Nuclear Reactor Laboratory
UWNR University of Wisconsin-Madison
1513 University Avenue, Room 1215 ME, Madison, WI 53706-1687, Tel: (608) 262-3392, FAX: (608) 262-8590
email: reactor@engr.wisc.edu, http://reactor.engr.wisc.edu

June 16, 2010 RSC 1048

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Docket 50-156, License R-74
Response to Request for Additional Information
for License Renewal to Facility License No. R-74
University Of Wisconsin Nuclear Reactor
TAC No. ME1585 (Technical RAI)

Dear Sirs:

By letter, dated May 3, 2010, the Commission has requested additional information in order to complete the review for the University of Wisconsin Nuclear Reactor's (UWNR) request to renew facility license number R-74.

Enclosed are the responses to the request for additional information. The responses are provided in the same order as the Commission's requests. The format of the enclosure is to restate the request followed by the response. The original request is counter shaded to aid in the separation between request and response.

I certify under penalty of perjury that the foregoing is true and correct.

Sincerely,

Executed on: 6/16/2010

Robert J. Agaise
Reactor Director

Enclosure

A020
NRR

Responses to License Renewal Request for Additional Information

1. NUREG 1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 1, Section 4.3, Reactor Tank or Pool states that the applicant should present all information about the pool necessary to ensure its integrity and should assess the possibility of uncontrolled leakage of contaminated primary water. Please discuss the pool leakage to the ground experienced at University of Wisconsin Nuclear Reactor (UWNR) in terms of the following:

   a. Please include in the discussion a description of the typical radioactivity content of the pool, the frequency with which the pool water is analyzed for radioactivity, the pathways known or expected to be contributing to the leakage, any trends associated with the leakage, and the physical means with which UWNR can detect small releases from the pool directly to the environment.

Licensee's Response:

Reactor pool water is analyzed monthly for radioactivity. No activity with a half-life greater than 24 hours has ever been detected in pool water samples except for tritium, at typical concentrations of 1.3E-4 µCi/ml which is approximately 10% of the effluent release limit in 10 CFR 20 Appendix B Table 2. Radioactivity with a half-life less than 24 hours is routinely produced from full power operations including Na-24 ($T_{1/2}$=14.95hr, activated from aluminum structure), Mg-27 ($T_{1/2}$=9.45min, activated from aluminum structure), N-16 ($T_{1/2}$=7.13sec, activated from oxygen in water), and O-19 ($T_{1/2}$=26.9sec, activated from oxygen in water). In addition to analyzing routine monthly water samples, non-routine water sample analysis is initiated if other indications of increased radioactivity in the pool water exist, such as abnormally high continuous air monitor activity (which takes its suction directly above the pool water surface) or demineralizer area radiation monitor activity.

A 40 year history of monthly pool water make up is depicted in figure 1 below. Water is routinely added to the pool to make up for losses due to evaporation, sampling, and even thermal contraction of the water. The average make-up volume, excluding those periods of time of known leakage, is approximately 450 gallons per month. The standard deviation in the make-up rate is 150 gallons and is due to variations in seasonal temperatures, humidity, the number of full power operations in a month and even the number of days in the month. Because the original cooling system utilized a cooling tower as the ultimate heat sink, the cyclic trend of summer temperatures can be seen to increase the pool make-up due to increased evaporation as a result of increased pool water temperatures.

During previous pool leaks it has been shown that the cyclic change in pool water temperature during full power runs led to excessive flexing of the aluminum pool liner given the large coefficient of thermal expansion for aluminum. As the aluminum liner stretched and contracted, minor cracks in the pool liner welds were formed, and then re-sealed as the temperature stabilized. Historically the only known leak path has been through cracks in the corner welds around the thermal column, through the pool concrete and into the compacted fill below.

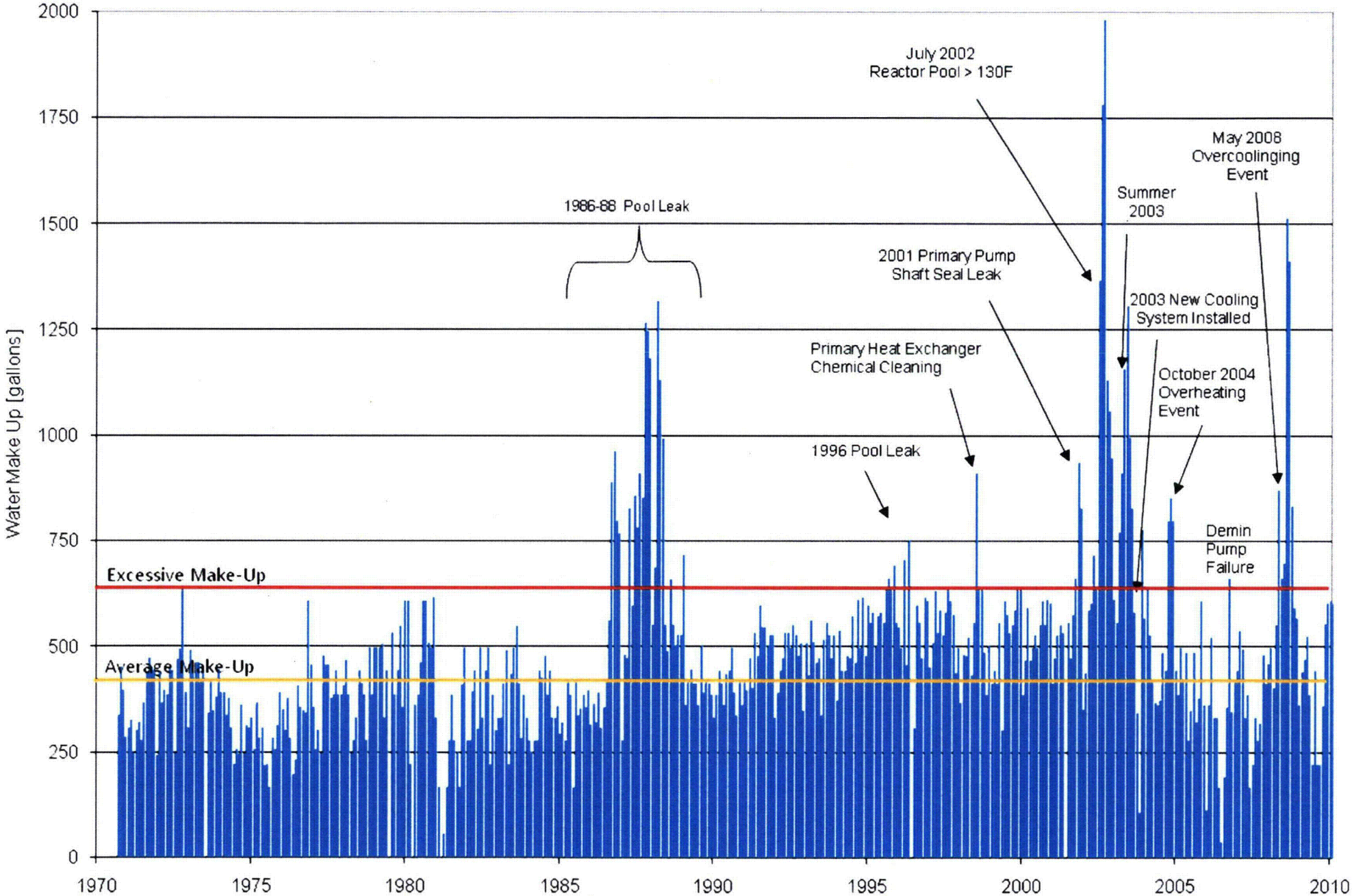


Figure 1, Historical Pool Water Makeup Trend

Following the development of the 2002 leak, the cyclic nature of the increased leakage as a function of increased pool water temperature is obvious by observing the cessation of the leak in December 2002 and the reoccurrence in the summer of 2003. This trend of thermal cycling the aluminum liner and increased leakage is also demonstrated by an over-heating event in 2004 and an over-cooling event in 2008 that occurred at the facility following installation of the new cooling system.

Minor leaks are most easily detected by observing the monthly volume of make-up water added to the pool. Make-up volumes that exceed 600 gallons a month are excessive. This is evident in figure 1, where every leakage event correlates with make-up water volumes in excess of 600 gallons. Evaporation rates at the pool surface are also measured over a 24 hour period and can be correlated with the volume of monthly make-up water; however, these 24 hour evaporative rates have an uncertainty of +/- 5 gallons per day. Any volume of water exceeding normal losses is assumed to be lost to the environment and is reported as a direct environmental release.

b. With this information, please then discuss how current UWNR policies and procedures associated with the pool leakage meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 20.1302(a) to monitor releases to the environment. Additionally, please discuss any plans for physically addressing the known leakage paths, including any leakage rates that will be used as a decision point for taking further action.

Licensee's Response:

The monitoring of environmental releases per 10 CFR 20.1302(a) is satisfied by routine analysis of pool water activity and making the assumption that all water loss exceeding normal losses is a direct environmental release. Isotopes with a half-life shorter than 24 hours are neglected because the leak path is into the compacted fill below the pool concrete, and the SAR section 13.1.9 estimates that water leaking directly into the ground would take approximately 55 years to travel to the nearest city well where it could be exposed to the public.

The root cause of the environmental releases was determined to be thermal cycling of the reactor pool liner as discussed above in response to RAI 1.a. To address the large pool water temperature swings (which would range from 75-125°F) a new cooling system was installed in September 2003 which has enough capacity to maintain pool water temperature at a steady 80°F. Since then there have been no significant increases in pool water temperature resulting in minor pool leaks with the exception of an over-heating event in October 2004 and over-cooling event in May 2008. The over-heating event occurred when an operator failed to turn the cooling system on after performing a normal reactor startup to full power. During the over-cooling event, an operator decreased reactor power from 100% to 5% without turning off the cooling system. The cooling system is designed to reject 1MW and will automatically control a variable frequency drive to maintain a steady temperature; however, the VFD has a minimum frequency of 20Hz and was not able to reduce cooling capacity any further at such a low heat load. Both instances initiated a brief recurrence of the minor pool leak. Following the over-heating event a procedure change was initiated, however following the second event a "System Temperature High/Low" annunciator was added to the console alarm panel in order to warn the operator to turn off the cooling system or increase reactor power. There have been no further pool leaks.

Even though the pool is not currently leaking, make-up water volume, evaporation rate and pool water activity continue to be monitored. It is the facility's goal to permanently eliminate the leak; however, due to financial and ALARA considerations, increased monitoring has been implemented while a cost effective, low dose solution is sought. However, the Reactor Safety Committee (RSC) in 2003 mandated reactor shutdown and an immediate repair if the following action levels are reached.

Action Level 1
Pool water make-up greater than 2200 gallons per month. (The basis for this action level is that 2200 gallons per month is equal to 73 gallons per day which is approximately 80% of the rated still capacity.)

Action Level 2
Pool water activity approaching 80% of 10 CFR 20, Appendix B, Table II water effluent concentration limits for isotopes with half lives greater than 24 hours.

2. Regulation 10 CFR Part 20, Appendix B Table 2 lists the maximum allowable concentration value for Ar-41 at 1.0E-8 Ci/m^3 equivalent to the radionuclide concentrations which, if inhaled or ingested continuously over the course of a year, would produce a total effective dose equivalent of 50 mrem. Section 11.1.1.1.2 of the UWNR Safety Analysis Report (SAR) estimates that in a typical year of operation the maximum concentration to which the public would be exposed would be about 3.31E-9 Ci/m^3 resulting in a maximum dose to the public of 0.6 mrem/yr. The estimated value of 3.31E-9 Ci/m^3 is about one-third lower than the 10 CFR 20 Appendix B Table 2 value and therefore the expected dose would be higher. Please discuss the dose conversion calculations.

Licensee's Response:

In section 11.1.1.1.2, the EPA COMPLY code calculated dose of 0.6 mrem/yr is assuming operation of the ventilation system, whereas the maximum concentration cited above which the public would be exposed, 3.31E-9 Ci/m^3, is assuming the ventilation system is inoperable.

However, following the methodology of comparing to 10 CFR 20 Appendix B Table 2 limits, as calculated in the SAR Appendix A the maximum ground-level concentration with operation of the ventilation system and the maximum hypothetical Ar-41 release rate of 13.3 µCi/sec is 1.25E-9 Ci/m^3. This value is 12.5% of the 10 CFR Part 20 Appendix B Table 2 limit of 1.0E-8 Ci/m^3, and therefore should theoretically result in a dose of 6.3 mrem/yr which is about 10 times the value of 0.6 mrem/yr reported in the SAR. The EPA COMPLY code calculation was performed at the nearest residence 133 meters to the west, and using averaged wind rose data ranging from 4.01 to 5.30 m/s. Wind speed and frequency records were obtained from the International Station Meteorological Climate Summary jointly produced by the National Oceanic and Atmospheric Administration, the United States Air Force, and the United States Navy. The data specifically compiled for Madison was obtained from the National Weather Service and was for the period of record from 1948 to 1995. The calculations in the SAR Appendix A were performed at the location of maximum ground-level concentration and using the minimum reported monthly average wind speed of 3.54 m/s.

However, the methodology for calculating air effluent releases was updated and approved in the LEU Conversion SAR. Using the updated methodology, the maximum ground-level concentration with operation of the ventilation system is 4.78E-10 Ci/m^3. This value is 4.8% of the 10 CFR Part 20 Appendix B Table 2 limit and therefore should theoretically result in a dose of 2.4 mrem/yr. Furthermore, the approved LEU Conversion SAR in section 13.1.4 includes an updated methodology for calculating whole-body dose from immersion in a radioactive cloud. Using this methodology, with an effective dose coefficient for Ar-41 of 0.2405 rem-m^3/Ci-s, the dose from being exposed to the maximum ground-level concentration with operation of the ventilation system is calculated to be 3.6 mrem/yr.

The EPA COMPLY code calculation was also repeated with the receptor at the site boundary rather than the nearest residence. The calculated dose was 67 mrem/yr. However, this assumes the maximum hypothetical Ar-41 release rate of 13.3 µCi/sec with continuous operation year-round resulting in a hypothetical annual release of 420 Ci. The highest recorded annual release is only 3.04 Ci which would result in an EPA COMPLY code calculated dose of only 0.5 mrem/yr.

