THE OHIO STATE UNIVERSITY

The Ohio State University
Nuclear Reactor Laboratory

1298 Kinnear Road
Columbus, OH 43212

Phone 614-688-8220
Fax 614-292-2209

04-Sep-2007

Daniel E. Hughes, Project Manager
Research and Test Reactors Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

RE: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE OHIO STATE UNIVERSITY RESEARCH REACTOR APPLICATION FOR RE-LICENSING (TAC NO. MA7724)

Mr. Hughes,

Please find enclosed our response to your letter dated 05-July-2007 requesting additional information to supplement the OSURR re-licensing application. If you have any questions, please contact Andrew Kauffman at 614-688-8220 or kauffman.9@osu.edu.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on 04-Sep-2007.

Sincerely,

Thomas E. Blue

Thomas Blue, Director
OSU Nuclear Reactor Lab
The Ohio State University
(License R-75, Docket 50-150)

c. W.A. "Bud" Baeslack III, Dean, College of Engineering
Andrew C. Kauffman, Associate Director, OSU Nuclear Reactor Lab

A020

REQUESTED ADDITIONAL INFORMATION REGARDING THE OHIO STATE UNIVERSITY RESEARCH REACTOR (OSURR) - DOCKET NO. 50-150

This document is the main body of the response to the Request for Additional Information (RAI) from the Nuclear Regulatory Commission dated 05-Jul-2007. In this document and its attachments, we have attempted to address the clarifications asked for in the requesting document. In addition, we have included additional improvements to the Technical Specifications of our facility. Where inconsistencies exist between this document and previous documents (e.g. the 21-Aug-2002 response to a Request for Additional Information), statements, information, and commitments in this document supersede what is in the former documents.

Safety Analysis Report, Chapter 8

Per a phone conversation with William Kennedy on 30-Aug-2007, this information will be sent in a subsequent mailing. A mailing regarding this information will be sent to the NRC by 01-Oct-2007.

As the subsequent mailing will include information that was to be in the first attachment to this document, there is no Attachment A included with this document; however, there are still Attachments B and C.

Technical Specifications

Attachment B to this response document lists proposed changes to the Technical Specifications (TSs), ordered by section number. Attachment C contains the updated TSs including all changes listed in Attachment B. This version supersedes previous submitted versions.

Unless otherwise specified below, responses to specific requests for clarification or additional information posed in the *Request for Additional Information* are included in the list given in Attachment B. Needed responses that are not covered in Attachment B follow immediately in this main document. The responses below are preceded by the requesting text from the RAI in bold italics.

Section 1.3: Incorporate the relevant portions of the supplement to your application submitted August 21, 2002.

This refers to a commitment in our 21-Aug-2002 submittal to add definitions for the terms "fast scram" and "slow scram". Instead of doing this, we propose that replacing both of these terms with "scram" is a better course of action. The terminology of fast and slow scrams is an historical way to describe different means for the safety system to scram the reactor. A fast scram cuts current to the control rod electro-magnets by electronic means through the biasing of a current-controlling device, and a slow scram cuts current to the control rod electro-magnets by electromechanical means through the opening of relay contacts controlling the operation of the magnet current supply modules. While a slow scram may have been slightly slower (on the order of milliseconds) in the era of vacuum tubes, it is now a somewhat anachronistic term that has proven to be confusing to those not familiar with our system. Since both result in cutting electric current to the control rod electro-magnets to quickly drop rods into the core, differentiating between them does not add value to the TSs. (Regardless of whether the scram is fast or slow, the limit on "Control Rod Scram Time" must be met.) Specifying that a scram must occur is what is crucial. Therefore, we will not add the terms "fast scram" and "slow scram" to our definitions. Instead, we will simply refer to "scram" in our Technical Specifications. This is reflected in Attachments B and C to this response.

Section 3.1.1(5): Clarify the range of temperatures referred to by the phrase "all normal operating temperatures."

As seen in Table 4.1 of the SAR under "Thermal Characteristics", normal operating temperatures for moderator range from ambient temperature (~ 20 °C) up to the maximum water temperature (~ 60 °C).

Section 3.2.1: Incorporate the relevant portions of the supplement to your application submitted August 21, 2002. Specifically, update the reference to the SAR found in the Bases for this proposed TS, and ensure consistency with Section 1.3 of the proposed TSs.

This is referring (other than the needed reference update) to a commitment in our 21-Aug-2002 submittal to add and synchronize definitions for "scram time", "drop time", and "insertion time". Instead of doing this, we propose that replacing all of these terms with the single term "control rod scram time" and standardizing the definition between Sections 1.3 and 3.2.1 is preferable and should eliminate confusion about terms. This is reflected in Attachments B and C to this response. The definition given in the 21-Aug-2002 submittal will be used.

Section 3.2.3: Incorporate the relevant portions of the supplement to your application submitted August 21, 2002. Specifically, ensure that scram terminology is consistent with Section 1.3 and other relevant sections of the proposed TSs.

Please see response above for Section 1.3

Section 3.4: Incorporate the relevant portions of the supplement to your application submitted August 21, 2002.

Section 3.4 of our current TSs is titled “Confinement Isolation”. As this is not a typical term (typically either ‘containment’ or ‘confinement’ is specified), clarification was requested in an 05-Feb-2002 RAI. In our 21-Aug-2002 response, we offered to change the term “confinement isolation” to “building isolation” to avoid confusion stemming from the current terminology.

However, as ANSI/ANS-15.1-1990 recommends that this section of the TSs defines how containment or confinement is achieved for reactor operations, it is preferable to get rid of the terms “confinement isolation” and “building isolation” and specify how confinement is achieved for operations. Therefore we will not be incorporating the change discussed in the 21-Aug-2002 response; instead we will make changes to Section 3.4 of the TS as seen in Attachments B and C of this document.

Section 6.2.4(4): Revise this proposed TS to bring it into conformance with the language of 10 CFR 50.59. As noticed in the Federal Register (64 FR 53613, October 4, 1999), the NRC views this change as editorial in nature.

As seen in Attachments B and C to this document, Section 6.2, “Review and Audit”, has been changed to mirror the guidance in ANSI/ANS-15.1-1990. The new text has been corrected to eliminate use of the outdated phrase “unreviewed safety question”, but this has been done in the new Section 6.2.3(1).

Section 6.2.4(7): Incorporate the relevant portions of the supplement to your application submitted August 21, 2002. Specifically, update the reference to the proposed TS.

In the 21-Aug-2007 submittal, we committed to update a reference in this section of the TSs from “6.6.4” to “6.6.2”. This has been done in the updated TSs seen in Attachment C. However, it is in Section 6.2.3(7) of the updated TSs, due to the changes in Section 6.2 to mirror the guidance in ANSI/ANS-15.1-1990.

Section 6.2.5(4): Revise this proposed TS to reflect Amendment No. 16 to Facility License No. R-75, issued December 4, 1995, that removed license condition 3.F, “Physical Security Plan.” The proposed TS should be consistent with the requirements of 10 CFR 73.67(f).

As seen in Attachments B and C to this document, Section 6.2 of the TSs (Review and Audit) has been rewritten to mirror the guidance given in ANSI/ANS-15.1-1990. As the guidance does not suggest an audit of security implementation plans and procedures, this has been left out of the revised TSs.

Attachment B

Proposed changes to The Ohio State University Research Reactor (OSURR) Technical Specification (TSs)

Notes:

1) The base document to which these changes refer is the current TSs document, not the proposed TSs submitted with the Relicensing Application on 15-Dec-1999. Doing so makes this list a complete compilation of changes rather than a list of changes that reference a document already including changes.
2) Where inconsistencies exist between this document and previous documents (e.g. the 21-Aug-2002 response to a Request for Additional Information), statements, information, and commitments in this document supersede what is in the former documents.
3) Due to additions and deletions in the TSs document, some changes may occur in section numbering. Unless otherwise noted, section numbers referenced in this changes document refer to the numbering in the current version of the TSs.
4) The proposed changes have been ordered by section number, and the justification for each proposed change follows immediately after on an indented line.

Changes and justifications:

1. § 1.1 – Removed comma in the first line of the second paragraph
    - Corrected typo
2. § 1.1 – Changed reference from ANSI/ANS-15.1-1982 to ANSI/ANS-15.1-1990
    - Updated guidance document used
3. § 1.2.2 – Changed reference from ANSI/ANS-15.1-1982 to ANSI/ANS-15.1-1990
    - Updated guidance document used
4. § 1.3 – Changed "Commission" to "Nuclear Regulatory Commission" in the definition of "Administrative Controls"
    - Consistency of terminology
5. § 1.3 - Removed definition of "Containment"

- This term is not used in our Technical Specifications, and it does not apply to our facility.

6. § 1.3 – Removed definition of "Commission"

   - This shorthand for the NRC is not used in the TSs, so the definition was unnecessary.

7. § 1.3 – Removed definition of "Direct Supervision"

   - This term is not used in the TSs, so the definition was unnecessary.

8. § 1.3 – Removed definition for "Exclusion Area" and replaced with "Controlled Area". Used definition from 10-CFR-20.1003

   - The term "Exclusion Area" is not applicable to research reactors

9. § 1.3 – Added definition for "Fuel Element, Blank"

   - This term is used in the TSs and was not defined

10. § 1.3 – Added definition for "Fuel Element, Partial"

    - This term is used in the TSs and was not defined

11. § 1.3 – Replaced term "Control Rod Fuel Element" with "Fuel Element, Control Rod" and updated definition

    - Improved organization and definition

12. § 1.3 – Replaced term "Standard Fuel Element" with "Fuel Element, Standard" and updated definition

    - Improved organization and definition

13. § 1.3 – Added entry from "Indicated Value" in definitions cross-listed with the term "Measured Value"

    - This term is also used.

14. § 1.3 – Removed entry for "Licensee" under definitions

    - This was not really a definition, as it merely stated that The Ohio State University was the licensee. This is made sufficiently clear in Section 1.2.

15. § 1.3 – Updated definition for "Limiting Conditions for Operation"

    - Make consistent with ANSI/ANS-15.1-1990

16. § 1.3 – Removed entry for “Nuclear Regulatory Commission” in definitions and replaced with entry for “NRC”

    - Old format had an acronym as a definition for a term instead of vice versa

17. § 1.3 – Removed entry for “Onset of Nucleate Boiling” in definitions and replaced with entry for “ONB”

    - Old format had an acronym as a definition for a term instead of vice versa

18. § 1.3 – Updated definitions for “Operable” and “Operating”

    - Made consistent with ANSI/ANS-15.1-1990

19. § 1.3 – Removed definition for “Reactivity Limits”

    - The existing definition was incomplete in that it only referred to limits on excess reactivity. Rather than changing the definition, it was deleted since it is self explanatory. (In addition, it is not a suggested definition in ANSI/ANS-15.1-1990.)

