THE PENNSYLVANIA STATE UNIVERSITY

UNIVERSITY PARK, PENNSYLVANIA 16802

College of Engineering
Penn State Breazeale Reactor

Area Code 814
865-6351

DKT 50-005

August 16, 1985

Mr. Cecil O. Thomas
Standardization and Special Projects Branch
Division of Licensing
United States Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Request for Additional Information

Dear Mr. Thomas:

Your letter of July 24, 1985 requested answers for several questions regarding our application for renewal of the operating license of the PSBR. Enclosed is a copy of the questions submitted by you together with their answers. In addition, we are also enclosing a copy of our Technical Specifications changed to conform with ANSI/ANS 15.1 (1982).

If you have any questions concerning the enclosed information, please contact either Samuel H. Levine, Director or Ira B. McMaster, Deputy Director at (814) 865-3110.

Sincerely yours,

C. L. Hosler

Charles L. Hosler
Vice President for Research
Dean of the Graduate School

CLH:SHL/skr

Enclosures

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<u>PENNSYLVANIA STATE UNIVERSITY FORMAL QUESTIONS</u>

1. **What is the metal-to-water ratio of the core? Including the ZrH_x matrix, what is the average H/U-235 ratio in the core?**

   The metal-to-water ratio of the core is 2.0. The average H/U-235 ratio is 373 for the 8.5 wt% fuel and 258 for the 12 wt% fuel.

2. **What is β effective for the Pennsylvania State Breazeale Reactor (PSBR)?**

   0.007 as given in GA-7882 "Kinetic Behavior of TRIGA Reactors" by G. B. West, et.al., March (1967).

3. **For the PSBR current core, what are the reactivity worths of the individual control rods (shim, safety, regulating, and transient) for a typical PSBR core loading? What is the total number of fuel elements for the core loading? Are your shutdown margin and total excess reactivity consistent?**

   Core information for PSBR Loading #36

Rod Designation	Worth
Shim	$2.18
Safety	4.07
Regulating	2.17
Transient	2.74
TOTAL	11.16

   Fuel Inventory - Loading #36

Number of Elements	Element Type
78	8.5 wt%
13	12 wt%
1	12 wt% - instrumented
3	8.5 wt% - control rod followers
Total 95	

   To date the shutdown margin and total excess reactivity of all our cores have been consistent.

4. **In addition to the reactor scrams required by your Technical Specifications, describe any other scrams you have at the PSBR.**

   If any of the six radiation monitors described in Table 7-1 page VII-2 of the SAR exceed their alarm settings, a reactor scram and a building evacuation alarm will occur. The monitors are: Reactor

Bridge East and West, Reactor Bay Air East and West, Co-60 Bay, and Beam Hole Laboratory.

Depressing the BUILDING EVACUATION ALARM button located on the console will initiate a reactor scram in addition to an evacuation alarm.

Finally, there are four external scram buttons associated with the PSBR. Two are located in the reactor bay (east and west walls) and two are located in the beam hole laboraotry (east and west walls). Depressing any of these four external scram buttons will initiate a reactor scram.

Thus, there are eleven devices (the six monitors, four external buttons, and the building evacuation button) which will energize the EXTERNAL SCRAM and are not required by our Technical Specifications.

5. **What are the relative pressures of the primary and secondary cooling systems in the heat exchangers? What is the heat exchanger tube material?**

Primary inlet pressure = 16 psig
Secondary outlet pressure = 21 psig
Tube material is type T304 stainless steel

6. **What are the flow rates of the normal and emergency ventilation systems?**

The flow rate of the normal ventilating system is 2,000 cubic feet per minute or greater. The flow rate of the emergency ventilating system is 2,500 cubic feet per minute or greater.

7. **How are the floor drains within the reactor pool closed so that inadvertent draining of the pools is precluded.**

Refer to Figure 4-1, page IV-3 of the SAR. Pipes connected to the pool floor drains pass through a valve pit below the floor of the beam hole laboratory and then to a storm sewer. There are two inline valves in each pipe with a tee connection between them. The outer valves (65 & 66) on each pipe are closed and padlocked, accessible only to a licensed senior operator (i.e., a ND 325A key). The inner valves on each pipe (62 & 64) are kept closed except when transferring water. The tee connections between the inline valves also have valves (61 & 63) that are kept closed except when transferring water. These two pipes are then joined at a tee which connects to the storage tank transfer pump loop between two other valves (71 & 74) to control flow direction, and also to another valve (68) which accepts a fire hose (not connected) that can be used to fill the pool from the building water supply. The fire hose can be connected when necessary during an emergency, i.e., large leak in the pool.