3. NUREG-1537 Section 12.8 provides guidance for including a brief discussion of security planning in the SAR. The UWNR SAR Section 12.8 states, "The plan will require revision as a result of this Safety Analysis Report, since some figures from the previous Safety Analysis Report are included by reference." In a letter dated March 31, 2009, UWNR updated the Security Plan in accordance with 10 CFR 50.54(p)(2). Please verify that UWNR has no intention to further revise the UWNR Security Plan as a result of this license renewal; or submit a revised Security Plan for approval as a supplement to the License Renewal Application.

Licensee's Response:

The changes to the security plan referenced in the SAR section 12.8 were incorporated into the security plan revision dated March 31, 2009 in accordance with 10 CFR 50.54(p)(2). No further changes to the security plan are required as a result of the license renewal.

4. NUREG-1537 Chapter 13 states that non-power reactors should analyze events that could affect their safe operation or shutdown including evolution of scenarios and evaluating the consequences of postulated events. Section 13.1.3.3 describes the radiation levels in unrestricted areas due to the unshielded reactor core after a postulated large loss-of-coolant accident event and the consequent maximum dose rate to the member of the public in the Mechanical Engineering Building. One of the assumptions used in the model is that members of the public would evacuate in a specific amount of time. Please discuss whether there is a general evacuation procedure, how the members of the public are instructed to follow the plan, how it is insured that all members of the public actually evacuate the building, and whether there is any evacuation exercises practiced periodically.

Licensee's Response:

The general evacuation procedure, UWNR 150 "Reactor Accident, Fission Product Release, or Major Spill of Radioactive Materials," identifies the areas to be evacuated and describes the evacuation alarm system which alerts members of the public to evacuate.

Members of the public are instructed on the plan by use of red-framed evacuation notices which are posted throughout the building along with floor maps indicating evacuation routes. The notice reads:

> "In the event of an accident at the Nuclear Reactor Laboratory, an evacuation signal may be given. This signal will be a slow whoop from horns located throughout the evacuation zone. In addition, backlit panels will give a flashing indication of:
>
> RADIATION ALARM LEAVE THIS AREA
>
> When the horn sounds, this area is to be evacuated and all personnel shall proceed to the north wing of the Mechanical Engineering and then down to the University Avenue main floor lounge/lobby. Alternatively, personnel can proceed to any location farther away from the Reactor Laboratory.

An evacuation drill is performed annually to verify operation of the evacuation alarm system and to provide training for both reactor staff and building occupants. Prior to the evacuation drill the building occupants are informed of the upcoming drill and reminded of the appropriate evacuation routes. During the drill the reactor staff assures people are evacuating and provides additional training to building occupants as needed. Historically evacuation drills have demonstrated that building occupants are evacuated within 5 minutes.

In the event of an evacuation alarm, the site boundary is verified evacuated by the operating staff, with assistance from the University of Wisconsin Police Department (UWPD) as necessary. Access into the evacuated site boundary is controlled by the operating staff and UWPD.

5. Regulation 10 CFR 50.36(b) requires that technical specifications (TS) are derived from the analyses and evaluation included in the SAR, and amendments thereto. Section 14 of the 2008 SAR (Revision 2) includes the TS without incorporating the changes approved and implemented by Amendment 17, the HEU to LEU Conversion Order. Please discuss how you plan to incorporate Amendment 17 changes into the TS as part of the license renewal application.

Licensee's Response:

The changes to the technical specifications related to the LEU conversion, approved as Amendment 17 to the license, have been incorporated into a revision to the proposed technical specifications submitted as part of the 2000 license renewal application. The proposed technical specifications are included as Attachment 1 to this response.

6. NUREG-1537 states that the format and content of the TS follow that of American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1, "The Development of Technical Specifications for Research Reactors." ANSI/ANS-15.1-2007 provides a definition of "reactor secured." Please evaluate UWNR TS 1.3.1.2(a) against the standard definition of "reactor secured."

Licensee's Response:

TS 1.3 definition of "Reactor Secured" 2 is revised to conform to ANSI/ANS-15.1-2007. Furthermore, the multiple sub-sections of definitions TS 1.3.1 through TS 1.3.4 have been combined into a single alphabetized section as TS 1.3 to conform to ANSI/ANS-15.1-2007.

Previously proposed TS 1.3.1 "Reactor Secured" 2:

a. The reactor is shut down,
b. The console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area, and
c. No work is in progress involving in-core fuel handling or refueling operations, maintenance of the reactor or its control mechanisms, or insertion or withdrawal of in-core experiments with a reactivity worth exceeding 0.7 %Δk/k.

Currently proposed TS 1.3 "Reactor Secured" 2:

a. All shim-safety blades are fully inserted,
b. The reactor is shut down,
c. The console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area, and
d. No work is in progress involving in-core fuel handling or refueling operations, maintenance of the reactor or its control mechanisms, or insertion or withdrawal of in-core experiments with a reactivity worth exceeding 0.7 %Δk/k.

See Attachment 1.

7. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007, Section 1.3 provides a definition of Reference Core Condition and Excess Reactivity. UWNR TS 1.3 defines "Cold Critical" at 125°F with no specific condition for Xenon reactivity. Please evaluate the definition of "Cold Critical" against the ANSI/ANS-15.1 standard definition for "Reference Core Condition" and consider developing a definition that can be included in the Limiting Condition for Operation (LCO) for Excess Reactivity and Shutdown Margin in TS 3.1.1 and TS 3.1.2 respectively, or justify the current definition.

Licensee's Response:

TS 1.3 definition of "Cold Critical" is revised and new definitions for "Reference Core Condition" and "Excess Reactivity" are added to conform to ANSI/ANS-15.1-2007. Furthermore, the multiple sub-sections of definitions TS 1.3.1 through TS 1.3.4 have been combined into a single alphabetized section as TS 1.3 to conform to ANSI/ANS-15.1-2007.

Previously proposed TS 1.3.1 "Cold Critical":

The reactor is in the cold critical condition when it is critical with the fuel and bulk water temperatures both below 125°F.

Currently proposed TS 1.3 "Cold Critical":

The reactor is in the cold critical condition when it is critical in the reference core condition.

Newly proposed TS 1.3 "Reference Core Condition":

The reactor is in the reference core condition when the fuel and bulk water temperatures are both below 125°F and the reactivity worth of xenon is negligible (<0.2 %Δk/k).

Newly proposed TS 1.3 "Excess Reactivity":

Excess reactivity is that amount of reactivity that would exist if all control elements were fully withdrawn from the core in the cold critical condition.

Furthermore, TS 3.1.2.3 is revised to be consistent with the currently proposed definitions.

Previously proposed TS 3.1.2.3:

The reactor in the cold condition without xenon.

Currently proposed TS 3.1.2.3:

The reactor in the reference core condition.

See Attachment 1.

8. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007, Section 1.3 provides definitions for key terminology utilized in TSs. Please include a definition of "Confinement, Excess Reactivity, Operating, Scram Time, and Shall, Should and May" in UWNR TS 1.3 or provide a basis for not defining these terms.

Licensee's Response:

TS 1.3 is revised to include the definitions requested. Furthermore, the multiple sub-sections of definitions TS 1.3.1 through TS 1.3.4 have been combined into a single alphabetized section as TS 1.3 to conform to ANSI/ANS-15.1-2007.

Newly proposed TS 1.3 "Confinement":

Confinement is an enclosure of the facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled pathways. This is room 1215 of the Mechanical Engineering Building.

Newly proposed TS 1.3 "Excess Reactivity" (copied from response to RAI No. 7):

Excess reactivity is that amount of reactivity that would exist if all control elements were fully withdrawn from the core in the cold critical condition.

Newly proposed TS 1.3 "Operating":

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

Newly proposed TS 1.3 "Scram Time":

The time from the initiation of a scram signal to the time that the slowest scrammable control element reaches its fully inserted position.

Newly proposed TS 1.3 "Shall, Should, and May":

The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission, neither a requirement nor a recommendation.

See Attachment 1.

9. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007, Section 6.7.2 discusses special reporting requirements for operational occurrences. UWNR TS 1.3(6) lists a reportable occurrence as "Abnormal and significant degradation in reactor fuel or cladding which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both." ANSI/ANS-15.1 Section 6.7.2 does not include the stipulation for exceeding exposure limits. Please consider removing the stipulation and including all instances of abnormal and significant fuel or cladding damage (excluding minor leaks) as reportable occurrences or provide a basis for including this stipulation.

Licensee's Response:

TS 1.3 definition of "Reportable Occurrence" was based on the standard ANSI/ANS-15.1-1990 approved at the time of license renewal application submittal which included the stipulation for exceeding exposure limits during degradation in reactor fuel or cladding. This definition is revised to conform to the current standard ANSI/ANS-15.1-2007.

Previously proposed TS 1.3.1 "Reportable Occurance" 6:
Abnormal and significant degradation in reactor fuel or cladding which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

Currently proposed TS 1.3 "Reportable Occurance" 6:
Abnormal and significant degradation in reactor fuel or cladding, or coolant boundary (excluding minor leaks) where applicable.

See Attachment 1.

10. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. UWNR TS 3.1 states an overall objective for TS in Section 3.1. TS 3.1.1, 3.1.2 and 3.1.3 do not have specific objectives. TS 3.1.4 and TS 3.1.6 have specific objectives. Please clarify which objectives are applicable to which TS in Section 3.1.

Licensee's Response:

TS 3.1 is revised to eliminate generic applicability and objective statements for TS 3.1 and include specific statements for TS 3.1.1 through 3.1.6.

Previously proposed TS 3.1 (Reactor Core Parameters) Applicability (now deleted):
These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods. They apply for all modes of operation.

Previously proposed TS 3.1 (Reactor Core Parameters) Objective (now deleted):
The objective is to assure that the reactor can be shut down at all times and to assure that the fuel temperature safety limit will not be exceeded.

Newly proposed TS 3.1.1 (Excess Reactivity) Applicability:
This specification applies to the reactivity condition of the reactor and the reactivity worths of control elements and applies for all modes of operation.

Newly proposed TS 3.1.1 (Excess Reactivity) Objective:
The objective is to assure that the reactor can be shut down at all times.

Newly proposed TS 3.1.2 (Shutdown Margin) Applicability:
This specification applies to the reactivity condition of the reactor and the reactivity worths of control elements and applies for all modes of operation.

Newly proposed TS 3.1.2 (Shutdown Margin) Objective:
The objective is to assure that the reactor can be shut down at all times.

Newly proposed TS 3.1.3 (Pulse Limits) Applicability:
This specification applies to the reactivity worth of the transient rod and pulse interlocks based on power level. It applies to pulse mode operation.

Newly proposed TS 3.1.3 (Pulse Limits) Objective:
The objective is to assure that the fuel temperature safety limit will not be exceeded.

Applicability and Objective statements for TS 3.1.4 (Core Configurations) and 3.1.6 (Fuel Parameters) remain unchanged. See Attachment 1.

11. NUREG-1537 Part 1, Chapter 3, Section 3.1 Design Criteria for Structures, System and Components states that one of the design criteria to be considered should be the redundancy of reactor protective and safety features, so that any single failure will not prevent safe reactor shutdown. The Fuel Temperature channel specification in TS 3.2.4 and TS 3.2.8 discusses an exception related to the availability of replacement instrumented fuel elements (IFE). Specifically, the Setpoint and Function statement for the Fuel Temperature Safety Channel specification allows continued operation of an operational core in the absence of an operable IFE if the Linear Power Level scram setpoints are reduced to 110 percent full power.

Please further discuss the basis and need for this exception, with regard to the reduction in redundancy and defense-in-depth with no Fuel Temperature safety channel and Linear Power Scram channels reduced to 110 percent full power. Additionally discuss how the exception, if utilized, would meet the requirements of TS 2.2 for measuring the fuel temperature at the IFE.

Licensee's Response:

As part of the refueling to the TRIGA FLIP core from 1973-1980, four IFEs were acquired, each containing three thermocouples. By the time the license renewal application was being prepared, all three thermocouples in two of the IFEs had burned out as well as one thermocouple on the remaining installed IFE, with one un-irradiated spare. A second thermocouple on the remaining installed IFE was unreliable and suspected bad, leaving only a single reliable thermocouple measuring fuel temperature in the core. Furthermore, production of TRIGA FLIP fuel had ceased and therefore acquiring replacement IFEs, even if funding were available, was not possible. This was the basis for requesting the exemption allowing continued operation with no operable thermocouples for operational cores only, if the power level scram set-point was reduced to 110% to compensate. Therefore, new or experimental core configurations for which fuel temperatures could not be measured would not be allowed but operational cores for which fuel temperatures had already been measured would still be permitted (see TS 1.3 "Operational Core").

However, since submitting the original license renewal application in 2000, the core has been converted to TRIGA LEU 30/20 fuel. Two IFEs are currently installed in the core, with two un-irradiated spares on hand. After completing all startup testing including fuel temperature mapping, no thermocouples have burned out or are giving any indication of failing. Therefore it is no longer anticipated that all available IFE thermocouples would burn out in the expected operational life of the reactor core, so the exemption allowing continued operation with no operable thermocouples has been removed from the currently proposed technical specifications.

However, the TRIGA fuel manufacturer has announced its intention to shutdown the TRIGA fuel production facility in the near future, and if in the future all available IFE thermocouples burn out and replacements cannot be acquired, a separate license amendment will be submitted at that time requesting an exemption similar to that previously proposed.

12. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 3.2(1) specifies that the operability for control elements be defined using Scram Times. Please discuss whether UWNR TS 3.2.1 is consistent with the standard guidance.

Licensee's Response:

The operability requirement for control element scram times is already addressed in TS 3.2.2, however TS 3.2.1 is revised for clarification.

Previously proposed TS 3.2.1:

> The reactor shall not be operated unless at least three control elements are functioning and scrammable.

Currently proposed TS 3.2.1:

> The reactor shall not be operated unless at least three control elements are operable and scrammable in accordance with TS 3.2.2.