20. § 1.3 – Updated definition of “Reactivity Worth of an Experiment”

    - Made consistent with ANSI/ANS-15.1-1990

21. § 1.3 – Removed entry for “Reactor Operations Committee” in definitions and replaced with entry for “ROC”

    - Old format had an acronym as a definition for a term instead of vice versa

22. § 1.3 – Updated definition of “Reactor Secured”

    - Made consistent with ANSI/ANS-15.1-1990

23. § 1.3 – Updated definition of “Regulating Rod”

    - Definition previously made reference to an automatic servo controller, and we no longer have this.

24. § 1.3 – Removed definition for “Reportable Occurrence”

    - The definition merely referred the reader to Section 6.5.2 of the Technical Specification, making the definition unnecessary.

25. § 1.3 – Removed entry for “Safety Analysis Report” in definitions and replaced with entry for “SAR”

    - Old format had an acronym as a definition for a term instead of vice versa

26. § 1.3 – Updated definition of "Safety Limits"

   - Made consistent with ANSI/ANS-15.1-1990

27. § 1.3 – Replaced definition of "Scram Time" with "Control Rod Scram Time" and made the definition match that given in the 21-Aug-2002 submittal to the NRC.

   - Make the TSs internally consistent and meet the requirement of a previous commitment.

28. § 1.3 – Updated definition of "Surveillance Time Intervals" to include the interval "five-year" and to make the other interval names match (including replacing the term "two-year" with "biennial").

   - Make consistent with ANSI/ANS-15.1-1990

29. § 2.2 – Removed the Specification, "Steady state power level shall not exceed 500 kW thermal."

   - This is not a Limiting Safety System Setting. As this limit is explicitly stated in the facility license, the specification may be deleted instead of moving it to another part of the TSs

30. § 2.2 – Removed the sentence in Bases, "One may also reference SAR Sections 4.8.1, 4.8.2 for an estimate of cladding temperature during steady state operation at 500 kW (56.5 °C)."

   - This statement purpose was to support the Specification, "Steady state power level shall not exceed 500 kW thermal." Since the specification has been removed, this statement is unnecessary.

31. § 2.2 – Changed LSSS specification for scram on maximum reactor power to state that "Reactor safety systems settings shall initiate automatic protective action at or before an indicated reactor power of 600 kW" instead of stating that "Reactor safety systems settings shall initiate automatic protective action so that reactor thermal power level shall not exceed 600 kW (120% of full power) during a transient.".

   - The former statement was not supported by the SAR. The SAR Design Basis Accident analysis shows that power will peak higher than 600 kW but that the Safety Limit on Fuel Temperature will not be exceeded.

32. § 3.1.1 – Changed wording of *Objective* to say, "To ensure the capability for safe shutdown of the reactor and that the safety limits are not exceeded."

   - More accurate wording

33. § 3.1.1 – Added clause to specification to exempt operations for the determination of reactor reactivity worth values.

 - Previously, we have assumed that this exemption was understood, as the reactor must be operated to determine the values that are being used to compare to limits for reactor operations. We wanted to make this explicit to avoid any potential for a misunderstanding.

34. § 3.1.1(3) – Removed Specification and Basis for limit on regulating rod worth

 - Since the basis for this specification was strictly related to safety of the automatic servo control system, it is no longer relevant without an automatic servo control system

35. § 3.1.1(4) – Removed reference to "regulating rod fuel element"

 - There are four control rod fuel elements, but there is no specific "regulating rod fuel element"

36. § 3.1.1(4) – Removed reference to "instrumented fuel element"

 - We have no instrumented fuel elements.

37. § 3.1.1(4) – Changed term "control fuel element" to "control rod fuel element"

 - Match term used in definition

38. § 3.1.1(4) – Added term "partial fuel element" to list of what can go in core grid positions.

 - Previously, the definition of "standard fuel element" included any core element that was not a control rod element. Since this has now been separated into "standard", "partial", and "blank", these terms need to be reflected as appropriate in Specifications.

39. § 3.2.1 – Changed section name from "Control Rod Drop Times" to "Control Rod Scram Time". Also changed wording of specification to use this terminology.

 - Consistency of terms within the TSs (i.e. consistent with term in "Definitions").

40. § 3.2.1 – Updated Applicability to align with the definition given for "Control Rod Scram Time" in Section 1.3 as well as the definition offered in the 21-Aug-2002 submittal to the NRC

 - Internal consistency in the TSs and meet the requirement of a previous commitment.

41. § 3.2.1 – Updated Specification to specify that the 3.2.1 Control Rod Scram Time is to be measured for fully-withdrawn rods

- ➢ Clarification necessitated by the change to the definition given in "Applicability".

42. § 3.2.1 – Added hyphen between the words "short period" in the Bases.

- ➢ Proper punctuation

43. § 3.2.1 – Updated the reference to the SAR from "Section 4.3.3" to "Section 8.4.3.3".

- ➢ Typo correction

44. § 3.2.2 – Changed the Specification for Maximum Reactivity Insertion Rate from 0.02% to 0.05% Δk/k/second.

- ➢ The limit of 0.02% Δk/k/second is historical and not directly supported by the SAR. The only mention of reactivity insertion rate in the SAR is the following text in Section 4.4.1, "The maximum rate of reactivity addition from the control rod system is estimated to be about 0.02 % Δk/k/second."

  However, the SAR analyzes an instantaneous insertion of 0.93% Δk/k in the Design Basis Accident in Section 8.4.3.2 without breach of the Safety Limit. Therefore, a maximum reactivity insertion rate that allows sufficient time for the operator to react before this amount of reactivity is inserted will protect the Safety Limit according to the analysis in § 8.4.3.2. The value of 0.05% was chosen to be consistent with the maximum reactivity insertion rate for experiments in Section 3.7.1 of the TSs. It would take over 18 seconds at this rate to reach the total reactivity insertion analyzed in the Design Basis Accident.

45. § 3.2.2 – Added information to the Basis.

- ➢ Include the rationale shown above for a maximum rate of 0.05% delta k/k per second.

46. § 3.2.3 – In the table of "Minimum Number of Scram Channels", replaced references to "fast scrams" and "slow scrams" with just references to "scrams".

- ➢ This terminology is an historical way to describe different means for the safety system to scram the reactor. A fast scram cuts current to the control rod electro-magnets by electronic means through the biasing of a current-controlling device, and a slow scram cuts current to the control rod electro-magnets by electromechanical means through the opening of relay contacts controlling the operation of the magnet current supply modules. While a slow scram may have been slightly slower (on the order of milliseconds) in the era of vacuum tubes, it is now a somewhat anachronistic term that has proven to be confusing to those not familiar with our system. Since both result in cutting electric current to the control rod electro-magnets to quickly drop rods into the core, differentiating between them does not add value to the TSs. (Regardless of whether the scram is fast or slow, the limit on "Control Rod Scram Time" must be met.) Specifying that a scram must occur is what is crucial.

47. § 3.2.3(1) – Changed Basis for this Specification to simply state, "Assures safety limit is not exceeded".

   - The extra text that previously followed this statement is irrelevant.

48. § 3.2.3(4) – Changed text in Function description to say "primary and secondary coolant system pumps" instead of "coolant system pumps"

   - Improved clarity

49. § 3.2.3(7) – Removed entry for scram on "Startup Cal-Use" switch

   - This instrument was replaced in 1991 with an instrument without this switch, so it is always in "Use" mode.

50. § 3.2.3(7) – Added entry for scram on "Effluent Monitor Counter" being in its operating mode

   - This switch for "Count / Stop" for this module has been wired into the safety system, as the module needs to be accumulating counts from the effluent monitoring system while the reactor is operating.

51. § 3.2.3(7) – Added entries for scrams on switches on the reactor power level safety modules and the reactor period safety module that select between operating and test mode.

   - These safety modules must be set in their operating mode (non-test mode) while the reactor is operating to perform their function. The system is wired to reflect this, but it was not stated previously in the TSs.

52. § 3.2.3(7) – Removed the scram for "Period Generator Switch" position set to "Off"

   - The setting of this switch is unrelated to reactor safety. The period generator module's function is to provide test signals for two modules that determine reactor period. These modules must be set by switches into test mode for tests to occur, and switching either module to test mode results in a scram. Therefore the reactor is precluded from running when either channel that the period generator module feeds is analyzing a test signal from the period generator. This scram requirement is likely an historic relic that no longer serves a safety function. As the guidance document ANSI/ANS-15.1-1990 states that, "Only those operational parameters and equipment requirements directly related to preserving that safe envelope shall be listed", this scram requirement should be removed.

53. § 3.2.3(7) – Changed the specification to say that particular switches must be in their operating (non-test and non-off) position instead of specifying positions for individual switches

   - The former wording was very specific to labeling on current instrumentation modules. This new wording accomplishes the same function while potentially negating the need for Tech Spec updates because a replacement module is labeled differently (e.g. for replacements that would otherwise fall under 10-CFR-50.59).

54. § 3.2.3(7) – Changed wording from "LOG-Period Amp" to " Period Amp"

   - Better description of circuit. The current period amp circuit is in a module named Log N Period Amplifier, which houses both the LogN and period circuits. There are separate switches for test mode for the LogN circuit and the period circuit. Using the term "LOG-Period Amp" has the potential to cause confusion.

55. § 3.2.3(7) – Rearranged the order of switches listed

   - Better organization

56. § 3.2.3(7) – Changed the number of switches listed under "Minimum Required"

   - Reflect the changes listed above for Section 3.2.3(7)

57. § 3.2.3(8) – Changed wording from "Recorders" to "Time-trace displays".

   - The primary function of these equipment channels for ensuring safe operation is to display a recent time trace of the appropriate signal. To date, they have been referred to as "recorders", as the time-trace displays have been in the form of ink traces on recorder paper. This wording change should preclude potential loss of this primary functionality as technology changes and equipment is replaced.

58. § 3.2.3(10) – Changed the wording "Compensated Ion Chambers" to "Neutron-Sensitive Ionization Chambers" and changed "Minimum Required" from 2 to 4.

   - In addition to needing sufficient high voltage for the chambers supplying signals to the Linear and LogN Channels, sufficient high voltage is necessary for the chambers supplying signals to the power level safety channels. Our safety system is wired to reflect this, but it was not previously stated in the TSs.

59. § 3.2.3(10) – Under "Function", added the word "bias" so that it now reads, "Scram if bias voltage drops below operational specifications"

   - Improved clarity

60. § 3.2.3(11) – Removed entry for scram associated with servo deviation and reduced "Minimum Required" from 4 to 3. The associated basis was also removed.

   - We no longer have servo control system associated with the regulating rod.

61. § 3.2.3(11) – Changed wording from "Safety Set Points on Recorders" to "Safety Set Points Associated with Time-Trace Displays", and changed "Function" description from "Scram if associated recorder values are exceeded" to "Scram if any value listed below is exceeded"

   - Remove references to "Recorders". See change listed above for Section 3.2.3(8).