Directions for emergency filling and for transferring water between the pool and the storage tank are posted in the area and written in procedures. The posted directions and procedures refer to valves by numbers and each valve is clearly marked with 2" high numbers on large tags. Thus the locked valves, procedures, and administrative control preclude advertent draining of the pool.

8. **Discuss the sensitivity of the fission product monitor. How is it calibrated to quantify fission product leakage from fuel?**

A Tracerlab Model MWP-1A Fission Products Water Monitoring System is used to monitor the pool water for fission products as described on pages IV-2 to IV-5 of the application. A 1½" diameter by 1" NaI scintillation detector is used to give a count rate of gamma rays with energy greater than about 0.8 MeV, primarily from I-132, I-134, I-135, and I-136. The fission product monitor is used only as a qualitative indicator of possible fission product contamination in the pool water. It is not calibrated for fission product concentration. The manufacturer's specifications indicate a count rate of 10,000 cpm (full scale on the normally used range) for I-135 at 2 x 10^{-5} μCi/ml for a period of 30 hours. This is equivalent to the leakage of about 27 mCi of I-135 over a 30 hour period. A release of 54 mCi of I-135 should produce a count rate of 10,000 cpm in about 1 hour.

9. **What is the maximum potential radiation exposure rate in an external radiation beam? What measures are taken to prevent exposure of personnel?**

The maximum radiation potential exposure rate from #4 beam port, with beam port empty, 18" from Beam Hole Lab side is ~ 3,000 R/hr @ 1 MW. This is based on a measurement at 200 KW of 650 R/hr on October 6, 1982.

Measures to prevent personnel exposure include:

a. A radiation survey is required for all new or changes to beam port or vertical tube irradiations even if the change involves additional shielding.

b. The radiation survey includes time and/or position limitations for experimenters and a copy is kept with the experiment authorization which is available to the reactor operator.

c. For beam port experiments, television montoring of the beam port area is required at the reactor console.

d. For beam port experiments when a beam port shield door is open, a light beam alarm circuit is activated to alarm in the control room if the beam area is entered.

e. For beam port irradiation, entrance to Beam Hole Lab from the outside is controlled by locking and labeling interior fence. This fence controls access to the outside of the building in the beam area.

f. Access in the beam hole room is restricted by a chain with proper radiation area marking across walkway to beam area.

For vertical tubes used with the reactor, the potential exposure rate is less than the above value for #4 beam port. The reactor bridge and/or other accessable areas are appropriately roped off and labeled as radiation or high radiation areas when such tubes are in use.

10. Discuss the inadvertent flooding of a beam port air void. What is the maximum potential change in reactivity?

Presently the reactor is coupled to the beam port through the D_2O tank only. There is 18" of heavy water between the reactor and a 6" diameter by 9" long air void inside the D_2O tank. If this air void is filled with water, the reactivity effect on the reactor would be negligable compared to the +54¢ reactivity of the D_2O. Flooding of the air void between the D_2O tank and the beam port and/or the beam port itself would have even less effect. There are several vertical irradiation tubes used at the PSBR. Their reactivity worths are as follows:

6.5" diameter with cadmium, lead, and boron filters at core face - 18¢
6.5" diameter with cadmium liner on Instrument Bridge at core face - 4¢
3" diameter with cadmium liner at core face - 48¢
3" diameter with no liner at core face - 23¢
1" x 3" rectangular tube with cadmium liner at core face - 36¢

The maximum potential change in reactivity due to the flooding of an existing experimental tube would be +23¢ from a 3" bare aluminum irradiation tube since the worth of the cadmium lined tubes would not be affected by flooding.

Should a beam port be coupled to the reactor via a beam port air void its worth would be much less than the 2.31% $\Delta k/k$ ($3.30) allowed by the Technical Specifications for a single secured experiment.

11. Provide information about Health Physics Office involvement in the review and approval process for experiments. That is, how is it assured that health physics concerns are appropriately addressed by researchers using the reactor facility.

Rules and procedures are in effect at The Pennlsylvania State University to provide for the safe use of radioactive material by University personnel. It is the intent of the University that these

rules and procedures allow as much needed flexibility as possible for experimenters while assuring that the release of radioactive material and the exposure of personnel to ionizing radiation be kept to a minimum. It is the policy of the University that the release of radioactive material and the exposure to ionizing radiation be kept as far below the regulatory limits as is reasonably achievable.

