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Introduction

The regulatory requirement to develop an upgraded safety basis for a DOE nuclear facility was realized in January 2001 by issuance of a revision to Title 10 of the Code of Federal Regulations Section 830 (10 CFR 830).[1] Subpart B of 10 CFR 830, "Safety Basis Requirements," requires a contractor responsible for a DOE Hazard Category 1, 2, or 3 nuclear facility to either submit by April 9, 2001 the existing safety basis which already meets the requirements of Subpart B, or to submit by April 10, 2003 an upgraded facility safety basis that meets the revised requirements.[1]

10 CFR 830 identifies Nuclear Regulatory Commission (NRC) Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants"[2] as a safe harbor methodology for preparation of a DOE reactor documented safety analysis (DSA). The regulation also allows for use of a graded approach. This report presents the methodology that was developed for preparing the reactor accident analysis portion of the Advanced Test Reactor Critical Facility (ATRC) upgraded DSA. The methodology was approved by DOE for developing the ATRC safety basis as an appropriate application of a graded approach to the requirements of 10 CFR 830.

Advanced Test Reactor Critical Facility Description

The Advanced Test Reactor Critical Facility (ATRC) is a nuclear mockup of the higher-powered Advanced Test Reactor (ATR). Both are located at the Test Reactor Area at the Idaho National Engineering and Environmental Laboratory (INEEL). The mission of the ATRC is to support operation of the ATR by performing low power tests which predetermine nuclear characteristics of higher-powered test configurations planned for application in the ATR.

The ATRC is a heterogeneous pool-type reactor that typically operates at very low power levels. The core is approximately 4-feet high consisting of 40 fuel elements positioned such that they form a serpentine shape between and around nine flux traps. The core is moderated by light water and uses a beryllium reflector. Cooling is provided by natural convection of light water.

All reactor operation is performed under manual control from a reactor console located near the reactor pool. The only automatic reactivity control functions are associated with shutdown actuations for reactor protection. Reactivity insertion by manual control is restricted by system interlocks that limit simultaneous withdrawal of neck shims, outer shim control cylinders, and safety rods. Safety rod, neck shim, and outer shim control cylinder positions are displayed on the reactor console.

The ATRC automatic reactor shutdown system functions include scram actuation on an indicated high neutron level or a low reactor power period. The low period function is effectively an indication of a rapid power increase. Each of these functions are actuated when the measured parameter exceeds a predefined setpoint value. A manual scram can also be initiated by reactor operators. A reactor scram shuts down the core by rapidly inserting the safety rods. The safety rods are suspended above the core and held by electromagnets. A scram actuation results in loss of electrical current to the magnets, allowing the safety rods to free fall with gravitational acceleration into the core.

ATRC DSA Upgrade Accident Analysis Methodology Development

10 CFR 830 identifies approved (safe harbor) methodologies for various categories of DOE nuclear facilities. The safe harbor methodology identified for a DOE reactor is NRC Regulatory Guide 1.70. The rule also allows some latitude, through application of a graded approach, in selecting an appropriate methodology. Use of a graded approach means that the required level of analysis and documentation may take into consideration such things as the importance to safety, the life cycle and status of the facility, characteristics of the facility, and the potential severity of the consequences of postulated occurrences. 10 CFR 830 requires that documentation for application of a graded approach to an approved safe harbor methodology be submitted to DOE.

Regulatory Guide 1.70 is the NRC standard method for preparing a safety analysis report (SAR) for a commercial nuclear power plant. For most commercial plants in operation in the U.S. today this document or its previous revisions provided the format and content guidance for preparing the preliminary and final safety analysis reports. Large DOE reactors, including the ATR, have also followed this guidance in preparing an upgraded DSA. However, for a facility such as the ATRC, which is a low-power pool-type reactor, much of the content required by Regulatory Guide 1.70 is not applicable. The ATRC, for example, has no primary coolant system, emergency core cooling system, or secondary steam system. Considering characteristics of the ATRC and the potential severity of the consequences of postulated occurrences, the guidance provided in Regulatory Guide 1.70 is excessive.