See Attachment 1.

13. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 3.2(8) includes guidance on establishing permitted bypassing of channels for the purposes of calibrations and maintenance. Please discuss whether UWNR TS 3.2 should include acceptable conditions for bypassing channels for this purpose.

Licensee's Response:

Bypassing channels is already addressed in TS 3.2.7. The numbering of sub-sections under TS 3.2 was based on the standard ANSI/ANS-15.1-1990 approved at the time of license renewal application submittal. The current standard ANSI/ANS-15.1-2007 added a sub-section 3.2.5, "Minimum Channels Needed for Reactor Operation", which changed the numbering of following sub-sections. The specification in ANSI/ANS-15.1-2007 3.2.5 is already addressed in TS 3.2.8, "Control Systems and Instrumentation Required for Operation." Furthermore, bypassing of scram channels is not authorized according to the proposed TS 3.2.7. See Attachment 1.

14. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 3.3 provides guidance for leak or loss of coolant detection. TS 3.3 specification 5 states pool level alarms if level drops "one foot or less." Please consider revising to state "one foot or more."

Licensee's Response:

TS 3.3 is revised for clarification. Furthermore, previously proposed specifications for pool design features are already addressed in TS 5.2 and are deleted from TS 3.3. The specification for pool water temperature was added with the LEU Conversion Amendment No. 17 (see RAI No. 5).

Previously proposed TS 3.3:

1. The reactor core shall be cooled by natural convective water flow.
2. The pool water inlet pipe to the demineralizer shall not extend more than 15 feet into the top of the reactor pool when fuel is in the core. The outlet pipe from the demineralizer shall be equipped with a check valve and siphon breaker to prevent inadvertent draining of the pool.
3. Diffuser and other auxiliary systems pumps shall be located no more than 15 feet below the top of the reactor pool.
4. All other piping and pneumatic tube systems entering the pool shall have siphon breakers and valves or blind flanges which will prevent draining more than 15 feet of water from the pool.
5. A pool level alarm shall indicate loss of coolant if the pool level drops one foot or less below normal level.
6. The reactor shall not be operated if the conductivity of the pool water exceeds 5 microohms/cm (<0.2 MegOhm-cm) when averaged over a period of one week.
7. The reactor shall not be operated if the radioactivity of pool water exceeds the limits of 10 CFR Part 20 Appendix B Table 3 for radioisotopes with half-lives >24 hours.

Currently proposed TS 3.3:

1. A pool level alarm shall indicate loss of coolant if the pool level drops one foot or more below normal level.
2. A pool water temperature alarm shall indicate if water temperature reaches 130°F.
3. The reactor shall not be operated if the conductivity of the pool water exceeds 5 microohms/cm (<0.2 MegOhm-cm) when averaged over a period of one week.
4. The reactor shall not be operated if the radioactivity of pool water exceeds the limits of 10 CFR Part 20 Appendix B Table 3 for radioisotopes with half-lives >24 hours.

See Attachment 1.

15. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 3.4 provides guidance for establishing a LCO on Confinement. Specifically, the standard includes discussion of conditions that require Confinement and equipment required for Confinement to be established. Please discuss whether UWNR TS 3.4 is consistent with the standard guidance.

Licensee's Response:

TS 3.4 is revised to conform to ANSI/ANS-15.1-2007. The previous specifications for minimum free volume and minimum exhaust height are removed because they are already addressed in TS 5.1.

Previously proposed TS 3.4:

1. The reactor shall be housed in a closed room designed to restrict leakage. The minimum free volume shall be 2,000 cubic meters.
2. All air or other gas exhausted from the reactor room and associated experimental facilities shall be released to the environment a minimum of 26.5 meters above ground level.

Currently proposed TS 3.4:

3.4.1 Operations that require confinement:
Confinement is required for reactor operation or any movement of irradiated fuel or fueled experiments with significant fission product inventory outside of containers, systems, or storage areas.

3.4.2: Equipment to achieve confinement:
To achieve confinement, the ventilation system must be operating in accordance with TS 3.5.

See Attachment 1.

16. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 3.5 provides guidance to ensure the minimum number of ventilation fans for normal operation is defined. Please evaluate TS 3.5 in relation to this standard to include an explanation of the basis and need for the exemption allowing two days of reactor operation with the ventilation system inoperable.

Licensee's Response:

TS 3.5 is revised to eliminate the exemption allowing two days of reactor operation with the ventilation system inoperable. In addition, the specification is revised to clarify when the ventilation system is operating to conform to standard ANSI/ANS-15.1-2007.

Previously proposed TS 3.5:

The reactor shall not be operated unless the laboratory ventilation system is in operation, except for periods of time not to exceed two days, to permit repairs of the system.

Currently proposed TS 3.5:

The reactor shall not be operated unless the ventilation system is operating. The ventilation system is considered operating if:

1. One stack exhaust fan is operating,
2. Exhaust flow-rate is at least 9600 scfm,
3. Exhaust filter total pressure drop is less than 2.5 inches of water column.

See Attachment 1.

17. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 3.7.1 provides a time limit for alternate methods of radiation monitoring with a channel out of service. Please review the asterisk condition in TS 3.7.1 and consider adding a time limit consistent with the standard guidance or provide a basis for not including a time limit.

Licensee's Response:

TS 3.7.1 is revised to conform to standard ANSI/ANS-15.1-2007.

Previously proposed TS 3.7.1 Table note:

*For periods of time for maintenance to the radiation monitoring channels, the intent of this specification will be satisfied if they are replaced with portable gamma sensitive instruments having their own alarms or which shall be kept under visual observation.

Currently proposed TS 3.7.1 Table note:

*For periods of time, not to exceed 1 week, for maintenance to the radiation monitoring channels, the intent of this specification will be satisfied if they are replaced with portable gamma sensitive instruments having their own alarms or which shall be kept under visual observation.

See Attachment 1.

18. Regulation 10 CFR 50.36(c)2(ii)B, Criterion 2 states that a LCO must be established for process variable that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. In addition ANSI/ANS-15.1-2007 section 3.8.1(2) guidance includes establishing a LCO for the sum of the absolute values of the reactivity worth of all experiments. TS 3.8.1.1 and TS 3.8.1.2 establish limits for secured and non-secured single experiments respectively, but there is no reactivity limit specification for the sum of all experiments in the reactor. Please discuss whether there is a need for a LCO regarding the sum of all experiments in the reactor to ensure that the total maximum reactivity worth limit is not exceeded.

Licensee's Response:

TS 3.8.1 is revised to conform to standard ANSI/ANS-15.1-2007.

Previously proposed TS 3.8.1:

1. The reactivity worth of any single non-secured experiment shall not exceed 0.7 %Δk/k.
2. The reactivity worth of any single secured experiment shall not exceed 1.4 %Δk/k.

Currently proposed TS 3.8.1:

1. The sum of the absolute values of the reactivity worths of all non-secured experiments does not exceed 0.7 %Δk/k.
2. The reactivity worth of any single secured experiment does not exceed 1.4 %Δk/k.
3. The sum of the absolute values of the reactivity worths of all experiments, both secured and non-secured, does not exceed the maximum excess reactivity specified in TS 3.1.1.

See Attachment 1.

19. Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors," Section C.1.c.(3) states that the "materials of construction and fabrication and assembly techniques should be so specified and used that assurance is provided that no stress failure can occur at stresses twice those anticipated in the manipulation and conduct of the experiment or twice those which could occur as a result of unintended but credible changes of, or within, the experiment." UWNR TS 3.8.2 allows explosive materials in quantities less than 25 mg to be irradiated in the reactor in a container "provided that the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the container." Please discuss how UWNR will ensure a safety factor of two in TS 3.8.2.

Licensee's Response:

TS 3.8.2 is revised to conform to Regulatory Guide 2.2.

Previously proposed TS 3.8.2.1:

Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 milligrams may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the container.

Currently proposed TS 3.8.2.1:

> Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 milligrams may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half the design pressure of the container.

See Attachment 1.

20. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 4.2 (5) provides guidance related to annual calibration requirements for Scram channels. UWNR TS 4.2(5) discusses "channel tests" but does not mention calibration requirements. Please discuss whether UWNR TS 4.2(5) is consistent with the standard guidance.

Licensee's Response:

TS 4.2.5 is revised to conform to ANSI/ANS-15.1-2007.

Newly proposed TS 4.2.5.c:

> A channel calibration of items (1) and (2) in Table 3.2.4 shall be performed annually.

Note that item (1) is the fuel temperature channel and item (2) is the linear power level channels. See Attachment 1.

21. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 4.2 (9) provides guidance related to annual surveillances of reactor safety system interlocks. Please discuss whether UWNR TS 4.2 is consistent with the standard guidance.

Licensee's Response:

TS 4.2 was based on the standard ANSI/ANS-15.1-1990 approved at the time of license renewal application submittal, which did not include sub-section 4.2(9). Surveillance of interlocks is already addressed in TS 4.2.5.a. These interlock surveillances are conducted in accordance with the procedure UWNR 110 "Daily Reactor Pre-Startup Checklist" which includes the pulse mode control interlock. While TS 4.2(9) specifies that the pulse mode control interlock is required to be operable in pulse mode only, TS 4.1.3 requires semiannual pulsing of the reactor. Therefore the pulse mode control interlock must be verified operable at least semi-annually. See Attachment 1.

22. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 4.4 provides guidance for functional testing of Confinement. Please discuss the basis for determining UWNR TS 4.4 is not required.

Licensee's Response:

The only requirement to achieve confinement according to TS 3.4 is operation of the ventilation system. TS 4.4 is revised to conform to ANSI/ANS-15.1-2007.

Previously proposed TS 4.4:
No surveillances are required.

Currently proposed TS 4.4:
The ventilation system shall be verified operable in accordance with TS 4.5 quarterly.

See Attachment 1.

23. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 4.5 provides guidance for surveillances on ventilation system filter efficiency measurements and an operability check of any emergency exhaust systems. Please discuss whether the UWNR TS 4.5 is consistent with the standard guidance.

Licensee's Response:

ANSI/ANS-15.1-2007 states specific systems from section 3 specifications will establish the minimum performance level, and the companion section 4 surveillance specifications will prescribe the frequency and scope of surveillance to demonstrate such performance. TS 3.5 states the minimum requirements of one exhaust fan operating, exhaust flow-rate, and exhaust filter pressure drop. TS 4.5 states that the ventilation system shall be verified operable quarterly. Even though TS 3.5 has been revised to identify the minimum performance levels, TS 4.5 is still consistent with ANSI/ANS-15.1-2007.

24. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 4.7.2 (2) provides guidance for surveillance requirements covering environmental monitoring, specifically "sampling of soil, vegetation, or water in the vicinity of the facility." Please discuss whether UWNR TS 4.7.2 is consistent with the standard guidance.

Licensee's Response:

TS 4.7.2 was based on the standard ANSI/ANS-15.1-1990 approved at the time of license renewal application submittal, which was not specific to address off-site monitoring. However, environmental monitoring as described in ANSI/ANS-15.1-2007 is satisfied. All liquid releases to the sewer and air effluents to the stack are monitored and verified to be below effluent limits. Pool water is routinely analyzed for radioactivity according to TS 4.3, and any water make-up beyond normal evaporative losses is monitored and verified to be below effluent limits. Environmental TLD badges are also located at various positions off-site to monitor exposure.

25. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 5.1 provides guidance for including a "description of the site and of the facility including location and exclusion or restricted areas." Please discuss whether UWNR TS 5.1 is consistent with the standard guidance.

Licensee's Response:

TS 5.1 is revised to conform to ANSI/ANS-15.1-2007. Two new specifications are added to define the operations and site boundaries.

Previously proposed TS 5.1.1:

The reactor shall be housed in a closed room designed to restrict leakage. The minimum free volume shall be 2,000 cubic meters.

Previously proposed TS 5.1.2:

All air or other gas exhausted from the reactor room and the Beam Port and Thermal Column Ventilation System shall be released to the environment a minimum of 26.5 meters above ground level.

Newly proposed TS 5.1.3:

The operations boundary shall be the Reactor Laboratory, room 1215 of the Mechanical Engineering Building. The operations boundary shall be a restricted area.

Newly proposed TS 5.1.4:

The site boundary shall be that portion of the center and east wings of the Mechanical Engineering Building south of the north lobby, plus the portion of Engineering Drive south of the designated areas of the building. The site boundary may be a non-restricted area.