University faculty or staff wishing to use radioactive material must request permission to do so by submitting the form "Request for Authorization to use Radioactive Material" to the University Isotopes Committee. The request should be submitted in the name of the invididual(s) who is directly responsible for the direction of the work. That person will be named the laboratory supervisor and is responsible for supervising the work in such a manner to assure that all radiation safety practices and procedures are observed. The forms are submitted to the Chairman of the Committee or the Health Physics Office. It is suggested that applicants discuss the proposed use with a member of the Committee or the Health Physics staff prior to submission of their request.

Requests for authorizations will be reviewed by the Health Physics Office and returned to the Chairman of the University Isotopes Committee with a recommendation for approval, disapproval, or approval with conditions.

The applicant will be notified of the Committee action by the return of a copy of the authorization request signed by the Committee Chairman and noting the action taken. A copy will be returned to the Health Physics Office for filing and a copy of approved requests will be sent to the University research reactor (PSBR), if it is listed as a supplier of the radioactive material or the place of use.

The release of radioactive material from the reactor pool to researchers is the responsibility of the Director of the PSBR. The PSBR staff is assisted by a Health Physics personnel in these releases as appropriate to assure that all state, federal, and University procedures are observed and that personnel exposures and release of radioactive materials to the environs is as low as reasonably achievable.

Health Physics considerations involved with experiments performed at the PSBR facility are also the responsibility of the Director of the PSBR. The expertise of the University Health Physics office is available to assist in assuring that all experiments are performed within the rules and procedures set forth by the various governing agencies.

Although they are usually performed by Health Physics Office Personnel, all area surveys, contamination smear surveys, and environmental monitoring surveys are the responsiblity of the PSBR Director.

12. **Explain how it is assured that radiation control activities are performed properly. Include information about how requirements for the performance of these activities are integrated into reactor (license R-02) procedures and information about how it is determined that these activities are performed consistently and acceptably.**

Radiation control at the PSBR is the responsibility of the Director of the PSBR. Written PSBR procedures, approved and initiated by a representative of the University Health Physics Office, are to be in effect at the facility for the following operations where health physics concerns are of primary importance:

1. Release of Irradiated Samples
2. Evacuation
3. Fire or Explosion
4. Gaseous Release
5. Medical Emergencies
6. Civil Disorder
7. Bomb Threat
8. Threat or Theft of Special Nuclear Material
9. Industrial Sabotage
10. Experiment Evaluation and Authorization
11. Reactor Operation Using a Beam Port
12. D_2O Handling
13. Health Physics Orientation Requirements
14. Hot Cell Entry Procedure

13. **Provide a summary of the annual personnel exposures (the number of persons receiving a total annual exposure within the designated exposure ranges, similar to the report described in 10 CFR 20.407(b)) for the last 5 years of operation.**

The following table lists annual doses to the staff of the reactor and the health physics office for a five year period. The data are from monthly film badge records for penetrating radiation.

	Year				
Persons with:	1980	1981	1982	1983	1984
Less than reportable	2	8	13	17	6
Up to 99 mrem	15	10	8	5	17
100 - 249 mrem	1	1	0	0	0
greater than 249 mrem	0	0	0	0	0
TOTAL	18	19	21	22	23

14. **How is it assured that radiation detection instrumentation performs acceptably? That is, what standards are used to establish and implement calibration and operability - check procedures.**

Radiation measuring instruments both fixed and portable are calibrated periodically as perscribed by the Technical Specifications using a working standard. Both the radiation detection and the measuring instruments are source checked for operability, and alarm action where appropriate on a weekly basis. Existing procedures call for source check of all portable instruments prior to their being used.

15. **In figure 7.2, (SAR), what are functions of the CALIBRATE switches. Compare these functions with the definition of "Calibration" in ANSI/ANS 15.1 (1982).**

The CALIBRATE switch in the log countrate channel is a five position switch and is normally in the operate position. The other four positions, 10^2, 10^3, 10^4, and 10^5 counts per second, provide a calibration signal from a crystal controlled multivibrator to assure that the indicated values are within 70 to 140% of the calibration points. This signal is introduced at the input to the count rate amplifier. It does not test the preamplifier or the pulse amplifier in this channel. This constitutes a channel test by ANSI/ANS 15.1 definitions.

When the CALIBRATE switch is in any position other than OPERATE, the low countrate bistable prevents control rods from being moved and the SOURCE annunciator illuminates.

The LINEAR RECORDER switch in the linear power channel is a three position switch normally in the OPERATE position. When it is turned to CALIBRATE, it provides a 100 MV signal to the input of the linear amplifier which should indicate 100% $\pm$ 1.5% on the recorder chart. This is a channel test by ANSI/ANS 15.1 definitions. The third position provides a signal to test the 1.1 range bistable and the linear power scram. This is a channel check by ANSI/ANS 15.1 definition.

