United States
Nuclear Regulatory Commission

Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors

Format and Content

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NUREG - 1537 PART 1

Office of Nuclear Reactor Regulation
Division of Reactor Program Managment

4 REACTOR DESCRIPTION

In this chapter of the SAR, the applicant should discuss and describe the principal features, operating characteristics, and parameters of the reactor. The analysis in this chapter should support the conclusion that the reactor is conservatively designed for safe operation and shutdown under all credible operating conditions. Information in this chapter of the SAR should provide the design bases for many systems, subsystems, and functions discussed elsewhere in the SAR and for many technical specifications.

4.1 Summary Description

In this section the applicant should briefly summarize the design and functional characteristics of the reactor. The applicant should present the principal safety considerations in the selection of the reactor type as well as the design principles for the components and systems that address those considerations. This section should contain summary tables of important reactor parameters and sufficient drawings and schematic diagrams to explain and illustrate the main reactor design features.

The applicant should briefly address the following features of the reactor:

- thermal power level
- fuel type and enrichment
- pool or tank type
- forced and/or natural-convection cooling
- type of coolant, moderator, and reflector
- principal features for experimental programs
- pulsing or steady power
- novel concepts requiring substantial new development

4.2 Reactor Core

In this section the applicant should present all design information and analyses necessary to demonstrate that the core can be safely operated. The major core components to be described are fuel, neutron moderator, neutron reflector, control elements, neutron startup source, incore cooling components, and any incore experimental facilities. The source or basis of the information presented should be given.

4.2.1 Reactor Fuel

In this section the applicant should describe the reactor fuel system. Included should be the design features selected to ensure that the fuel and cladding can withstand all credible environmental and irradiation conditions during their life cycle at the reactor site. The discussions should address the incore fuel operating conditions. Handling, transport, and storage of fuel should be discussed in Chapter 9, "Auxiliary Systems," of the SAR. Drawings and tables of design specifications and operating characteristics of the fuel should be presented.

Most non-power reactors contain heterogenous fuel elements consisting of rods, plates, or pins, which are addressed in the following sections. Homogeneous fuels should be described and analyzed in a comparable way. Information should be current; supported by referenced tests, measurements, and operating experience; and compared with additional applicant experience where applicable. The information should include the following:

- Chemical composition, enrichment, uranium loading, and important metallurgical features of the fissile material in the basic fuel unit. The information should indicate dispersion, alloy, cermet, sintering, and such special properties as burnable poisons or neutron moderators.

- Description of the basic fuel unit, including plates, rods, pins, or pellets. This information should include dimensions, fabrication methods, and cladding or encapsulation methods. Special features, such as moderators or reflectors, external geometrical designs to enhance cooling capability, and inherent safety or feedback provisions should be discussed.

- Material and structural information such as dimensions, spacings, fabrication methods, compatibility of materials, and specifications with tolerances. All types of fuel elements to be used should be described, including full elements, partial elements, control rod elements, instrumented elements, and special elements for experimental facilities. Features that ensure accurate and secure positioning and adequate cool nt flow should be described.

- Information on material parameters that could affect fuel integrity, such as melting, softening, or blistering temperatures; corrosion; erosion; and mechanical factors, such as swelling, bending, twisting, compression, and shearing.

- Physical properties with significance in regard to safety and fuel integrity that are important for the thermal-hydraulic analyses, such as heat capacity, thermal conductivity, gas evolution or diffusion, occluded or encapsulated void volume, fuel burnup limits, capability to retain fission products, swelling resistance, and buildup of oxides.

- If the reactor is designed for pulsing, any special attributes of the fuel that contribute to pulsing safety.

- A brief history of the fuel type, with references to the fuel development program, including summaries of performance tests, qualification, and operating history. A brief history of the actual fuel elements to be used, including fabrication, previous irradiation conditions, storage environments, surveillance procedures, and qualification tests.

- Mechanical forces and stresses, hydraulic forces, thermal changes and temperature gradients, internal pressures including that from fission products and gas evolution, and radiation effects including the maximum fission densities and fission rates that the fuel units and elements are designed to accommodate.

Limits on operating conditions for the fuel should be supported by information and analyses. These limits are specified to ensure that the integrity of the fuel elements and their cladding or fission product barrier will not be impaired. They should form the design bases for this and other chapters of the SAR, for the reactor safety limits, and for other fuel-related technical specifications.

4.2.2 Control Rods

In this section the applicant should give information on the control rods, including all rods or control elements that are designed to change reactivity during reactor operation. The physical, kinetic, and electromechanical features demonstrating that the rods can fulfill their control and safety functions should be described. Results of computing control rod reactivity worths may be presented in this section, but details of the calculation of reactivity effects should appear in Section 4.5, "Nuclear Design," of the SAR. The information in this section should include the following:

- The number and types of rods (e.g., shim, safety, regulating, transient), their designed locations in the core, and their designed reactivity worths. The considerations and bases for redundancy and diversity should be provided. Limits on core configuration should be discussed.

