perform safety evaluations of the event sequences will also be subject to a future licensing submittal. Section 3 of the TR describes the methodology that UIUC plans to implement to determine the appropriate safety classification of the UIUC MMR SSCs.

The NRC staff notes that TR section 1.5 states that preliminary lists of PIEs and SSC safety classifications are provided in TR appendices A and B, respectively, but that these appendices are for informational purposes only. Therefore, the staff did not consider the preliminary information in TR appendices A and B to be within the scope of its review, and makes no regulatory findings associated with the preliminary information in TR appendices A and B.

In addition, the NRC staff notes that the results generated by UIUC through its implementation of the methodologies described in the TR will be subject to future review as part of future licensing application(s). Therefore, ultimate approval of the implementation of the methodology, the event sequences that are identified, and the classifications that are assigned to UIUC MMR SSCs will be subject to the appropriate future licensing submittal(s), and is not provided in this SE. The staff accordingly imposed Limitation 1 on its approval of this TR. Limitation 1 is listed in the Limitation section of this SE.

Event Sequence Identification Methodology

Section 2 of the TR describes the event sequence identification methodology, which is a process used to identify and assess PIEs throughout the development of the UIUC MMR design, screen the identified PIEs, and define and group the resulting event sequences.

PIE Identification

As discussed in TR section 2.3.1, the event sequence identification process begins with identifying PIEs using a multi-phase approach. The "initial phase" considers historical and regulatory sources, and engineering judgement applied to the UIUC MMR conceptual design to establish a preliminary set of PIEs (included for informational purposes in TR appendix A). The next phase employs a "top-down" approach as the design becomes mature enough to support the approach. Specifically, the top-down phase uses MLDs to analyze three high-level fundamental safety functions: reactivity control, reactor heat removal, and control of release of radioactive material that could exceed public dose limits. The insights from these analyses are then used to confirm or extend the PIEs identified during the initial phase, leading to more specific initiating events.

TR section 2.3.1 states that the third phase of PIE identification employs a "bottom-up" approach when the design is sufficiently detailed such that failures can be more readily determined. This phase verifies and expands on the details that resulted from the top-down phase using unique methodologies for each type of PIE identified. The PIE types, defined in TR section 2.2.1, include piping system breaches, transients, and internal and external hazards. TR section 2.3.1 states that the bottom-up methodology related to piping system breaches assesses design description information to determine whether a breach can result in an AOO or accident conditions. For transients, the bottom-up phase consists of performing failure modes and effects analyses (FMEAs) to identify PIEs. TR section 2.3.1 states that details on the FMEA methodology and results will be provided in a future UIUC MMR OL application. The internal and external hazards methodologies will both rely on hazard analysis used to identify hazard-induced PIEs. TR section 2.3.1 states that details on the internal and external hazard analyses will also be provided in a future UIUC MMR OL application.

The NRC staff reviewed the above information and finds that the multi-phase approach for identifying PIEs described in the TR is acceptable on the basis that implementation of the approach can reasonably be expected to result in a sufficiently comprehensive list of PIEs that could be used to adequately identify limiting event sequences and appropriately identify required ESFs in alignment with the guidance described in NUREG-1537. Additionally, the staff notes that the iterative nature of the approach allows the list of PIEs to be expanded, as needed, as the details of the UIUC MMR design continue to be determined and/or finalized.

As noted above, the TR states that specific information regarding MLD, FMEA, and internal and external hazards analysis methodologies was not included in this TR but will be included in a future UIUC MMR OL application and be subject to future licensing review. The NRC staff acknowledges that the final methodologies and results obtained would be included in the review of a UIUC OL application. However, the staff notes that preliminary information regarding MLDs, FMEAs, and hazards analyses, beyond what is provided in this TR, may also be appropriate to be included as part of any future UIUC MMR CP application. The staff notes that such information could help to ensure that the proposed preliminary design adequately considers the appropriate PIEs and resultant event sequences, including those materially influenced by internal and/or external hazards.

PIE Screening

As discussed in TR section 2.3.2, each PIE identified following the three-phase approach is then screened for credibility on a deterministic basis. PIEs related to SSC failures are evaluated based on the specifics of the failure and/or the engineering design rules applied to the SSCs. For PIEs related to external hazards, the specifics of the external hazard assessment will also be considered. PIEs that are determined to not be credible are screened out of the remainder of the process. PIEs that do not have sufficient supporting information to screen out will be retained as part of the licensing basis, unless further information later supports a determination that the PIE is not credible.

The NRC staff reviewed the above information and finds that the approach of screening PIEs for credibility is acceptable because it is consistent with NUREG-1537 guidance that only credible events should be considered and grouped when identifying limiting event sequences. Additionally, the staff notes that NUREG-1537 guidance regarding non-credible events is limited to establishing an MHA, which is not within the scope of this TR review. The staff expects that, consistent with the approach discussed in NUREG-1537, the screening of PIEs for credibility can reasonably be performed on a deterministic basis through the application of appropriate engineering principles. However, the staff also expects that any future licensing submittal detailing the results of the PIE screening process would likely need to describe the deterministic bases used to conclude that screened PIEs are not credible. The staff also notes that the acceptability of the screening process results would be subject to the review of that submittal.

Defining and Grouping Event Sequences

As discussed in TR section 2.3.3, after PIEs are identified and screened, the remaining credible PIEs are used to determine which SSCs are required to mitigate the consequences of the PIEs. The safety classification methodology is used to establish which SSCs are considered SR. The SR SSCs associated with the PIEs then determine the plant response to each PIE (or combination of PIEs) to define event sequences. Safety analyses will be performed on the event sequences, considering the worst-case single failure of an active component. Additionally, as discussed in TR section 2.3.4, the event sequences will be grouped into categories that largely align with those listed in NUREG-1537, with some adaptation to accommodate the MMR design, as summarized in TR table 2-1. UIUC states in TR section 2.3.4 and its response to RAI-2 (Reference 2) that limiting event sequence(s) will be identified for each event sequence group. Event sequences bounded by a limiting event sequence will rely upon the same SSCs; event sequences that rely on differing sets of SSCs will have different limiting event sequences identified. Therefore, event sequence groups may result in multiple limiting event sequences, such that all SSCs performing safety functions will be identified during implementation of the safety classification methodology.

The NRC staff reviewed the above information and finds the described approach of defining and grouping event sequences to be acceptable on the basis that it aligns with the approach and guidance provided in NUREG-1537, Parts 1 and 2, chapters 6 and 13, including the guidance that accident scenarios should be categorized by type and likelihood of occurrence, and that limiting events should be identified for each group. The staff's review of the safety classification methodology for UIUC MMR SSCs (TR section 3) is discussed in the next SE section.

The NRC staff notes that TR table 2-1 includes an MHA but the methodology to determine and analyze that MHA is not included in the TR. However, TR section 1.4 explains that the methodology to identify and evaluate the MHA will be subject to a future licensing submittal. Therefore, the staff finds that UIUC's general approach of identifying and analyzing an MHA is acceptable on the basis that it aligns with NUREG-1537 guidance, but the staff makes no determination regarding any specific MHA sequence or methodology for identification or evaluation of the MHA in this SE.

Safety Classification of SSCs

Section 3 of the TR defines two safety classification groups, SR and NSR, and describes UIUC's methodology used to classify each UIUC MMR SSC into one of the groups. Additionally, section 3 of the TR defines the term "safety function" and lists three fundamental safety functions for SR SSCs in the UIUC MMR.

Safety Functions

The term "safety function" is defined in TR section 3.2.1 as "a specific purpose that must be accomplished for safety for a facility or activity to prevent or to mitigate radiological consequences of normal operation, anticipated operational occurrences, and accident conditions." The NRC staff finds that this definition is acceptable on the basis that it aligns with previous such definitions associated with other existing research reactors, and that the definition provides enough specificity to ensure that appropriate safety functions can be identified.

Three fundamental safety functions are identified in TR section 3.2.1 for the UIUC MMR SR SSCs: control of reactivity (including reactor shutdown), removal of heat from the reactor, and control of release of radioactive material that could exceed public dose limits. The NRC staff finds that the fundamental safety functions identified are acceptable on the basis that they appear to represent a reasonable set of functions that could be used to support the UIUC MMR safety classification methodology and definition of SR. Specifically, the staff finds that the three safety functions identified appear to form a sufficiently comprehensive basis to evaluate the performance of SSCs during PIEs and identify those SSCs that should be classified as SR, consistent with the guidance in NUREG-1537. Additionally, the staff notes that, while not specifically applicable to the UIUC MMR, the 10 CFR 50.2 definition of SR SSCs includes a list of functions that is similar, although tailored to LWRs. This similarity between the approaches further supports the staff's finding that the UIUC approach is reasonable and acceptable.