62. § 3.2.3(13) – Changed wording from "Backup Shutdown Mechanisms" to "Shim/Safety Rod Magnet Current"

   - More descriptive title.

63. § 3.2.3(13) – Changed "Function" description from "Rod drop will occur for any control rod which has excess magnet current > 100 ma" to "Rod drop will occur for any Shim/Safety rod which has magnet current > 100 ma"

   - Better clarity of description.

64. § 3.4 – Changed reference from, "revised SAR of September 1987" to "SAR"

   - Fix outdated reference

65. § 3.4 – Changed section title from "Confinement Isolation" to "Confinement"

   - ANSI/ANS-15.1-1990 recommends that this section specifies how "containment" or "confinement" is achieved for operations. Our facility is configured for confinement.

66. § 3.4 – Changed references from "capability to isolate" the building to "capability to provide confinement for" the building in Applicability and Bases.

   - ANSI/ANS-15.1-1990 recommends that this section specifies how "containment" or "confinement" is achieved for operations. Our facility is configured for confinement.

67. § 3.4 – Changed term "Ventilation fan" to "Exhaust fan"

- Make terminology consistent with other portions of the TSs

68. § 3.4 – Combined items (2) through (4) into a single item: "All exterior doors and windows closed".

    - What is important for maintaining confinement for the building is all exterior doors and windows being closed. The former means of stating this was needlessly wordy.

69. § 3.4(2) – Added qualifier on doors and windows closed allowing for ingress and egress

    - This has always been assumed but not made explicit before.

70. § 3.4 – Changed Bases wording to say, "By having the capability to provide confinement for the Reactor Building, exposure of the public to airborne radioactivity may be limited to the extent analyzed in the SAR"

    - Improved wording

71. § 3.6.1(5) – Changed specification for length of time during which portable survey instruments may be substituted for normally-installed radiation monitors to reflect the guidance given in ANSI/ANS-15.1-1990. The specification now reads, "When required monitors are inoperable, portable instruments, surveys, or analyses may be substituted for any of the normally-installed monitors in Section 3.6.1 above for periods of one week or for the duration of a reactor run in cases where the reactor is continuously operated.".

    - Make specification consistent with ANSI/ANS-15.1-1990.

72. § 3.6.1(2) – Removed Specification and Basis for the rabbit monitoring system. Renumbered subsequent items in Section 3.6.1.

    - When the reactor power limit was increased to 500 kW in 1990, the staff felt that it would be useful to have a system separate from the effluent-monitoring system to monitor Ar-41 solely produced in the rabbit irradiation facility. The rationale was that while we had calculations to predict (or at least provide an envelope for) Ar-41 production in various facilities, we might come near the daily or annual limits for Ar-41 releases and want to separately measure the amount produced in the rabbit facility. In practice, this has not turned out to be the case. We have never come close to reaching the limits for Ar-41 releases, and the rabbit monitoring system has provided no extra useful information to us. Any Ar-41 that is produced in the rabbit facility (whose exhaust is near the effluent sampling tube) is measured by the effluent-monitoring system as well as the rabbit monitoring system. Without the rabbit monitoring system, we would still monitor all Ar-41 releases. As the guidance document ANSI/ANS-15.1-1990 states that, "Only those operational parameters and equipment requirements directly related to preserving that safe envelope shall be listed", this Specification should be removed.

73. § 3.6.2(1) – Changed the Specification to say, "The concentration of radioactive liquids released into the sanitary sewer shall not exceed the limits as specified in 10CFR20.2003."

➢ Better reflect regulations

74. § 3.6.2(2) – Changed terminology for Objective and Specification from "MPC" to "Effluent Concentration limit".

➢ Reflect terminology in 10-CFR-20 Appendix B, Table 2. In the 1999 submittal of the TSs, the phrase "Annual Average Concentration" (AAC) was used in place of MPC to reflect the terminology used in 10CFR20.1302. This has been changed to use of EC to satisfy the request made in the Request for Additional Information to which this document is a response.

75. § 3.6.2(2) – Added reference to "10CFR20 Appendix B, Table 2" to accompany the reference to "10CFR20.1302"

➢ Completeness

76. § 3.6.2(2) – Replaced the phrase "at the point of release into the unrestricted area" with "at ground level below the point of release into the unrestricted area"

➢ Improved clarity. The old wording is somewhat vague, and the new wording is what was intended when the Specification was originally written. This is evidenced by the analysis in Section 6.3.5 of the SAR to which the Basis for this Specification references. (I.e. a dilution factor is used to account for the fact that the exhaust fan is 32 feet above the ground.)

77. § 3.6.2(3) – Changed terminology for Objective and Specification from "MPC" to "DAC". Changed averaging time in Specification from "7 consecutive days" to a "2000 hour work year".

➢ Reflect current regulations.

78. § 4.2.1(2) – Changed term "rod drop time" to "Control-rod scram time" in Specifications and Bases

➢ Consistent terminology with Sections 1.3 and 3.2.1

79. § 4.2.2(5)b – Changed wording to say, "Adequate pool water level indication"

➢ Better description of channel

80. § 4.2.2(5)c – Changed wording to say, "Indication of coolant system pumps operating"

- Better description of channel

81. § 4.2.2(5)d – Changed wording to say, "Indication that there is flow in the primary coolant loop"

- Better description of channel

82. § 4.4 – Changed surveillance frequency in Specification and Basis to Quarterly

- Consistency with ANSI/ANS-15.1-1990

83. § 4.4 – Changed objective to say, "To assure that building confinement capability exists"

- Improved wording

84. § 4.4 – Changed wording in Specification to say, "... to assure that the building exhaust fan is operable and all exterior doors and windows have closure capability.

- Improved wording.

85. § 4.5 – Removed Specification (1) which said, "Ventilation fans and closures shall be checked for proper operation on a quarterly basis". Also, removed numbering from list of Specifications since only one was left.

- As the only ventilation fan is the building exhaust fan and the "closures" are the exterior doors and windows, this Specification is merely restating the Specification in Section 4.4. Also this provides better alignment with ANSI/ANS-15.1-1990 recommendations, which suggests a quarterly check of emergency exhaust systems. As the only emergency exhaust system function our system has is shutoff, the remaining Specification meets this.

86. § 4.5 – Changed Objective to say, "To assure that the ventilation shutoff functions satisfactorily".

- Improved description to align with Specification.

87. § 4.5 – Changed Bases to say, "This surveillance will assure that the building can be isolated quickly if necessary to prevent uncontrolled escape of air-borne radioactivity to the unrestricted environment."

- Improved description to align with Specification.

88. § 4.6.2 – Removed Section 4.6.2, which specifies surveillance requirements for the rabbit vent monitor. Renumbered subsequent items in Section 4.6.

- As described above, the specification for monitoring with the rabbit monitoring system is to be removed from Section 3.6.1. Therefore the requirement for surveillance on this system is unnecessary.

89. § 4.6.4 – Changed Specification to say that, "Beta-gamma and neutron survey meters shall be checked with a source for operability quarterly and shall be calibrated annually". Updated Bases to match.

- Align with ANSI/ANS-15.1-1990 recommendation for surveillance requirements for radiation monitoring systems.

90. § 5.1.1 – Removed the last sentence that referenced The Ohio State University Research Center, and added the phrase, "located on the West Campus of The Ohio State University" to the first sentence.

- The former term is seldom used anymore. The new reference is more appropriate.

91. § 5.1.2 – Changed "Exclusion Area" to "Controlled Area"

- The new term is appropriate for research reactors

92. § 5.1.2 – Changed the controlled area to "The fence surrounding the Reactor Building"

- There is now a fence with locking gates that surrounds the Reactor Building and is more appropriate than what was formerly listed as the exclusion area.

93. § 5.2.2 - Changed 3rd sentence from, "If the outside air temperature is $\leq$ 78°F then an outside fan-forced drycooler is sufficient to remove all heat generated at 500 kW" to "Heat is removed from the first by an outside fan-forced dry cooler."

- The former statement was based on calculations done before the reactor power was up-rated in 1990, but operational experience has not born this statement out. Since this statement is merely part of the facility description (as opposed to a specification), the reference to a particular temperature was removed.

94. § 5.3 – Changed approximate number of fuel elements needed for reactor operation to eighteen

- Loss of excess reactivity from operation history has necessitated addition of more fuel. Excess reactivity after this addition of fuel is still within the Tech Spec limit for excess reactivity.

95. § 5.3 – Changed term "core plugs" to "blank elements"

- ➢ Provide consistent terminology. The term "core plugs" is not defined in Section 1.3 and is not used elsewhere in the TSs, whereas "blank elements" is.

96. § 5.4 – Removed description of fuel storage racks located in the Bulk Shielding Facility pool.

- ➢ These racks were analyzed and built to store HEU fuel that was removed from the reactor when it was switched to LEU fuel. They have not been analyzed for LEU, so description is extraneous to the current fuel.

97. § 6.1 – Updated Figure 6.1 to reflect current administrative organization structure for the OSURR.

- ➢ Make TSs consistent with Amendment #18 to the Facility Operating License for the OSURR.

98. § 6.1 – Updated Figure 6.1 to reflect changes in Section 6.2 (ROC to report to Level 1 Management now)

- ➢ Align with ANSI/ANS-15.1-1990 recommendations

99. § 6.1 – Updated Figure 6.1 to reflect changes in position titles for those above the Director of the Radiation Safety Section. Paths of responsibility have not been changed, so therefore independence from OSURR management is not affected; only position titles have changed.

- ➢ Make figure consistent with current position titles

100. § 6.1.3 – Changed staffing requirements to mirror the guidance in ANSI/ANS-15.1-1990, with the following exceptions: (1) Did not specify that an SRO is required for relocation of any experiment with reactivity worth greater than one dollar, and (2) did not specify that an SRO is required for recovery from a significant power reduction.

- ➢ Align with ANSI/ANS-15.1-1990 recommendations and meet the requirements of 10-CFR-50.54. The exceptions were made because (1) we are limited to total experimental worth of less than one dollar, and (2) we are not a high-power reactor, so this does not apply

101. § 6.1.4 – Updated referenced standard from ANSI/ANS-15.4-1977 to ANSI/ANS-15.4-1988.

- ➢ Newer version of guidance document.

102. § 6.2 – Changed "Review and Audit" section to mirror the guidance in ANSI/ANS-15.1-1990.

- Align with ANSI/ANS-15.1-1990 recommendations.

103. § 6.2.3(1) – Changed the wording to eliminate use of the phrase "unreviewed safety question" and instead use terminology consistent with the current version of 10-CFR-50.59. (Note: the section number for this item applies to the new numbering, as the former Section 6.2 was significantly changed.)

- Consistency with current regulation language

104. § 6.3.1(7) – Changed this requirement for procedures to say, "Implementation of the Emergency Plan, reactor operator training and requalification requirements, and the security requirements of 10-CFR-73.67."