- The structural and geometric description, including the shape, size, materials, cladding, fabrication methods, and specifications with tolerances for the rods. This should include the type and concentration of neutron absorber, or emitter, if applicable. Also, calculations of changes in reactivity worth due to burnup and assessment of radiation damage, heating effects, and chemical compatibility with the coolant and other core components should be given. If the control rods have followers, the design, composition, and reactivity effects of the follower should be discussed.

- The design of mechanical supports for the active component, the method of indicating and ensuring reproducible positioning in the core, and the drive mechanism of each type of rod. This information should include the source of motive power, usually electrical, and the systems ensuring scram capability. For a reactor designed for pulsing, the transient rod should be described in detail, including its drive mechanisms and the methods for calibration, pulse reproducibility, and prevention of inadvertent pulsing.

- The kinetic behavior of the rods, showing either the positive or negative rate of reactivity change, in the normal drive and scram modes of operation. This information should be supplied for all rods, including transient rods in a reactor designed for pulsing. The applicant should show that the control rod design conforms with the shutdown margin requirements.

- The scram logic and circuitry, interlocks and inhibits on rod withdrawal, trip release and insertion times, and trip or scram initiation systems should be summarized here and described in detail in Chapter 7, "Instrumentation and Control Systems."

- Special features of the control rods, their core locations, power sources, drive or release mechanisms designed to ensure operability and capability to provide safe reactor operation and shutdown under all conditions during which operation is required in the safety analysis if there is a single failure or malfunction in the control system itself. Such features may include mechanisms to limit the speed of rod movement.

- Technical specification requirements for the control rods and their justification. These are the limiting conditions for operation, surveillance requirements, and design features as discussed in Chapter 14, "Technical Specifications," of this format and content guide.

4.2.3 Neutron Moderator and Reflector

In this section the applicant should discuss the materials and systems designed to moderate the neutrons within the fuel region and reflect leakage neutrons back into the fuel region. The information should include the materials, geometries, designs for changes or replacement, provisions for cooling, radiation damage considerations, and provisions for experimental facilities or special uses. Multiple-use systems and features such as moderator coolant, fuel moderator, and reflector shield should be described. If moderators or reflectors are encapsulated to prevent contact with coolant, the effect of failure of the encapsulation should be analyzed. It should be possible to operate the reactor safely until failed encapsulations are repaired or replaced. If reactor operations cannot be safely continued, the reactor should be placed and maintained in a safe condition until encapsulations are repaired or replaced. Technical specification requirements should be proposed and justified for the moderator and reflector in accordance with the guidance in Chapter 14 of this format and content guide. The nuclear design of the moderator and reflector should be discussed in Section 4.5 of the SAR.

4.2.4 Neutron Startup Source

In this section the applicant should present design information about the neutron startup source and its holder. The applicant should show that the source will produce the necessary neutrons to allow a monitored startup with the reactor instrumentation. The information should include the neutron strength and spectrum, source type and materials, its burnup and decay lifetime, and its regeneration characteristics. Other necessary information includes the material and geometry of the holder, the method of positioning the source in the core, and the core locations in which the source is designed to be used. Utilization information and such limitations as radiation heating or damage and chemical compatibility with coolant and other core components should be discussed. Any technical specification limits on the source, such as the maximum power level the reactor can be run with the source in place (for plutonium-beryllium sources and other source types that can act as fuel), or surveillance requirements to ensure source integrity should be proposed and justified in this section of the SAR in accordance with the guidance in Chapter 14 of this format and content guide.

4.2.5 Core Support Structure

In this section the applicant should present design information about the mechanical structures that support and position the core and its components. The information should include the following:

- The design considerations that ensure that all necessary loads and hydraulic forces can be conservatively supported with and without the buoyant forces of the reactor water.

- The methods by which core components are accurately and reproducibly positioned and secured, including specification tolerances, as well as features of the grid plate such as fuel holddown grids, fuel element spacers, and control rod guides and supports.

- The materials of construction, including considerations for radiation damage, corrosion, erosion, chemical compatibility with coolant and core components, potential effects on reactivity, induced radioactivities, and maintenance.

- Design features of the core support structure that accommodate other systems and components such as radiation shields, beam ports or other experimental facilities, coolant pipes, coolant plenums or deflectors, and nuclear detectors.

- For a movable core support, design information describing the motive power system, the system for ensuring position, and interlocks that prevent or control motion while the reactor is critical, while forced cooling is required, or while other activities that prohibit core support movement are to be conducted, if such a system is required (e.g., experimental facility operations).

- Technical specifications that control important design features, limiting conditions for operation, and surveillances as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

4.3 Reactor Tank or Pool

The cores of most non-power reactors are immersed in water within a tank or pool. In this section the applicant should present all information about the tank or pool necessary to ensure its integrity. The information should include the following:

- Design and considerations to ensure that no hydrodynamic, hydrostatic, mechanical, chemical, and radiation forces or stresses could cause failure or loss of integrity of the tank during its projected lifetime over the range of design characteristics.