Safety Classification Groups

As discussed in TR section 3.2.2, the UIUC MMR safety classification methodology includes two classifications: SR and NSR. SSCs that have an impact on safety and are relied upon to remain functional to meet at least one of the three fundamental safety functions during and following all event sequences that are part of the plant design basis are considered SR. All other SSCs are considered NSR. In its response to RAI-1 (Reference 2), UIUC clarified that shutdown from normal operation is encompassed by the fundamental safety function of "control of reactivity," and, therefore, SSCs relied upon for normal shutdown (and not only those needed, for example, to maintain the reactor shutdown during and following an event sequence) would be included as SR. The NRC staff finds that the definitions provided for the terms SR and NSR are acceptable on a basis similar to that discussed in the previous SE section related to safety functions. Specifically, the staff finds that the definitions reasonably align with research reactor precedents, provide adequate specificity, can be expected to adequately support the implementation of the UIUC MMR safety classification methodology, and generally reflect the analogous 10 CFR 50.2 definition of SR SSCs for LWRs.

Safety Classification Methodology

TR section 3.3 discusses the overall safety classification methodology for the UIUC MMR. The methodology used to classify UIUC MMR SSCs begins with the identification of PIEs, and then limiting event sequences, as described in TR section 2. As discussed in TR section 3.3.2, the limiting event sequences will then be evaluated considering the three fundamental safety functions to identify the SSCs that are required to achieve those safety functions. As discussed in TR section 3.3.3, all SSCs that are identified as required to provide one or more of these fundamental safety functions will be classified as SR. The remaining SSCs will be classified as NSR.

In its response to RAI-2.a (Reference 2), regarding how UIUC's limiting event sequences would be determined to be limiting, UIUC clarified that radiological consequence or other surrogate characteristics (i.e., fuel temperature, helium pressure, etc.) would be used as a figure of merit.

The NRC staff reviewed the above information and finds that the overall safety classification methodology described by UIUC is acceptable on the basis that it aligns with the definitions provided in the TR and the guidance provided in NUREG-1537 and, therefore, can be expected to result in UIUC MMR SSC classifications that can be consistent with the regulations and guidance discussed in the Evaluation Criteria section of this SE.

An example of an event sequence evaluation (associated with a pipe break PIE) is provided in TR section 3.3.2. The NRC staff notes that the example appears to demonstrate event sequence evaluation methodology described in the TR in a reasonable manner, but the staff makes no determination on the specific details of the example PIE itself due to the preliminary nature of the information. In addition, TR section 1.5 states that this example is provided for information purposes only.

LIMITATION

The NRC staff imposes the following limitation on the acceptance of this TR:

1. The NRC staff finds the methodology described acceptable on the basis that the presented approach appears to be a reasonable way to identify event sequences and classify SSCs. However, the staff notes that the methodology is iterative in nature as the design process continues, and has not yet been fully implemented, and the resulting SSC safety classifications, event sequences, and PIEs have not yet been finalized. Therefore, the staff's approval is limited to the methodology approach itself. Approval of the specific implementation of the methodology and the resulting event sequences and SSC classifications obtained as they pertain to the UIUC MMR would be subject to future licensing submittals.

CONCLUSION

The NRC staff concludes that UIUC TR "University of Illinois Urbana-Champaign High Temperature Gas-Cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology" provides acceptable methodologies to identify PIEs and limiting event sequences, and to classify UIUC MMR SSCs depending on their involvement in the identified limiting event sequences. The staff based this conclusion on the alignment between the proposed approach and the applicable guidance in NUREG-1537. Specifically, the staff determined that the methodologies described in the TR represent a well-defined approach that could reasonably be expected to be implemented in a manner that would align with the intent of the NUREG-1537 guidance and be consistent with the requirements of the portions of 10 CFR 50.34 discussed in the Evaluation Criteria section of this SE. Approval of this TR is subject to Limitation 1 in the above SE section.

REFERENCES

1. Letter from UIUC to NRC, "Submittal of University of Illinois Topical Report, 'Event Sequence Identification and SSC Safety Classification Methodology,'" dated September 7, 2023 (ML23250A318).

2. Letter from UIUC to NRC, "Written communication as specified by 10 CFR 50.4 regarding responses to the 'Request for Additional Information – University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology,' Topical Report, dated December 19, 2023," dated January 17, 2024 (ML24017A307).

3. Letter from UIUC to NRC, "Submittal of the revised 'University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology' Topical Report, Release 02," dated February 20, 2024 (ML24053A336).

4. NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 1, "Format and Content," and Part 2, "Standard Review Plan and Acceptance Criteria," dated February 1996 (ML042430055 and ML042430048).

5. Letter from UIUC to NRC, "University of Illinois Topical Report Submission, 'University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Applicability of Nuclear Regulatory Commission Regulations," dated December 9, 2022 (ML22343A282).

6. Letter from UIUC to NRC, "USNRC Project No. 99902094: UIUC Regulatory Engagement Plan revision submission," dated June 26, 2023 (ML23178A259).

7. NRC Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," Revision 0, dated April 2018 (ML17325A611).

8. Letter from UIUC to NRC, "Submittal of 'University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria' Topical Report," dated November 15, 2023 (ML23319A407).

Principal Contributors: D. Beacon, NRR
L. Kingsbury, NRR

Date: March 7, 2024

From: Edward Helvenston
Sent: Tuesday, December 19, 2023 5:41 PM
To: Brooks, Caleb
Cc: Grunloh, Timothy P; lfoyto@illinois.edu; Kelly Sullivan; Patrick Boyle; Paulette Torres; Greg Oberson (He/Him); Holly Cruz (She/Her/Hers); Dan Beacon (He/Him)
Subject: Request for Additional Information Regarding UIUC Event Sequence Identification and SSC Safety Classification Methodology
Attachments: UIUC Safety Classification and Event Sequence Identification Methodology TR RAIs (final).pdf

Dear Dr. Brooks:

By letter dated September 7, 2023 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML23250A318), University of Illinois at Urbana-Champaign (UIUC) submitted Revision 0 of the topical report (TR), "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology," for the U.S. Nuclear Regulatory Commission (NRC) staff's review.

The U.S. Nuclear Regulatory Commission (NRC) staff identified additional information needed to continue its review of the TR, as described in the enclosed request for additional information (RAI). As discussed, provide a response to the RAI or a written request for additional time to respond, including the proposed response date and a brief explanation of the reason, by January 19, 2024. Following receipt of the complete response to the RAI, the NRC staff will continue its review of the topical report.

If you have any questions regarding the NRC staff's review or if you intend to request additional time to respond, please contact me at (301) 415-4067 or by electronic mail at Edward.Helvenston@nrc.gov.

Sincerely,

Ed Helvenston, U.S. NRC
Non-Power Production and Utilization Facility Licensing Branch (UNPL)
Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU)
Office of Nuclear Reactor Regulation (NRR)
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Project No. 99902094
EPID: L-2022-NFN-0011
Enclosure: As stated
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Concurrence on RAI Questions

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DATE	12/12/2023	12/12/2023	12/19/2023

Hearing Identifier: NRR_DRMA
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Received Date: 12/19/2023 5:41:00 PM
From: Edward Helvenston

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MESSAGE	1812	12/19/2023 5:41:00 PM
UIUC Safety Classification and Event Sequence Identification Methodology TR RAIs (final).pdf	91908	

Options
Priority: Normal
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:

OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

EVENT SEQUENCE IDENTIFICATION AND SAFETY CLASSIFICATION

METHODOLOGY TOPICAL REPORT

UNIVERSITY OF ILLINOIS AT URBANA-CHAMPAIGN

PROJECT NO. 99902094

Based on its review, the NRC staff requires the following additional information to continue its review of the University of Illinois at Urbana-Champaign (UIUC) topical report (TR).

1) The TR "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology," Revision 0 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23250A318), in Section 3.1, discusses how the proposed definition of safety-related (SR) UIUC has proposed for its reactor will "meet the intent" of the 10 CFR 50.2 definition of SR structures, systems, and components (SSCs). However, the staff notes that the proposed definition of SR in TR Section 3.2.2 specifies SSCs that "are relied upon … during and following all event sequences…" and that "event sequences" are defined in TR Sections 2.2.2 and 2.3.3 as conditions based on limiting postulated initiating events (PIEs), which based on the examples provided in TR Appendix A, all appear to be off-normal events. The 10 CFR 50.2 definition of SR SSCs includes, in part, SSCs that "are relied upon … during and following design basis events to assure … 2) the capability to shut down the reactor and maintain it in a safe shutdown condition…". The staff notes that, per 10 CFR 50.49 (which the staff notes is not applicable to research reactors, but which does apply to nuclear power plants that use the 10 CFR 50.2 definition of SR SSCs), design basis events are considered "conditions of normal operation, including anticipated operational occurrences…". As such, the staff considers the intent of the 10 CFR 50.2 definition of SR SSC to include SSCs required to achieve safe shutdown from normal operations.