- The former wording implied that there was an official "Physical Security Plan" for which there should be implementing procedures. Amendment 16 removed a condition for this from the license. Therefore, the new wording reflects that the security requirements of 10-CFR-73.67 must be implemented in procedures instead.

105. § 6.6.1, § 6.6.2(1), § 6.6.2(2), § 6.6.2(3), § 6.6.2(4) – Changed text for where reports should be sent to align with the following text in 10-CFR-50.4(b)(1), "to the NRC's Document Control Desk (if on paper, the signed original), with a copy to the appropriate Regional Office".

- Consistency with appropriate regulations

Attachment C

APPENDIX A

TO

FACILITY OPERATING LICENSE NO. R-75

Technical Specifications

And Bases For

The Ohio State University

Pool-Type Nuclear Reactor

Columbus, Ohio

Docket No. 50-150

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Scope

This document constitutes the Technical Specifications for Facility License No. R-75 and supersedes all prior Technical Specifications. Included are the "Specifications" and the "Bases" for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, are included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

This document was written to be in conformance with ANSI/ANS-15.1-1990. The content of the Technical Specifications includes: Definitions, Safety Limits, Limiting Safety System Settings, Limiting Conditions for Operation, Surveillance Requirements, Design Features, and Administrative Controls.

1.2 Application

1.2.1 Purpose

These Technical Specifications have been written specifically for The Ohio State University Research Reactor (OSURR).

The Technical Specifications represent the agreement between the licensee and the U.S. Nuclear Regulatory Commission on administrative controls, equipment availability, and operational parameters.

Specifications are limits and equipment requirements for safe reactor operation and for dealing with abnormal situations. They are typically derived from the Safety Analysis Report (SAR). These specifications represent a comprehensive envelope for safe operation. Only those operational parameters and equipment requirements directly related to preserving that safe envelope are listed.

1.2.2 Format

The format of this document is in general accordance with ANSI/ANS-15.1-1990.

1.3 Definitions

Administrative Controls - those organizational and procedural requirements established by the Nuclear Regulatory Commission and/or the facility management.

ALARA - as low as is reasonably achievable.

Channel - the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

Channel Calibration - an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the measured parameter. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip settings, and shall be deemed to include a channel test.

Channel Check - a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

Channel Test - the introduction of a signal into the channel for verification that it is operable.

Cold Clean Core - when the core is at ambient temperature and the reactivity worth of xenon is negligible.

Confinement - a closure on the overall facility which controls the movement of air into it and out of it through a controlled path.

Control Rod - a device fabricated from neutron absorbing material which is used to establish neutron flux changes.

Control Rod Scram Time - elapsed time from the receipt of a safety signal to when a shim/safety rod is fully inserted.

Controlled Area - an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason.

Controls - mechanisms used to regulate the operation of the reactor.

Core - the general arrangement of fuel elements and control rods.

Critical - when the effective multiplication factor (k_{eff}) of the reactor is equal to unity.

Excess Reactivity - that amount of reactivity that would exist if all control rods were removed from the core.

Experiment - any operation, or any apparatus, device, or material installed in or near the core or which could conceivably have a reactivity effect on the core and which itself is not a core component or experimental facility, intended to investigate non-routine reactor parameters or radiation interaction parameters of materials.

Experimental Facility - any structure or device associated with the reactor that is intended to guide, orient, position, manipulate, or otherwise facilitate completion of experiments.

Explosive Material - any material that is given an Identification of Reactivity (Stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, Identification System for Fire Hazards of Materials, or is enumerated in the Handbook for Laboratory Safety published by the Chemical Rubber Company (1967).

Facility - the Reactor Building including offices and laboratories.

Fuel Element, Blank - A core element with *no* fuel plates. The bottom ends of these elements are closed to minimize coolant flow bypassing core elements with fuel. Also called "Filler Fuel Element".

Fuel Element, Control Rod - a fuel element with less than the full number of plates that is capable of holding a control rod.

Fuel Element, Partial - a fuel element with the full number of plates and less than 100% of the nominal fuel element loading

Fuel Element, Standard - a fuel element with the full number of plates and 100% of the nominal fuel element loading

Fueled Experiment - any experiment that contains U-235 or U-233 or Pu-239, not including the normal reactor fuel elements.

Indicated Value - See *Measured Value*.

Limiting Conditions for Operation (LCO) - administratively established constraints on equipment and operational characteristics that shall be adhered to during operation of the facility. The LCOs are the lowest functional capability or performance level required for safe operation of the facility.

Limiting Safety System Settings (LSSS) - settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

Measured Value - the value of a parameter as it appears on the output of a channel.

Movable Experiment - one for which it is intended that all or part of the experiment may be moved in relation to the core while the reactor is operating.

NRC - Nuclear Regulatory Commission.

ONB - onset of nucleate boiling.

Operable - a component or system is capable of performing its intended functions in a normal manner.

Operating - a component or system is performing its intended function.

Protective Action - the initiation of a signal or the operation of equipment within the reactor safety system in response to a variable or condition of the reactor facility having reached a specified limit.

Reactivity Worth of an Experiment - value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

Reactor - the combination of core, permanently installed experimental facilities, control rods, and connected control instrumentation.

Reactor Operating - the reactor is operating whenever it is not secured or shutdown.

ROC - Reactor Operations Committee.

Reactor Operator (RO) - an individual who is licensed to manipulate the controls of the reactor in accordance with 10CFR55.

Reactor Safety Systems - those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

Reactor Secured - the reactor is secured when:
(1) Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflections, or
(2) The following conditions exist:
  a. All shim/safety rods are fully inserted, and
  b. The console key switch is in the OFF position and the key is removed from the lock, and

c. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, and
d. No experiments are being moved or serviced that that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment.

Reactor Shutdown - when the reactor is subcritical by at least 1% delta k/k in the cold clean core condition.

Regulating Rod - a low reactivity-worth control rod used primarily to maintain an intended power level.

Restricted Area - area to which access is controlled for purposes of protection of individuals from exposure to radiation and radioactive materials.

SAR - Safety Analysis Report.

Safety Channel - a measuring or protective channel in the reactor safety system.

Safety Limits - limits on important process variables that are found to be necessary to reasonably protect the integrity of the principle physical barriers that guard against the uncontrolled release of radioactivity. The principle physical barrier is often the fuel cladding.

Scram - the rapid insertion of the shim/safety rods into the reactor for the purpose of quickly shutting down the reactor.

Secured Experiment - any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected from the normal environment of the experiment or by forces which can result from credible malfunctions.

Senior Reactor Operator (SRO) - an individual who is licensed to direct the activities of reactor operators. Such an individual may also operate the controls of the reactor pursuant to 10CFR55.

Shall, Should, and May - the word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, which is neither a requirement nor a recommendation.

Shim/Safety Rods - high-reactivity worth control rods used primarily to provide coarse reactor control. They are connected electro-magnetically to their drive mechanisms and have scram capabilities.

Shutdown Margin - the shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems with the most reactive shim/safety rod and the regulating rod in the most reactive position (fully withdrawn) and that the reactor will remain subcritical without further operator action.

Startup Source - a spontaneous source of neutrons which is used to provide a channel check of the startup (fission chamber) channel and provide neutrons for subcritical multiplication during reactor startup.

Surveillance Time Intervals - The average over any extended period for each surveillance time interval shall be closer to the normal surveillance time, e.g. for the biennial interval the average shall be closer to two years rather than 30 months.

Five-year	(interval not to exceed 6 years).
Biennial	(interval not to exceed 30 months).
Annual	(interval not to exceed 15 months).
Semiannual	(interval not to exceed 7-1/2 months).
Quarterly	(interval not to exceed 4 months).
Monthly	(interval not to exceed 6 weeks).
Weekly	(interval not to exceed 10 days).
Daily	(shall be done during the same working day).

Any extension of these intervals shall be occasional and for a valid reason and shall not affect the average as defined.

True Value - the actual value of a parameter.

Unscheduled Shutdowns - any unplanned shutdown of the reactor caused by actuation of the reactor safety systems, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation. They do not include those shutdowns resulting from expected testing operations, or planned shutdowns, whether initiated by controlled insertion of control rods or planned manual scrams.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTINGS (LSSS)

2.1 Safety Limit

Applicability: This specification applies to the melting temperature of the aluminum fuel cladding.

Objective: The objective is to assure that the integrity of the fuel cladding is maintained.

Specification: The reactor fuel temperature shall be less than 550 °C.

Bases: The melting temperature of aluminum is 660 °C (1220 °F). The blister threshold temperature for U_3Si_2 dispersion fuel has been measured as approximately 550 °C. (ANL/RERTR/TM-10, October 1987, NRC NUREG 1313). Because the objective of this specification is to prevent release of fission products, any fuel whose maximum temperature reaches 550 °C. is to be treated as though the safety limit has been reached until shown otherwise.

2.2 Limiting Safety System Settings

Applicability: This specification applies to the following items associated with core thermodynamics:

(1) Reactor Thermal Power Level and

(2) Reactor Coolant Inlet Temperature.

Objective: To assure that the fuel cladding integrity is maintained.

Specification:

(1) Reactor safety systems settings shall initiate automatic protective action at or below an indicated reactor power of of 600 kW.

(2) Reactor safety systems settings shall initiate automatic protective action so that core inlet water temperature shall not exceed 35 °C.

Bases: The criterion for these safety system settings is established as the fuel integrity. If the temperature of the clad is maintained below that for blister threshold then cladding integrity is maintained. This is the case for a power level of 600 kW and a core inlet temperature of 35 °C (normal inlet temperature is ≈ 20-25 °C).

The maximum credible accident analysis is provided in Section 8.4.3.2 of the Safety Analysis Report. The maximum credible accident assumes steady state operation at 600 kW and initiation of a scram at 750 kW. The maximum temperature of the cladding reaches 91 °C (SAR 8.4.3.3).

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Parameters

3.1.1 Reactivity

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worths of the shim/safety rods and regulating rod under any operating conditions.

Objective: To ensure the capability for safe shutdown of the reactor and that the safety limits are not exceeded.

Specification: With the exception of operations performed solely for determination of reactor reactivity worth values, the reactor shall be operated only if the following conditions exist:

(1) The reactor core shall be loaded so that the excess reactivity, including the effects of installed experiments does not exceed 2.6% delta k/k under any operating condition.

(2) The minimum shutdown margin under any operating condition with the maximum worth shim/safety rod and the regulating rod full out shall be no less than 1.0% delta k/k.

(3) All core grid positions internal to the active fuel boundary shall be occupied by a standard, partial, control rod, or blank fuel element; or by an experimental facility.

(4) The moderator temperature coefficient shall be negative and shall have a minimum absolute reactivity value of at least $2x10^{-5}/^{\circ}C$ across the active core at all normal operating temperatures.

(5) The moderator void coefficient of reactivity shall be negative and shall have a minimum value of at least $2.8x10^{-3}$ /1% void across the active core.