Additional guidance for application of the 10 CFR 830 rule was more recently issued in DOE G 421.1-2, “Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830”.[3] This guide also acknowledges that the Regulatory Guide 1.70 methodology is more appropriate for a large reactor. DOE G 421.1-2 further identifies

ANSI/ANS-15.21-1996, "American National Standard Format and Content for Safety Analysis Reports for Research Reactors,"[4] as an industry standard that provides guidance for developing the DSA for a small research reactor. DOE G 421.1-2 also cautions that Regulatory Guide 1.70 and ANSI/ANS-15.21-1996 were not developed for DOE facilities. As a result, these documents do not include all topics required by DOE. DOE G 421.1-2 suggests that DOE Standard 3009-94, "Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facilities Documented Safety Analysis,"[5] although developed for non-reactor facilities, also be consulted to supplement the methodology in ANSI/ANS-15.21-1996 in order to ensure that all required topics are addressed.

Accident Analysis Methodology Applying a Graded Approach

ANSI/ANS-15.21-1996 is a standard prepared by the American Nuclear Society that provides guidance for small research reactors. In preparing this standard, the American Nuclear Society Standards Committee considered the aspects of a small research reactor.[4] These reactors are generally non-evasive facilities with limited fission product inventories and stored energy (decay heat). They also have much simpler equipment and components that make up both the reactor core and cooling system. Therefore, a graded approach to the content, level of description, and level of analysis was considered when developing this standard.

The accident analysis guidance in ANSI/ANS-15.21-1996 provides a comprehensive format for developing and organizing the accident analysis portion of the ATRC DSA. The proposed chapter subsections include analysis methods, reactor characteristics, selection of initiating events, evaluation of each fault sequence or accident scenario, and conclusions.

DOE developed Standard 3009-94 for use by contractors in meeting DOE expectations. This preparation guide provides a standardized format and content for non-reactor nuclear facilities with an emphasis on existing Hazard Category 2 facilities. The INEEL further developed an internal supplemental guide that provides additional information on performing and documenting hazard and accident analyses for non-reactor nuclear facilities following DOE Standard 3009-94.

The hazard analysis method presented in DOE Standard 3009-94, which is used to identify all potential facility hazards and support identification of potential reactor accidents, includes classifying those accidents by their likelihood of occurrence. Grouping postulated accidents by their event frequency is a standard approach used by both DOE and the NRC. This concept is not captured in ANSI/ANS-15.21-1996. DOE Standard 3009-94 provides event frequency categories that are applicable to classifying accidents for the ATRC. The categories include anticipated events, unlikely events, extremely unlikely events, and beyond extremely unlikely or beyond design basis events. To avoid confusion between ATRC and ATR operations, the terminology was revised somewhat to be consistent with the existing definitions in the ATR SAR. That is, accidents are defined as Condition 1, 2, 3, 4 or beyond design basis. The accident criteria, discussed later, are also defined according to event frequency categories.

Condition 1 (normal operation) activities and events are those that are expected to occur regularly in the course of normal reactor operation, refueling, and maintenance. This includes planned operations such as reactor startup, shutdown, lobe power balance adjustments, and experiment positioning. This is not an accident category and as such is not specifically identified in DOE Standard 3009-94 as a hazard or accident. Rather, this category encompasses the day-to-day activities of the facility. Reactor operating parameters are controlled within a conservative set of initial conditions. Facility procedure updates supporting the DSA upgrade will be required in order to control the initial reactor conditions for normal operation within the accident analysis assumptions. Conditions for normal operation are also evaluated and analyzed to ensure that facility worker day-to-day radiological exposure is within acceptable limits. This ensures day-to-day safe operation of the facility.

Condition 2 (anticipated) events include those events with an estimated annual likelihood of occurrence $>10^{-2}$ and $\leq 10^{-1}$ per year (10^{-1}/yr $\geq$ frequency $> 10^{-2}$/yr).[5,6] For the ATRC, anticipated events also include reactor upset conditions with an event frequency $>10^{-1}$/yr. Hence, upset conditions with a likelihood of occurrence $>10^{-2}$ are categorized as anticipated events. This is in order to distinguish between reactor accidents (upset conditions) and normal operation activities. Condition 2 events may occur several times during the lifetime of the facility. Events resulting from a single failure in a reactor control system generally fall into this frequency range.

Condition 3 (unlikely) events include those events with an estimated annual likelihood of occurrence $>10^{-4}$ and $\leq 10^{-2}$ per year (10^{-2}/yr $\geq$ frequency $> 10^{-4}$/yr).[5,6] These events are not anticipated to occur during the lifetime of the facility. Generally, events of this type result from simultaneous (multiple) system or component failures.

Condition 4 (extremely unlikely) events include those events with an estimated annual likelihood of occurrence $>10^{-6}$ and $\leq 10^{-4}$ per year (10^{-4}/yr $\geq$ frequency $> 10^{-6}$/yr).[5,6] These events will probably not occur during the lifetime of the facility. Generally, events of this type result from compounded or multiple system and/or component failures.