   Is UIUC's definition of SR in TR Section 3.2.2 intended to include SSCs required for safe shutdown from normal operations, as well as those that are relied upon during and following event sequences? If so, clarify the definition or explain how it meets this intent.

2) TR Section 3.3.1 states "[t]he limiting PIEs will be identified and serve as an input in the safety classification methodology." However, the application of this statement in the context of the TR is not fully clear. Please clarify the statement in TR Section 3.3.1. Specifically:

   a. Describe what a "limiting PIE" is and how a PIE is determined to be limiting (i.e., what figure(s) of merit would be used).
   b. Describe how "limiting PIEs" align with the event sequence categories described in TR Section 2.3.4. Does each category have a single limiting PIE (e.g., as described in

NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors” (ML042430055 and ML04243055)) or multiple limiting PIEs, and how is that decided?

c. When classifying SSCs based on limiting PIEs alone, how will consideration be given to PIEs that were not considered limiting, but rely on different SSCs than the limiting case? For example, if SSC A is relied upon to mitigate PIE X, and SSC B is relied upon to mitigate PIE Y, how does the methodology classify SSC A, if PIE Y is identified as the "limiting PIE" of the two (particularly for potential cases when the failure of SSC A following PIE X could result in measurable consequences)?

UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN

The Grainger College of Engineering
Department of Nuclear, Plasma, & Radiological Engineering
Suite 100 Talbot Laboratory, MC-234
104 S. Wright St.
Urbana, IL 61801

January 17, 2024

Docket No. 99902094
U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: Written communication as specified by 10 CFR 50.4 regarding responses to the "Request for Additional Information – University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology," Topical Report, dated December 19, 2023

By letter dated September 7, 2023 (ML23250A319), the University of Illinois Urbana-Champaign (UIUC) submitted the Topical Report (TR), "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology" (ML23250A320), to the U.S. Nuclear Regulatory Commission (NRC) for acceptance review and generation of a review schedule.

By email dated October 4, 2023 (ML23276A600), the NRC provided UIUC the completeness determination for the TR (ML23276A599). Based on a preliminary review of the TR, the NRC staff determined that it provided sufficient information for the NRC staff to begin a detailed technical review.

By email dated December 19, 2023 (ML23354A009), the NRC staff requested additional information and clarification in the form of two (2) request for additional information (RAI). The RAI, and UIUC's responses to the RAI, are enclosed as Attachment 1. Also enclosed as Attachment 2 is a mark-up of the TR pages as a result of the RAI responses, as recommended by Section 2.4.3 of LIC-500, Rev. 9, "Topical Report Process."

The responses to the RAI or a mark-up of the TR pages do not contain any commercially sensitive information and can be posted for unrestricted access by the public. Questions or other requests related to the RAI responses should be directed to Caleb Brooks at csbrooks@illinois.edu or (217) 265-0519.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 17, 2024.

Sincerely,

Caleb S. Brooks, Ph.D.

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Attachments:

1. Responses to the "Request for Additional Information – University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology" Topical Report, dated December 19, 2023
2. Mark-up of the pages of the "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology" Topical Report, as a result of the RAI responses

Cc:
University of Illinois Urbana-Champaign
Tim Grunloh
Les Foyto

Ultra Safe Nuclear Corporation
Daniel Stout
Zackary Rad
Jeff Perry

U.S. Nuclear Regulatory Commission
Edward Helvenston

ATTACHMENT 1

Responses to the "University of Illinois at Urbana-Champaign – Request for Additional Information – University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology," Topical Report, dated December 19, 2023

RAI-1*: The topical report (TR) "University of Illinois Urbana-Champaign High Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology," Revision 0 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23250A318), in Section 3.1, discusses how the proposed definition of safety-related (SR) the University of Illinois at Urbana-Champaign (UIUC) has proposed for its reactor will "meet the intent" of the 10 CFR 50.2 definition of SR structures, systems, and components (SSCs). However, the staff notes that the proposed definition of SR in TR Section 3.2.2 specifies SSCs that "are relied upon ... during and following all event sequences…" and that "event sequences" are defined in TR Sections 2.2.2 and 2.3.3 as conditions based on limiting postulated initiating events (PIEs), which based on the examples provided in TR Appendix A, all appear to be off-normal events. The 10 CFR 50.2 definition of SR SSCs includes, in part, SSCs that "are relied upon ... during and following design basis events to assure ... 2) the capability to shut down the reactor and maintain it in a safe shutdown condition…". The staff notes that, per 10 CFR 50.49 (which the staff notes is not applicable to research reactors, but which does apply to nuclear power plants that use the 10 CFR 50.2 definition of SR SSCs), design basis events are considered "conditions of normal operation, including anticipated operational occurrences…". As such, the staff considers the intent of the 10 CFR 50.2 definition of SR SSC to include SSCs required to achieve safe shutdown from normal operations.*

Is UIUC's definition of SR in TR Section 3.2.2 intended to include SSCs required for safe shutdown from normal operations, as well as those that are relied upon during and following event sequences? If so, clarify the definition or explain how it meets this intent.

Yes, UIUC's definition of safety-related is intended to include SSCs required for safe shutdown from normal operation, as well as those that are relied upon during and following event sequences. A safety-related SSC will be available during normal operation to assure reactor shutdown capability.

Reactor shutdown is encompassed by the fundamental safety function 'control of reactivity' outlined in Section 3.2.1 of the TR.

UIUC will add the following language to Section 3.2.1: "Control of reactivity encompasses reactor shutdown."

ATTACHMENT 1

RAI-2*: TR Section 3.3.1 states "[t]he limiting PIEs will be identified and serve as an input in the safety classification methodology." However, the application of this statement in the context of the TR is not fully clear. Please clarify the statement in TR Section 3.3.1. Specifically:*

a. *Describe what a "limiting PIE" is and how a PIE is determined to be limiting (i.e., what figure(s) of merit would be used).*

   A Limiting PIE is meant to be equivalent to Limiting Event (sequence) discussed in NUREG-1537. UIUC will update Limiting PIEs to Limiting Event Sequences in the TR.

   Radiological consequence or other surrogate characteristics (i.e. fuel temperature, helium pressure, etc.) will be the figure of merit when determining Limiting Event Sequences.

b. *Describe how "limiting PIEs" align with the event sequence categories described in TR Section 2.3.4. Does each category have a single limiting PIE (e.g., as described in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" (ML042430055 and ML04243055)) or multiple limiting PIEs, and how is that decided?*

   A limiting event sequence will represent events using the same set of responding SSCs to perform safety functions within the category. Therefore, more than one limiting event sequence may exist for each category.

   UIUC will update the TR to include explanatory language in Section 2.3.4.

c. *When classifying SSCs based on limiting PIEs alone, how will consideration be given to PIEs that were not considered limiting, but rely on different SSCs than the limiting case? For example, if SSC A is relied upon to mitigate PIE X, and SSC B is relied upon to mitigate PIE Y, how does the methodology classify SSC A, if PIE Y is identified as the "limiting PIE" of the two (particularly for potential cases when the failure of SSC A following PIE X could result in measurable consequences)?*

   Event sequences bound by a limiting event sequence will rely on the same set of SSCs to perform safety functions. Therefore, all SSCs performing safety functions will be captured by safety classification methodology.

ABBREVIATIONS & ACRONYMS

This list contains the abbreviations and acronyms used in this document.

Abbreviation or Acronym	Definition
AP	Adjacent Plant
ASME	American Society of Mechanical Engineers
CNSC	Canadian Nuclear Safety Commission
CP	Construction Permit [per 10 CFR 50]
CPA	Construction Permit Application [per 10 CFR 50]
DID	Defense-in-Depth
D-LOFC	Depressurized loss of flow cooling
ECCS	Emergency core cooling system
EPRI	Electric Power Research Institute
FCM™	Fully Ceramic Micro-Encapsulated
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HALEU	High Assay Low-Enriched Uranium
HPB	Helium Pressure Boundary
HTGR	High Temperature Gas-Cooled Reactor
HTS	Heat Transport System
HVAC	Heating, Ventilation, and Air Conditioning
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Controls
IHX	Intermediate Heat Exchanger
LWR	Light Water Reactor
MHA	Maximum Hypothetical Accident
MHTGR	Modular High Temperature Gas Reactor
MLD	Master Logic Diagram
MMR™	Micro Modular Reactor™
MSS	Molten Salt System
MW	Megawatts
NEIMA	Nuclear Energy Innovation and Modernization Act [115-439 (01/14/2019)]
NP	Nuclear Plant
NRC	[U.S.] Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Document
OL	Operating License [in accordance with 10 CFR 50]
OLA	Operating License Application [in accordance with 10 CFR 50]
PDC	Principal Design Criteria
P&ID	Piping and Instrumentation Diagram
PIE	Postulated Initiating Event
P-LOFC	Pressurized Loss of Flow Cooling
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
PSE	Planned Special Exposure

Commented [JC1]: Adjusted row height

This TR considers regulations and guidance for preparing CPA and OLA for a research reactor facility in accordance with 10 CFR 50 (Reference 2) and NUREG-1537. The facility will be licensed under the provisions of 10 CFR 50.21(c) as a Class 104, non-power utilization facility. The MMR is also a non-light water reactor (non-LWR). These characterizations limit applicability of some NRC regulations. Reference 4 discusses the applicable NRC regulations for the MMR at UIUC.