The PERCENT POWER switch in the % power channel is also a three position switch. It is normally in the OPERATE postion; however, when it is turned to CALIBRATE it provides a fixed current to the input of the linear amplifier corresponding to a meter indication of 100% $\pm$ 2%. This is a channel test by ANSI/ANS 15.1 definition. The third position provides a signal to test the % power scram which comes from the optical relay on the % power meter. This is a channel check by ANSI/ANS 15.1 definition.

The CALIBRATE switch in the log power channel is a three position switch. It is normally in the OPERATE position; however, when it is turned to the HI or LO calibrate positions, a constant current signal is provided to the input of the log power amplifier which indicates 10^5 watts or 1 watt respectively on the log power scale.

The tolerance limit in this channel is 70 to 140% of the calibration point. This is a channel test by ANSI/ANS 15.1 definition.

The PERIOD switch in the period channel is a three position switch normally in the OPERATE position. When it is turned to the CALIBRATE position, a fixed voltage is supplied to the period amplifier which results in a period indication of five seconds. This is a channel test by ANSI/ANS 15.1 definition. The BISTABLE TEST position checks the operation of the 3 second period scram bistable and the resulting period scram. This is a channel check by ANSI/ANS 15.1 definition.

The LINEAR RECORDER switch, the PERCENT POWER switch, the CALIBRATE switch, and the PERIOD switch are all spring loaded to return to the OPERATE position.

16. In table 7-1 (SAR), some detector ranges and settings are given in C/m. Discuss and give relationship between C/m and some exposure or dose parameter, or MPC.

The count rate observed from the activity on an air monitor filter is dependent upon air flow rate, filter collection efficiency, beta energy, number of beta emissions per disintegration, air concentration, background, natural airborne activity, length of time the sample has been collected, and half-life of the radioisotope. The alarm set point should be set at a level low enough to detect hazardous airborne concentrations of those radioisotopes with low MPC values which are likely to occur. The set point should, however, be high enough to eliminate false alarms from varying concentrations of naturally occurring airborne activity and slight changes in background radiation levels. The count rate at time t after sampling was started can be determined from the following formula:

$$C = \frac{MPC}{\lambda} (I - e^{-\lambda t}) \, F \cdot E_C \cdot E_F \cdot R_\beta \cdot 2.22 \times 10^6$$

where C = observed count rate cpm

MPC = maximum permissible airborne concentration of the radioactive material in $\mu Ci/cm^3$

t = time since start of sampling in min

F = air flow rate in cm^3/min

E_C = overall detection efficiency, counts per beta

R_β = beta particles emitted per disintegration

E_F = filter collection efficiency

$2.22x10^6$ = dpm/μCi

For the air monitors, the air flow rate is 4 ft^3/min or $1.13x10^5$ cm^3/min, E_F is assumed to be 1.0, E_c is assumed to be .0766, and R_β is assumed to be 1.0 for all beta emitters or:

$$C = 1.92x10^{10} \frac{R_\beta \cdot MPC}{\lambda} (I-e^{-\lambda t})$$

Listed below are several radioisotopes of interest and the time it would take to reach a count rate of 5,000 cpm assuming an airborne concentration equal to the MPC for 40-hour occupational exposure.

Isotope	MPC μCi/cm^3	Half-life Minutes	Minutes to Reach 5,000 cpm
Sr-90	$1x10^{-9}$	$1.46x10^7$	263
Cs-137	$6x10^{-8}$	$1.58x10^7$	4.4
Na-24	$1x10^{-6}$	898	0.26
I-131	$9x10^{-9}$	$1.16x10^4$	29
I-132	$2x10^{-7}$	138	1.3
I-133	$3x10^{-8}$	$1.26x10^3$	8.8
I-134	$5x10^{-7}$	52	0.5
I-135	$1x10^{-7}$	402	2.6
Si-31	$6x10^{-6}$	157	.04
Mn-56	$8x10^{-7}$	154	.33
P-32	$7x10^{-8}$	$2.06x10^4$	3.7

The above chart shows that an alarm set point of 5,000 or even 10,000 cpm is acceptable for the air monitors. For Sr-90, the alarm set point of 5,000 cpm would be reached in 4.4 hours at the 40 hr MPC level. The count rate due to natural airborne activity is usually about 500 cpm but might approach 2500 cpm or more in unusual circumstances such as a severe or prolonged atmospheric inversion condition.