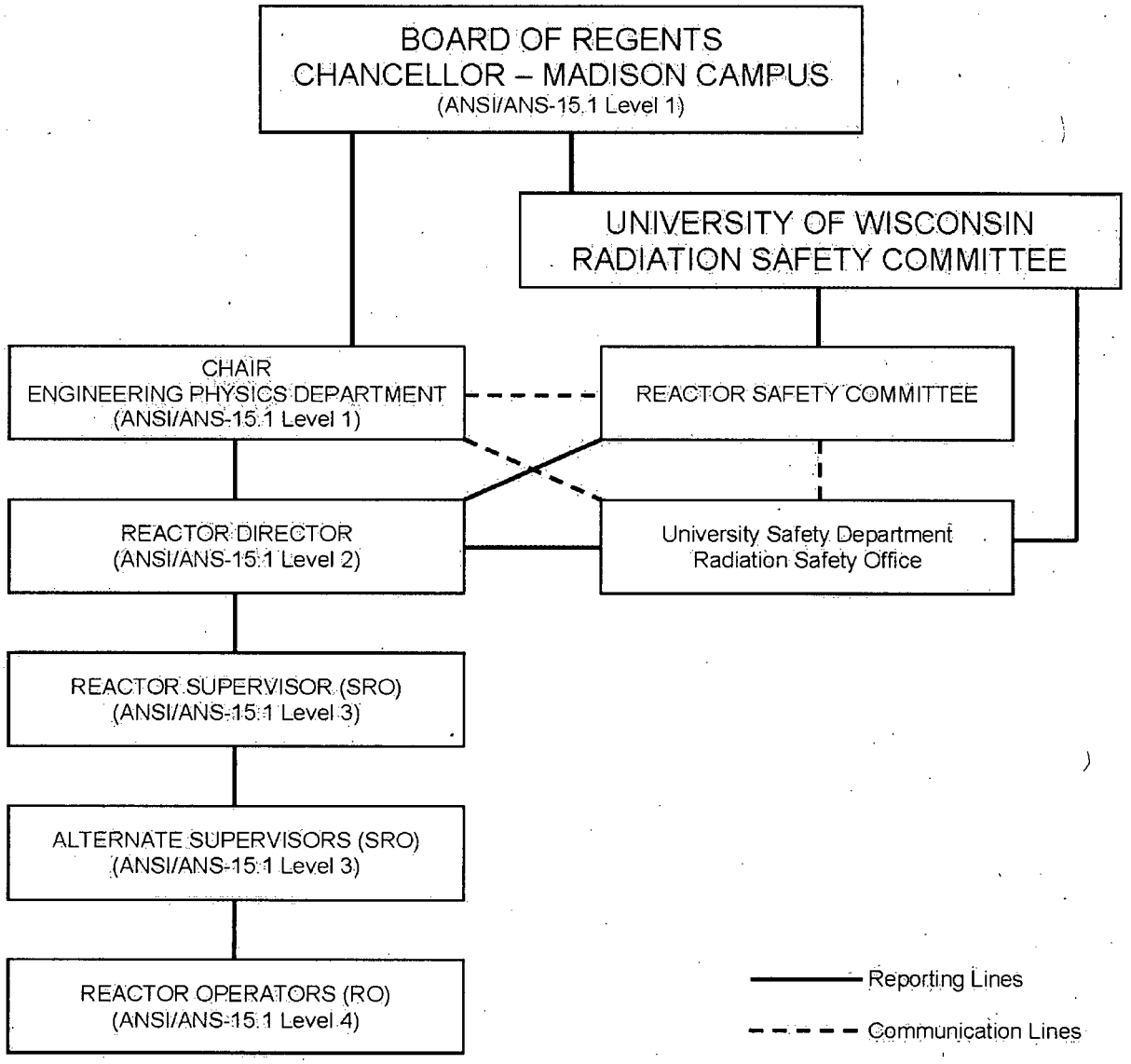
See Attachment 1.

26. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 Section 6.1.1 provides guidance related to organizational structure. UWNR TS 6.1.1 states that the Radiation Safety Office reports to "both committees as well as to the Reactor Director." However, the organizational chart has the Radiation Safety Office reporting to a level above the ANSI/ANS-15.4 Level 1 position. Please clarify the reporting structure for the Radiation Safety Office at UWNR.

Licensee's Response:

The University Radiation Safety Committee (URSC) is the body that authorizes use of ionizing radiation on campus and is responsible for the oversight of all radioactive material on campus. This authority is delegated to the URSC by the Chancellor who receives authority from the Board of Regents, the ultimate holder of the reactor license. The Radiation Safety Office is delegated by the URSC to implement, on a day-to-day basis, the authority of the URSC. The Radiation Safety Officer (RSO) is in charge of the Radiation Safety Office and is a member of the URSC. The Reactor Safety Committee (RSC) is a standing sub-committee of the URSC and the RSC chair is a member of the URSC. The RSO is also a member of the RSC. Therefore the Radiation Safety Office operates under the authority of the URSC,and reports to the URSC, department chair, RSC and Reactor Director on review and audit functions at the facility. Each of these organizations has the authority to stop work at the reactor laboratory.

Certain level 1 responsibilities of the Board of Regents of the University of Wisconsin, the holder of the reactor license, are delegated to the Engineering Physics Department chair. The organizational chart is revised for clarification below. See also Attachment 1.


BOARD OF REGENTS
CHANCELLOR – MADISON CAMPUS
(ANSI/ANS-15.1 Level 1)
UNIVERSITY OF WISCONSIN
RADIATION SAFETY COMMITTEE
CHAIR
ENGINEERING PHYSICS DEPARTMENT
(ANSI/ANS-15.1 Level 1)
REACTOR SAFETY COMMITTEE
REACTOR DIRECTOR
(ANSI/ANS-15.1 Level 2)
University Safety Department
Radiation Safety Office
REACTOR SUPERVISOR (SRO)
(ANSI/ANS-15.1 Level 3)
ALTERNATE SUPERVISORS (SRO)
(ANSI/ANS-15.1 Level 3)
REACTOR OPERATORS (RO)
(ANSI/ANS-15.1 Level 4)
Reporting Lines
Communication Lines

27. NUREG 1537, Part 1, Section 12.1.3, Staffing, states that the applicant should discuss the availability of senior reactor operators during routine operation and should meet, at a minimum, the requirements of 10 CFR 50.54(m)(1). ANSI/ANS-15.1-2007, Section 6.1.3(1) specifies the minimum staffing when the reactor is not secured and specifically calls for a Senior Reactor Operator to be readily available on call. The standard further specifies "on call" as within 30 minutes or 15 miles of the facility. Please discuss whether UWNR TS 6.1.3(1) is consistent with the standard guidance.

Licensee's Response:

TS 6.1.3.1.c is revised to conform to ANSI/ANS-15.1-2007.

Previously proposed TS 6.1.3.1.c:

A designated senior reactor operator shall be readily available at the facility or on call.

Currently proposed TS 6.1.3.1.c:

A designated senior reactor operator shall be readily available at the facility or on call. On call means the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility.

See Attachment 1.

28. NUREG-1537, Chapter 12.1 Conduct of Operations, Organizations states that the organization shall meet the non-power reactor standard ANSI/ANS 15.1-2007. ANSI/ANS 15.1.6.2.2, Review and Audit Groups, Quorums states not less than one-half of the membership where operating staff does not constitute a majority is considered as quorums. TS 6.2.1 does not specify the composition of the Safety Review Committee (SRC), just the number of members. Please discuss the composition of the SRC and the number of members from operating staff.

Licensee's Response:

The Reactor Safety Committee (RSC) charter precludes reactor operating staff from being members of the committee.

29. NUREG-1537 states that the format and content of the TS follow that of ANSI/ANS 15.1. ANSI/ANS-15.1-2007 provides guidance for audit functions and states "In no case shall the individual immediately responsible for the area perform an audit in that area." TS 6.2.4 states "reactor staff shall perform annual reviews of the requalification program, the security plan, and the emergency plan and its implementing procedures." Please discuss the audit program at UWNR and how it meets the criteria of an independent review as outlined by the standard guidance.

Licensee's Response:

It is recognized that TS 6.2.4 does not specify an independent review of the areas cited above. TS 6.2.4 is revised to clearly specify the requested independent review.

Previously proposed TS 6.2.4:

A Health Physicist from the University of Wisconsin Safety Department Radiation Safety Office shall represent the University Radiation Safety Committee and shall conduct an inspection of the facility at least monthly to assure compliance with the regulations of 10 CFR Part 20. The services and inspection function of the Health Physics Office shall also be available to the Reactor Safety Committee, and will extend the scope of the audit to cover license, technical specification, and procedure adherence.

The committee shall audit operation and operational records of the facility. If the committee chooses to use the staff of the Health Physics organization for the audit function, the reports of audit results will be distributed to the committee and included as an agenda item for committee meetings.

Reactor staff shall perform annual reviews of the requalification program, the security plan, and the emergency plan and its implementing procedures.

Currently proposed TS 6.2.4:

A Health Physicist from the University of Wisconsin Safety Department Radiation Safety Office shall represent the University Radiation Safety Committee and shall conduct an inspection of the facility at least monthly to assure compliance with the regulations of 10 CFR Part 20. The services and inspection function of the Health Physics Office shall also be available to the Reactor Safety Committee, and will extend the scope of the audit to cover license, technical specification, and procedure adherence.

The committee shall audit operation and operational records of the facility, requalification program, security plan, and emergency plan and its implementing procedures. If the committee chooses to use the staff of the Health Physics organization for the audit function, the reports of audit results will be distributed to the committee and included as an agenda item for committee meetings.

See Attachment 1.

30. Regulation 10 CFR 20.1101(a) requires that each licensee develop, document, and implement a radiation protection program. NUREG-1537, Chapter 12.1 Conduct of Operations, Organizations states that the organization shall meet the non-power reactor standard ANSI/ANS 15.1-2007. ANSI/ANS-15.1 Section 6.3 states that the facility shall implement a radiation protection program. Please discuss whether TS 6.3 meets the criteria in 10 CFR 20.1101(a) and ANSI/ANS-15.1 Section 6.3.

Licensee's Response:

TS 6.3 specifies that the reactor laboratory shall meet the requirements of the University Radiation Safety Regulations. The University Radiation Safety Regulations meet the requirements of 10 CFR 20.1101(a) and ANSI/ANS-15.11-1993 (R2004) as specified in ANSI/ANS-15.1-2007.

31. UWNR TS 6.8.2 specifies one cycle as retention time for operator qualification or requalification. Regulation 10 CFR 55.59(c)(5) requires that it be a training cycle. Please specify what cycle is being discussed.

Licensee's Response:

TS 6.8.2 is revised to confirm to ANSI/ANS-15.1-2007 and 10 CFR 55.59(c)(5).

Previously proposed TS 6.8.2, Records to be Retained for at Least One Cycle:
Operator qualification and re-qualification records.

Currently proposed TS 6.8.2, Records to be Retained for at Least One Certification Cycle:
Record of retraining and requalification of certified operations personnel shall be maintained at all times the individual is employed or until the certification is renewed. For the purposes of this technical specification, a certification is an NRC issued operator license.

See Attachment 1.

Following discussions with the NRC on 6/2/2010, two additional requests were made regarding the proposed technical specifications beyond those already submitted as RAIs.

First, it was observed that each technical specification in chapter 5 was missing the basis. Therefore the applicability, objective, and basis for each technical specification in chapter 5 is included in the currently proposed technical specifications. Each applicability, objective, and basis is based on existing wording in the currently approved technical specifications (Amendment No. 17 to the license). See Attachment 1.

Second, it was noted in the technical specifications chapter 6 that there was no requirement for retaining certain records as required by 10 CFR 50.36(c). Specifically, notification of an exceeded safety limit, notification that an automatic safety system did not function as required, and notification of a failure to meet limiting conditions for operation. These three records were added to TS 6.8.3, Records to be Retained for the Lifetime of the Reactor Facility.

Newly proposed TS 6.8.3 items 5-7:

5. Notification that safety limit was exceeded.
6. Notification that automatic safety system did not function as required.
7. Notification of failure to meet limiting conditions for operation.

See Attachment 1.

Attachment 1

Newly Proposed UWNR Technical Specifications

UWNR TECHNICAL SPECIFICATIONS

TS 1 INTRODUCTION

TS 1.1 Scope

This section of the SAR for license renewal of the University of Wisconsin Nuclear Reactor constitutes the proposed Technical Specifications for that facility as required by 10 CFR 50.36. This document includes the basis to support the selection and significance of the specifications. Each basis is included for information purposes only, and is not part of the Technical Specifications in that it does not constitute requirements or limitations which the licensee must meet in order to meet the specifications. Dimensions, measurements, and other numerical values given in these specifications may differ slightly from actual values due to construction and manufacturing tolerances or normal degree of accuracy or of instrument readings.

These specifications are re-formatted from the technical specifications in force in 1999 as amended in 2008 for the conversion to LEU fuel (Amendment No. 17)[1]. Changes reflect only changes required by name changes or to include information not in the original technical specifications. In addition, certain additions required by NUREG-1537 are included. All substantive changes were denoted by redlining in the 2000 license renewal submittal Rev 0, but currently only changes since the last revision are redlined (indicated by vertical line in margin).

TS 1.2 Format

Content and section numbering is in accordance with section 1.2.2 of ANSI/ANS 15.1.

TS 1.3 Definitions

The terms used herein are explicitly defined to ensure uniform interpretation of the Technical Specifications.

CHANNEL CALIBRATION:

A channel calibration consists of comparing a measured value from the measuring channel with a corresponding known value of the parameter so that the measuring channel output can be adjusted to respond with acceptable accuracy to known values of the measured variable.

CHANNEL CHECK:

A channel check is a qualitative verification of acceptable performance by observation of channel behavior.

CHANNEL TEST:

A channel test is the introduction of a signal into the channel to verify that it is operable.

COLD CRITICAL:

The reactor is in the cold critical condition when it is critical in the reference core condition.

CONFINEMENT:

Confinement is an enclosure of the facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled pathways. This is room 1215 of the Mechanical Engineering Building.

CORE LATTICE POSITION:

A core lattice position is that region in the core (approximately 3" by 3") over a grid hole. It may be occupied by a fuel bundle, an experiment or experimental facility, or a reflector element.

EXCESS REACTIVITY:

Excess reactivity is that amount of reactivity that would exist if all control elements were fully withdrawn from the core in the cold critical condition.

EXPERIMENT:

Experiment shall mean:

1. Any apparatus, device or material which is not a normal part of the reactor core or experimental facility, or

2. Any activity external to the biological shield using a beam of radiation emanating from the reactor core, or

3. Any operation designed to measure reactor parameters or characteristics, or any activity external to the biological shield using a beam of radiation emanating from the reactor core:

Classification of experiments shall be:

1. Routine experiments. Routine experiments are those which have previously been performed at the facility.

2. Modified routine experiments. Modified routine experiments are those which have not been performed previously but are similar to the routine experiments in that the hazards are neither greater nor significantly different than those for the corresponding routine experiments.

3. Special experiments. Special experiments are those which are not routine or modified routine experiments.

EXPERIMENT SAFETY SYSTEMS:

Experiment safety systems are those systems, including their associated input circuits, which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information which requires manual protective action to be initiated.

EXPERIMENTAL FACILITIES:

Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems and any other in-pool irradiation facilities.

FUEL BUNDLE:

A fuel bundle is a cluster of three or four fuel elements secured in a square array by a top handle and a bottom grid plate adaptor.

FUEL ELEMENT:

A fuel element is a single TRIGA fuel rod of LEU 30/20 type.

INSTRUMENTED ELEMENT:

An instrumented element is a special fuel element in which thermocouples are embedded for the purpose of measuring fuel temperatures during reactor operation.