Bases:

(1) The maximum allowed excess reactivity of 2.6% delta k/k provides sufficient reactivity to accommodate fuel burnup, xenon buildup, experiments, control requirements, and fuel and moderator temperature feedback (Section 4.2 of the SAR). Also, calculations show that this excess reactivity assures that the maximum temperature of the surface of the cladding will be well below the blister threshold of the U_3Si_2 fuel during a design basis accident (SAR 8.4.3.2).

(2) The minimum shutdown margin ensures that the reactor can be

shutdown from any operating condition and remain shutdown after cooling and xenon decay even with the highest worth rod and the regulating rod fully withdrawn.

(3) The requirement that all grid positions be filled during reactor operation assures that the volume flow rate of primary coolant which bypasses the heat producing elements will be within the range specified in Section 4.8 of the SAR. Furthermore, the possibility of accidentally dropping an object into a grid position and causing increase of reactivity is precluded.

(4) A negative moderator temperature coefficient of reactivity assures that any moderator temperature rise will cause a decrease in reactivity. The U_3Si_2 fuel also has a significant negative temperature coefficient of reactivity due to the Doppler broadening of neutron capture resonances in ^{238}U, but no credit is taken for this effect in our safety analyses.

(5) A negative void coefficient of reactivity helps provide reactor stability in the event of moderator displacement by experimental devices or other means.

3.2 Reactor Control and Safety System

3.2.1 Control Rod Scram Time

Applicability: This specification applies to the elapsed time from the receipt of a safety signal to when a shim/safety rod is fully inserted.

Objective: To ensure that the reactor can be shutdown within a specified period of time.

Specification: The reactor will not be operated unless the control rod scram time for a fully-withdrawn rod for each of the three shim/safety rods is less than 600 msec.

Bases: Control rod scram times as specified ensure that the safety limit will not be exceeded in a short-period transient.

The analysis for this is given in Section 8.4.3.3 of the SAR.

3.2.2 Maximum Reactivity Insertion Rate

Applicability: This applies to the maximum positive reactivity insertion rate by the most reactive shim/safety rod and the regulating rod simultaneously.

Objective: To ensure the reactor is operated safely and the safety limit is not exceeded due to a short period.

Specification: The reactor will not be operated unless the maximum reactivity insertion rate is less than 0.05% delta k/k per second.

Basis: This maximum reactivity insertion rate assures that the Safety Limit will not be exceeded during a startup accident due to a short period generated by a continuous linear reactivity insertion.

The SAR analyzes an instantaneous insertion of 0.93% $\Delta k/k$ in the Design Basis Accident in Section 8.4.3.2 without breach of the Safety Limit. Therefore, a maximum reactivity insertion rate that allows sufficient time for the operator to react before this amount of reactivity is inserted will protect the Safety Limit according to the analysis in § 8.4.3.2. It would take over 18 seconds at this rate to reach the total reactivity insertion analyzed in the Design Basis Accident.

3.2.3 Minimum Number of Scram Channels

Applicability: This specification applies to the reactor safety system channels.

Objective: To stipulate the minimum number of reactor safety system channels that shall be operable to ensure the Safety Limits are not exceeded by ensuring the reactor can be shutdown at all times.

Specification: The reactor shall not be operated unless the safety system channels described in the following table are operable.

	Reactor Safety System Component	**Minimum Required**	**Function**
1.	Core H_2O Inlet Temp.	1	Scram if temp. $\geq$ 35°C
2.	Reactor Thermal power level (Safety Channels)	2	Scram if thermal power $\geq$ 600 kW, as indicated on calibrated ionization chamber channels.
3.	Reactor Period	1	Scram if period $\leq$ 1 sec
4.	Reactor Thermal power level/coolant system pumps	1	Scram if primary and secondary coolant system pumps not on by $\geq$ 120 kW thermal power

Reactor Safety System Component	Minimum Required	Function
5. Coolant Flow Rate	1	Scram if coolant system has no flow (primary) by $\geq$ 120 kW thermal power
6. Pool Water Level	1	Scram if pool level $\leq$ 20 feet (15 feet above core)
7. Switches a. Magnet Power Key switch b. Effluent Monitor Counter switch c. Effluent Monitor Compressor power switch d. LOG-N Amp calibrate or test mode switch e. Period Amp calibrate or test mode switch f. Reactor power-level safety modules (2) calibrate or test mode switch g. Reactor period safety module calibrate or test mode switch	8	Scram if any listed switch is not set properly. Switches to select between operating mode and non-operating mode (e.g. on/off) must be set to operating mode. Switches to select between operating mode and a test or calibrate mode (e.g. Norm/Test) must be set to operating mode.
8. Time-Trace Displays a. LOG-N b. Linear Level c. Start-Up d. Period e. Effluent Monitor	5	Scram if power is lost to any one of the listed time-trace displays
9. Manual Scrams a. Control Room Console b. Pool Top Catwalk c. BSF Catwalk d. Rabbit/BP Area e. Thermal Column/BP Area	5	Scram upon activation of any one manual scram switch
10. Neutron-Sensitive Ionization Chambers	4	Scram if bias voltage drops below operational specifications

Reactor Safety System Component	Minimum Required	Function
11. Safety Set Points Associated with Time-trace Display Signals	3	Scram if any value listed below is exceeded
a. Period		$\leq$ 5 sec
b. Linear Level		$\geq$ 120% of licensed power
c. Start-Up		$\leq$ 2 cts/sec (may be bypassed if K_{eff} < 0.9)
12. Safety System	2	Scram in case of a safety amp fault or if system is discontinuous
13. Shim/Safety Rod Magnet Current	3	Rod drop will occur for any Shim/Safety rod which has magnet current $\geq$ 100 ma

Bases:

1. Assures safety limit is not exceeded
2. Assures safety limit is not exceeded
3. Assures safety limit is not exceeded
4. Assures coolant system pumps are functional before raising power > 120 kW
5. Assures there is always primary coolant flow when greater than 120 kW
6. Assures there is enough primary coolant for natural convection cooling
7. Assures nuclear instrumentation is in proper mode for operation
8. Assures information is available for observation by the reactor operator during operation, and is recorded if required as a record of reactor operations
9. Assures that the reactor can be shut down by the reactor operator in the control room or at other locations near experimental facilities if deemed necessary by other reactor staff

10. Assures shutdown if nuclear instrumentation fails

11. Assures backup shutdown capability from short period or high power level. Assures shutdown if count rate is too low to provide meaningful startup information. The startup interlock may be bypassed if K_{eff} is $\leq$ 0.9

12. Assures all components of the safety system are installed and operational

13. Assures that any control rod exhibiting excess magnet current will be released and fall to the bottom due to gravity

3.3 Coolant System

3.3.1 Pump Requirements

Applicability: This specification applies to the operation of pumps for both the primary and secondary coolant loops.

Objective: To ensure that both pumps are functioning whenever the reactor is operated above 120 kW.

Specification: The reactor will not be operated above 120 kW unless both the primary and secondary coolant pumps are activated and there is flow in the primary coolant loop.

Bases: Having both pumps operating and flow in the primary loop will ensure there is adequate cooling of the primary coolant so the Safety Limit is not exceeded.

3.3.2 Coolant Level

Applicability: This specification applies to the height of the water in the Reactor Pool above the core.

Objective: To ensure there is adequate primary coolant in the Reactor Pool and sufficient biological shielding above the core.

Specification: The reactor shall not be operated unless there is 20 feet of water in the reactor pool and 15 feet of water above the core.

Bases: With the pool full of water to a level of 20 feet there is adequate primary coolant for natural convection cooling. With 15 feet of water above the core there is sufficient shielding at the licensed power level. Section 7.1.1.4 of the SAR discusses this shielding.

3.3.3 Water Chemistry Requirements

Applicability: This specification applies to the purity of the primary coolant water.

Objective: To minimize corrosion of the cladding on the fuel elements, and to reduce the probability of neutron activation of ions in the water.

Specification:

(1) The conductivity of the pool water shall not exceed the limit of 2.0 μ mho/cm.

(2) The pH of the pool water shall not exceed 8.0.

Bases: Operation in accordance with these specification ensures aluminum corrosion is within acceptable limits, and that the concentration of dissolved impurities that could be activated by neutron irradiation remains within acceptable limits.

3.3.4 Leak, or Loss of Coolant Detection

Applicability: This specification applies to the capability of detecting and preventing the loss of primary coolant.

Objective: To ensure there is adequate primary coolant in the Reactor Pool and sufficient biological shielding above the core when the reactor is operating.

Specification: The pool water level shall be at least 15 feet above the top of the fuel in the core.

Bases: The same system that functions to scram the reactor on low pool level will also be used as the detection system for this specification. Design criteria of the cooling system to prevent large losses of pool water due to siphoning are discussed in Section 3.2.2.1 of the SAR.

3.3.5 Primary and Secondary Coolant Activity Limits

Applicability: This specification applies to the buildup of radioactive materials in the secondary coolant system.

Objective: To ensure there is a level low enough so as not to exceed 10CFR20 limits if coolant is released to the sanitary sewer system.

Specification: The primary and secondary coolant system shall be monitored for the buildup of radioactivity and analyzed at least semiannually for increase in the concentration of radionuclides.

Basis: The basis for this specification is to ensure releases are legal and consistent with the ALARA principal.

3.4 Confinement

Applicability: This specification applies to the capability to provide confinement for the reactor building.

Objective: To prevent the exposure of the public to airborne radioactivity exceeding the limits of 10CFR20 and the ALARA principle.

Specification: The reactor shall not be operated unless the following conditions are met:

(1) Exhaust fan operating

(2) With exceptions for ingress and egress, all exterior doors and windows closed.

Bases: By having the capability to provide confinement for the Reactor Building, exposure of the public to airborne radioactivity may be limited to the extent analyzed in the SAR.

3.5 Ventilation Systems

Applicability: This specification applies to all heating, ventilating, and air conditioning systems that exhaust building air to the outside environment.

Objective: To provide for normal ventilation and the reduction of airborne radioactivity within the reactor building during normal reactor operation and to provide a way to turn off all vent systems quickly in order to isolate the building for emergencies.

Specification:

(1) An exhaust fan with a capacity of at least 1000 cfm shall be operable whenever the reactor is operating.

(2) This fan, as well as all other heating, ventilating, and air conditioning systems shall have the capability to be shut off from a single switch in the control room.

Bases: In the unlikely event of a release of fission products or other airborne radioactivity, the ventilation system will reduce radioactivity inside the reactor building or be able to be isolated. An analysis of fission product release is found in section 8.4.4 of the SAR.

3.6 Radiation Monitoring Systems and Radioactive Effluents

3.6.1 Radiation Monitoring

Applicability: This specification applies to the availability of radiation monitoring equipment which shall be operable during reactor operation.