- Design and dimensions to ensure sufficient shielding water to protect personnel and components, as well as sufficient depth to ensure necessary coolant flow and pressures. (Also see Sections 4.4 and 4.6 and Chapter 11, "Radiation Protection Program and Waste Management," of this format and content guide.)

- Designs and description of materials, including dimensions, supporting structures, chemical compatibility with the coolant and other reactor system components, radiation fields and any consequences of radiation damage, protection from corrosion in inaccessible regions, and capability to replace components, if necessary.

- Locations of penetrations and attachment methods for other components and pipes. The relationships of these penetrations to core and water surface elevations should be discussed. Safety-related features that prevent loss of coolant should be discussed and related to Sections 4.4 and 4.6 and to the loss-of-coolant-accident scenarios analyzed in Chapter 13, "Accident Analyses," as applicable.

- Planned methods for assessing radiation damage, chemical damage, or deterioration during the projected lifetime. In this section the applicant should assess the possibility of uncontrolled leakage of contaminated primary coolant and should discuss preventive and protective features.

- Technical specifications that control important design features, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

4.4 Biological Shield

In this section the applicant should present information about the principal biological shielding designed for the reactor. The information should include the following:

- The design bases for the radiation shields (e.g., water, concrete, or lead), including the projected reactor power levels and related source terms and the criteria for determining the required protection factors for all applicable nuclear radiation activity. Information about conformance with the regulations for radiation exposure and the facility ALARA (as low as is reasonably achievable) program should appear in Chapter 11. The design basis should include the designed reactor power levels, the associated radiation source terms, and other radiation sources within the pool or tank that require shielding.

- The design details and the methods used to achieve the design bases. The applicant should discuss the protection of personnel and equipment functions. The information should specify the general size and shape of the shields and the methods used to ensure structural strength, rigidity, and functional integrity. The applicant should discuss the distribution of shielding factors between liquid (water) and solid (concrete, lead, etc.) materials. If loss of shield integrity could cause a loss-of-coolant accident, the features to prevent the loss of integrity should be described.

- The materials used and their shielding coefficients and factors, including a detailed list of constituents and their nuclear and shielding properties. The applicant should discuss radiation damage and heating or material dissociation during the projected lifetime of the reactor; induced radioactivity in structural components; potential radiation leakage or streaming at penetrations, interfaces, and other voids; shielding at experimental facilities; and shielding for facilities that store fuel and other radioactive materials within the reactor pool or tank.

- The assumptions and methods used to calculate the shielding factors, including references to and justification of the methods. Detailed results of the shielding calculations should give both neutron and gamma-ray dose rates at all locations that could be occupied. The applicant should calculate shield penetrations and voids, such as beam ports, thermal columns, and irradiation rooms or vaults, as well as the shielding of piping and other components that could contain radioactive materials or allow radiation streaming.

- Methods used to prevent neutron irradiation and activation of ground water or soils surrounding the reactor shield that could enter the unrestricted environment. The applicant should estimate the maximum activity should such activation occur and describe remedial actions.

- Technical specifications that control important design features, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

Regulatory Guide 2.1, "Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors" is given as Appendix 4.1.

4.5 Nuclear Design

In this section the applicant should give information on the nuclear parameters and characteristics of the reactor core and should analyze the kinetic behavior of the reactor for steady-state and transient operation throughout its life cycle of allowed cores and burnup as discussed in the safety analysis. The descriptions, analyses, and results should address all safety issues in the design and operation of the reactor and should support the conclusion that the reactor can be built and operated without unacceptable risk to the health and safety of the public. A detailed description of the analytical methods used in the nuclear design should be given. Computer codes that are used should be described in detail as to the name and type of code, the way it is used, and its validity on the basis of experiments or confirmed predictions of operating non-power reactors. Code descriptions should include methods of obtaining parameters such as cross sections. Estimates of the accuracy of the analytical methods should be included. Tables and figures should be used as necessary to present information clearly.

4.5.1 Normal Operating Conditions

In this section the applicant should present information on the core geometry and configurations. Operating core configurations should be compact, with no vacancy in which fuel could be inserted within the core periphery. The limiting core configuration for a reactor is the core that would yield the highest power density using the fuel specified for the reactor. All other core configurations should be demonstrated to be encompassed by the safety analysis of the limiting core configuration. Further information on power density limitations should be given in Sections 4.5.3 and 4.6. The information in the SAR should include the following:

- The number, types, and locations of all core components on the grid plate, including fuel, control rods, neutron reflectors, moderators, incore experimental components, and core-associated cooling components. If this information appears elsewhere in the SAR, the section where it is located should be referenced.