It should be noted that the above table assumes a filter collection efficiency of 100%. This is certainly not true for the iodine isotopes, for which the filter efficiency could be much less. However, even at 10% efficiency, the time to reach the alarm point would be much less than 40 hours.

17. **What are the minimum qualifications for such positions as the PSBR Director, and the head of the health physics services? Please include appropriate wording in the Technical Specifications.**

The Director of the PSBR should have as minimum qualifications an advanced degree in science or engineering and 2 years experience in

reactor operation. Five years additional experience in directing reactor operation may be used to substitute for an advanced degree.

The minimum qualifications for the University Health Physicist position are the equivalent of a graduate degree in radiation protection, 3-5 years experience with a broad byproduct material license and certification by the American Board of Health Physics or eligibility for certification.

18. **At what level of administration is the ALARA policy established? How is it promulgated? Please provide a copy.**

The ALARA program at Penn State University is established at the office of the Vice-President for Research. It is promulgated through the University Health Physics Office, the head of which reports directly to the office of the Vice-President for Research.

The attached inter-office correspondences, attachment B, one from 1971 and a more recent one from the current Vice-President confirm this fact. The ALARA policy is also stated in section 1 "Rules and Procedures for Use of Radioactive Material at The Pennsylvania State University by the University Isotopes Committee July 1980".

19. **Explain the purpose and scope and frequency of the facility audits, and compare with ANSI/ANS 15.1 (1982). Please include appropriate wording in the Technical Specifications to show frequency of audits, scope, by what office are they established, to whom are results reported, etc.**

A Penn State Reactor Safeguards Committee (PSRSC) exists to provide an additional measure of safety in the operation of the nuclear reactor at the Penn State Breazeale Reactor facility and in experiments utilizing special nuclear material and source material at The Pennsylvania State University.

The primary domain of jurisdiction of the PSRSC shall be the safety evaluation of in-core nuclear reactor experiments and the periodic review and evaluation of the physical integrity of the core. Expressly:

a. The PSRSC shall be concerned with those reactor experiments which by their unusual nature, potential hazard or unprecedented complexity, could endanger health, life, and property in and about the PSBR .
b. Reactor-oriented experiments which contain unusual dangers in addition to those listed above shall also be submitted to the PSRSC for evaluation. The PSRSC may also evaluate experiments which are not directly associated with the reactor environs or the fuel storage vaults at the request of the University Isotopes Committee (UIC), as noted below. Favorable review of an experiment by the PSRSC shall not obligate the staff of the PSBR to carry out the experiment.

c. The PSRSC may aid the reactor staff with the review and evaluation of any indications that changes are taking place in the core which influence its integrity.
d. The PSRSC shall provide for periodic review and audit of facility operation, including both equipment and operating personnel. The audit shall be performed at least once each year by individual(s) who are not part of the PSBR operating staff. An audit of operations should include an examination of operating and maintenance log books, personnel monitoring records, observation of and interview with members of the operating staff, evaluation of the results of surveillance tests and inspections, and a review of compliance with NRC regulations, Technical Specifications, license requirements and internal procedures.

STRUCTURE OF THE COMMITTEE

1. The PSRSC shall be composed of not less than seven and not more than ten members, including the chairman.
2. The Director or Acting Director of the reactor shall be a member of the PSRSC but shall not be a chairman.
3. All members of the PSRSC shall be knowledgeable in subject matter related to reactor operations.
4. At least one member of the PSRSC shall have health physics expertise and at least one member shall be well versed in reactor theory.
5. In addition, one member shall either be a member of the UIC or designated by the Chairman of the PSRSC as liaison between the PSRSC and the UIC.
6. The PSRSC shall be appointed by the Dean of the College of Engineering, acting for the Vice President for Research and Graduate Studies.
7. Each member, except the Director or Acting Director of the reactor, who is an ex-officio member, shall be appointed to a 3-year term.
8. Appointments for consecutive terms shall be limited to a maximum of two such terms except for the PSRSC member with health physics expertise.
9. Appointments shall be staggered with the intent that a maximum of three new members shall be added to the PSRSC each year.
10. Candidates for appointments to the PSRSC shall be selected with the advise and counsel of the Director or Acting Director of the reactor and the members of the PSRSC.

DETAILS OF MEETING

1. Meetings shall be called by the Chairman whenever an experiment is being delayed awaiting review.
2. The PSRSC shall meet quarterly ($\pm$ 25%) to review experiment records of the reactor and to discuss physical integrity of the core with the reactor staff.
3. Six members of the PSRSC shall constitute a quorum providing it includes representation of the reactor staff.