Objective: To assure that monitoring equipment is available to evaluate radiation levels in restricted and unrestricted areas and to be consistent with ALARA.

Specification:

(1) When the reactor is operating, the building gaseous effluent monitor shall be operating and have a readout and alarm in the control room. It may be used in either the "normal" mode or "sniffer" mode.

(2) When the reactor is operating, the following Area Radiation Monitors (ARMs) shall be operating and have both local and control room readouts and alarms.

a. Pool Top
b. Primary Cooling System
c. Beam Port/Rabbit Area
d. Thermal Column Area

(3) Portable survey instrumentation shall be available whenever the reactor is operating to measure beta-gamma exposure rates and neutron dose rates.

(4) When required monitors are inoperable, portable instruments, surveys, or analyses may be substituted for any of the normally-installed monitors in Section 3.6.1 above for periods of one week or for the duration of a reactor run in cases where the reactor is continuously operated.

Bases:

(1) The gaseous effluent monitor will detect Ar-41 levels in the reactor building. During "normal" mode operation it will sample and monitor air just before it is released from the reactor building. (SAR 6.3.1) During "sniffer" mode of operation it may be used for short periods to monitor in and around experimental facilities to determine local Ar-41

levels.

(2) The ARMs provide a continuing evaluation of the radiation levels within the Reactor Building (SAR 3.7) and provide a warning if levels are higher than anticipated.

(3) The availability of survey meters enables the Reactor Staff to independently confirm radiation levels throughout the building.

(4) In the event of instrument failure short term substitutions will enable the safe continued operation of the Reactor.

3.6.2 Radioactive Effluents

Applicability: This specification applies to the monitoring of radioactive effluents from the facility.

Objectives:

(1) To ensure that liquid radioactive releases are safe and legal.

(2) To ensure that the release of Ar-41 beyond the site boundary does not result in concentrations above the Effluent Concentration limit for unrestricted areas (10CFR20.1302; 10CFR20 Appendix B, Table 2).

(3) To assure that the release of Ar-41 in the restricted area does not result in concentrations above the DAC.

Specifications:

(1) The concentration of radioactive liquids released into the sanitary sewer shall not exceed the limits as specified in 10CFR20.2003.

(2) The concentration of Ar-41 at ground level below the point of release into the unrestricted area shall not exceed the unrestricted area Effluent Concentration limit (10CFR20.1302; 10CFR20 Appendix B, Table 2) when averaged over one year or ten times the Effluent Concentration limit when averaged over one day.

(3) The concentration of Ar-41 in the restricted area shall not exceed the DAC when averaged over a 2000 hour work year.

Bases:

(1) The basis for this specification is found in Section 6.2 of

the Safety Analysis Report.

(2) The basis for this specification is found in Section 6.3 of the Safety Analysis Report.

(3) The basis for this specification is found in Section 6.3 of the Safety Analysis Report and 10CFR20.1003.

3.7 Experiments

3.7.1 Reactivity Limits

Applicability: This specification applies to experiments to be installed in or near the reactor and associated experimental facilities.

Objectives: To prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specification:

(1) The absolute value of the reactivity worth of any single secured experiment shall not exceed 0.7% delta k/k.

(2) The absolute value of the reactivity worth of any single movable experiment shall not exceed 0.4% delta k/k.

(3) The absolute value of the reactivity worth of all movable experiments shall not exceed 0.6% delta k/k.

(4) The absolute value of the reactivity worth of experiments having moving parts shall be designed to have an insertion rate less than 0.05% delta k/k per second.

(5) The absolute value of the reactivity worth of any movable experiment that may be oscillated shall have a reactivity change of less than 0.05% delta k/k.

(6) The total reactivity worth of all experiments shall not be greater than 0.7% delta k/k.

Bases:

(1) The bases for specifications 1, 2, 3, and 6 are found in Section 8.4.3.2 of the SAR which evaluates a step insertion of reactivity from an experiment.

(2) The bases for specifications 4 and 5 allows for certain reactor kinetics experiments to be performed but still limits the rate of change of reactivity insertions to levels that

have been analyzed. Section 8.4.3.2 of the SAR evaluates a step insertion of reactivity from an experiment.

3.7.2 Design and Materials

Specification:

(1) No experiment shall be installed that could shadow the nuclear instrumentation, interfere with the insertion of a control rod, or credibly result in fuel element damage.

(2) All materials to be irradiated in the reactor shall be either corrosion resistant or doubly encapsulated within corrosion resistant containers.

(3) Explosive materials shall not be allowed in experiments, except for neutron radiographic exposures of items performed outside of the core and experimental facilities. The amount of explosive material contained in capsules used for radiographic exposures shall not exceed 5 grains of gunpowder.

Bases:

(1) Specification 1 assures no physical interference with the operation of the reactor detectors, control rods, or physical damage to fuel element will take place.

(2) Limiting corrosive materials in Specification 2, and explosives in Specification 3 reduces the likelihood of damage to reactor components and/or releases of radioactivity resulting from experiment failure.

(3) Limiting explosive materials to neutron radiographic exposures done outside of the core and experimental facilities reduces the likelihood of damage resulting for this experimental failure.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Reactor Core Parameters

4.1.1 Excess Reactivity and Shutdown Margin

Applicability: This specification applies to surveillance requirements for determining the excess reactivity of the reactor core and its shutdown margin.

Objective: To assure that the excess reactivity and shutdown margin limits of the reactor are not exceeded.

Specifications:

(1) Whenever a net change in core configuration, for which the predicted change in reactivity is > 0.2% delta k/k, involving grid position is made, both excess reactivity and shutdown margin shall be determined.

(2) Both shutdown margin and excess reactivity shall be determined annually.

Bases: A determination of excess reactivity is needed to preclude operating without adequate shutdown margin. Moving a component out of the core and returning it to its same location is not a change in the core configuration and does not require a determination of excess reactivity.

4.1.2 Fuel Elements

Applicability: This specification applies to surveillance requirements for determining the physical condition of the reactor fuel.

Objective: To ensure that visible deterioration, corrosion, or other physical changes to the fuel elements are detected in a timely manner.

Specification: All fuel elements, both in-core and out, shall be visually inspected at least once every five years, by inspecting at least one fifth of the elements annually.

Basis: If the water purity is continuously maintained within specified limits, it is projected that chemical corrosion of the fuel clad will proceed slowly. However, faults in the basic materials or fabrication could lead to loss of cladding integrity.

4.2 Reactor Control and Safety Systems

4.2.1 Control Rods

Applicability: This specification applies to the surveillance requirements for the shim safety rods and the regulating rod.

Objective: To assure that all rods are operable.

Specifications:

(1) The reactivity worth of the shim safety rods and regulating rod shall be determined annually and prior to the routine operation of any new core configuration.

(2) Shim safety control rod scram times and drive times and regulating rod drive time shall be determined annually or after maintenance or modification is completed on a mechanism.

(3) The shim safety rods and regulating rod shall be visually inspected annually for indication of corrosion and indication of excessive friction with guides.

Bases: The reactivity worth of the rods is measured to assure the required shutdown margin and reactivity insertion rates are maintained. It also provides a means for determining the reactivity of experiments. Measuring annually will provide corrections for burnup and after core changes assures that altered rod worths will be known prior to continued operations.

The visual inspection of the rods and measurements of control rod scram times and drive times are made to assure the rods are capable of performing properly. Verification of operability after maintenance or modification of the control system will ensure proper reinstallation.

4.2.2 Reactor Safety System

Applicability: This specification applies to the surveillance requirements for the Reactor Safety System.

Objective: To assure the reactor safety system channels will remain operable and prevent safety limits from being exceeded.

Specification:

(1) A channel check of each measuring channel shall be performed daily when the reactor is operating.

(2) A channel test of each measuring channel shall be performed prior to each day's operation or prior to each operation extending more than one day.

(3) A channel calibration of the reactor power level measuring channels shall be made annually. (Linear Level and LOG-N.)

(4) A channel calibration of the Level and Period Safety Channels shall be made annually. Channel tests are done on these before each day's operation.

(5) A channel calibration of the following shall be made annually

a. Core inlet temperature measuring system
b. Adequate pool water level indication
c. Indication of coolant system pumps operating
d. Indication that there is flow in the primary coolant loop

(6) The control room manual scram shall be verified to be operable prior to each day's operation. All other manual scram switches shall be tested annually.

(7) Other scram channels shall be tested/calibrated annually.

(8) Any instrument channel replacement shall be calibrated after installation and before utilization.

(9) Any instrument repair or replacement shall have a channel test prior to reactor operation.

Bases: The daily channel tests and checks will assure that the scram channels are operable. Appropriate annual tests or calibrations will assure that long term functions not tested before daily operation are operable.

4.3 Coolant System

4.3.1 Primary Coolant Water Purity

Applicability: This specification applies to the conductivity of the primary coolant water.

Objective: To assure high quality pool water.

Specification: The conductivity and pH of the pool water shall be measured weekly.

Bases: This assures that changes that might increase the corrosion rate are detected in a timely manner and that the concentrations of impurities that might be made radioactive do not increase significantly.

4.3.2 Coolant System Radioactivity

Applicability: This specification applies to the radioactive material in the primary coolant or secondary coolant.

Objective: To identify radionuclides as potential sources of release to the sanitary sewer system.

Specification: Primary and secondary coolant shall be analyzed for radioactivity quarterly or before release.

Bases: Radionuclide analysis of the pool water or secondary coolant allows for determination of any significant buildup of fission or activation products and helps assure that radioactivity is not permitted to escape to the tertiary system in an uncontrolled manner.

4.4 Confinement

Applicability: This specification applies to the surveillance requirements for building confinement.

Objective: To assure that building confinement capability exists.

Specification: A quarterly test shall be made to assure that the building exhaust fan is operable and all exterior doors and windows have closure capability.

Bases: Quarterly surveillance of this equipment will verify that the confinement of the reactor bay can be maintained if needed.

4.5 Ventilation System

Applicability: This specification applies to the surveillance requirements for the building ventilation system.

Objective: To assure that the ventilation shutoff functions satisfactorily.

Specification: The shutoff switch for all fans and air conditioning systems shall be tested on a quarterly basis.

Bases: This surveillance will assure that the building can be isolated quickly if necessary to prevent uncontrolled escape of air-borne radioactivity to the unrestricted environment.

4.6 Radiation Monitoring Systems and Radioactive Effluents

4.6.1 Effluent Monitor

Applicability: This specification applies to the surveillance requirement of the effluent monitor.

Objective: To assure the effluent monitor is operational and providing accurate effluent readings.

Specification: The effluent monitor shall have a channel calibration annually and a channel test before each days operation.

Bases: The calibration will assure effluent release estimates are accurate and the test will assure the monitor is operable whenever the reactor is operating.

4.6.2 Area Radiation Monitors (ARMs)

Applicability: This specification applies to the area radiation monitoring equipment.

Objective: To assure that radiation monitoring equipment is operable whenever the reactor is operating.