- Descriptions of planned core configurations during the life of the reactor, showing how a compact core is ensured.

- Discussions and analyses of the reactor operating characteristics. The applicant should give in detail the effects of changes in configuration and fuel burnup. If applicable, the applicant should analyze safety-related considerations for all requested operating modes (e.g., steady power and pulsing).

- Changes in core reactivity with fuel burnup, plutonium buildup, and poisons, both fission products and those added by design, if applicable.

- Analyses of the reactor kinetic behavior and the design requirements and dynamic features of the control rods that allow controlled operation for all possible reactor conditions.

- Analyses of the basic reactor criticality physics, including the interacting effects of fuel, neutron moderators and reflectors, control rods, and incore or in-reflector components such as experimental facilities.

- Discussion of the safety considerations for different core configurations, including a limiting core configuration that would yield the highest power densities and fuel temperatures achievable with the planned fuel.

- The individual reactivity worths of fuel elements, reflector components, incore and in-reflector components, experimental components, and control rods in allowed positions. If experimental facilities or components could be voided or flooded, the reactivity effects and safety considerations should be included.

- The calculated core reactivities for all core configurations. including the limiting configuration that would yield the highest possible power density.

- Discussion of the administrative and physical constraints to prevent inadvertent addition of positive reactivity.

- Technical specifications that control important design features, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

4.5.2 Reactor Core Physics Parameters

In this section the applicant should discuss the core physics parameters and show the methods and analyses used to determine them. The information should include the following:

- Analysis methods and values for neutron lifetime and effective delayed neutron fraction. The applicant should describe the effects of reactor operating characteristics and fuel burnup.

- Analysis methods, values, and signs for coefficients of reactivity (e.g., fuel and moderator temperature, void, and power). The applicant should describe the effects of reactor operating characteristics and fuel burnup. This analysis, along with the analysis in Chapter 13, should show that reactivity coefficients are sufficiently negative to prevent or mitigate damaging reactor transients.

- The axial and radial distributions of neutron flux densities, justifications for the methods used, and comparisons with applicable measurements. The applicant should describe changes in flux densities with power level, fuel burnup, core configurations, and control rod positions. The information on neutron flux density should include peak-to-average values for thermal-hydraulic analyses. The applicant should validate these calculations by comparing them with experimental measurements and other validated calculations.

- Technical specifications that control important design features, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

4.5.3 Operating Limits

The applicant should present the following information on reactor operating limits:

- Reactivity conditions, excess reactivity, and negative reactivity for combinations of control rods inserted that are analyzed for the limiting core and operating cores during the life of the reactor. The applicant should discuss operational and safety considerations for excess reactivity.

- The excess reactivity based on reactor temperature coefficients, poisons, and experiment worths. The applicant should justify the upper limit on excess reactivity to ensure safe reactor operation and shutdown.

- The amount of negative reactivity that must be available by control rod action to ensure that the reactor can be shut down safely from any operating condition and maintained in a safe shutdown state. The analyses should assume that the most reactive control rod is fully withdrawn (one stuck rod), non-scrammable control rods are at their most reactive position, and normal electrical power is unavailable to the reactor. The applicant should discuss how shutdown margin will be verified. The analyses should include all relevant uncertainties and error limits.

- The limiting core configuration that is possible with the planned fuel in this reactor. The limit should be imposed by the maximum neutron flux density and thermal power density compatible with coolant availability. The safety limits and limiting safety system settings for the reactor should be derived from this core configuration. The detailed analyses should be included in Section 4.6. Normal operating conditions and credible events, such as a stuck control rod, should be considered.

- A transient analysis assuming that an instrumentation malfunction drives the most reactive control rod out in a continuous ramp mode in its most reactive region. This analysis can also be based on a credible failure of a movable experiment. It should show that the reactor is not damaged and fuel integrity is not lost.

- The redundancy and diversity of control rods necessary to ensure reactor control for the considerations noted above.

- Technical specifications for safety limits, limiting safety system settings, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

4.6 Thermal-Hydraulic Design

In this section the applicant should present the information and analyses necessary to show that sufficient cooling capacity exists to prevent fuel overheating and loss of integrity for all anticipated reactor operating conditions, including pulsing, if applicable. The applicant should address the coolant flow conditions for which the reactor is designed and licensed, forced or natural-convection flow, or both. A detailed description of the analytical methods used in the thermal-hydraulic design should be provided. Computer codes that are used should be described in detail as to the name and type of code, the way it is used, and its validity on the basis of experiments or confirmed predictions of operating non-power reactors. Estimates of the accuracy of the analytical methods should be included. The information should include the following:

- The coolant hydraulic characteristics of the core, including flow rates, pressures, pressure changes at channel exits and entrances, and frictional and buoyant forces. The applicant should address individual heated channels as well as the core as a whole for all flow conditions in the primary coolant system. The transition from forced to natural-convection flow for all forced-flow reactors should be calculated, and the applicant should prepare calculations for an event during which normal electrical power is lost.