4. Records of any experiments shall be made available to the PSRSC for examination upon request.

Changes or amendments to the operating procedures of the PSRSC may be made by a two-thirds affirmative vote of the PSRSC membership subject to the approval of the Director or Acting Director of the reactor.

20. **Discuss the thermalhydraulics significance of arranging the PSBR fuel elements in a hexagonal pattern as indicated in your Section 3 of your SAR (Page III-3).**

The significance is not in the fact that the PSBR has a hexagonal pattern but that there exists a larger spacing between fuel elements for the PSBR than the standard TRIGA circular pattern. The metal-to-water (M/W) ratio for the hexagonal pattern is 2 whereas for the circular pattern the M/W is approximately 3.(1) In addition, and of importance is the fact that the central core position in the PSBR is filled with water whereas a fuel element occupies this core position in the standard TRIGA. After a LOCA, the larger spacing between fuel elements in the PSBR will result in more effective cooling by natural air circulation and result in lower fuel temperatures. This will be particularly significant when the water hole in the central core position is included.

(1) GA-4339, (Rev) "TRIGA Mark III Reactor Description", page 2-2, December 1963.

21. **On page IX-41 of your SAR, you reference a 1960 report (reference 20) for fission product radioactivity. Please compare the relevant data from that report with more recent data.**

The reference 20 (Katcoff, et.al.) referred to on page IX-41 provides the fission product release fractions and decay constants for analyzing the release of fission products to the environment. These data are convenient to use and are compared to the more recent data of P.M. Grant et.al., "Gamma Transition Energies of the Major Fissions Products," UCI Report No. 1971-1 (1971).

The data for I-135 is also compared with Duderstadt and Hamilton's value used in their text "Nuclear Reactor Analysis" John Wiley & Sons, Inc. (1976).

The iodine nuclides are the significant fission products and they are compared in the table below. The UCI report lists the cumulative yield of all nuclides having the same atomic mass; therefore, it is assumed that all nuclides decay into the iodine nuclide.

Nuclide	Yield % UCI	Yield % Ref. 20	Duderstadt	Half-Life UCI	Half-Life Ref. 20	Duderstadt
I-131	3.15	3.1		8.07d	8.041d	
I-132	4.45	4.38		2.34h	2.29h	
I-133	6.69	6.9		20.9h	20.8h	
I-134	7.09	7.8		52.m	52.6m	
I-135	6.3	6.1	6.386	6.72h	6.858h	6.70h

Of the above nuclides, I-131 and I-133 contribute over 90% of the total dose to the thyroid. For these two nuclides, the dose from Ref. 20 is low by 1.6% for I-131 and high by 3.1% for I-133. In general, the new data agrees with that given in Ref. 20 within $\pm$ 5%. This produces an insignificant error in the calculations.

22. **There are certain sections of your Technical Specifications (section 6, for example) that are not yet fully consistent with ANSI/ANS 15.1 (1982). Please compare, and include appropriate changes in your Technical Specifications.**

The Technical Specifications have been completely redacted to comply with ANSI/ANS 15.1 (1982). The new Technical Specifications are attached to this document.

23. **What is the minimum required distance from the top of the pool water level to the top (or bottom) of the core? What is the basis for this limit?**

The required distance from the top of the pool water level to the top of the core is a function of core power. The water is 23 ft. above the bottom of the core during normal operation. When the water level drops 26 cm (0.85 ft) below this height, i.e., approximately 22 ft. above the bottom of the core, an alarm sounds at the reactor console and at Police Services. If the water drops below this level, the allowed water height depends on the operating power of the reactor. At 1 MW the radiation monitor on the reactor bridge will scram the reactor before the water level drops below 21 ft. above the bottom of the core. When the reactor is shut down, the water level may be lowered to approximately 19.5 ft. above the bottom of the core to facilitate draining one side of the pool.

If a LOCA occurs while operating the reactor at 1 MW and the reactor is scrammed by the radiation monitor, it will take approximately 1250 sec to drain the pool from the 21 ft. above the bottom of the core. The present Safety Analysis Report gives 1360 sec for a height of 23 ft.; however, the effect on the safety analysis of shortening this time to 1250 sec is not significant.

24. **Please analyze the maximum step insertion available for the reactivity available to the reactor operator at a maximum licensed power level.**

If the reactor is operating at 1 MW, the fuel temperature reads 498 C. The maximum reactivity now available for pulsing is $3.30. If the reactor were to be pulsed by the sudden insertion of this $3.30 while operating at 1 MW, the fuel temperature would reach a maximum of 1147 C. This is to be compared with operating at 1.15 MW and pulsing the maximum allowed $3 reactivity producing a maximum fuel temperature of 1146 C.