This TR discusses the methodology to identify credible event sequences. A future TR will discuss the deterministic methodology used to (1) identify a Maximum Hypothetical Accident (MHA) that bounds the dose consequence associated the identified credible event sequence, (2) calculate the dose consequence associated with the MHA, and (3) analyze the limiting credible event sequence to demonstrate that the MHA dose consequence is bounding. This methodology will be consistent with NUREG-1537.

This TR also discusses the methodology for safety classification of SSCs. A future TR will provide the principal design criteria (PDC) for the UIUC MMR which establish necessary design, fabrication, construction, testing, and performance requirements for safety-related SSCs.

Commented [JC2]: Editorial update

DID will be considered throughout the design process for the MMR and will be discussed in detail in the CPA and the OLA for the MMR.

1.5. NRC ACTION REQUESTED

UIUC is requesting NRC review and approval of the event sequence methodology in Section 2.0 and the safety classification methodology in Section 3.0 used for the MMR at UIUC.

Per the NRC draft guidance provided in Reference 3, a preliminary list of postulated initiating events (PIEs) and SSC classifications are provided in Appendix A and B, respectively. These appendices, and the pipe break classification example in Section 3.0 are provided for information purposes to assist this TR review and not requested for approval at this time.

2.0 IDENTIFICATION OF EVENT SEQUENCES

2.1. REGULATORY FOUNDATION FOR EVENT SEQUENCE IDENTIFICATION METHODOLOGY

This section provides a summary of the applicable NRC regulatory requirements regarding the identification of event sequences for the MMR.

NRC reactor regulations mandate that safety analysis to assess the adequacy of the design during anticipated transient conditions must be performed and provided to the NRC. Specifically, 10 CFR 50.34(a)(4) states that a CPA under Part 50 must include:

> *"A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety* ***during normal operations and transient conditions anticipated during the life of the facility****, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents."*
> [emphasis added]

Safety analysis of the event sequences identified using the methodology described in this section of the TR is intended to meet the requirements outlined in 10 CFR 50.34(a)(4). As discussed in Section 1.4~~1.3~~, the safety analysis methodology used for the UIUC MMR deployment will be provided in a future TR.

Commented [JC3]: Section reference corrected

2.2. DEFINITIONS RELATED TO EVENT SEQUENCE IDENTIFICATION

2.2.1. Postulated Initiating Event

A PIE is defined in UIUC MMR licensing basis as:

> A postulated event identified in design as capable of leading to anticipated operational occurrences or accident conditions.
>
> Note: A postulated initiating event is not an entire sequence itself; it is the event that initiates a sequence.

PIE types include:

- piping system breaches,
- transients (i.e., non-pipe breach reactor events such as reactivity additions),
- internal hazard induced PIEs (e.g., reactor building fire), and
- external hazard induced PIEs (e.g., severe weather).

2.2.2. Event Sequence

An event sequence is defined as:

were created for LWR technologies, Table 2-1~~Table 2-1~~ provides the corresponding event sequence category for the MMR. Event sequence categories may be added and/or removed as the event sequence list matures.

Table 2-1. Grouping of MMR Event Sequences

NUREG-1537 Accident Categories	MMR Event Sequence Categories
MHA	MHA
Insertion of Excess Reactivity	Insertion of Excess Reactivity
Loss of Coolant	D-LOFC (depressurized loss of forced cooling) with air ingress
Loss of Coolant Flow	P-LOFC (pressurized loss of forced cooling)
Mishandling or Malfunction of Fuel	Mishandling or Malfunction of Fuel
Experiment Malfunction	N/A (No in-core experiments)
Loss of Normal Electrical Power	Loss of Normal Electrical Power
External Events	Internal and External Hazards
Mishandling or Malfunction of Equipment	Mishandling or Malfunction of Equipment

Event sequences will also be grouped under a limiting event sequence, such that the group of event sequences will rely on the same set of responding SSCs to perform safety functions.

- Control of reactivity,
- Removal of heat from the reactor, and
- Control release of radioactive material that could exceed public dose limits.

Note, control of reactivity encompasses reactor shutdown.

3.2.2. Safety Classification Groups

The following two classification groups are used for the UIUC MMR safety classification methodology:

- **Safety-Related (SR):** SSCs that have an impact on safety and are relied upon to remain functional to meet the three safety functions in Section 3.2.1 during and following all event sequences part of the plant design-basis.
- **Non-Safety-Related (NSR):** SSCs not required to remain functional to meet the three safety function in Section 3.2.1.

3.3. METHODOLOGY FOR SSC SAFETY CLASSIFICATION

This section defines the approach used for determining the safety classification and applicability of design requirements for SSCs. The purpose of safety classification is to ensure that SSCs are designed, fabricated, inspected, tested, operated, and maintained based on their roles in preventing and/or mitigating event sequences. This process is iterative during the design. The preliminary SSC safety classifications are provided in Appendix B.

3.3.1. Identify ~~Limiting PIEs~~ Limiting Event Sequences Relevant to Safety Classification

The ~~limiting PIEs~~ limiting event sequences will be identified and serve as an input in the safety classification methodology.

3.3.2. Identify SSCs Required to Achieve Safety Functions During Event Sequences

The next step in the classification process is to identify the SSCs that are required to achieve each of the safety functions for the ~~limiting PIEs~~limiting event sequences. This analysis is performed by separately evaluating each ~~limiting PIE~~limiting event sequence. Each analysis considers all three safety functions: 1) control of reactivity, 2) removal of heat from the reactor, and 3) control release of radioactive material that could exceed public dose limits.

Pipe Breach Example

Table 3-1~~Table 3-1~~ identifies the systems that contain SSCs necessary to perform the safety functions for a hypothetical pipe breach of the HPB. A narrative of the pipe breach analysis, discussing how determinations of safety-related SSCs, is provided in

Table 3-1. Safety Function Analysis for Pipe Breach ~~PIE~~Event Sequence

	Systems																									
	JK – Reactor Core	**JR – RPS**	**JA – Reactor Vessel**	**JB – Core Support Structure**	**JD – RCSS**	**KA – RCCS**	**UJ – Citadel**	**B0 – NP Elect. Aux. Power**	**JT – SPDS**	**JY – Monitoring**	**XS – Access Control**	**Y0 – Comm. & Info. Systems**	**C0 – NP Control, Data, & Instr.**	**JG – Molten Salt**	**JS – RCS**	**JE – HTS**	**KB – He Purification Supply**	**KL – HVAC**	**KM/N/P – Waste Treatment**	**KT – Drains**	**KU – Sampling**	**UK – Nuclear Building**	**VA – NP Storage**	**XF – Earthing**	**XG – Fire System**	**XK – Chilled Water**
Provides Safety Function?	Y	Y	Y	Y	Y	Y	Y	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N

Safety Function Analysis:

Y = Yes; System contains an SSC that is credited to provide at least one safety function; therefore, is safety-related

N = No; System does not contain an SSC that is credited to provide at least one safety function

Control of Reactivity:

The Reactor Protection System (RPS) and Reactivity Control and Shutdown System (RCSS) are required to be safety-related to trip the reactor after the break and maintain a subcritical state post reactor-trip.

The Reactor Core is required to be safety-related because the fuel and graphite structures provide core geometry to allow insertion of the control rods of the reactivity control and shutdown system to insert under gravity. To ensure core geometry it is essential that the core is retained in position, hence the Core Support Structure, Vessel and Citadel provide the structural support for the core to ensure core geometry and rod insertion.

Removal of Heat from Reactor:

When a pipe breach occurs, forced cooling from the helium is quickly lost. Only passive SSCs are credited to provide the safety function to remove decay heat from the reactor for this ~~PIE~~event sequence.

Passive components of the Reactor Cavity and Cooling System (RCCS), i.e., water in the RCCS standpipes, are required to be safety related to provide heat capacity to remove the heat following the reactor trip.