Specification: A channel test of the ARMs shall be completed before each day's operation and a channel calibration shall be completed annually.

Bases: Calibration annually will insure the required reliability and a check on days when the reactor is operated will detect obvious malfunctions in the system.

4.6.3 Portable Survey Instrumentation

Applicability: This specification applies to the portable survey instrumentation available to measure beta-gamma exposure rates and neutron dose rates.

Objective: To assure that radiation survey instrumentation is operable whenever the reactor is operating.

Specification: Beta-gamma and neutron survey meters shall be checked with a source for operability quarterly and shall be calibrated annually.

Bases: Checks with a source will detect obvious detector deficiencies and an annual calibration will assure reliability.

5.0 DESIGN FEATURES

5.1 Site and Facility Description

5.1.1 Facility Location

The reactor and associated equipment is housed in a building at 1298 Kinnear Road, Columbus, Ohio, located on the West Campus of The Ohio State University. The minimum free air volume of the building housing the reactor will be $\geq$ 70,000 ft^3. There is an exhaust fan with dampers providing for control of release of airborne radioactivity.

5.1.2 Controlled and Restricted Area

The fence surrounding the Reactor Building shall describe the controlled area. The restricted area as defined in 10CFR20 shall consist of the Reactor Building.

5.2 Reactor Coolant System

5.2.1 Primary Coolant Loop

Natural convective cooling is the primary means of heat removal from the core. Water enters the core at the bottom and flows upward through the flow channels in the fuel elements.

5.2.2 Secondary and Tertiary Coolant Loops

The secondary coolant loop removes heat from the primary coolant. The secondary coolant (ethylene glycol and water) passes through two separate heat exchangers to remove heat if necessary. Heat is removed from the first by an outside fan-forced dry cooler. City water flow through the secondary side of an additional heat exchanger makes up the tertiary loop. It provides additional cooling for the secondary coolant.

5.3 Reactor Core and Fuel

Up to 30 positions on the core grid plate are available for use as fuel element positions. Control rod fuel elements occupy four of these positions and one is reserved for the Central Irradiation Facility flux trap. Several arrangements for the cold, clean, critical core have been investigated. Approximately eighteen standard fuel elements in addition to the control rod fuel elements are currently required. Partial elements, blank elements, and graphite elements may be utilized in various combinations to achieve the proper K excess.

The reactor fuel is The DOE Standard uranium-silicide (U_3Si_2) with a U-235 enrichment of less than 20%. It is flat plate fuel with a "meat" thickness of 0.020" and aluminum cladding of 0.015". Standard fuel elements have a total of 16 fueled plates and 2 outer pure aluminum plates. The control rod fuel elements have eight of the inner fuel plates removed to allow the control rods to enter. Pure aluminum guide plates are on the inside of this gap. The outer two plates for each control rod assembly are fueled. Partial elements are also available with 25, 40, 50, and 60 percent of the nominal loading of a standard element. These partial fuel elements are prefabricated by the vendor with fixed numbers of plates.

(1) References: NRC NUREG 1313
ANL/RERTR/TM-10
ANL/RERTR/TM-11

5.4 Fuel Storage

The fuel storage pit, located below the floor of the reactor pool and at the end opposite from the core, shall be flooded with water whenever fuel is present and shall be capable of storing a complete core loading. When fully loaded with fuel and filled with water K_{eff} shall not exceed 0.90, and natural convective cooling shall ensure that no fuel temperatures reach a point at which ONB is possible.

5.5 Fuel Handling Tools

All tools designed for or capable of removing fuel from core positions or storage rack positions shall be secured when not in use by a system controlled by the supervisor of reactor operations, or the senior reactor operator on duty.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The Ohio State University Research Reactor is a part of the College of Engineering administered by the Engineering Experiment Station. The organizational structure is shown in Figure 6.1.

6.1.2 Responsibility

The Director of the Engineering Experiment Station (Level 1) is the contact person for communications between the U.S. Nuclear Regulatory Commission and The Ohio State University.

The Director of the Nuclear Reactor Laboratory (Level 2) will have overall responsibility for the management of the facility.

The Associate Director (or Manager of Reactor Operations) (Level 3) shall be responsible for the day-to-day operation and for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and Technical Specifications. During periods when the Associate Director is absent, his responsibilities are delegated to a Senior Reactor Operator (Level 4).

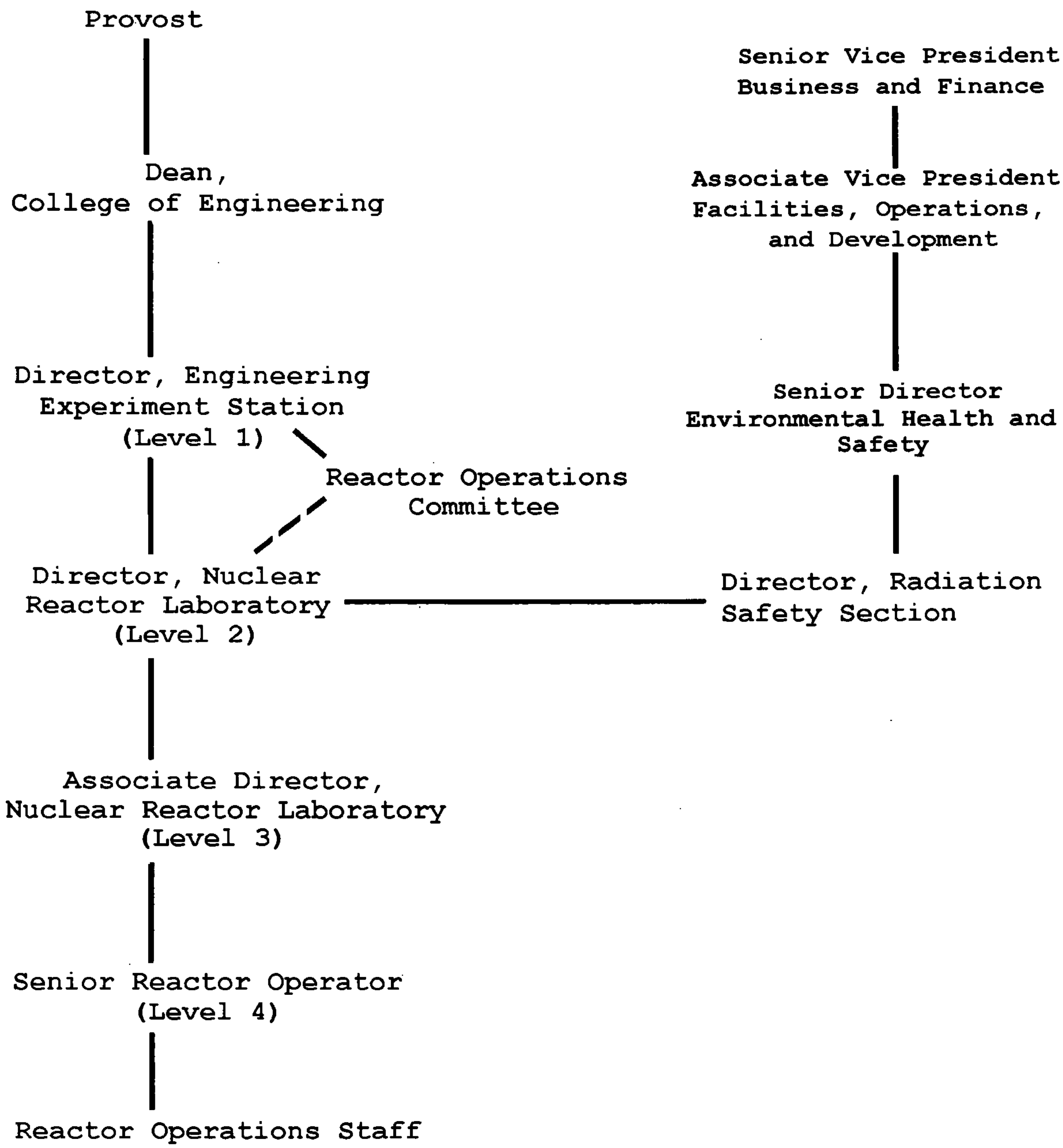
6.1.3 Staffing

(1) The minimum staffing when the reactor is not secured shall be:

(a) A certified reactor operator in the control room

(b) A second designated person present at the facility complex able to carry out prescribed written instructions. Unexpected absence for as long as two hours to accommodate a personal emergency may be acceptable provided immediate action is taken to obtain a replacement.

(c) A Senior Reactor Operator shall be readily available on call. "Readily Available on Call" means an individual who (1) has been specifically designated and the designation known to the operator on duty, (2) keeps the operator on duty informed of where he may be rapidly contacted and the phone number, and (3) is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g. 30 minutes or within a 15-mile radius).



Solid Lines ——— Paths of Direct Responsibility
Dashed Lines - - - - Paths of Information

Figure 6.1: Administrative Organization

(2) A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the reactor operator. The list shall include:

(a) Management personnel

(b) Radiation safety personnel

(c) Other operations personnel

(3) Events requiring the presence at the facility of a senior reactor operator:

(a) Initial startup and approach to power

(b) All fuel or control-rod relocations within the reactor core region

(c) Recovery from an unplanned or unscheduled shutdown. (In these instances, documented verbal concurrence from a senior reactor operator is required.)

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988.

6.2 Review and Audit

There shall be a Reactor Operations Committee (ROC) for independent review of the safety aspects of reactor operations to assure the facility is operating in a manner consistent with public safety and within the terms of the facility license.

A member or members of this committee or another qualified person or persons shall audit safety aspects of reactor operations as described in Section 6.2.4 of this document.

6.2.1 Composition and Qualifications of the ROC

The ROC shall be composed of a minimum of three members, who should collectively represent a broad spectrum of expertise in appropriate fields (i.e. having professional backgrounds in engineering, physical, biological, or medical sciences, as well as knowledge of and interest in applications of nuclear technology and ionizing radiation). Members and alternates shall be appointed by and report to Level 1 management. Individuals may be either from within or outside the operating organization. Qualified and approved alternates may serve in the absence of regular members.

6.2.2 ROC Meetings

ROC functions shall be conducted in accordance with the following:

(1) Meetings shall be held at least once per calendar year and more frequently as circumstances warrant, consistent with effective monitoring of facility activities.

(2) A meeting quorum shall consist of at least half of the membership where the operating staff does not constitute a majority.

(3) The ROC may appoint a subcommittee from within its membership to act on behalf of the full committee on those matters which cannot await the next meeting. The ROC shall review the actions taken by the subcommittee at the next regular meeting.

(4) Meeting minutes shall be distributed to ROC members before the next meeting and shall be reviewed at the next meeting.

6.2.3 Review Function

The ROC shall review the following:

(1) Determination that proposed changes in equipment, systems, tests, experiments, or procedures do not require a license update as described in 10-CFR-50.59.