If the reactivity inserted during a pulse is greater than $3.32 when the reactor is operating at 1 MW, the fuel temperature will exceed the 1150 C safety limit. Hence, the maximum allowed pulse reactivity allowed in the Technical Specifications is $3.30 (not $3.40).

25. **Your current Technical Specifications allow for the maximum total worth of all experiments at one time to be limited to $\leq$ $2.00. Justify the deletion of this limit from your new proposed Technical Specifications, or retain it.**

We are changing this Technical Specification to be consistent with the Safety Analysis and other sections of the Technical Specifications. The new Technical Specifications will allow a single secured experiment to have a maximum worth of $3.30 (2.31%) and the total absolute worth in the reactor of all experiments to be less than $3.70 (2.59%). These values are consistent with the allowed reactivity worth associated with the transient rod and permitted by the results of the Safety Analysis.

26. **What is the maximum steady power level possible with the transient rod inserted.**

The minimum worth of the transient rod is approximately $2.70. If the core had its full $7.00 of excess reactivity, this would leave a maximum of $4.30 reactivity remaining in the core. The maximum steady power level for $4.30 excess reactivity is approximately 1.3 MW.

27. **Please respond to the questions about your Operator Requalification Program given in Attachment A.**

Our operator and senior reactor operator requalification training program was approved by NRC in a letter dated 1971. In addition, our requalification program has been reviewed several times by NRC inspectors and found to be in compliance with regulations. We will, nevertheless, answer the questions to the best of our ability.

ATTACHMENT A

ATTACHMENT A

Request for Additional Information
Pennsylvania State University
Operator and Senior Operator
Requalification Training Program

SECTION C - WRITTEN EXAMINATION

C.1.a

This section of the program states only that "Principles of Reactor Operation," is one of the subject areas to be covered in a comprehensive written examination. Both 10 CFR 55, Appendix A and ANSI/ANS 15.4 state that course content shall include Nuclear Theory and Principle of Operation. It is not clear whether or not Nuclear Theory is included in licensee's course content.

Page 1 of our operator and senior operator qualification program shows that the written examination covers "Principles of Reactor Operation." The reactor is a nuclear device and could not be explained without including nuclear theory and principle of operation. This can be verified by reviewing the enclosed test as an example.

C.1.f

Licensee does not state what type of procedures are included in subject areas for examination. Both review documents require that normal, abnormal, and emergency procedures be part of the course content for requalification programs.

Refer again to page 1 of the operator and senior operator requalification where the written examination covers "Procedures, Technical Specifications and Government Regulations". This means that all procedures are covered (normal, abnormal, and emergency procedures). These procedures are in our procedures manual.

C.3

This section of the program description states that licensees receiving a grade of less than 70% (overall) shall be given an accelerated training program to remedy deficiencies. There is no indication of what form this accelerated program takes. Licensee should state how the accelerated program is carried out, e.g., lecture, tutoring, or self-study.

The accelerated program is carried out in a way that is best for the individual case. The accelerated program depends on the score the individual made. If he/she missed by a few points, self-study suffices; if he/she missed by many points, tutoring and lecturing may be required. The accelerated program is designed to provide the most effective corrective action.

SECTION E - ON-THE-JOB TRAINING

E.1

Although this section states that operators shall perform at least ten reactivity changes during the two year requalification program, and senior operators shall perform or supervise ten reactivity changes, there is nothing to indicate what is included in these reactivity changes. Both review documents require that these reactivity changes include startups, shutdowns, and significant reactivity changes. Licensee must include this information in the description of on-the-job training for the requalification program.

The licensee will include a description of "acceptable reactivity change," in the on-the-job training section of our requalification program.

ATTACHMENT B

Reactor File

THE PENNSYLVANIA STATE UNIVERSITY

INTER-OFFICE CORRESPONDENCE

Date: October 20, 1971

From: Warren F. Witzig WFW

To: Dr. Paul Althouse, Vice President

Via: Dean N. J. Palladino

To confirm our discussion this afternoon, I am requesting an appropriation of $33,500 to purchase and install equipment for water and gas handling equipment for the Breazeale Reactor. Since the AEC's announcement of a policy "of as low as practicable" for radioactive effluents, the University Health Physicist, the Reactor Director and I have been discussing ways of implementing such a policy. A copy of the proposed rule quantifying this policy, dated June 9, 1971, is attached.