IRRADIATION:

Irradiation shall mean the insertion of any device or material that is not a normal part of the core or experimental facilities into an experimental facility so that the device or material is exposed to a significant amount of the radiation available in that irradiation facility.

LEU 30/20 CORE:

A LEU 30/20 core is an arrangement of TRIGA LEU 30/20 fuel in the reactor grid plate.

LIMITING SAFETY SYSTEM SETTINGS:

Limiting safety system settings are settings for automatic protective devices related to those variables having significant safety functions.

MEASURED VALUE:

The measured value is the magnitude of that variable as it appears on the output of a measuring channel.

MEASURING CHANNEL:

A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output device which are connected for the purpose of measuring the value of a variable.

NON-SECURED EXPERIMENT

Any experiment not meeting the criteria of a secured experiment.

OPERABLE:

A system, device, or component shall be considered operable when it is capable of performing its intended functions in a normal manner.

OPERATING:

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

OPERATIONAL CORE:

An operational core is an LEU 30/20 core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable pulse reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

PULSE MODE (PU)

Pulse mode operation shall mean any operation of the reactor with the mode selector switch in the pulse position.

REACTOR OPERATION:

Reactor operation is any condition wherein the reactor is not secured.

REACTOR SAFETY SYSTEMS:

Reactor safety systems are those systems, including their associated input circuits, which are designed to initiate a reactor scram for the primary purpose of protecting the reactor or to provide information which requires manual protective action to be initiated.

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 upon optimum available conditions of moderation and reflection, or

2. The following conditions exist:

   a. All shim-safety blades are fully inserted,

   b. The reactor is shut down,

   c. The console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area, and

   d. No work is in progress involving in-core fuel handling or refueling operations, maintenance of the reactor or its control mechanisms, or insertion or withdrawal of in-core experiments with a reactivity worth exceeding 0.7% $\Delta K/K$.

REACTOR SHUTDOWN:

The reactor is shut down when the reactor is subcritical by at least 0.7% $\Delta k/k$ of reactivity.

REFERENCE CORE CONDITION:

The reactor is in the reference core condition when the fuel and bulk water temperatures are both below 125°F and the reactivity worth of xenon is negligible (<0.2 $\%\Delta k/k$).

REGULATING BLADE:

The regulating blade is a low worth control blade that need not have scram capability. Its position may be varied manually or by the servo-controller.

REPORTABLE OCCURRENCE:

A reportable occurrence is any of the following that occur during reactor operation:

1. Operation with any safety system setting less conservative than specified in the technical specifications;

2. Operation in violation of a Limiting Condition for Operation listed in Section 3;

3. Operation with a required reactor or experiment safety system component in an inoperative or failed condition which could render the system incapable of performing its intended safety function;

4. Any unanticipated or uncontrolled change in reactivity greater than 0.7% $\Delta K/K$, excluding reactor trips from a known cause;

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 a condition which could result in operation of the reactor outside the specified safety limits; and

6. Abnormal and significant degradation in reactor fuel or cladding, or coolant boundary (excluding minor leaks) where applicable.

SAFETY CHANNEL:

A safety channel is a measuring channel in the reactor safety system.

SAFETY LIMITS:

Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

SCRAM TIME:

The time from the initiation of a scram signal to the time that the slowest scrammable control element reaches its fully inserted position.

SECURED EXPERIMENT:

A secured experiment shall mean any experiment that is held firmly in place by a mechanical device or by gravity, that is not readily removable from the reactor, and that requires one of the following actions to permit removal:

1. Removal of mechanical fasteners

2. Use of underwater handling tools

3. Moving of shield blocks or beam port containers.

SHALL, SHOULD, AND MAY:

The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission, neither a requirement nor a recommendation.

SHIM-SAFETY BLADE:

A shim-safety blade is a control blade having an electric motor drive and scram capabilities. Its position may be varied manually or by the servo-controller.

SHUTDOWN MARGIN:

Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating condition (assuming the most reactive scrammable control element and any non-scrammable control elements remain full out), and the reactor will remain subcritical without further operator action.

SQUARE WAVE MODE (SW)

Square wave mode operation shall mean any operation of the reactor with the mode selector switch in the square wave position.

STEADY STATE MODE (SS)

Steady state mode operation shall mean operation of the reactor with the mode selector switch in the manual or automatic positions.

TRANSIENT ROD:

The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. Its position may be varied manually or by the servo-controller. It may have a voided or solid aluminum follower.

TS 2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

TS 2.1 Safety Limits

Applicability

This specification applies to fuel element temperature and steady-state reactor power level.

Objective

The objective is to define the maximum fuel element temperature and reactor power level that can be permitted with confidence that no fuel element cladding failure will result.

Specification

1. The temperature in a TRIGA LEU 30/20 fuel element shall not exceed 1150°C under any conditions of operation.

2. The reactor steady-state power level shall not exceed 1500 kW under any conditions of operation.

Basis

A loss of integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding if the fuel temperature exceeds the safety limit. The pressure is caused by air, fission product gases, and hydrogen from dissociation of the fuel moderator. The magnitude of this pressure is determined by the fuel moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The safety limit for the TRIGA LEU 30/20 fuel element is based on data which indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided the temperature does not exceed 1150°C and the fuel cladding is water cooled[2].

It has been shown by experience that operation of TRIGA reactors at a power level of 1500 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1500kW. The LEU Conversion SAR[1] section 4.7.8 shows by analysis that a power level of 1500 kW corresponds to a peak fuel temperature of 665°C. Thus a Safety Limit on power level of 1500 kW provides an ample margin of safety for operation.

TS 2.2 Limiting Safety System Settings

Applicability

This specification applies to the scram setting which prevents the safety limit from being reached.

Objective

The objective is to prevent the safety limits from being reached.

Specification

1. The limiting safety system setting for fuel temperature shall be 400°C as measured in an instrumented fuel element with a pin power peaking factor between 0.87 and 1.16, or 500°C as measured in an instrumented fuel element with a pin power peaking factor of at least 1.16.

2. The limiting safety system setting for reactor power level shall be 1.25 MW.

Basis

1. The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. Analyses performed in section 4.7.6 of the LEU Conversion Analysis[1] show that with the IFE in a core location with a pin power peaking factor of at least 0.87, the maximum fuel temperature would be no greater than 678°C if the IFE thermocouple reaches 400°C providing a margin of 472°C to the safety limit. The same analyses also show that with the IFE in a core location with a pin power peaking factor of at least 1.16, the maximum fuel temperature would be no greater than 678°C if the IFE thermocouple reaches 500°C providing a margin of 472°C to the safety limit.

   In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting off the "tail" of the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

2. Analysis in section 4.7 of the Conversion Analysis SAR[1] shows that at 1.3 MW, the peak fuel temperature in the core will be approximately 604°C so that the limiting power level setting provides an ample safety margin to accommodate errors in power level measurement and anticipated operational transients.

TS 3 LIMITING CONDITIONS FOR OPERATION

TS 3.1 Reactor Core Parameters

TS 3.1.1 Excess Reactivity

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control elements and applies for all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times.

Specification

The excess reactivity shall not exceed 5.6% $\Delta k/k$.

Basis

As shown in chapter 4 of the SAR, this amount of excess reactivity will provide the capability to operate the reactor at full power with experiments in place. The primary limitation providing reactivity safety, however, is the shutdown margin requirement discussed in the next specification.

TS 3.1.2 Shutdown Margin

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control elements and applies for all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times.

Specification

The reactor shall not be operated unless the shutdown margin provided by control rods shall be greater than 0.2% $\Delta k/k$ with:

1. the highest worth non-secured experiment in its most reactive state,
2. the highest worth control element and the regulating blade (if not scrammable) fully withdrawn, and
3. the reactor in the reference core condition.

Basis

The value of the shutdown margin assures that the reactor can be shut down from any operating condition even if the highest worth control element should remain in the fully withdrawn position. If the regulating blade is not scrammable, its worth is not used in determining the shutdown reactivity.

TS 3.1.3 Pulse Limits

Applicability

This specification applies to the reactivity worth of the transient rod and pulse interlocks based on power level. It applies to pulse mode operation.

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded.

Specification

1. The reactivity to be inserted for pulse operation shall be determined and mechanically limited such that the reactivity insertion will not exceed 1.4% $\Delta k/k$.

2. Pulses shall not be initiated at power levels exceeding 1 kilowatt.

Basis

1. The LEU Conversion SAR[1] section 4.7.10 shows by analysis that a 1.4 $\%\Delta k/k$ limitation on pulse reactivity will result in a maximum fuel temperature of 790°C. This leaves a margin to the 1150°C Safety Limit of 360°C, and a margin of 40°C to the 830°C operational limit recommended by General Atomics, "Pulsing Temperature Limit for TRIGA LEU Fuel," GA-C26017 (December, 2007).

2. The temperature rise from pulse initiation is in addition to the temperature in the fuel at the time the pulse is initiated. Limiting the initial power level to 1 kW assures that excessive temperatures will not be reached.

TS 3.1.4 Core Configurations

Applicability

This specification applies to the configuration of fuel and in-core experiments.

Objective

The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specification

1. The core shall be an arrangement of TRIGA LEU 30/20 uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

2. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

3. The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water.

4. Fuel shall not be inserted or removed from the core unless the reactor is subcritical by more than the calculated worth of the most reactive fuel assembly.

5. Control elements shall not be manually removed from the core unless the core has been shown to be subcritical with all control elements in the full out position.

Basis

1. TRIGA cores have been in use for years and their characteristics are well documented. LEU cores including 30/20 fuel have also been operated at General Atomics and Texas A&M and their successful operational characteristics are available. In addition, the analysis performed at Wisconsin indicates that the LEU 30/20 core will safely satisfy all operational requirements. See chapters 4 and 13 of the LEU Conversion Analysis SAR[1].

2. Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high power density.

3. The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

4-5. Manual manipulation of core components will be allowed only when a single manipulation can not result in inadvertent criticality.

TS 3.1.5 Reactivity Coefficients

Does not apply to TRIGA reactors.

TS 3.1.6 Fuel Parameters

Applicability

This specification applies to the dimensional and structural integrity of the fuel elements.

Objective

The objective is to assure that the reactor will not be operated with defective fuel elements installed.

Specification

The reactor shall not be operated with damaged fuel except for purposes of identifying the damaged fuel. A fuel element shall be considered damaged and must be removed from the core if:

1. In measuring the transverse bend, its sagitta[3] exceeds 0.125 inch over the length of the cladding;

2. In measuring the elongation, the length of the cladding exceeds its original length by 0.125 inch;

3. A clad defect exists as indicated by detection of release of fission products.

4. The fuel has not been visually inspected within the previous 15 months.

5. The burnup of uranium-235 in the UzrH fuel matrix exceeds 50 percent of the initial concentration.[4,5]

Basis

The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel element bowing that could result in touching has occurred without deleterious effects. The elongation limit has been specified to assure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to assure adequate coolant flow through the top grid plate.

TS 3.2 Reactor Control and Safety Systems

TS 3.2.1 Operable Control Rods

Applicability

This specification applies to the number of operable control elements that must exist in order to operate the reactor.

Objective

The objective of this requirement is to insure that the reactor may be shut down from any condition of operation.

Specification

The reactor shall not be operated unless at least three control elements are operable and scrammable in accordance with TS 3.2.2.

Basis

In most cores the limits on shutdown margin actually dictate the number of operable control elements required. Non-pulsing cores do not require presence of a transient control rod if the shutdown margin requirements are met by the control blades.

TS 3.2.2 Reactivity Insertion Rates (Scram time)

Applicability

This specification applies to the time required for the scrammable control elements to be fully inserted from the instant that a safety channel variable reaches the Safety System Setting.

Objective

The objective is to achieve prompt shutdown of the reactor to prevent fuel damage.

Specification

The scram time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control element reaches its fully inserted position shall not exceed 2 seconds.

Basis

This specification assures that the reactor will be promptly shut down when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to assure the safety of the reactor.

TS 3.2.3 Other Pulsed Operation Limitations

Limitations other than those on core configuration and pulsed reactivity insertion limits are not required on this reactor.

TS 3.2.4 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

The objective is to specify the minimum number of reactor safety channels that must be operable for safe operation.

Specification

The reactor shall not be operated unless the safety channels described in Table 3.2.4 are operable.

Table 3.2.4 Reactor Safety System Channels

Safety Channel	Setpoint and Function	Number operable in specified mode		
		SS	SW	PU
Fuel Temperature	Scram if fuel temperature exceeds 400°C in the fuel temperature safety channel for an instrumented fuel element pin power peaking factor of 0.87-1.16, or 500°C for an instrumented fuel element pin power peaking factor greater than 1.16.	1	1	1
Linear Power Level	Scram if power > 125% full power	2	2	-
Manual Scram	Manually initiated scram	1	1	1
Preset Timer	Transient rod scram 15 seconds or less after pulse	-	-	1
Reactor water level	Scram if < 19 feet above top of core	1	1	1
High Voltage Monitor	Scram on loss of high voltage to neutron and gamma ray power level instrument detectors	1	1	1
Reactor water temperature	Scram if > 130°F	1	1	1

Basis

The fuel temperature and power scrams provide protection to ensure that the reactor is shut down before the safety limit on fuel temperature is reached.

The manual scram allows the operator a means of rapid shutdown in the event of unsafe or abnormal conditions.

The preset timer assures reduction of reactor power to a low level after a pulse.

The reactor pool water level scram assures shutdown of the reactor in the event of a serious leak in the primary system or pool.

The high voltage monitor prevents operation of the reactor with other systems inoperable due to failure of the detector high voltage supplies.

The reactor pool water temperature scram prevents operation of the reactor in an un-analyzed condition.

TS 3.2.5 Interlocks

Applicability

This section applies to the interlocks which inhibit or prevent control element withdrawal or reactor startup.

Objective

The objective of these interlocks is to prevent operation under unanalyzed or imprudent conditions.

Specification

The reactor shall not be operated in the indicated modes unless the interlocks in Table 3.2.5 are operable.