The Reactor Core, Core Support Structure, Reactor Vessel, and Citadel are all required to be safety-related as these SSCs' geometry and heat capacity are credited to reject heat radially from the fuel to the surrounding soil and bedrock in the longer term to remove decay heat.

Control release of radioactive material that could exceed public dose limits:

The Reactor Core is required to be safety-related to meet this safety function. Specifically, the barriers to fission product release (i.e., retention layers in TRISO particles and FCM) are credited to confine radioactive material.

All other systems are non-safety related.

3.3.3. Assign SSCs to Classification Groups

After the safety function analysis is completed for the ~~limiting PIEs~~limiting event sequences, any SSC that is required to provide a safety function ~~for a PIE~~ is classified as safety-related. All other SSCs are classified as non-safety-related.

3.3.4. Apply Engineering Design Rules

The PDC, which will be provided in a future TR, will establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components. Details of engineering design rules applied to safety-related SSCs to satisfy the PDC will be provided in the CPA.

Document Number	**:**	**IMRDD-MMR-23-03-A**
Release	**:**	**02**
Status	**:**	**Issued**
Issue Date	**:**	**2024/02/28**
NRC Project No.	**:**	**99902094**

CONFIGURATION CONTROL SUMMARY

Document Revision History

Document No.	Rel.	Date	Prepared By	Revision Description
IMRDD-MMR-23-03	01	2023/08/24	MPR Associates, Inc.	Initial Issue for NRC review
IMRDD-MMR-23-03	02	2024/02/28	Michael Hamer, USNC	Revision to include UIUC responses to the NRC's RAIs received on 12-Dec-2023 [ML23354A009]

Revision Summary for Release 02	
RAI-1	UIUC's definition of "safety-related" is intended to include SSC's required for safe shutdown from normal operation, as well as those relied upon during and following event sequences. Section 3.2.1 was revised to add the following statement: "Control of reactivity encompasses reactor shutdown."
RAI-2	Section 2.3.4, "Grouping of Event Sequences", was revised to clarify how "limiting PIEs" align with the event sequence categories and to address how more than one limiting event sequence may exist for each category. Section 2.3.4 was revised to add the following statement: "Event sequences will also be grouped under a limiting event sequence, such that the group of event sequences will rely on the same set of responding SSCs to perform safety functions." Sections 3.3.1, 3.3.2 & 3.3.3: Occurrences of "limiting PIEs" or "PIE" have been replaced with "limiting event sequences" or "event sequence" respectively. Table 3-1 Title: "PIE" has been replaced with "Event Sequence". Section 4.0 References: Added reference #14 for the UIUC RAI Response Letter to the USNRC [ML24017A308, ML24017A309, ML24017A310] Minor inconsequential editorial corrections included throughout the TR as needed.

Document Approvals

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Approvals	Name/Organization	Title	Signature	Date
Reviewer	Tim Grunloh UIUC	Project Associate Director	DocuSigned by: Timothy Grunloh 8C98154D67E24DD...	20-Feb-24
Approver	Caleb Brooks UIUC	Associate Professor & Project Lead	DocuSigned by: Caleb Brooks EB6137AF6F3C41A...	20-Feb-24

EXECUTIVE SUMMARY

The University of Illinois Urbana-Champaign (UIUC) is proposing to construct a research reactor (also known as a non-power utilization facility) using high temperature gas-cooled reactor (HTGR) technology. UIUC plans to build and operate Ultra Safe Nuclear Corporation's (USNC) Micro Modular Reactor™ (MMR™) HTGR-based design.

This topical report (TR) describes the following processes to be used for the licensing of the MMR at UIUC:

1. The identification of credible event sequences that will be considered for the MMR design, and
2. The safety classification of systems, structures, and components (SSCs) appropriate for their function(s) in meeting the design basis.

Details of how the design addresses these aspects will be provided in the preliminary and final safety analysis reports (PSAR and FSAR, respectively).

This TR presents a simplified, deterministic approach to identify event sequences and classify SSCs that aligns with the guidance provided in NUREG-1537 (Reference 1). The standard MMR design developed by USNC uses probabilistic risk assessment (PRA) insights to inform the generic plant design; however, the UIUC deployment will not use PRA in the licensing basis and will assess design acceptability with regard to the radiological limits applicable for research reactors.

The methodologies presented in this TR are iterative and will be used numerous times as the plant design matures.

UIUC is requesting NRC review and approval of the event sequence methodology in Section 2.0 and the safety classification methodology in Section 3.0 used for the MMR at UIUC. Preliminary lists of PIEs and SSC Safety Classifications (Appendix A and B, respectively) and the pipe breach classification example in Section 3.0 are provided for information only, NRC approval is not requested.

CONTENTS

ABBREVIATIONS & ACRONYMS

This list contains the abbreviations and acronyms used in this document.

Abbreviation or Acronym	Definition
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ASME	American Society of Mechanical Engineers
CNSC	Canadian Nuclear Safety Commission
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CPA	Construction Permit Application [per 10 CFR 50]
DID	Defense-in-Depth
D-LOFC	Depressurized loss of flow cooling
ECCS	Emergency core cooling system
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PIE	Postulated Initiating Event
P-LOFC	Pressurized Loss of Flow Cooling
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
PSE	Planned Special Exposure

Abbreviation or Acronym	Definition
PWR	Pressurized Water Reactor
RCCS	Reactor Cavity Cooling System
RCSS	Reactor Control and Shutdown System
RPS	Reactor Protection System
SPDS	Safety Parameter Display System
SSCs	Systems, Structures, and Components
TEDE	Total Effective Dose Equivalent
TR	Topical Report
TRISO	Tri-Structural Isotropic
UCO	Uranium Oxycarbide
UIUC	University of Illinois Urbana – Champaign
U.S.	United States
USNC	Ultra Safe Nuclear Corporation (i.e., the reactor design vendor)

1.0 INTRODUCTION

USNC has undertaken a project to build an MMR at UIUC to serve as a research reactor (i.e., a Class 104 utilization facility) in accordance with 10 CFR 50.21(c).

1.1. PURPOSE

The purpose of this TR is to describe the UIUC MMR methodologies used for:

1. The identification of credible event sequences that will be considered for the MMR design, and
2. The safety classification of SSCs appropriate for their function(s) in meeting the design basis.

A definition for the term event sequence is provided in Section 2.2.

1.2. UIUC MMR DEPLOYMENT BACKGROUND

The MMR is a non-power research HTGR that will be licensed to a maximum operating power limit of 10 MWth for deployment to the UIUC. The MMR is an HTGR limited to the research reactor power limit for the UIUC deployment. The reactor will be fueled with High Assay Low-Enriched Uranium (HALEU) at an enrichment from 5% to 19.75% ^{235}U in the form of tri-structural isotropic (TRISO) particles embedded in silicon carbide Fully Ceramic Micro-Encapsulated (FCM™) pellets that are stacked in columns in solid hexagonal graphite blocks. The MMR is designed for passive safety response to event sequences in the design basis and relies on functional containment as the primary means to limit release of radioactivity to the environment.

UIUC will apply for a Construction Permit per 10 CFR 50 per the schedule provided in Reference 13. As a non-water, non-power reactor, the UIUC MMR does not match the underlying assumptions that form the basis for many NRC regulations (as discussed and assessed in Reference 4).

In Reference 3, the NRC discusses approaches to improving the timeliness and efficiency of advanced reactor licensing reviews through early interactions. A key action is submitting TRs for staff review and approval. One such TR should discuss the “proposed process for selection of licensing basis events and classification and treatment of SSCs…”.

1.3. MMR TECHNOLOGY BACKGROUND

The MMR uses technology and safety capabilities considerably different from Light Water Reactor (LWR) technology that is the focus of many regulations. For example, the MMR does not require an active or passive emergency core cooling system (ECCS) to rapidly replenish primary coolant to recover the fuel in the event of a rupture of the primary pressure boundary. Large safety margins are provided by both the fuel and the reactor design.

- The fuel is comprised of TRISO particles, which provide a highly effective fission product retention capability. In response to an Electric Power Research Institute

(EPRI) TR on the performance of TRISO fuel, the U.S. NRC issued a Safety Evaluation Report (SER), Reference 5, with some limitations and conditions that are considered in the MMR design. The fission product retention capability of TRISO fuel particles contributes to a “functional containment” whereby the TRISO particles serve as the first containment barrier when operated within the range of qualification parameters.