Beyond design basis accidents are required by DOE in order to provide a perspective of the residual risk associated with the operation of the facility[5]. These include all other accidents with a likelihood of occurrence less than 10^{-6} per year. Because of their very low event frequency, analysis of beyond design basis accidents is not required to provide assurance of public health and safety. Accordingly, they serve as the bases for cost-benefit considerations if consequences exceeding evaluation guidelines are identified.

An event frequency evaluation of all reactor control and protection functions was proposed in order to identify a reasonable beyond design basis reactor accident. The representative beyond design basis accident was deterministically analyzed to determine its postulated worst case consequences. The analysis of this accident was presented in the DSA.

Accident Analysis Acceptance Criteria

Accident analysis acceptance criteria are used for comparison to the calculated consequences of each accident sequence analyzed in the safety basis. They provide a quantitative method for demonstrating that accident consequences are within acceptable limits. Accident acceptance criteria are also defined according to the anticipated likelihood of occurrence. That is, since a

Condition 2 accident is expected to occur more often than a Condition 4 accident, more restrictive acceptance criteria are required to be met.

ANSI/ANS-15.21-1996 does not identify acceptance criteria for reactor accidents. DOE Standard 3009-94 has qualitative guidelines relative to developing a hazard analysis, but also does not define specific accident criteria for deterministic analyses. DOE ID Order 420.D, "Requirements and Guidance for Safety Analysis" Attachment III, *Evaluation Guidelines for Nonreactor Nuclear Facilities*[6] specifies the radiological acceptance criteria for non-reactor nuclear facilities. The Code of Federal Regulations, specifically 10 CFR 100 "Reactor Site Criteria"[7] specifies acceptance criteria for siting of reactors that have historically been used as acceptance criteria for DOE reactors. The criteria in 10 CFR 100 are consistent with the off-site criteria for extremely unlikely events in DOE ID Order 420.D, however, DOE ID 420.D specifies more restrictive criteria for more likely events. The more restrictive criteria specified in DOE ID 420.D was proposed for application to the ATRC. The criteria in the ATR SAR for normal operation (Condition 1) is also proposed for application in the ATRC upgraded DSA. This approach results in the radiological acceptance criteria described below.

The following acceptance criteria were proposed for the accident analysis of the ATRC DSA upgrade.

Condition 1 (normal operation) activity off-site exposures shall not exceed the limits of 100 mrem/year effective dose equivalent and 10 mrem/year effective dose equivalent from airborne releases. On-site exposures to facility personnel shall not exceed 5.0 rem/year total effective dose equivalent.

Condition 2 (anticipated) accident off-site exposures to members of the public shall not exceed 0.5 rem total effective dose equivalent. On-site personnel exposures to facility workers or co-located workers shall not exceed 5.0 rem total effective dose equivalent.

Condition 3 (unlikely) accident off-site exposures to members of the public shall not exceed 5.0 rem total effective dose equivalent. On-site personnel exposures to facility workers or co-located workers shall not exceed 25 rem total effective dose equivalent.

Condition 4 (extremely unlikely) accident off-site exposures to members of the public shall not exceed 25 rem total effective dose equivalent. On-site personnel exposures to evacuating workers (excluding personnel directly at the location of the accident) shall not exceed 100 rem total effective dose equivalent.

The above criteria are radiological and do not place any direct requirements on fuel integrity. Demonstrating that fuel integrity is maintained during a design basis accident is not a DOE requirement; however, it may be used to conservatively demonstrate that there are no adverse radiological consequences (i.e., if the cladding integrity is maintained there is no significant fission product release). For many reactor accident analyses this is a convenient and preferred approach.

The fuel can be shown to remain undamaged and in a coolable geometry by demonstrating that the melting temperature of the cladding is not exceeded. For the ATRC fuel, the melting temperature is taken to be the solidus temperature of aluminum alloy 6061; that is 855 K (1079°F). This limit can be translated into an energy deposition limit during a reactor transient by conservatively assuming that the core heats up adiabatically with the total core energy deposited into the fuel (i.e., no heat transfer to the coolant). For conservatism, the fuel temperature rise is limited to half of the difference between the normal fuel and coolant temperature (approximately room temperature) and the fuel melting temperature. Based on these conditions, a specific protective criteria for core energy deposition during an accident was developed which ensures that the reactor fuel plate cladding remains undamaged. This criterion applies to all postulated Condition 1, 2, 3, and 4 reactor events.