- The thermal power density distribution in the basic fuel units and heat fluxes into the coolant of each channel and along the channel, derived from the fuel loading and neutron flux characteristics discussed above.

- Calculations and the thermal-hydraulic methodology for the transfer of heat to the coolant. The applicant should take into account uncertainties in thermal-hydraulic and nuclear parameters and such engineering factors as plate thickness, gap width, and the buildup of cladding oxides. The calculations should be based on fuel measurements and procurement specifications, as well as operating history and conditions. The calculational methodology should be applicable to the thermal-hydraulic operating conditions, and the applicant should justify its use.

- The calculations and experimental measurements to determine the coolant conditions ensuring that fuel and cladding integrity are not lost. The applicant should calculate at least the limiting core configuration. Operating conditions should include steady fission power, shutdown decay heat, planned pulses, and transients analyzed in Chapter 13. The applicant should take into account operational and fuel characteristics from the beginning to the end of fuel life.

- For the core geometry and the coolant thermal-hydraulic characteristics, a discussion to establish the fuel heat removal conditions that ensure fuel integrity such as fuel surface saturation temperature, onset of nucleate boiling, departure from nucleate boiling and/or flow instability. The discussion should show correlations among these factors and justify their use in deriving safety limits and limiting safety system settings for the technical specifications.

- The design bases for the primary coolant system, emergency core cooling system, and other systems designed to maintain fuel integrity, which should also be discussed in Chapter 5, "Reactor Coolant Systems." The analyses here and in Chapter 13 should describe loss-of-coolant scenarios for forced-flow reactors. Natural-convection cooling that removes decay heat to ensure thermal stability should also be discussed. Flow blockages should be analyzed in Chapter 13.

- Detailed analyses for a pulsing reactor containing descriptions of the core configurations; the bases of the feedback coefficients; the calculational model and assumptions; the thermal-hydraulic evolution during a pulse; core, transient rod, and fuel characteristics that determine the shape and magnitude of a pulse; and the safety considerations that establish limits to pulse sizes. Any changes in fuel parameters resulting from steady-power operation that could affect pulse characteristics should be analyzed. These changes could include burnup, hydrogen migration, cladding oxidation, and decrease in burnable poison, as applicable. The analyses should form the bases for technical specifications that limit reactor operating conditions, process variables, and pulse rod reactivity worths.

Appendix 4.1

Regulatory Guide 2.1

Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors

May 1973

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 2.1

SHIELD TEST PROGRAM FOR EVALUATION OF INSTALLED BIOLOGICAL SHIELDING IN RESEARCH AND TRAINING REACTORS

A. INTRODUCTION

Subdivision (b)(6) (iii) of section 50.34, "Contents of applications technical information," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires an applicant for a license to include in his final safety analysis report plans for preoperational testing and initial operation. This regulatory guide describes a shield test program that is generally acceptable for evaluation of installed biological shielding in research and training reactors

B. DISCUSSION

Subcommittee ANS 6, Shielding, of the American Nuclear Society Standards Committee has developed a standard that describes an operational shield test program which may be used in evaluating the installed biological shielding in research and training reactors. This standard was approved by the American National Standards Committee N18, Nuclear Design Criteria, and its Secretariat. It was subsequently approved and designated ANSI N18.9-1972 by the American National Standards Institute (ANSI) on September 15, 1972

C. REGULATORY POSITION

The requirements and guidelines contained in ANSI N18.9-1972, "Program for Testing Biological Shielding in Nuclear Reactor Plants,"[1] approved September 15, 1972, are generally acceptable and, with due consideration for the unique characteristics of each research and training reactor, provide an adequate basis for conducting a shield test program during preoperational and startup testing for evaluation of installed biological shielding in research and training reactors subject to the following.

1. Section 3 2.4 of ANSI 18 9-1972 defines accessible areas, controlled areas, and unlimited access areas. Section 3.2 5 defines Maximum Permissible Dose rate. Nothing in these paragraphs should imply that exposures need not be controlled to the requirements of 10 CFR Part 20, "Standards for Protection Against Radiation."

2. Section 5 2 of ANSI 18 9-1972 states that procedures for implementing the minimum shield test program shall be prepared. These procedures should be designed so that exposures to personnel performing the test program are as low as practicable. These procedures should also be designed so that safety hazards to personnel performing the shield test program are properly identified For example, gas monitoring should be required where gases or vapors could affect the accessibility of an area.