- The TRISO particles in MMR fuel are encased in an FCM pellet of SiC that provides an additional layer of retention for fission products, thereby achieving functional containment.
- The low power density of active fuel region leads to slow fuel heat-up during loss of heat removal events.
- Low thermal power results in a small inventory available for release of the most limiting short-lived fission products for public safety, such as ^{131}I and ^{85}Kr. The increased inventory of long-lived fission products associated with a long core life is addressed by the defense-in-depth (DID) approach to functional containment.
- The low power rating also reduces the decay heat that must be removed in postulated accidents, simplifying passive decay heat removal.
- Heat transfer fluid used for core cooling during normal operation is an inert, chemically stable, single-phase gas (helium).
- Safety-related core cooling is passive and capable of maintaining fuel and component temperatures below limits with no helium, electrical power, or operator action.
- Intermediate heat transfer is performed by a molten salt loop that effectively isolates the reactor from transients in the adjacent plant power conversion system.
- The reactor is below ground. Although it does not have nor need a leak-tested containment building, it is surrounded by a concrete structure (the citadel) that provides DID for retention of fission products to the environment and provides protection against external hazards.

Table 1-1 provides a high-level summary of key design features of the MMR and, in comparison, to current LWRs.

Table 1-1. MMR Key Features and Differences from Operating LWRs

Feature	MMR	LWR	Remarks
Operating power level	Maximum research reactor allowed power (10 MW) for UIUC deployment	3000 to 4400 MW(t) (AP1000 3415 MW(t))	Full MMR power is less than decay heat of large LWR more than 24 hours after shutdown; short-lived fission product inventory is small
Heat transfer fluid	Helium – inert gas; single phase under all conditions; low stored energy	Water – also serves as moderator; scrubs fission products; high stored energy; undergoes phase change that causes high pressure and temperature in surrounding structure	Water coolant causes corrosion, and blowdown can damage safety systems by impingement, pressure, moisture, and temperature
Containment	Functional: TRISO integrity at high temperatures, with supplemental passive barriers for defense-in-depth, continuously confirmed by radiological monitoring while operating	Large containment building: subject to high pressure and temperature; many penetrations requiring active isolation, periodic leak testing, and maintenance	Functional containment is a set of barriers that effectively limit physical transport of radioactive material to the environment and serve as basis for the revised Principal Design Criteria (PDC) in RG 1.232 (Reference 6)
Safety-related ac power systems	None	Class 1E ac distribution and emergency diesel generators	MMR safety is provided by passive systems
Fuel form	Uranium oxycarbide (UCO) TRISO particles encased in FCM pellets in hexagonal graphite fuel blocks	Uranium dioxide pellets encased in zirconium alloy tubes	Insubstantial fission product release from MMR during operation or accidents
Fuel (^{235}U) enrichment	HALEU ≤ 19.75% ^{235}U	LEU < 5% ^{235}U	Both are low-enriched uranium; MMR higher enrichment provides for longer core life
Fuel damage temperature	> 3272˚F (1800˚C)	2200˚F (1204˚C)	Zirconium-water reactions start at about 1800˚F in LWRs
Emergency replenishment of coolant	None; fuel limits met for unmitigated primary system blowdown	ECCS needed	Must quickly recover LWR fuel with water if loss of coolant occurs
Hydrogen management	External/internal flooding might release hydrogen (graphite-water reaction)	Zirconium-water reaction produces hydrogen if clad exceeds 2200˚F	Acceptance criteria limit mass of LWR fuel clad reacted
Primary system corrosion mechanisms	While helium itself is non-corrosive, contaminants must be controlled to low levels to avoid degradation of graphite and other materials	Various types of stress corrosion cracking; boric acid corrosion (PWRs)	Helium is inert whereas hot water is corrosive unless water chemistry is carefully controlled

1.4. SCOPE

UUIC will be applying for a 10 CFR 50 Construction Permit (CP) and subsequent Class 104 Operating License (OL), using NUREG-1537 (Reference 1). This TR is part of pre-submittal activities to support NRC review of the CPA and Operating License Application (OLA).

UIUC is the license applicant and owner/operator of the MMR non-power utilization facility, with USNC as reactor designer/vendor/original equipment manufacturer and fuel supplier.

This TR considers regulations and guidance for preparing CPA and OLA for a research reactor facility in accordance with 10 CFR 50 (Reference 2) and NUREG-1537. The facility will be licensed under the provisions of 10 CFR 50.21(c) as a Class 104, non-power utilization facility. The MMR is also a non-light water reactor (non-LWR). These characterizations limit applicability of some NRC regulations. Reference 4 discusses the applicable NRC regulations for the MMR at UIUC.

This TR discusses the methodology to identify credible event sequences. A future TR will discuss the deterministic methodology used to (1) identify a Maximum Hypothetical Accident (MHA) that bounds the dose consequence associated the identified credible event sequence, (2) calculate the dose consequence associated with the MHA, and (3) analyze the limiting credible event sequence to demonstrate that the MHA dose consequence is bounding. This methodology will be consistent with NUREG-1537.

This TR also discusses the methodology for safety classification of SSCs. A future TR will provide the principal design criteria (PDC) for the UIUC MMR which establish necessary design, fabrication, construction, testing, and performance requirements for safety-related SSCs.

DID will be considered throughout the design process for the MMR and will be discussed in detail in the CPA and the OLA for the MMR.

1.5. NRC ACTION REQUESTED

UIUC is requesting NRC review and approval of the event sequence methodology in Section 2.0 and the safety classification methodology in Section 3.0 used for the MMR at UIUC.

Per the NRC draft guidance provided in Reference 3, a preliminary list of postulated initiating events (PIEs) and SSC classifications are provided in Appendix A and B, respectively. These appendices, and the pipe break classification example in Section 3.0 are provided for information purposes to assist this TR review and not requested for approval at this time.

2.0 IDENTIFICATION OF EVENT SEQUENCES

2.1. REGULATORY FOUNDATION FOR EVENT SEQUENCE IDENTIFICATION METHODOLOGY

This section provides a summary of the applicable NRC regulatory requirements regarding the identification of event sequences for the MMR.

NRC reactor regulations mandate that safety analysis to assess the adequacy of the design during anticipated transient conditions must be performed and provided to the NRC. Specifically, 10 CFR 50.34(a)(4) states that a CPA under Part 50 must include:

> *"A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety* ***during normal operations and transient conditions anticipated during the life of the facility****, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents."* [emphasis added]

Safety analysis of the event sequences identified using the methodology described in this section of the TR is intended to meet the requirements outlined in 10 CFR 50.34(a)(4). As discussed in Section 1.4, the safety analysis methodology used for the UIUC MMR deployment will be provided in a future TR.

2.2. DEFINITIONS RELATED TO EVENT SEQUENCE IDENTIFICATION

2.2.1. Postulated Initiating Event

A PIE is defined in UIUC MMR licensing basis as:

> A postulated event identified in design as capable of leading to anticipated operational occurrences or accident conditions.
>
> Note: A postulated initiating event is not an entire sequence itself; it is the event that initiates a sequence.

PIE types include:

- piping system breaches,
- transients (i.e., non-pipe breach reactor events such as reactivity additions),
- internal hazard induced PIEs (e.g., reactor building fire), and
- external hazard induced PIEs (e.g., severe weather).

2.2.2. Event Sequence

An event sequence is defined as:

1) a PIE or combination of PIEs that initially perturbs the plant,

2) the resulting response following the PIE(s), and

3) the resulting well-defined end state.

For the deterministic licensing approach used for the UIUC MMR deployment, only SSCs classified as safety-related using the methodology in Section 3.0 are credited to respond to PIEs. Additionally, the worst-case failure of any active component is assumed (i.e., single-failure criterion) when defining event sequences.

2.3. EVENT SEQUENCE IDENTIFICATION METHODOLOGY

2.3.1. Identify PIEs

The PIE identification methodology is split into three phases: (1) Initial phase, (2) Top-down phase, and (3) Bottom-up phase based on the maturity of the UIUC MMR design. The PIE list provided in Appendix A is based on the Initial phase methodology.

Initial Phase

An initial list of PIEs is first developed to provide conceptual information to the design process. This initial PIEs list is developed using historical information, regulatory documents, and engineering judgement.

The following historical and regulatory sources were considered and assessed against the MMR design to identify applicable PIEs, and screen out non-applicable PIEs where the technology was fundamentally different from that of the MMR, or not applicable:

- Fort St. Vrain Safety Analysis Report (Reference 8),
- MHTGR Licensing Basis Events (Reference 9), and
- CNSC, IAEA, and US NRC publications (References 10, 11, and 12).

Additional PIEs were also proposed, considered, and screened via engineering judgment.

For internal and external hazards, generic industry accepted hazard lists are used as a starting point for hazard identification.

The initial list of PIEs will be reconciled with the PIEs list using the Top-down and Bottom-up phase methodologies. Initial PIEs that are not captured by these two methodologies will be retained to ensure industry experience is considered.