Accident Sequence Identification

In addition to grouping postulated accidents according to the likelihood of occurrence, accidents are also grouped by the type of event. For example, there are multiple conditions that may occur that would result in a reactivity insertion to the core and lead to a core over-power excursion. These events may be grouped together and only the limiting accident, within a given event frequency category (i.e., Condition 1, 2, 3, or 4), fully analyzed and presented in the DSA. This practice, which minimizes the required analysis, is suggested in both DOE Standard 3009-94 and ANSI/ANS-15.21-1996.

Appendix C of ANSI/ANS-15.21-1996 provides a list of possible accident sequences for a reactor facility. This list is available to help identify potential accident sequences applicable to the ATRC. The existing ATRC safety basis and experience from operation of the facility were also considered in identifying potential accidents. This report does not identify specific reactor accidents for the ATRC, rather types of accidents that should be considered for each event frequency category. For each accident type and category, a corresponding limiting accident applicable to the ATRC was identified. During this process it was determined that a detailed analysis was not required for some accidents in order to demonstrate that the acceptance criteria were met. Types of reactor accidents applicable to the ATRC are listed below.

<u>Condition 1 - Normal Operation</u>

- Manual Reactivity or Core Changes During Normal Reactor Operation
- Radiological Exposure to Workers During Normal Reactor Operation
- Potential Challenges to Fuel Integrity During Normal Reactor Operation

<u>Condition 2 - Anticipated Accidents</u>

- Control System Failure, Equipment Failure, or Operator Error Resulting in a Continuous Reactivity Insertion Accident
- Control System Failure, Equipment Failure, or Operator Error Resulting in a Step Reactivity Insertion Accident
- Fuel Element or Experiment Damage
- Loss of Canal Inventory (slow or partial)
- Loss of Commercial Power

Condition 3 – Unlikely Accidents

- Control System Failure, Equipment Failure, or Operator Error Resulting in a Continuous Reactivity Insertion Accident
- Control System Failure, Equipment Failure, or Operator Error Resulting in a Step Reactivity Insertion Accident
- Fuel Element or Experiment Damage
- Loss of Canal Inventory (rapid or complete)

Condition 4 – Extremely Unlikely Accidents

- Control System Failure, Equipment Failure, or Operator Error Resulting in a Continuous Reactivity Insertion Accident
- Control System Failure, Equipment Failure, or Operator Error Resulting in a Step Reactivity Insertion Accident
- Fuel Element or Experiment Damage

Beyond Design Basis Accidents

- Hypothetical Severe Fuel Damage Event

The general accidents proposed above may appear repetitive, but in reality should lead to developing discrete limiting accidents within each accident frequency category. For example, a single failure in a reactor control function may result in a relatively low positive reactivity insertion into the core. This type of event would likely be categorized as a Condition 2 or anticipated accident. This same failure combined with a second failure, or an independent more complex system failure, may result in a more severe reactivity insertion accident with more severe consequences. The more complex and severe reactivity insertion accident would likely be categorized as a Condition 3 (unlikely) or Condition 4 (extremely unlikely) accident.

Accident Analysis Format and Content for the ATRC Upgraded DSA

The DOE Standard 3009-94 and ANSI/ANS-15.21-1996 guidance documents each identify an acceptable format to follow in the facility DSA. The detailed format identifies section and subsection headings, which in turn define the overall content to be presented in the DSA. The proposed format and content for the ATRC, presented below, is primarily based on that provided in ANSI/ANS-15.21-1996. The ANSI/ANS-15.21-1996 format has been modified to include accident groupings by event frequency.

X.0 Accident Analysis
- X.1 Introduction
  - X.1.1 Types of Accidents Evaluated (Grouping)
  - X.1.2 Methodologies Used
  - X.1.3 Limiting Accidents