3 Section 6 of ANSI N18.9-1972 specifies tests that should be conducted for evaluation of installed biological shielding. This section further specifies use of survey meters when conducting the required tests. The shield test program should also include provisions for gamma and neutron film mapping of critical areas where personnel exposure may occur due to streaming, cracks, or gaps in the shielding too small to detect by survey meters, e g areas in the vicinity of beam holes, irradiation ports, or shielding areas directly aligned with the core

4 Section 9 of ANSI N18 9-1972 states that instruments used in carrying out the minimum shield test program shall have been calibrated prior to use in the test program and immediately after each survey The shield test program should also include provisions for

[1] Copies may be obtained from American Nuclear Society, 244 East Ogden Avenue, Hinsdale, Illinois 60521

calibrating all radiation survey monitors (both portable and installed) against a source emitting radiation of approximately the same type and intensity as that expected to be measured during the survey.

5. Sections 9.2 and 9.3 of ANSI 18.9-1972 provide requirements for survey instruments. In addition to these requirements, a survey instrument's range should be consistent with the actual dose range expected. For measurements conducted while a reactor is operating in the pulsed mode, appropriate instrumentation, such as film packets, which will properly respond to and measure radiation during the pulsed mode of operation should be provided.

13 ACCIDENT ANALYSES

In the other chapters of the SAR, the applicant should discuss and analyze the safety considerations and functional requirements at a non-power reactor facility for the design bases that ensure safe reactor operation and shutdown and acceptable protection for the public, the operations and user staff, and the environment. In those chapters, the applicant should not only discuss potential equipment malfunctions, deviations of process variables from normal values, and potential effects of external phenomena on the facility, but should also describe how equipment will work when needed in accident situations. In Chapter 13 of the SAR, the applicant should submit information and analyses that show that the health and safety of the public and workers are protected and that the applicant has considered potential radiological consequences in the event of malfunctions and the capability of the facility to accommodate such disturbances. The major purpose of this chapter is for the applicant to demonstrate that the facility design features, safety limits, limiting safety system settings, and limiting conditions for operation have been selected to ensure that no credible accident could lead to unacceptable radiological consequences to people or the environment.

The issue of what standards to use in evaluating accidents at a research reactor was discussed in an Atomic Safety and Licensing Appeal Board (ASLAB) decision issued May 18, 1972, for the research reactor at Columbia University in New York City. The ASLAB stated that "as a general proposition, the Appeal Board does not consider it desirable to use the standards of 10 CFR Part 20 for evaluating the effects of a postulated accident in a research reactor inasmuch as they are unduly restrictive for that purpose. The Appeal Board strongly recommends that specific standards for the evaluation of an accident situation in a research reactor be formulated." The staff has not found it necessary to follow the board's recommendation to develop separate criteria for evaluating research reactor accidents because most research reactors to date have been able to conform to the conservative criteria of 10 CFR Part 20.

The principal safety issues that differentiate test reactors from research reactors are the reactor site requirements and the doses to the public that could result from a serious accident. For a research reactor, the results of the accident analysis have generally been compared with 10 CFR Part 20 (10 CFR 20.1 through 20.602 and appendices for research reactors licensed before January 1, 1994, and 10 CFR 20.1001 through 20.2402 and appendices for research reactors licensed on or after January 1, 1994). For research reactors licensed before January 1, 1994, the doses that the staff has generally found acceptable for accident analysis results are less than 5 rem whole body and 30 rem thyroid for occupationally exposed persons and less than 0.5 rem whole body and 3 rem thyroid for members of the public. For research reactors licensed on or after January 1, 1994, occupational exposure is discussed in 10 CFR 20.1201 and public exposure is discussed in 10 CFR 20.1301.

In several instances, the staff has accepted very conservative accident analyses with results greater than the 10 CFR Part 20 dose limits discussed above.

If the facility conforms to the definition of a test reactor, the results should be compared with 10 CFR Part 100. As discussed in the footnotes to 10 CFR 100.11, the doses given in 10 CFR Part 100 are reference values and are not intended to imply that the dose numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, they are values that can be used for evaluating reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of exposure of the public to radiation.

The accidents analyzed should range from such anticipated events as a loss of normal electrical power to a postulated fission product release with radiological consequences that exceed those of any accident considered to be credible. This limiting accident is named the maximum hypothetical accident (MHA) for non-power reactors; the details are reactor specific. Because the MHA is not expected to occur, the scenario need not be entirely credible. The initiating event and the scenario details need not be analyzed, but the potential consequences should be analyzed and evaluated.

The information on credible postulated accidents should achieve the following objectives:

- Ensure that enough events have been considered to include any accident with significant radiological consequences. Rejection of a potential event should be justified in the discussions.

- Categorize the initiating events and scenarios by type and likelihood of occurrence so that only the limiting cases in each group must be quantitatively analyzed.

- Develop and apply consistent, specific acceptance criteria for the consequences of each postulated event.

Each postulated event should be assigned to one of the following categories, or grouped consistently according to the type and characteristics of the particular reactor:

- MHA
- insertion of excess reactivity (ramp, step, startup, etc.)
- loss of coolant
- loss of coolant flow
- mishandling or malfunction of fuel

- experiment malfunction
- loss of normal electrical power
- external events
- mishandling or malfunction of equipment

The accident events in each group should be evaluated systematically to identify the limiting event selected for detailed quantitative analysis. Limiting events in each category should have potential consequences that exceed all others in that group. As noted above, the MHA selected should bound all credible potential accidents at that facility, yet should be an event that is not likely to occur during the life of the facility.