Table 3.2.5 Interlocks

Channel	Setpoint and Function	Number operable in specified mode		
		SS	SW	PU
Log Count Rate	Prevent control element withdrawal when neutron count rate < 2 per second	1	1	1
Transient Rod Control	Prevent application of air to fire transient rod unless drive is at IN limit.	1	0	0
Log N Power Level	Prevent application of air to fire transient rod when power level is above 1 kW and transient rod is not full in.	1	1	1
Pulse Mode Control	Prevents withdrawal of control blades while in pulse mode.	0	0	1

Basis

The Log count rate interlock does not allow control element withdrawal unless the neutron count rate is high enough to assure proper instrument response during reactor startup.

The Transient Rod Control interlock prevents inadvertent addition of excessive amounts or reactivity in steady-state modes.

The Log N interlock prevents firing of the transient rod at power levels above 1.0 kW if the transient rod drive is not in the full down position. This effectively prevents inadvertent pulses which might cause fuel temperature to exceed the safety limit on fuel temperature.

The pulse mode control blade withdrawal interlock prevents reactivity addition in pulse mode other than by firing the transient rod.

TS 3.2.6 Backup Shutdown Mechanisms

Backup shutdown mechanisms are not required for this reactor.

TS 3.2.7 Bypassing Channels

Applicability

This specification applies to the interlocks in Table 3.2.5.

Objective

The objective is to indicate the conditions in which an interlock may be bypassed.

Specification

The Log Count Rate interlock in Table 3.2.5 may be bypassed:

During fuel loading in order to allow control element withdrawal necessary for the fuel loading procedure or

When Log Power Level and Linear Power Level channels are on-scale.

Basis

During early stages of fuel loading the count-rate on the source range channel will be below the interlock setpoint. The bypass allows control element movements necessary for loading fuel with control elements partially withdrawn and for performing inverse multiplication determinations of control element worth and core reactivity status. Once the other power indications are available the startup count rate channel is no longer required, so the interlock no longer serves any purpose.

TS 3.2.8 Control Systems and Instrumentation Required for Operation

Applicability

This specification applies to the information which must be available to the reactor operator during reactor operation.

Objective

The objective is to require that sufficient information is available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless measuring channels listed in Table 3.2.8 are operable.

Table 3.2.8 Instrumentation and Controls Required for Operation

Channel	Function	Number operable in specified mode		
		SS	SW	PU
Fuel Temperature	Input for fuel temperature scram.	1	1	1
Linear Power Level	Input for safety system power level scram	2	2	0
Log Power Level	Wide range power indication, permissive for initiation of Pulse Mode	1	1	0
Startup Log Count Rate	Wide range power indication, permissive for control element withdrawal	1*	1*	0
Pulsing Power Level	Pulse power level indication	0	0	1

* Required during startup only until the Log Power Level and Linear Power Level channels are on-scale. See TS 3.2.7.

Basis

Fuel temperature indicated at the control console gives continuous information on the process variable which has a specified safety limit.

The power level monitors assure that reactor power level is adequately monitored for all modes of operation.

TS 3.3 Reactor Pool Water Systems

Applicability

This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding and to prevent damage to in-pool components by corrosion.

Specification

1. A pool level alarm shall indicate loss of coolant if the pool level drops one foot or more below normal level.

2. A pool water temperature alarm shall indicate if water temperature reaches 130°F.

3. The reactor shall not be operated if the conductivity of the pool water exceeds 5 micromhos/cm (<0.2 MegOhm-cm) when averaged over a period of one week.

4. The reactor shall not be operated if the radioactivity of pool water exceeds the limits of 10 CFR Part 20 Appendix B Table 3 for radioisotopes with half-lives >24 hours.

Basis

1. Loss of coolant alarm, after one foot of loss, requires corrective action. This alarm is observed in the reactor control room and outside the reactor building.

2. The thermal-hydraulic analysis in the SAR assumes a pool water temperature of 130°F. If the temperature exceeds 130°F then the alarm will prevent continued operation in an un-analyzed condition.

3. The conductivity limit assures that materials within the pool will not be degraded and that the radioactivity of the pool water will be minimized.

4. Analyses in section 12.2.9 of the Safety Analysis Report show that limiting the activity to this level will not result in any person being exposed to concentrations greater than those permitted by 10 CFR Part 20.

TS 3.4 Confinement

Applicability

These specifications apply to the room housing the reactor and the ventilation system controlling that room.

Objective

The objective is to provide restrictions on release of airborne radioactive materials to the environs.

TS 3.4.1 Operations That Require Confinement

Specification

Confinement is required for reactor operation or any movement of irradiated fuel or fueled experiments with significant fission product inventory outside of containers, systems, or storage areas.

Basis

During reactor operation or movement of irradiated fuel there is the potential for a release of radioactivity from the fuel clad. Confinement will limit the consequences to the public from such a release.

TS 3.4.2 Equipment to Achieve Confinement

Specification

To achieve confinement, the ventilation system must be operating in accordance with TS 3.5.

Basis

With the ventilation system operating any potential fission product release will be swept out of the lab and exhausted from a monitored and elevated release point to limit the consequences to the public from such a release.

TS 3.5 Ventilation Systems

Applicability

This specification applies to the operation of the reactor laboratory ventilation system.

Objective

The objective is to assure that the ventilation system is in operation to mitigate the consequences of the possible release of radioactive materials resulting from reactor operation.

Specification

The reactor shall not be operated unless the ventilation system is operating. The ventilation system is considered operating if:

1. One stack exhaust fan is operating,
2. Exhaust flow-rate is at least 9600 scfm,
3. Exhaust filter total pressure drop is less than 2.5 inches of water column.

Basis

It is shown in the SAR Chapter 11 that Argon-41 release at zero stack height results in concentrations less than the concentrations permitted for non-restricted areas. However, the calculations indicate that operation of the ventilation system significantly reduces the concentration to which the public would be exposed. Exposures in the event of a fuel element cladding leak are also calculated based on non-operation of the ventilation system, but are significantly reduced with the ventilation system running. Therefore, operation of the reactor with the ventilation system running will minimize exposure to the public from routine operation and hypothetical accidents.

TS 3.6 Emergency Power

Emergency power systems are not required for this facility.

TS 3.7 Radiation Monitoring Systems and Effluents

TS 3.7.1 Monitoring Systems

Applicability

This specification applies to the radiation monitoring information which must be available to the reactor operator during reactor operation.

Objective

The objective is to assure that sufficient radiation monitoring information is available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the radiation monitoring channels listed in Table 3.7.1 are operable.

Table 3.7.1 Radiation Monitoring Systems

Radiation Monitoring Channels*	Function	Number
Area Radiation Monitor	Monitor radiation levels within the reactor room	3
Exhaust Gas Radiation Monitor	Monitor radiation levels in the exhaust air stack	1
Exhaust Particulate Radiation Monitor	Monitor radiation levels in the exhaust air stack	1
Environmental Radiation Monitors	TLD dosimeters evaluated on a quarterly basis record exposure in area surrounding the stack	4

* For periods of time, not to exceed 1 week, for maintenance to the radiation monitoring channels, the intent of this specification will be satisfied if they are replaced with portable gamma sensitive instruments having their own alarms or which shall be kept under visual observation.

Basis

The radiation monitors provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings. The environmental monitors are placed in areas immediately surrounding the reactor laboratory to record actual dose that would have been delivered to a person continually present in the area.

TS 3.7.2 Effluent (Argon-41) Discharge Limit

Applicability

This specification applies to the concentration of Ar-41 which may be discharged from the facility.

Objective

The objective is to assure that the health and safety of the public are not endangered by the discharge of Ar-41.

Specification

The concentration of Ar-41 in the effluent gas from the facility, as diluted by atmospheric air in the lee of the facility as a result of the turbulent wake effect, shall not exceed 1×10^{-8} μCi/ml averaged over one year.

Basis

10 CFR Part 20 Appendix B, Table II specifies a limit of 1×10^{-8} μCi/ml for Ar-41. Chapter 13 of the LEU Conversion SAR calculates that the maximum ground-level concentration from operation of the ventilation system is 3.6×10^{-5} μCi/ml per Ci/sec discharged. A ground-level concentration of 1×10^{-8} μCi/ml would result from a discharge rate of 278 μCi/sec; the resulting stack exhaust concentration would be 6.14×10^{-5} μCi/ml. Chapter 11 of the SAR calculates that the maximum hypothetical Ar-41 release rate is only 13.3 μCi/ml.

TS 3.8 Experiments

Applicability

These specifications apply to experiments installed in the reactor and its experimental facilities.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

TS 3.8.1 Reactivity Limits

Specification

The reactor shall not be operated unless the following conditions governing experiments exist:

1. The sum of the absolute values of the reactivity worths of all non-secured experiments does not exceed 0.7 %Δk/k.

2. The reactivity worth of any single secured experiment does not exceed 1.4 %Δk/k.

3. The sum of the absolute values of the reactivity worths of all experiments, both secured and non-secured, does not exceed the maximum excess reactivity specified in TS 3.1.1.

Basis

1. This specification is intended to provide assurance that the worth of non-secured experiments will be limited to a value such that the safety limit will not be exceeded if the positive worth of all experiments were to be suddenly inserted (SAR Chapter 13).

2. The maximum worth of a single experiment is limited so that its removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit. Since experiments of such worth must be fastened in place, its removal from the reactor operating at full power would result in a relatively slow power increase such that the reactor protective systems would act to prevent high power levels from being attained. SAR accident analysis includes a sudden addition of 1.4 %$\Delta k/k$ from firing the transient control rod while operating at the power level scram point, a more severe transient than that which could result from removal of a fixed experiment with the same reactivity worth.

3. This specification provides assurance that by removing all installed experiments the maximum excess reactivity specified in TS 3.1.1 would not be exceeded.

TS 3.8.2 Materials

Specification

1. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 milligrams may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half the design pressure of the container.

2. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, or (3) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limit of Appendix B of 10 CFR Part 20.

3. In calculations pursuant to 2 above, the following assumptions shall be used:

   a. If the effluent from an experimental facility exhausts through a holdup tank which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.

   b. If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these particles can escape.

   c. For materials whose boiling point is above 130°F and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors can escape.

   d. An atmospheric dilution factor of 3.6×10^{-5} μCi/ml per Ci/s for gaseous discharges from the facility.

4. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies.

Basis

1. This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials.

2-3. These specifications are intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary of the UWNR. The dilution factor is based on computations reported in Chapter 11 and Appendix A of the Safety Analysis Report.

4. The 1.5 curie limitation on iodine 131 through 135 assures that in the event of failure of a fueled experiment leading to total release of the iodine, the exposure dose at the exclusion area boundary will be less than that allowed by 10 CFR Part 20 for an unrestricted area.

TS 3.8.3 Experiment Failure and Malfunctions

Specification

If a capsule fails and releases material which could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection of the capsule shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Reactor Director or designated alternate and determined to be satisfactory before operation of the reactor is resumed.

Basis

Operation of the reactor with a failed capsule is prohibited to prevent damage to the reactor fuel or structure. Failure of a capsule must be investigated to assure no damage has or will occur.

TS 3.9 Facility Specific LCOs

There are no facility specific LCOs at this facility.

TS 4 SURVEILLANCE REQUIREMENTS

In accordance with section 4.0 of Standard ANSI/ANS-15.1, the following terms for average surveillance intervals shall allow, for operational flexibility only, maximum times between surveillance intervals as indicated below unless otherwise specified within the specification.

- Five-year interval not to exceed six years.
- Biennial interval not to exceed two and one-half years.
- Annual interval not to exceed 15 months.
- Semiannual interval not to exceed seven and one-half months.
- Quarterly interval not to exceed four months.
- Monthly interval not to exceed six weeks.
- Weekly interval not to exceed ten days
- Daily interval must be done within the calendar day.

Scheduled surveillances, except those specifically required when the reactor is shut down, may be deferred during shutdown periods, but be completed prior to subsequent reactor startup unless operation is required for the performance of the surveillance. Scheduled surveillances which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown. If the reactor is not operational in a particular mode, surveillances required specifically for that mode may be deferred until the reactor becomes operational in that mode.

General Applicability

This specification applies to the surveillance requirements of any system related to reactor safety.

Objective

The objective is to verify the proper operation of any system related to reactor safety after maintenance or modification of the system.

Specification

Any additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the rod drive mechanism, or the reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications approved by the Reactor Safety Committee. A system shall not be considered operable until after it is successfully tested.

Basis

This specification relates to changes in reactor systems which could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications, then it can be assumed that they meet the presently accepted operating criteria.

TS 4.1 Reactor Core Parameters

Applicability

This specification applies to the surveillance requirements for measurements, tests, and calibrations of reactor core parameters.

Objective

The objective is to verify the core parameters which are directly related to reactor safety.

Specification

1. Excess reactivity
   Excess reactivity shall be determined at least annually and after changes in either the core, in-core experiments, or control elements for which the predicted change in reactivity exceeds the absolute value of the specified shutdown margin.

2. Shutdown margin
   The shutdown margin shall be determined at least annually and after changes in either the core, in-core experiments, or control elements.

3. Pulse limits
   The reactor shall be pulsed semiannually to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value.

4. Core configuration
   Each planned change in core configuration shall be determined to meet the requirements of Sections 3.1(4) and 5.3 of these specifications before the core is loaded.

5. Reactivity Coefficients
   Power defect and pulsing characteristics shall be measured during startup testing of cores containing different fuel compositions and compared to predictions in the Safety Analysis Report.

6. Fuel Parameters

   a. All fuel elements shall be inspected visually for damage or deterioration annually.

   b. Uninstrumented fuel elements which have been resident in the core during the previous year shall be measured for length and sagitta annually. Fuel elements shall not be added to a core unless a measurement of length and sagitta has been completed within the previous fifteen months.

   c. Fuel elements in the hottest assumed location, as well as representative elements in each of the rows, shall be measured for possible damage in the event there is indication that the Limiting Safety System Setting may have been exceeded.

Basis

1-2. Annual measurements, coupled with measurements made after changes that can affect reactivity values provide adequate assurance that core behavior will be as analyzed. The reactivity values in TRIGA LEU 30/20 fuel change very slowly with fuel burnup.

3. Semiannual verifications assure no changes in behavior are resulting from fuel characteristic changes.

4. Checking contemplated core configurations against requirements will prevent inadvertent loading of cores which do not meet power peaking restraints imposed by composition restrictions.