- X.2 Reactor Characteristics
  - X.2.1 Core Parameters
  - X.2.2 Assumed Reactor Protection Actions
- X.3 Analytical Methods [*section expanded to capture each computer code or model*]
  - X.3.1 Description of Analytical Model
  - X.3.2 Input and Output Data
  - X.3.3 Summary of Validation Results
  - X.3.4 Core and Systems Performance Modeling
  - X.3.5 Source Term and Consequence Modeling
- X.4 Selection of Initiating Events
  - X.4.1 Methodology
  - X.4.2 List of Initiating Events and Categorization
  - X.4.3 Limiting Events
- X.5 Evaluation of Fault Sequences and Accident Conditions for Normal Operation (Condition 1)
  - X.5.1 Specific Accident Title [*Section to be repeated for each Condition 1 accident discussed in the DSA. If a detailed analysis was not required, the section may be simplified.*]
    - X.5.1.1 Identification of Causes
    - X.5.1.2 Sequence of Events and System Operation
    - X.5.1.3 Transient Analysis
    - X.5.1.4 Damage States and Source Terms
    - X.5.1.5 Evaluation of Radiological Consequences
    - X.5.1.6 Summary of Specific Event Sequences
- X.6 Evaluation of Fault Sequences and Accident Conditions for Anticipated Events (Condition 2)
  - X.6.1 Specific Accident Title [*Section to be repeated for each Condition 2 accident discussed in the DSA. If a detailed analysis was not required, the section may be simplified.*]
    - X.6.1.1 Identification of Causes
    - X.6.1.2 Sequence of Events and System Operation
    - X.6.1.3 Transient Analysis
    - X.6.1.4 Damage States and Source Terms
    - X.6.1.5 Evaluation of Radiological Consequences
    - X.6.1.6 Summary of Specific Event Sequences
- X.7 Evaluation of Fault Sequences and Accident Conditions for Unlikely Events (Condition 3)
  - X.7.1 Specific Accident Title [*Section to be repeated for each Condition 3 accident discussed in the DSA. If a detailed analysis was not required, the section may be simplified.*]
    - X.7.1.1 Identification of Causes
    - X.7.1.2 Sequence of Events and System Operation
    - X.7.1.3 Transient Analysis
    - X.7.1.4 Damage States and Source Terms
    - X.7.1.5 Evaluation of Radiological Consequences
    - X.7.1.6 Summary of Specific Event Sequences

X.8 Evaluation of Fault Sequences and Accident Conditions for Extremely Unlikely Events (Condition 4)

X.8.1 Specific Accident Title [*Section to be repeated for each Condition 4 accident discussed in the DSA. If a detailed analysis was not required, the section may be simplified.*]

X.8.1.1 Identification of Causes

X.8.1.2 Sequence of Events and System Operation

X.8.1.3 Transient Analysis

X.8.1.4 Damage States and Source Terms

X.8.1.5 Evaluation of Radiological Consequences

X.8.1.6 Summary of Specific Event Sequences

X.9 Evaluation of Beyond Design Basis Event

X.9.1 Specific Accident Title [*Section to be repeated as needed.*]

X.9.1.1 Identification of Causes

X.9.1.2 Sequence of Events and System Operation

X.9.1.3 Transient Analysis

X.9.1.4 Damage States and Source Terms

X.9.1.5 Evaluation of Radiological Consequences

X.9.1.6 Summary of Specific Event Sequences

X.10 General Conclusions

X.10.1 Summary of Results

X.10.2 Acceptability of Reactor Design and Technical Specifications

X.10.3 Comparison with Acceptance Criteria

Conclusions

10 CFR 830 Subpart B requires that a DOE nuclear facility submit an upgraded DSA. There is no DOE developed guidance document that directly provides an accident analysis methodology for an upgraded reactor facility DSA. The combined guidance in ANSI/ANS-15.21-1996, DOE Standard 3009-94, and DOE ID Order 420.D has been used to develop the method presented herein. The accident categorization, acceptance criteria, and the format and content to be presented in the DSA was approved by DOE for application to the ATRC. The upgraded DSA was subsequently prepared and has been approved by DOE.

References

1) 10 CFR 830, *Nuclear Safety Management*, Code of Federal Regulations, Office of the Federal Register, January 2001.

2) Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*, U.S. Nuclear Regulatory Commission, Revision 3, November 1978.

3) DOE G 421.1-2, *Implementation Guide for use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830*, U.S. Department of Energy, October 2001.

4) ANSI/ANS-15.21-1996, *American National Standard Format and Content for Safety Analysis Reports for Research Reactors*, American National Standards Institute and American Nuclear Society, November 1996.

5) DOE Standard 3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear facility Safety analysis Reports*, U.S. Department of Energy, July 1994, Change Notice 1, January 2000.

6) DOE ID Order 420.D, *Requirements and Guidance for Safety Analysis*, U.S. Department of Energy, Idaho Operations Office, July 2000.

7) 10 CFR 100, *Reactor Site Criteria*, Code of Federal Regulations, Office of the Federal Register, December 1996.