13.1 Accident-Initiating Events and Scenarios

In this section of the SAR, the applicant should describe potential accident-initiating events and scenarios for non-power reactors. For documents on general accident scenarios and analysis, radiological consequences, and fuel types, see Section 13.4. The following sections contain suggestions for selecting and categorizing postulated accidents:

13.1.1 Maximum Hypothetical Accident

In general, the escape of fission products from fuel or fueled experiments and their release to the unrestricted environment would be the most hazardous radiological accident conceivable at a non-power reactor. However, non-power reactors are designed and operated so that a fission product release is not credible for most. Therefore, this release under accident conditions can reasonably be selected as the MHA, which bounds all credible accidents and can be used to illustrate the analysis of events and consequences during the accidental release of radioactive material. The applicant may choose to perform sensitivity analysis of the assumptions of the MHA. For example, reactor operating time before accident initiation may be examined to determine the change in MHA outcome if a more realistic assumption is made. However, these assumptions may form the basis for technical specification limits on the operation of the facility. The MHA could be any of the following:

- A specified fraction of fuel in the core melts. (How this occurs may or may not be specified.)
- Cladding is stripped from a specified fraction of the core fuel plates or elements.

- The fuel encapsulation bursts, releasing gaseous fission products to the pool or the air. (The failure of one fuel element in air is the MHA for a TRIGA reactor.)

- A fueled experiment melts or fails catastrophically in the pool or in the air.

13.1.2 Insertion of Excess Reactivity

In some cases, the insertion of excess reactivity can be an initiating event that leads to fuel or fueled experiment melting, which is the MHA. Insertion-of-excess-reactivity accidents can also be used to show that limiting conditions for operation on reactivity are justified. Some insertion-of-excess-reactivity events are the following:

- Rapid inadvertent insertion of a portion of all excess reactivity loaded into the reactor.

- Rapid removal of the most reactive control rod or shim rod.

- Rapid insertion of a fuel element into a vacancy in the core at the most reactive position.

- Ramp insertion of reactivity by drive motion of the most reactive control rod or shim rod, or ganged rods, if possible. (This event could occur during reactor startup procedures or when the reactor is at power.)

- Failure or other malfunction of an experiment that inserts excess reactivity. (This can be used to justify movable experiment reactivity limits.)

- Rapid increase in reactivity as a result of a change in operating parameters, such as a surge of cold coolant.

13.1.3 Loss of Coolant

In many non-power reactor designs, the loss-of-coolant accident (LOCA) is of no consequence because decay heat in the fuel is so small as to be incapable of causing fuel failure. In some higher power reactors (normally greater than 2 MW), an engineered safety feature, such as an emergency core cooling system, may need to be operable for some time after reactor shutdown to remove decay heat in the event of a LOCA. Some initiators of LOCAs are the following:

- failure or malfunction of some component in the primary coolant loop
- failure or malfunction of an experimental facility, such as a beam tube
- failure or leak of the reactor coolant boundary

13.1.4 Loss of Coolant Flow

This accident is usually most limiting for forced convection-cooled non-power reactors, where the forced flow is downward through the reactor core. The effects of loss of coolant flow should be considered for all non-power reactors. Upon loss of forced downward coolant flow through the core, coolant flow in the core must reverse to upward natural-convection cooling. During the flow reversal, heat transfer may be inadequate in the core. Loss of coolant flow may also occur if a foreign object obstructs a coolant flow path. Some initiators of loss of coolant flow are the following:

- loss of electrical power
- failure of a pump or other component in the primary coolant system
- blocking or significant decrease in flow in one or more fuel coolant channels

13.1.5 Mishandling or Malfunction of Fuel

This class of accidents represents fuel damage less severe than the MHA. Operation with water-logged fuel is an important consideration for pulsing reactors where the sudden addition of energy to the fuel due to a pulse may cause the water to turn quickly to steam and damage the fuel cladding. Initiating events in this class are the following:

- overheating of fuel during steady-power or pulsed operation
- dropping or otherwise damaging fuel in any location
- dropping, impact, or other malfunction of a non-fueled component
- operation (including pulsing) with damaged fuel, such as water-logged pin- or rod-type fuel

13.1.6 Experiment Malfunction

The conduct of experiments is one of the important functions of a non-power reactor. Experiments may contain fuel, explosives, and highly reactive materials. Failure or malfunction of experiments may initiate accidents. In some cases, particularly for lower power non-power reactors, failure or malfunction of an experiment may be the MHA, especially if fueled experiments are allowed by the facility license. Initiating events for this class of accidents include the following:

- loss of cooling capability or other malfunction in a fueled experiment resulting in liquefaction or volatilization of the fissile component

- loss of cooling capability in a strongly absorbing non-fueled experiment resulting in absorber failure and rapid increase in reactivity

- placement of an experiment component in an unplanned location, causing effects that were not evaluated

- failure of an experiment containing highly reactive contents

- failure of an experiment and release of corrosive materials into the reactor coolant

- detonation of an explosive experiment

13.1.7 Loss of Normal Electrical Power

This accident initiator could result from onsite or offsite power interruptions. Emergency power supplies, if provided, are assumed to operate. However, the applicant may want to analyze the effects of failure of emergency power.