Top-down Phase

In the top-down phase of PIE identification, the design is mature enough to follow a “top-down” approach in determining PIEs. The following three fundamental safety functions have been identified for the MMR:

- Control of reactivity,
- Removal of heat from the reactor, and
- Control release of radioactive material that could exceed public dose limits.

By analyzing challenges to these functions, the PIEs identified at the initial phase can be confirmed or extended. Given the increased design maturity, specific initiating events can be identified. This process starts with the effect and determines a cause. For example, a reactivity excursion is the “effect”, and improperly controlling reactivity is the possible cause that challenges the control of reactivity safety function. By cascading down the function tree, initiating events can be identified at the point where the function has been allocated to a human, automation system or hardware. Master logic diagrams (MLDs) will be used to perform the top-down phase methodology. Detailed description of the MLD methodology and results will be provided in the OLA.

Bottom-up Phase

In this phase it is possible to follow a “bottom-up” approach where the design is now defined in sufficient detail such that failures can more easily be determined. This phase acts as a verification and extension of the “top-down” approach. The bottom-up phase includes a unique methodology for each of PIE types listed in 2.2.1.

Pipe Breaches

Design descriptions are assessed to determine whether the breach of a piping system can result in anticipated operational occurrences or accident conditions. Generally, piping systems that contain reactor coolant and/or radioactive material, or breaches that could give rise to an internal hazard are retained for further analysis.

Specific pipe break locations are identified for retained piping systems. For helium pressure boundary (HPB) and molten salt system (MSS) breaches, all breach locations are retained. For other systems, breach locations can be qualitatively screened out depending on sources of radioactivity and hazard consequences/effect on plant.

Transients

Failure modes and effects analysis (FMEAs) will be performed to identify PIEs for transients (non-pipe breach reactor events). Detailed description of the FMEA methodology and results will be provided in the OLA.

Internal Hazards

An Internal Hazard Assessment identifies internal hazards associated with the standard MMR product, and project specific configuration aspects. A Hazard Analysis is performed, to identify hazard induced PIEs. Detailed description of the Internal

Hazard Assessment and Analysis methodology and results will be provided in the OLA.

External Hazards

An External Hazard Assessment identifies external hazards associated with the UIUC site. A Hazard Analysis is performed, to identify hazard induced PIEs. Detailed description of the External Hazard Assessment and Analysis methodology and results will be provided in the OLA.

2.3.2. Screen PIEs

In alignment with NUREG-1537 (Reference 1), only credible events, on a deterministic basis, are considered (an MHA will be identified that bounds the consequences of all credible events). Therefore, PIEs that are determined to be non-credible are screened out.

For PIEs related to an SSC failure (e.g., pipe breach), the engineering design rules for SSCs associated with each identified PIE will be reviewed. A determination will be made if the PIE is credible or not based on the specifics of the SSC failure and/or the engineering design rules applied to the SSC. For example, a double-ended guillotine break at the hot gas duct PIE would be screened out as non-credible because it's expected the hot gas duct is designed to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (Reference 7), which practically eliminates large ruptures from occurring.

For external hazards, a determination of PIE credibility will be made based on the specifics of the external hazard and/or available hazard frequency information from the External Hazard Assessment.

If sufficient information does not exist to screen out a PIE, the PIE is retained as part of the plant licensing basis until it can be properly assessed.

2.3.3. Defining Event Sequences

After PIEs are identified, the safety classification methodology in Section 3.0 is used to determine which SSCs are required to mitigate the consequences of the PIEs. SSCs that are classified as safety-related determine the plant response to each PIE (or combination of PIEs) to define event sequences deterministically.

Additionally, the subsequent worst-case single failure of an active component will be determined for each event sequence and assumed when performing safety analysis.

2.3.4. Grouping of Event Sequences

Once credible event sequences are identified, they are grouped using the accident categories provided in NUREG-1537 (Reference 1). Given that the accident categories were created for LWR technologies, Table 2-1 provides the corresponding event sequence category for the MMR. Event sequence categories may be added and/or removed as the event sequence list matures.

Table 2-1. Grouping of MMR Event Sequences

NUREG-1537 Accident Categories	MMR Event Sequence Categories
MHA	MHA
Insertion of Excess Reactivity	Insertion of Excess Reactivity
Loss of Coolant	D-LOFC (depressurized loss of forced cooling) with air ingress
Loss of Coolant Flow	P-LOFC (pressurized loss of forced cooling)
Mishandling or Malfunction of Fuel	Mishandling or Malfunction of Fuel
Experiment Malfunction	N/A (No in-core experiments)
Loss of Normal Electrical Power	Loss of Normal Electrical Power
External Events	Internal and External Hazards
Mishandling or Malfunction of Equipment	Mishandling or Malfunction of Equipment

Event sequences will also be grouped under a limiting event sequence, such that the group of event sequences will rely on the same set of responding SSCs to perform safety functions.

3.0 CLASSIFICATION OF SYSTEMS, STRUCTURES, AND COMPONENTS

3.1. REGULATORY FOUNDATION FOR CLASSIFICATION OF SYSTEMS, STRUCTURES, AND COMPONENTS

This section provides a summary of the applicable NRC regulatory requirements regarding the safety classification of SSCs for the MMR plant.

10 CFR 50.2 defines safety-related SSCs as those SSCs that are:

> *"…relied upon to remain functional during and following design basis events to assure:*
>
> 1) *The integrity of the reactor coolant pressure boundary;*
> 2) *The capability to shut down the reactor and maintain it in a safe shutdown condition; or*
> 3) *The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.…"*

The definition of safety-related SSCs in 10 CFR 50.2 is applicable to water-cooled reactors. Per Reference 4, Attachment 1, the UIUC MMR will "meet the intent" of the definition of safety-related SSCs. Alternative safety functions appropriate for the UIUC MMR are provided in Section 3.2.1 to meet the intent of 10 CFR 50.2.

As discussed in Section 2.1, 10 CFR 50.34(a)(4) states that a CPA under Part 50 must include:

> *"A preliminary analysis and evaluation of the design and* ***performance of structures, systems, and components*** *of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and* ***the adequacy of structures, systems, and components provided for the prevention*** *of accidents and the mitigation of the consequences of accidents."* [emphasis added]

There are corresponding requirements regarding the OLA.

3.2. DEFINITIONS RELATED TO SSC CLASSIFICATIONS

3.2.1. Safety Functions

A safety function is a specific purpose that must be accomplished for safety for a facility or activity to prevent or to mitigate radiological consequences of normal operation, anticipated operational occurrences, and accident conditions.

The UIUC MMR safety classification methodology considers the fundamental safety functions below in lieu of the functions for safety-related SSCs in 10 CFR 50.2.

- Control of reactivity,
- Removal of heat from the reactor, and
- Control release of radioactive material that could exceed public dose limits.

Note, control of reactivity encompasses reactor shutdown.

3.2.2. Safety Classification Groups

The following two classification groups are used for the UIUC MMR safety classification methodology:

- **Safety-Related (SR):** SSCs that have an impact on safety and are relied upon to remain functional to meet the three safety functions in Section 3.2.1 during and following all event sequences part of the plant design-basis.
- **Non-Safety-Related (NSR):** SSCs not required to remain functional to meet the three safety function in Section 3.2.1.

3.3. METHODOLOGY FOR SSC SAFETY CLASSIFICATION

This section defines the approach used for determining the safety classification and applicability of design requirements for SSCs. The purpose of safety classification is to ensure that SSCs are designed, fabricated, inspected, tested, operated, and maintained based on their roles in preventing and/or mitigating event sequences. This process is iterative during the design. The preliminary SSC safety classifications are provided in Appendix B.

3.3.1. Identify Limiting Event Sequences Relevant to Safety Classification

The limiting event sequences will be identified and serve as an input in the safety classification methodology.

3.3.2. Identify SSCs Required to Achieve Safety Functions During Event Sequences

The next step in the classification process is to identify the SSCs that are required to achieve each of the safety functions for the limiting event sequences. This analysis is performed by separately evaluating each limiting event sequence. Each analysis considers all three safety functions: 1) control of reactivity, 2) removal of heat from the reactor, and 3) control release of radioactive material that could exceed public dose limits.

Pipe Breach Example

Table 3-1 identifies the systems that contain SSCs necessary to perform the safety functions for a hypothetical pipe breach of the HPB. A narrative of the pipe breach analysis, discussing how determinations of safety-related SSCs, is provided in the following table. Note that the SSC names and classifications provided in this section are preliminary and provided as an example to demonstrate the safety classification methodology. The SSC names and classifications are subject to change as the design matures.