5. Measurements made during core startup testing are sufficient to assure core behavior will be as analyzed.

6. Annual inspection of the TRIGA fuel has been shown adequate to assure fuel element integrity through a long history of standard operation.

TS 4.2 Reactor Control and Safety Systems

Applicability

This specification applies to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems.

Objective

The objective is to verify the performance and operability of those systems and components which are directly related to reactor safety.

Specification

1. Reactivity worth of control elements
   The reactivity worth of control elements shall be determined upon substantiative changes in core composition or arrangement and annually thereafter.

2. Control element withdrawal and insertion speeds
   Control element drive withdrawal and insertion speeds shall be measured annually and following maintenance to the control element or the control element drive mechanism.

3. Transient Rod and Associated Mechanism
   The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated as necessary annually.

4. Scram times of control and safety elements
   The scram time for all scrammable control elements shall be measured annually and following maintenance to the control elements or their drives.

5. Scram and Power Measuring Channels

   a. A channel test of each Reactor Safety System measuring channel in Table 3.2.4 items (1) through (4) and the interlocks in Table 3.2.5 required for the intended modes of operation shall be performed within 24 hours before each day's operation or prior to each operation extending more than one day.

   b. A channel test of items (5), (6), and (7) in Table 3.2.4 shall be performed semi-annually.

   c. A channel calibration of items (1) and (2) in Table 3.2.4 shall be performed annually.

6. Operability Tests
   This concern is covered by the General Surveillance criterion at the beginning of this section.

7. Thermal Power Calibration-Forced Convection
   Not applicable to this reactor

8. Thermal Power Calibration-Natural Convection
   A Channel Calibration shall be made of the power level monitoring channels by the calorimetric method upon substantiative changes in core composition or arrangement and annually thereafter.

9. Control Element Inspection
   The control elements shall be visually inspected for deterioration biennially.

Basis

1. Control element worths change slowly unless the core arrangement is changed, so annual measurement is sufficient to assure safety.

2. Control element insertion or withdrawal speeds are fixed by the motor design and thus do not change except for extreme binding conditions within the drive.

3. Transient rod drive and air supply includes filtration and lubrication, so an annual check coupled with pre-startup checks is sufficient to assure operabilty.

4. Measurement of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control rods to perform properly.

5. The items 1 through 4 in the table are essential safety equipment and thus should be checked frequently, even though no failures have been observed by checkout in nearly 50 years of operation. Frequent testing is unnecessary for item 5, a simple float switch which is very unlikely to fail, and has performed for nearly 50 years without a failure. Testing item 6, the high voltage monitor scram, results in changing the voltage to the neutron detectors. This introduces step changes into the signal circuits of the measuring channels which can lead to long recovery times and a significant increase in failures of the measuring channels. Further, since the checkout of the linear safety channels is a source check, if high voltage were lost that check would not be possible if the voltage had been lost.

6. The general requirement for checks of equipment operability after maintenance or modification of systems will reveal any loss of safety functions due to the maintenance or modification.

8. The power level channel calibration will assure that the reactor will be operated at the proper power levels.

9. Annual checks in other TRIGA reactors and for nearly 50 years in this reactor have been sufficient to insure no failures due to deterioration.

TS 4.3 Coolant Systems

Applicability

This specification applies to the reactor pool water.

Objective

The objective is to assure the water quality and radioactivity is within the defined limits

Specification

The pool water conductivity and radioactivity shall be measured quarterly.

Basis

Pool water conductivity is continuously monitored, but would be manually monitored on a quarterly basis if the instruments failed. Radioactivity is indirectly monitored by an area radiation monitor near the demineralizer bed, so gross activity increases would be detected immediately. Experience with TRIGA reactors indicates the earliest detection of fuel clad leaks is usually from airborne activity, rather than pool water activity. The quarterly measurement can identify specific radionuclides.

TS 4.4 Confinement

Applicability

This specification applies to the reactor confinement.

Objective

The objective is to assure that air is swept out of confinement and exhausted through a monitored release point.

Specification

The ventilation system shall be verified operable in accordance with TS 4.5 quarterly.

Basis

Because the ventilation system is the only equipment required to achieve confinement, operability checks of the ventilation system meet the functional testing requirements for confinement.

TS 4.5 Ventilation Systems

Applicability

This specification applies to the building confinement ventilation system.

Objective

The objective is to assure the proper operation of the ventilation system in controlling releases of radioactive material to the uncontrolled environment.

Specification

It shall be verified quarterly and following repair or maintenance that the ventilation system is operable.

Basis

Over 30 years of experience with the previous ventilation system has demonstrated that testing the system quarterly is sufficient to assure the proper operation of the system and control of the release of radioactive material. The new ventilation system is expected to exceed the reliability of the previous system so quarterly testing is still appropriate.

TS 4.6 Emergency Electrical Power Systems

Not Applicable.

TS 4.7 Radiation Monitoring Systems and Effluents

TS 4.7.1 Radiation Monitoring Systems

Applicability

This specification applies to the surveillance requirements for the area radiation monitoring equipment and the stack air monitoring system.

Objective

The objective is to assure that the radiation monitoring equipment is operating and to verify the appropriate alarm settings.

Specification

The radiation monitoring and stack monitoring systems shall be calibrated annually and shall be verified to be operable by monthly source checks or channel tests.

Basis

Experience has shown that monthly verification of area radiation monitor operability and setpoints in conjunction with the downscale-failure feature of the instrument is adequate to assure operability. Annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span. Annual calibrations and monthly source or channel checks of the stack particulate and gaseous monitors, along with the high or low flow alarms associated with the monitor assure operability and accuracy.

TS 4.7.2 Effluents

Applicability

This specification applies to gaseous and liquid discharges from the reactor laboratory.

Objective

The objective is to assure that ALARA and 10 CFR Part 20 limits are observed.

Specification

Liquid radioactive waste discharged to the sewer system shall be sampled for radioactivity to assure levels are below applicable limits before discharge. Results of the measurements shall be recorded and reported in the Annual Report.

The total annual release of gaseous radioactivity to the environment shall be recorded and reported in the Annual Report.

Basis

Liquid waste releases are batch releases, so the liquid can be sampled before release. Air activity discharged is continuously recorded and the integrated release is reported.

TS 4.8 Experiments

No surveillances are required.

TS 4.9 Facility-Specific Surveillance

Not applicable. There is no facility-specific surveillance.

TS 5 DESIGN FEATURES

TS 5.1 Site and Facility Description

Applicability

This specification applies to the room housing the reactor and the ventilation system controlling that room.

Objective

The objective is to provide restrictions on release of airborne radioactive materials to the environs.

Specification

1. The reactor shall be housed in a closed room designed to restrict leakage. The minimum free volume shall be 2,000 cubic meters.

2. All air or other gas exhausted from the reactor room and the Beam Port and Thermal Column Ventilation System shall be released to the environment a minimum of 30.5 meters above ground level.

3. The operations boundary shall be the Reactor Laboratory, room 1215 of the Mechanical Engineering Building. The operations boundary shall be a restricted area.

4. The site boundary shall be that portion of the center and east wings of the Mechanical Engineering Building south of the north lobby, plus the portion of Engineering Drive south of the designated areas of the building. The site boundary may be a non-restricted area.

Basis

Calculations in Chapter 13 of the SAR demonstrate that the occupational doses in the event of the maximum hypothetical accident do not exceed limits if the lab volume is at least 2000 cubic meters. Furthermore, calculations in Chapter 13 that assume operation of the ventilation system assume a stack height of 30.5m. The Reactor Director has direct authority over all activities within room 1215 of the Mechanical Engineering Building. The Reactor Director may directly initiate emergency activities within the site boundary. The site boundary may be frequented by people unacquainted with reactor operations.

TS 5.2 Reactor Coolant System

Applicability

This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specification

1. The reactor core shall be cooled by natural convective water flow.

2. The pool water inlet pipe to the demineralizer shall not extend more than 15 feet into the top of the reactor pool when fuel is in the core. The outlet pipe from the demineralizer shall be equipped with a check valve and siphon breaker to prevent inadvertent draining of the pool.

3. Diffuser and other auxiliary systems pumps shall be located no more than 15 feet below the top of the reactor pool.

4. All other piping and pneumatic tube systems entering the pool shall have siphon breakers and valves or blind flanges which will prevent draining more than 15 feet of water from the pool.

5. A pool level alarm shall indicate loss of coolant if the pool level drops approximately one foot below normal level.

6. A pool water temperature alarm shall indicate if water temperature reaches 130°F.

Basis

1. The LEU Conversion SAR Section 4.7.8 shows by analysis that the natural convective cooling of the reactor core is sufficient to maintain the fuel in a safe condition up to at least a power level of 1500 kW (the power level Safety Limit).

2. The inlet pipe to the demineralizer is positioned so that a siphon action will drain less than 15 feet of water. The outlet pipe from the demineralizer discharges into a pipe entering the bottom of the pool through a check valve which prevents leakage from the pool by reverse flow from pipe ruptures or improper operation of the demineralizer valve manifold. In addition, the pipe has a loop equipped with a siphon breaker which prevents loss of pool water.

3. In the event of pipe failure and siphoning of pool water, the pool water level will drop no more than 15 feet from the top of the pool.

4. Other pipes which enter the pool have siphon breakers which prevent pool drainage. Valves are provided for pneumatic tube system lines and primary cooling system pipes. Other piping installed in the pool has blind flanges permanently installed.

5. Loss of coolant alarm, after one foot of loss, requires corrective action. This alarm is observed in the reactor control room and outside the reactor building.

6. The thermal-hydraulic analysis in the SAR assumes a pool water temperature of 130°F. If the temperature exceeds 130°F then the alarm will prevent continued operation in an un-analyzed condition.

TS 5.3 Reactor Fuel

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specification

The individual unirradiated TRIGA LEU 30/20 fuel elements shall have the following characteristics:

1. Uranium content: maximum of 30 Wt-% enriched to maximum of 19.95 Wt-% with nominal enrichment of 19.75 Wt-% Uranium 235.

2. Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65.

3. Natural erbium content (homogeneously distributed): nominal 0.9 Wt-%.

4. Cladding: 304 stainless steel, nominal 0.020 inch thick.

Basis

1. The fuel specification permits a maximum uranium enrichment of 19.95%. This is about 1% greater than the design value for 19.75% enrichment. Such an increase in loading would result in an increase in power density of less than 1%. An increase in local power density of 1% reduces the safety margin by less than 2% (Texas A&M LEU Conversion SAR, December 2005).

2. The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad (M.T. Simnad, "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," General Atomics Report E-117-833, February, 1980).

3. The fuel specification for a single fuel element permits a minimum erbium content of about 5.6% less than the design value of 0.90 Wt-%. (However, the quantity of erbium in the full core must not deviate from the design value by more than -3.3%). This variation for a single fuel element would result in an increase in fuel element power density of about 1-2%. Such a small increase in local power density would reduce the safety margin by less than 2% (Texas A&M LEU Conversion SAR, December 2005).

4. Stainless steel clad has been shown through decades of operation to provide a sufficient barrier against fission product release with minimal corrosion.

TS 5.4 Reactor Core

Applicability

This specification applies to the configuration of fuel and in-core experiments.

Objective

The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specification

1. The core shall be an arrangement of TRIGA LEU 30/20 uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

2. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

3. The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water.

Basis

1. TRIGA cores have been in use for years and their characteristics are well documented. LEU cores including 30/20 fuel have also been operated at General Atomics and Texas A&M and their successful operational characteristics are available. In addition, the analysis performed at Wisconsin indicates that the LEU 30/20 core will safely satisfy all operational requirements. See chapters 4 and 13 of the LEU Conversion Analysis SAR.

2. Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high power density.

3. The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

TS 5.5 Control Elements

Applicability

This specification applies to the control blades and transient control rod.

Objective

The objective is to assure that control elements are fabricated to reliably perform their intended control and safety function.

Specification

1. The safety blades shall be constructed of boral plate and shall have scram capability.

2. The regulating blade shall be constructed of stainless steel.

3. The transient rod shall contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The transient control rod shall have scram capability and may incorporate an aluminum or air follower.

Basis

The boral safety blades and stainless steel regulating blade used in the reactor have been shown to provide adequate reactivity worth, structural rigidity, and reliability to assure reliable operation and long life under operating conditions. The transient control rod materials and fabrication techniques have been used in many TRIGA reactors and have demonstrated reliable operation and long life.

TS 5.6 Fissionable Material Storage

Applicability

This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective

The objective is to assure that fuel which is being stored will not become critical and will not reach an unsafe temperature.

Specification

1. All fuel elements shall be stored in a geometrical array where the value of k-effective is less than 0.8 for all conditions of moderation.

2. Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel element or fueled device temperature will not exceed design values.

Basis

The limits imposed by specifications 5.6.1 and 5.6.2 are conservative and assure safe storage.

TS 6. ADMINISTRATIVE CONTROLS

TS 6.1 Organization

TS 6.1.1 Structure

The reactor facility shall be an integral part of the Engineering Physics Department of the College of Engineering of the University of Wisconsin-Madison. The reactor shall be related to the University structure as shown in **Figure 14-1**.

The Radiation Safety office performs audit functions for both the Radiation Safety Committee and the Reactor Safety Committee and reports to both committees as well as to the Reactor Director.

TS 6.1.2 Responsibility

The Reactor Director is responsible for all activities at the facility, including licensing, security, emergency preparedness, and maintaining radiation exposures as low as reasonably achievable.

The reactor facility shall be under the direct control of a Reactor Supervisor designated by the Reactor Director. The Reactor Supervisor shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, procedures, and the requirements of the Radiation Safety Committee and the Reactor Safety Committee.