13.1.8 External Events

This class of accident initiators represents some outside effect on the facility, be it natural or caused by humans. Some initiating events in this category are the following:

- meteorological disturbance, 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

13.1.9 Mishandling or Malfunction of Equipment

This class of accident initiators represents failures or errors that do not fall into one of the other categories. Some initiators in this category are the following:

- operator error at the controls

- other operator errors

- malfunction or loss of safety-related instruments or controls, such as amplifiers or power supplies
- electrical fault in control/safety rod systems
- malfunction of confinement or containment system
- rapid leak of contaminated liquid, such as waste or primary coolant

13.2 Accident Analysis and Determination of Consequences

In this section of the SAR, the applicant should discuss each event giving information consistently and systematically for gaining a clear understanding of the specific reactor and making comparisons with similar reactors. Many of the steps used to select the limiting event in each category may be semiquantitative. However, the analyses and determination of consequences of the limiting events should be as quantitative as possible. The following steps are suggested:

(1) State the initial conditions of the reactor and equipment. Discuss relevant conditions depending on fuel burnup, experiments installed, core configurations, or other variables. Use the most limiting conditions in the analyses.

(2) Identify the causes that initiate the event; the causes may include equipment malfunction, operator error, or a natural phenomenon or one caused by humans. Base the scenario on a single initiating malfunction, rather than on multiple causes.

(3) List the sequence of events, assumed equipment operation and malfunction, and operator actions until a final stabilized condition is reached. Discuss functions and actions assumed to occur that change the course of the accident or mitigate the consequences, such as reactor scrams or initiation of such engineered safety features as emergency core cooling. If credit is taken for mitigation of the accident consequences, discuss the bases used to determine that the systems are operable and discuss the system functions.

(4) Classify damage that might occur to components during the accident until the situation is stabilized. Discuss all components and barriers that could affect the transfer of radiation and radioactivity from the reactor to the public and that ensure continued stability of conditions after the accident.

(5) Prepare realistic analyses to demonstrate a detailed, quantitative evaluation of the accident evolution, including the performance of all barriers and the

transport of radioactive materials to the unrestricted area. Include the assumptions, approximations, methodology, uncertainties, degree of conservatism, margins of safety, and both intermediate transient and ultimate radiological conditions. Justify the methods used. Further, make sure the information is complete enough to allow the results to be independently reproduced or confirmed. Demonstrate the validation of the computational models, codes, assumptions, and approximations by comparison with measurements and experiments when possible. Describe in detail computer codes that are used as to the name and type of code, the way it is used, and its validity on the basis of experiments or confirmed predictions of operating non-power reactors. Include estimates of the accuracy of the analytical methods. In Chapter 11, "Radiation Protection Program and Waste Management," of the SAR, discuss the methods and assumptions used to analyze the release and dispersion of radioactive materials from normal operations. Adapt those methods as appropriate for accident analyses.

(6) Define and derive the radiation source terms, if any are involved. Include in the source terms the quantity and type of radionuclides that could be released, their physical and chemical forms, and the duration of potential releases. Describe potential radiation sources that could cause direct or scattered radiation exposure to the facility staff and the public.

(7) Evaluate the potential radiological consequences using realistic methods. Discuss the degree of conservatism in the evaluation. For example, include a discussion of the degree of conservatism introduced by the use of postulated release fractions or assumption of an infinite hemispherical cloud.

Include environmental and meteorological conditions specific for the facility site to illustrate consequences. Exposure conditions should account for the facility staff until the situation is stabilized (including staff evacuation and reentry), the most exposed member of the public in the unrestricted environment until the accident conditions are terminated or the person is moved, and the integrated exposure at the facility boundary and the nearest permanent residence. The radiological consequences should include external and internal exposures. Address contamination of land and water where applicable; include exposure control measures to be initiated.

13.3 Summary and Conclusions

In this section of the SAR, the applicant should summarize the important conclusions about the postulated accidents and the potential consequences. The applicant should compare the projected radiological consequences with the acceptance criteria discussed previously in this chapter The information should demonstrate that all reasonable measures have been incorporated into the facility design bases to prevent undue radiation exposures and contamination of the unrestricted environment. The discussions should show that engineered safety features have been incorporated where necessary to limit consequences to acceptable levels.

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