Table 3-1. Safety Function Analysis for Pipe Breach Event Sequence

	Systems																									
	JK – Reactor Core	JR – RPS	JA – Reactor Vessel	JB – Core Support Structure	JD – RCSS	KA – RCCS	UJ – Citadel	B0 – NP Elect. Aux. Power	JT – SPDS	JY – Monitoring	XS – Access Control	Y0 – Comm. & Info. Systems	C0 – NP Control, Data, & Instr.	JG – Molten Salt	JS – RCS	JE – HTS	KB – He Purification Supply	KL – HVAC	KM/N/P – Waste Treatment	KT – Drains	KU – Sampling	UK – Nuclear Building	VA – NP Storage	XF – Earthing	XG – Fire System	XK – Chilled Water
Provides Safety Function?	Y	Y	Y	Y	Y	Y	Y	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N	N

Safety Function Analysis:

Y = Yes; System contains an SSC that is credited to provide at least one safety function; therefore, is safety-related

N = No; System does not contain an SSC that is credited to provide at least one safety function

Control of Reactivity:

The Reactor Protection System (RPS) and Reactivity Control and Shutdown System (RCSS) are required to be safety-related to trip the reactor after the break and maintain a subcritical state post reactor-trip.

The Reactor Core is required to be safety-related because the fuel and graphite structures provide core geometry to allow insertion of the control rods of the reactivity control and shutdown system to insert under gravity. To ensure core geometry it is essential that the core is retained in position, hence the Core Support Structure, Vessel and Citadel provide the structural support for the core to ensure core geometry and rod insertion.

Removal of Heat from Reactor:

When a pipe breach occurs, forced cooling from the helium is quickly lost. Only passive SSCs are credited to provide the safety function to remove decay heat from the reactor for this event sequence.

Passive components of the Reactor Cavity and Cooling System (RCCS), i.e., water in the RCCS standpipes, are required to be safety related to provide heat capacity to remove the heat following the reactor trip.

The Reactor Core, Core Support Structure, Reactor Vessel, and Citadel are all required to be safety-related as these SSCs' geometry and heat capacity are credited to reject heat radially from the fuel to the surrounding soil and bedrock in the longer term to remove decay heat.

Control release of radioactive material that could exceed public dose limits:

The Reactor Core is required to be safety-related to meet this safety function. Specifically, the barriers to fission product release (i.e., retention layers in TRISO particles and FCM) are credited to confine radioactive material.

All other systems are non-safety related.

3.3.3. Assign SSCs to Classification Groups

After the safety function analysis is completed for the limiting event sequences, any SSC that is required to provide a safety function is classified as safety-related. All other SSCs are classified as non-safety-related.

3.3.4. Apply Engineering Design Rules

The PDC, which will be provided in a future TR, will establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components. Details of engineering design rules applied to safety-related SSCs to satisfy the PDC will be provided in the CPA.

4.0 REFERENCES

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A PRELIMINARY LIST OF PIES

A preliminary list of postulated initiating events and related assumptions for the UIUC MMR licensing basis are provided in this appendix. This list was generated by using the Initial Phase methodology described in Section 2.3.1. The final list of PIEs will be provided in the OLA.

A.1 INSERTION OF EXCESS REACTIVITY

- Molten Salt Maximum Flow (Overcooling Event)
  - Molten salt flow instantly changes from normal parameters to minimum inlet temperature and maximum flow rate simultaneously.
- Spurious withdrawal of Single Control Rod
  - Single, maximum worth, control rod is spuriously withdrawn from current position at limiting withdrawal speed.
- Spurious withdrawal of Control Rod Group
  - Three control rods are spuriously withdrawn from current position at limiting withdrawal speed.
- Simultaneous withdrawal of All Actuatable Rods
  - All control rods are spuriously withdrawn at the maximum withdrawal speed.
- Single Rod Ejection
  - Single, maximum worth, control rod is ejected from its current position instantaneously due to a failure in the pressure boundary and failure to restrain the control rod.

A.2 D-LOFC

Note that the impact of air ingress into the primary coolant loop must be evaluated for D-LOFC event sequences.

- Un-isolated small breach in HPB
- Un-isolated small pipe breach outside Citadel
- Small Intermediate Heat Exchanger (IHX) Leak
    - A small IHX leak occurs when pressurized helium leaks into the salt side.
- Breach in largest connecting pipe to pressure boundary

A.3 P-LOFC

- Blockage of Helium Coolant Channel
    - Blockage in helium flow in a single fuel element
- Loss of Direct Reactor Cooling (Loss of Primary Helium Flow)
- Molten Salt Flow Stop
- Loss of Secondary Heat Sink
- Total Loss of Forced Flow (Total Loss of Primary and Secondary Flow)

A.4 MISHANDLING OR MALFUNCTION OF FUEL

- Inadvertent Load and Operation of a Fuel Element in an Improper Position

A.5 INTERNAL AND EXTERNAL HAZARDS

Internal and external hazards considered during the design of the UIUC MMR are provided in this section.

Generic Internal Hazards

- Internal Fires
- Internal Explosions
- Internal Missiles and Pipe Breaches
- Internal Flooding
- Heavy Load Drop
- Electromagnetic Interference
- Release of Hazardous Substances inside the Plant

Generic External Hazards

- External Floods
- Hazardous Substances
- Aircraft Crash
- EMF including Solar Storms
- Biological Phenomena
- Meteorological disturbances (such as hurricane, tornado, or flood)
- Seismic event
- Mechanical impact or collision with building
- Event caused by humans, such as explosion or toxic release near the reactor building

B PRELIMINARY SSC SAFETY CLASSIFICATIONS

The preliminary systems and structures safety classifications are provided in Table B-1. The methodology discussed in Section 3.3 is used to generate the classifications. Sub-system safety classifications are provided in Table B-2.

Table B-1. Preliminary Systems and Structures Safety Classifications

System/ Structure Code	System/Structure	UIUC MMR Safety Classification
JK	Reactor Core	Safety-Related
JR	Reactor Protection System (RPS)	Safety-Related
JA	Reactor Vessel	Safety-Related
JB	Core Support Structure	Safety-Related
JD	Reactivity Control and Shutdown System (RCSS)	Safety-Related
KA	Reactor Cavity Cooling System (RCCS – Only Passive Cooling Components)	Safety-Related
UJ	Citadel	Safety-Related
B0	Nuclear Plant (NP) Electrical Auxiliary Power	Non-safety-related
C0	NP Controls, Data, and Instrumentation	Non-safety-related
JE	Heat Transport System (HTS)	Non-safety-related
JG	Molten Salt (MS) System	Non-safety-related
JS	Reactivity Control System (RCS)	Non-safety-related
JT	Safety Parameter Display System (SPDS)	Non-safety-related
JY	Radiation Monitoring System	Non-safety-related
KB	Helium Purification and Supply System	Non-safety-related
KL	HVAC System	Non-safety-related
KM/N/P	Waste Treatment	Non-safety-related
KT	Drains	Non-safety-related
XF	Earthing	Non-safety-related
XG	Fire System	Non-safety-related
XS	Access Control	Non-safety-related
Y0	Communication and Information Systems	Non-safety-related
KU	Sampling	Non-safety-related
UK	Nuclear Building	Non-safety-related
VA	NP Storage	Non-safety-related
XK	Chilled Water	Non-safety-related

Table B-2. Sub-system Safety Categorization

System/ Structure Code	Sub Systems	Safety Class
B0	Electrical – Normal	Non-Safety-Related
	Electrical – Essential loads	Non-Safety-Related
JA	Reactor Vessel	Safety-Related
	Cross duct	Non-Safety-Related
	Intermediate Heat Exchanger (IHX) Vessel	Non-Safety-Related
	Pressure Relief Valve (PRV)	Non-Safety-Related
	Reactor Vessel Support	Safety-Related
JD	Controls Rods	Safety-Related
	Control Rod Drive Mechanisms (CRDM)	Safety-Related
	Controller	Non-Safety-Related
JE	Helium Circulator	Non-Safety-Related
	IHX	Non-Safety-Related
	Hot Gas Duct (HGD)	Non-Safety-Related
JG & KO	MS Piping	Non-Safety-Related
	MS Valves	Non-Safety-Related
	MS Drain Tank	Non-Safety-Related
	MS Cold Pumps	Non-Safety-Related
	MS Hot/Cold Tanks	Non-Safety-Related
JR	Trip Breaker	Safety-Related
	Flux monitor	Safety-Related
	Pressure Monitor	Safety-Related
	Temp Monitor	Safety-Related
	He Flow Monitor	Non-Safety-Related
	Inverter supply	Safety-Related
	Control Cabinet	Safety-Related
	Readout Monitor	Non-Safety-Related
JY	Helium Activity Monitor	Non-Safety-Related
	Salt Activity Monitor	Non-Safety-Related
	Radiation Monitor	Non-Safety-Related