(2) All new procedures and major revisions thereto having safety significance; proposed changes in reactor facility equipment or systems having safety significance

(3) All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

(4) Proposed changes in technical specifications or license

(5) Violations of technical specifications or license; violations of internal procedures having safety significance.

(6) Operating abnormalities having safety significance.

(7) Reportable occurrences listed in Section 6.6.2 of this document

(8) Audit reports

A written report or minutes of the findings and recommendations of the review group shall be submitted to Level 1 management and ROC members in a timely manner after the review has been completed.

6.2.4 Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with cognizant personnel and observations of operations should be used also as appropriate. In no case shall the individual immediately responsible for an area perform an audit in that area. The following items shall be audited:

(1) Facility operations for conformance to the technical specifications and license, at least once per calendar year (interval between audits not to exceed 15 months)

(2) The requalification program for the operating staff, at least once every other calendar year (interval between audits not to exceed 30 months)

(3) The results of action taken to correct those deficiencies that may occur in the reactor facility, equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval between audits not to exceed 15 months)

(4) The reactor facility emergency plan and implementing procedures at least once every other calendar year (interval between audits not to exceed 30 months)

Deficiencies found that affect reactor safety shall be reported immediately to Level 1 management. A written report of audit findings should be submitted to Level 1 management and the full Reactor Operations Committee within three months of the audit's completion.

6.3 Procedures

6.3.1 Reactor Operating Procedures

Written procedures, reviewed and approved by the Director, or his/her designee, and reviewed by the ROC, shall be in effect and followed. The procedures shall be adequate to assure the safety of the reactor, but should not preclude the use of independent judgement and action should the situation require such. All new procedures and changes to existing procedures shall be documented by the NRL staff and subsequently reviewed by the ROC. At least the following items shall be covered:

(1) Startup, operation, and shutdown of the reactor,

(2) Installation, removal, or movement of fuel elements, control rods, experiments, and experimental facilities,

(3) Actions to be taken to correct specific and foreseen potential malfunctions of systems or components including responses to alarms, suspected cooling system leaks, and abnormal reactivity changes,

(4) Emergency conditions involving potential or actual release of radioactivity including provisions for evacuation, re-entry, recovery, and medical support,

(5) Preventive and corrective maintenance procedures for systems which could have an effect on reactor safety,

(6) Periodic surveillance of reactor instrumentation and safety systems, area monitors, and radiation safety equipment,

(7) Implementation of the Emergency Plan, reactor operator training and requalification requirements, and the security requirements of 10-CFR-73.67.

(8) Personnel radiation protection.

6.3.2 Administrative Procedures

Procedures shall also be written and maintained to assure compliance with Federal regulations, the facility license, and commitments made to the ROC or other advisory or governing bodies. As a minimum, these procedures shall include:

(1) Audits,

(2) Special Nuclear Material accounting,

(3) Operator requalification,

(4) Record keeping, and

(5) Procedure writing and approval.

6.4 Experiment Review and Approval

6.4.1 Definitions of Experiments

Approved experiments are those which have previously been reviewed and approved by the ROC. They shall be documented and may be included as part of the Procedures Manual. New experiments are those which have not previously been reviewed, approved, and performed. Routine tests and maintenance activities are not experiments.

6.4.2 Approved Experiments

All proposed experiments utilizing the reactor shall be evaluated by the experimenter and a licensed Senior Reactor Operator to assure compliance with the provisions of the utilization license, the Technical Specifications, and 10CFR Parts 20 and 50. If, in the judgement of the Senior Reactor Operator, the experiment meets with the above provisions, is an approved experiment, and does not constitute a threat to the integrity of the reactor, it may be approved for performance. When pertinent, the evaluation shall include considerations of:

(1) The reactivity worth of the experiment

(2) The integrity of the experiment, including the effects of changes in temperature, pressure, or chemical composition

(3) Any physical or chemical interaction that could occur with the reactor components, and

(4) Any radiation hazard that may result from the activation of materials or from external beams

6.4.3 New Experiments

Prior to performing an experiment not previously approved for the reactor, the experiment shall be reviewed and approved by the Reactor Operations Committee. Committee review shall consider the following information:

(1) The purpose of the experiment,

(2) The procedure for the performance of the experiment, and

(3) The safety evaluation previously reviewed by a licensed Senior Reactor Operator.

6.5 Required Actions

6.5.1 Action To Be Taken In the Event A Safety Limit Is Exceeded

(1) The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the NRC.

(2) The safety limit violation shall be promptly reported to the Director of the Reactor Laboratory.

(3) The safety limit violation shall be reported by telephone to the NRC within 24 hours.

(4) A safety limit violation report shall be prepared. The report shall describe the following:

a. Applicable circumstances leading to the violation including, when known, the cause and contributing factors,
b. Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public, and
c. Corrective action to be taken to prevent recurrence.

(5) The report shall be reviewed by the Reactor Operations Committee and shall be submitted to the NRC within 14 working days when authorization is sought to resume operation of the reactor.

6.5.2 Action To Be Taken In The Event Of A Reportable Occurrence

A reportable occurrence is any of the following conditions:

(1) Operating with any safety system setting less conservative than stated in these specifications,

(2) Operating in violation of a Limiting Condition for Operation established in Section 3 of these specifications.

(3) Safety system component malfunctions or other component or system malfunctions during reactor operation that could, or threaten to, render the safety system incapable of performing its intended function.

(4) An uncontrolled or unanticipated increase in reactivity in excess of 0.4% delta k/k,

(5) An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor, and

(6) Abnormal and significant degradation in reactor fuel and/or cladding, coolant boundary, or confinement boundary (excluding minor leaks) where applicable that could result in exceeding prescribed radiation exposure limits of personnel and/or the environment.

(7) Any uncontrolled or unauthorized release of radioactivity to the unrestricted environment.

In the event of a reportable occurrence, the following action shall be taken:

(1) The reactor conditions shall be returned to normal, or the reactor shall be shutdown, to correct the occurrence.

(2) The Director of the Reactor Laboratory shall be notified as soon as possible and corrective action shall be taken before resuming the operation involved.

(3) A written report of the occurrence shall be made which shall include an analysis of the cause of the occurrence, the corrective action taken, and the recommendations for measures to preclude or reduce the probability of recurrence. This report shall be submitted to the Director and the Reactor Operations Committee for review and approval.

(4) A report shall be submitted to the Nuclear Regulatory Commission in accordance with Section 6.6.2 of these specifications.

6.6 Reports

Reports shall be made to the Nuclear Regulatory Commission as follows:

6.6.1 Operating Reports

An annual report shall be made by September 30 of each year to the NRC's Document Control Desk (if on paper, the signed original), with a copy to the appropriate Regional Office, in accordance with 10-CFR-50.4, providing the following information:

(1) A narrative summary of operating experience (including experiments performed) and of changes in facility design, performance characteristics, and operating procedures related to reactor safety occurring during the reporting period.

(2) A tabulation showing the energy generated by the reactor (in kilowatt hours) and the number of hours the reactor was in use.

(3) The results of safety-related maintenance and inspections. The reasons for corrective maintenance of safety-related items shall be included.

(4) A table of unscheduled shutdowns and inadvertent scrams, including their reasons and the corrective actions taken.

(5) A summary of the Safety Analyses performed in connection with changes to the facility or procedures, which affect reactor safety, and performance of tests or experiments carried out under the conditions of Section 50.59 of 10CRF50.

(6) A summary of the nature and amount of radioactive gaseous, liquid, and solid effluents released or discharged to the environs beyond the effective control of the licensee as measured or calculated at or prior to the point of such release or discharge.

(7) A summary of radiation exposures received by facility personnel and visitors, including the dates and times of significant exposures.

6.6.2 Special Reports

(1) A telephone or telegraph report of the following shall be submitted as soon as possible, but no later than the next working day, to appropriate Regional Office:

(a) Any accidental offsite release of radioactivity above authorized limits, whether or not the release resulted in property damage, personal injury, or known exposure.

(b) Any exceeding of the safety limit as defined in Section 2.1 of these specifications.

(c) Any reportable occurrences as defined in Section 6.5.2 of these specifications.

(2) A written report shall be submitted within 14 days to the NRC's Document Control Desk (if on paper, the signed original), with a copy to the appropriate Regional Office, in accordance with 10CFR 50.4, of the following:

(a) Any accidental offsite release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury, or known exposure.

(b) Any exceeding of the safety limit as defined in Section 2.1.

(c) Any reportable occurrence as defined in Section 6.5.2 of these specifications.

(3) A written report shall be submitted within 30 days to the NRC's Document Control Desk (if on paper, the signed original), with a copy to the appropriate Regional Office in accordance with 10CFR 50.4, of the following:

(a) Any substantial variance from performance specifications contained in these specifications or in the SAR,

(b) Any significant change in the transient or accident analyses as described in the SAR, and

(c) Changes in personnel serving as Director, Engineering Experiment Station, Reactor Director, or Reactor Associate Director.

(4) A report shall be submitted within nine months after initial criticality of the reactor or within 90 days of completion of the startup test program, whichever is earlier, to the NRC's Document Control Desk (if on paper, the signed original), with a copy to the appropriate Regional Office, upon receipt of a new facility license, an amendment to license authorizing an increase in power level or the installation of a new core of a different fuel element type or design than previously used.

The report shall include the measured values of the operating conditions or characteristics of the reactor under the new conditions, and comparisons with predicted values, including the following:

(a) Total control rod reactivity worth,

(b) Reactivity worth of the single control rod of highest reactivity worth, and

(c) Minimum shutdown margin both at ambient and operating temperatures.

(d) Excess reactivity

(e) Calibration of operating power levels

(f) Radiation leakage outside the biological shielding

(g) Release of radioactive effluents to the unrestricted environment.

6.7 Records

Records or logs of the items listed below shall be kept in a manner convenient for review, and shall be retained for as long as indicated.

6.7.1 Records to be Retained for a Period of at Least Five Years

(1) normal plant operation,

(2) principal maintenance activities,

(3) experiments performed with the reactor,

(4) reportable occurrences,

(5) equipment and component surveillance activity,

(6) facility radiation and contamination surveys,

(7) transfer of radioactive material,

(8) changes to operating procedures, and

(9) minutes of Reactor Operations Committee meetings.

6.7.2 Records to be Retained for at Least One Requalification Cycle

Regarding retraining and requalification of licensed operations personnel, the records of the most recent complete requalification cycle shall be maintained at all times the individual is employed.

6.7.3 Records to be Retained for the Life of the Facility

(1) gaseous and liquid radioactive effluents released to the environment,

(2) fuel inventories and transfers

(3) radiation exposures for all personnel,

(4) changes to reactor systems, components, or equipment that may affect reactor safety,

(5) updated, corrected, and as-built drawings of the facility.

(6) records of significant spills of radioactivity, and status,

(7) annual operating reports provided to the NRC,

(8) copies of NRC inspection reports, and related correspondence