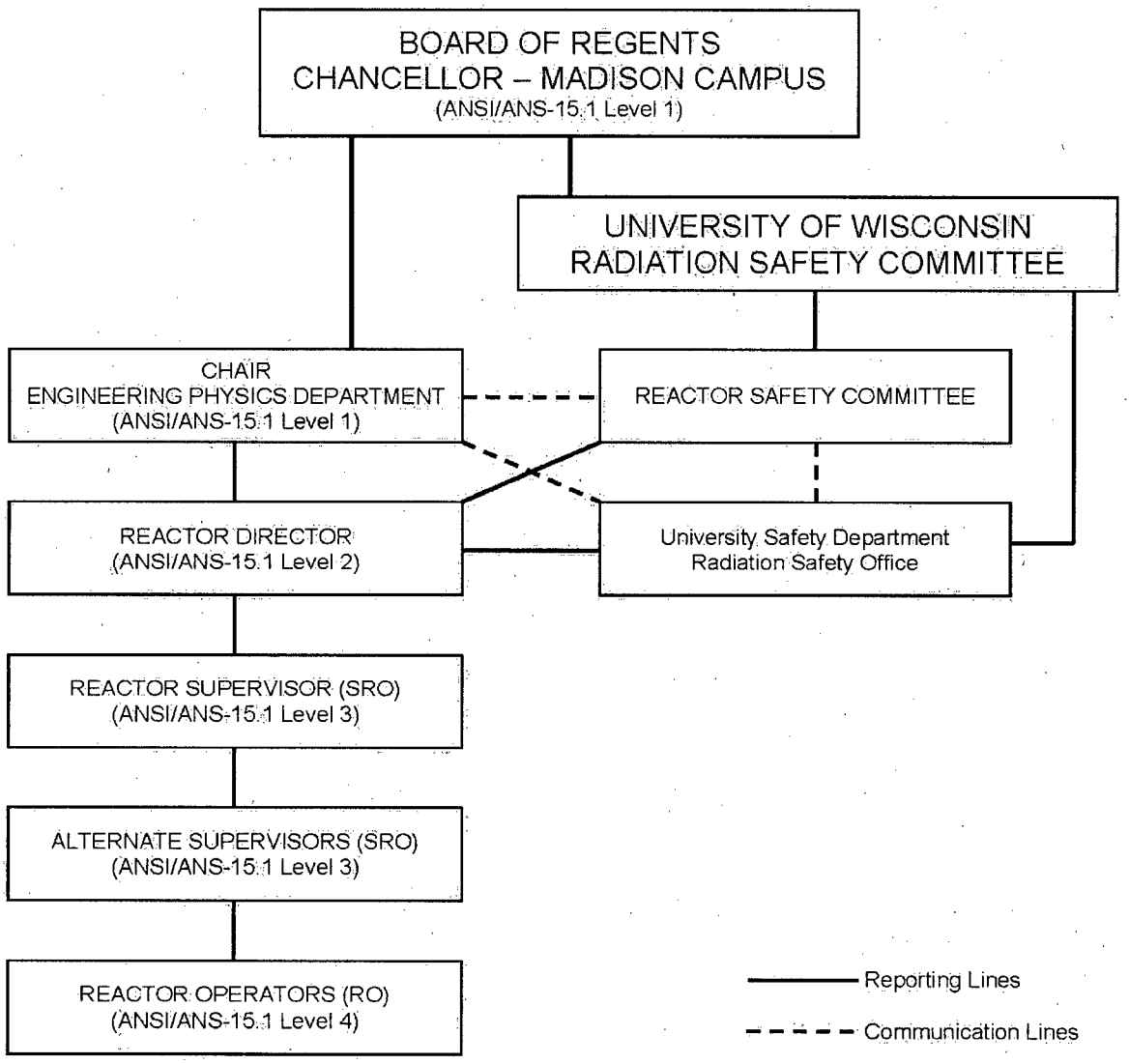


Figure 14-1, Organization Chart

TS 6.1.3 Staffing

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

   a. A licensed reactor operator in the control room (if senior operator licensed, may also be the person required in c).

   b. A second designated person present at the facility capable of carrying out prescribed written instructions.

   c. A designated senior reactor operator shall be readily available at the facility or on call. On call means the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility.

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

3. A licensed senior reactor operator shall be present at the facility for:

   a. Initial startup and approach to power.

   b. All fuel handling or control-element relocations.

   c. Relocation of any in-core experiment with a reactivity worth greater than 0.7% $\Delta K/K$.

   d. Recovery from unplanned or unscheduled shutdown or significant power reduction.

TS 6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of ANSI/ANS-15.4-1988 Sections 4-6.

TS 6.2 Review and Audit

There shall be a Reactor Safety Committee which shall review and audit reactor operations to assure that the facility is operated in a manner consistent with public safety and within the conditions of the facility license.

TS 6.2.1 Composition and Qualifications

The Committee shall be composed of a least six members, one of whom shall be a Health Physicist from the University of Wisconsin Safety Department Radiation Safety Office. The Committee shall collectively possess expertise in the following disciplines:

1. Reactor Physics;
2. Heat transfer and fluid mechanics;
3. Metallurgy
4. Instruments and Control Systems;
5. Chemistry and Radio-chemistry;
6. Radiation Safety.

TS 6.2.2 Charter and Rules

The Committee shall meet at least annually.

The Committee shall formulate written standards regarding the activities of the full committee; minutes, quorum, telephone polls for approvals not requiring a formal meeting, and subcommittees.

TS 6.2.3 Review Function

The responsibilities of the Reactor Safety Committee shall include, but are not limited to, the following:

1. Review and approval of experiments utilizing the reactor facilities;

2. Review and approval of all proposed changes to the facility, procedures, license, and technical specifications;

3. Determination of whether a proposed change, test or experiment would constitute an unreviewed safety question or a change in Technical Specifications;

4. Review of abnormal performance of plant equipment and operating anomalies having safety significance; and

5. Review of unusual or reportable occurrences and incidents which are reportable under 10 CFR Part 20 and 10 CFR Part 50.

6. Review of audit reports.

7. Review of violations of technical specifications, license, or procedures and orders having safety significance.

TS 6.2.4 Audit Function

A Health Physicist from the University of Wisconsin Safety Department Radiation Safety Office shall represent the University Radiation Safety Committee and shall conduct an inspection of the facility at least monthly to assure compliance with the regulations of 10 CFR Part 20. The services and inspection function of the Health Physics Office shall also be available to the Reactor Safety Committee, and will extend the scope of the audit to cover license, technical specification, and procedure adherence.

The committee shall audit operation and operational records of the facility, requalification program, security plan, and emergency plan and its implementing procedures. If the committee chooses to use the staff of the Health Physics organization for the audit function, the reports of audit results will be distributed to the committee and included as an agenda item for committee meetings.

TS 6.3 Radiation Safety

The Reactor Laboratory shall meet the requirements of the University Radiation Safety Regulations as submitted for the University Broad License, License Number 25-1323-01 and is subject to the authority of the state license.

The Reactor Director shall have responsibility for maintaining radiation exposures as low as reasonably achievable and for implementation of laboratory procedure for insuring compliance with 10 CFR Part 20 regulations.

TS 6.4 Procedures

Written operating procedures shall be adequate to assure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect for the following items:

1. Testing and calibration of reactor operating instrumentation and controls, control rod drives, area radiation monitors, and air particulate monitors;

2. Reactor startup, operation, and shutdown;

3. Emergency and abnormal conditions, including provisions for evacuation, reentry, recovery, and medical support;

4. Fuel element and experiment loading or unloading;

5. Control rod removal or replacement;

6. Routine maintenance of the control rod drives and reactor safety and interlock systems or other routine maintenance that could have an effect on reactor safety;

7. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms and abnormal reactivity changes; and

8. Civil disturbances on or near the facility site.

Substantive changes to the above procedures shall be made only with the approval of the Reactor Safety Committee. Temporary changes to the procedures that do not change their original intent may be made by the Senior Operator in control or designated alternate. All such temporary changes shall be documented and subsequently reviewed by the Reactor Safety Committee.

TS 6.5 Experiment Review and Approval

1. Routine experiments may be performed at the discretion of the senior operator responsible for operation without the necessity of further review or approval.

2. Prior to performing any experiment which is not a routine experiment, the proposed experiment shall be evaluated by the senior operator responsible for operation. The senior operator shall consider the experiment in terms of its effect on reactor operation and the possibility and consequences of its failure, including where significant, consideration of chemical reactions, physical integrity, design life, proper cooling, interaction with core components, reactivity effects, and interactions with reactor instrumentation.

3. Modified routine experiments may be performed at the discretion of the senior operator responsible for operation without the necessity of further review or approval provided that the evaluation performed in accordance with Section 6.5(2) results in a determination that the hazards associated with the modified routine experiment are neither greater nor significantly different than those involved with the corresponding routine experiment which shall be referenced.

4. No special experiment shall be performed until the proposed experiment has been reviewed and approved by the Reactor Safety Committee.

5. Favorable evaluation of an experiment shall conclude that failure of the experiment will not lead directly to damage of reactor fuel or interference with movement of a control element.

TS 6.6 Required Actions

TS 6.6.1 Action to be Taken in Case of Safety Limit Violation

In the event a safety limit is exceeded:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.

2. An immediate report of the occurrence shall be made to the Chairman, Reactor Safety Committee, and reports shall be made to the NRC in accordance with Section 6.7 of these specifications, and

3. A report shall be prepared which shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Committee (RSC) for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

TS 6.6.2 Action to be Taken in the Event of an Occurrence of the Type Identified in 6.7.2(1)b., and 6.7.2(1)c.

In the event of a reportable occurrence (see TS 1.3) the following actions shall be taken:

1. The reactor shall be shut down.

2. The Director or designated alternate shall be notified and corrective action taken with respect to the operations involved,

3. The Director or designated alternate shall notify the Chairman of the Reactor Safety Committee,

4. A report shall be made to the Reactor Safety Committee which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence, and

5. A report shall be made to the NRC in accordance with Section 6.7.2 of these specifications.

TS 6.7 Reports

TS 6.7.1 Operating Reports

1. An annual report covering the activities of the reactor facility during the previous calendar year shall be submitted (in writing to U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555) within six months following the end of each calendar year, providing the following information:

   a. A brief narrative summary of (1) operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;

   b. Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality;

   c. The number of emergency shutdowns and inadvertent scrams, including reasons therefor;

   d. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;

   e. A brief description, including a summary of the safety evaluations of changes in the facility or in the procedures and of tests and experiments carried pursuant to Section 50.59 of 10 CFR Part 50;

   f. A summary of radiation exposures received by facility personnel and visitors, including dates and time of significant exposures and a summary of the results of radiation and contamination surveys performed within the facility; and

   g. A description of any environmental surveys performed outside the facility.

h. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge;

(1) Liquid Effluents (summarized on a monthly basis)

Liquid radioactivity discharged during the reporting period tabulated as follows:

(a) Total estimated radioactivity released (in curies).

(b) The isotopic composition if greater than 1×10^{-7} microcuries/cc for fission and activation products.

(c) Total radioactivity (in curies), released by nuclide, during the reporting period based on representative isotopic analysis.

(d) Average concentration at point of release (in microcuries/cc) during the reporting period and the fraction of the applicable limit in 10 CFR Part 20.

(e) Total volume (in gallons) of effluent water (including diluent) during periods of release.

(2) Exhaust Effluents (summarized on a monthly basis)

Radioactivity discharged during the reporting period (in curies) for:

(a) Gases.

(b) Particulates with half lives greater than eight days.

(c) The estimated activity (in curies) discharged during the reporting period, by nuclide, for all gases and particulates based on representative isotopic analysis and the fraction of the applicable 10 CFR Part 20 limits for these values.

(3) Solid Waste

(a) The total amount of solid waste packaged (in cubic feet).

(b) The total activity involved (in curies).

(c) The dates of shipment and disposition (if shipped off site).

2. A report within 60 days after completion of startup testing of the reactor (in writing to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555) upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level describing the measured values of the operating conditions or characteristics of the reactor under the new conditions including:

   a. An evaluation of facility performance to date in comparison with design predictions and specifications, and

   b. A reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis.

TS 6.7.2 Special Reports

1. There shall be a report of any of the following not later than the following day by telephone or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report describing the circumstances of the event and sent within 14 days to U.S. Nuclear Regulatory commission, Attn: Document Control Desk, Washington, D.C. 20555:

   a. Any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;

   b. Any violation of a safety limit; and

   c. Any reportable occurrences as defined in TS 1.3 of these specifications.

2. A written report within 30 days in writing to the U.S. Nuclear Regulatory commission, Attn: Document Control Desk, Washington, D.C. 20555 of:

   a. Permanent changes in facility organization at Reactor Director or Department Chair level.

   b. Any significant change in the transient or accident analysis as described in the Safety Analysis Report;

TS 6.8 Records

TS 6.8.1 Records to be Retained for a Period of at least Five Years or for the Life of the Component Involved if Less than Five Years

1. Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which shall be maintained for a period of at least one year),

2. Principal maintenance activities,

3. Reportable occurrences,

4. Surveillance activities required by the Technical Specifications,

5. Reactor facility radiation and contamination surveys where required by applicable regulations,

6. Experiments performed with the reactor,

7. Fuel inventories, receipts, and shipments,

8. Approved changes in operating procedures,

9. Records of meeting and audit reports of the review and audit group.

TS 6.8.2 Records to be Retained for at Least One Certification Cycle

Record of retraining and requalification of certified operations personnel shall be maintained at all times the individual is employed or until the certification is renewed. For the purposes of this technical specification, a certification is an NRC issued operator license.

TS 6.8.3 Records to be Retained for the Lifetime of the Reactor Facility

Annual reports which contain the information in items 1 and 2 may be used as records for those items.

1. Gaseous and liquid radioactive effluents released to the environs,
2. Offsite environmental monitoring surveys required by technical specifications,
3. Radiation exposures for all personnel monitored,
4. Updated, corrected, and as-built drawings of the facility.
5. Notification that safety limit was exceeded.
6. Notification that automatic safety system did not function as required.
7. Notification of failure to meet limiting conditions for operation.

TS 7 REFERENCES

1. LEU Conversion SAR, 2008, as amended.

2. GA-9064, pages 3-1 to 3-23.

3. "Sagitta" refers to the bow of the element and means the maximum excursion of the clad surface from a chord connecting the two ends of the clad surface.

4. Simnad and West, 1986.

5. NUREG-1282.