While we recognize this proposed rule is for power reactors, the Universities' demonstrated concern for the environment leads us to feel that the University must meet these requirements, and perhaps even better them. With our present effluent handling equipment, it is uncertain that the University could carry out such a policy. To assure that the University can meet this future requirement, an expenditure of $33,500 is estimated and detailed on the second attachment. It is likely that an operator will be required for this equipment at the level of an engineering aide or technician upon completion of the installation of the equipment.

Knowing your general interest in our reactor operations, I have enclosed a copy of our staff's evaluation of the AEC's Division of Compliance most recent visit to the reactor.

WFW:rw
Enc.

cc: R. G. Cunningham, V. P. Research
F. J. Remick, Asst. to V. P. Research
R. W. Granlund, Health Physicist
S. H. Levine, Director, Reactor

THE PENNSYLVANIA STATE UNIVERSITY

INTER-OFFICE CORRESPONDENCE

Date: August 12, 1985

From: Charles L. Hosler, Vice President for Research C. L. Hosler

To: W. F. Witzig, F. J. Remick, Chairman of University Isotope Committee, and Chairman of the Safeguards Committee

This is to confirm that it is University policy to act in conformance with 10 CFR 20 in that in addition to complying with the requirements set forth in the regulations, the University will make every reasonable effort to maintain radiation exposures and releases of radioactive materials in effluents to unrestricted areas as low as is reasonably achievable (ALARA).

Please make sure that all activities dealing with radiation under your purview conform with this policy.

CLH:SHL/skr

REQUALIFICATION EXAM

Penn State Breazeale Reactor

17 November 1983

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C. General Operating Characteristics	20		
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F. Procedures, Tech Specs, and Gov't Regulations	20		
G. Radiation Control and Safety	20		
Total	140		

A. PRINCIPLES OF REACTOR OPERATIONS

(2) 1. Delayed neutrons:

a. Are insignificant as far as steady state reactor operation is concerned.

b. Come from the subsequent decay of certain fission products.

c. Shorten or decrease positive reactor periods.

d. Constitute a large percentage of the total neutron population.

e. None of the above.

(2) 2. Prompt neutrons:

a. Constitute a large percentage of the total neutron population.

b. Appear almost instantaneously after the fission event.

c. Cause sudden increases in linear and log power indications when they appear.

d. Cause short reactor periods when they appear.

e. All of the above.

(4) 3. Given a reactivity insertion of $0.25 into a critical reactor and a λ of 0.1 sec^{-1}, the resulting stable period is:

a. 80.6 sec   b. 3 sec   c. 30 sec

d. 300 sec   e. None of the above

(2) 4. As one inserts borated graphite control rods into the core:

a. The fission rate increases.

b. The fission rate decreases.

c. The external neutron source becomes larger.

d. The external neutron source becomes smaller.

e. None of the above.

(2) 5. Referring to the statement of Problem #4, the factor in the six-factor equation most affected is:

a. ε b. p c. P_f d. f e. None of the above

(2) 6. As one performs a critical experiment by systematically adding more fuel to a reactor core, the factor in the six-factor equation primarily affected i

a. ε b. p c. P_t d. η e. None of the above

(4) 7. If enough reactivity is inserted into a reactor which is critical at 100 watts to put it on a 30-second period, what will be the power in one minute?

a. 271.8 watts   b. 738.9 watts   c. 164.9 watts

b. 7389 watts   e. None of the above

(2) 8. During a pulse, the affect of delayed neutrons is:

a. To shorten reactor period.

b. To lengthen reactor period.

c. Even more dramatic than during steady state operation.

d. Is not even noted because of the relatively long time it takes them to appear.

e. None of the above.

B. Features of Facility Design

(2) 1. Water leakage from the reactor recirculation pump and heat exchanger primary pump goes to

a. Reactor pool

b. Underground waste hold-up tank

c. City sewer

d. Evaporator building waste hold-up tank

e. Processed water tank

(2) 2. The chemicals used in the regeneration of the reactor demineralizer are

a. H_2SO_4 (sulfuric acid) and HSO_3 (sulfonic acid)

b. NaOH (sodium hydroxide) and HSO_3 (sulfonic acid)

c. H_2SO_4 (sulfuric acid) and NaOH (sodium hydroxide)

d. HNO_3 (nitric acid) and NaOH (sodium hydroxide)

(2) 3. The visicon secondary pressure low alarm is a result of measuring differential pressure between which two points in the system?

a. secondary in, primary out

b. primary in, secondary out

c. primary in, primary out

d. primary out, secondary out

(2) 4. Which of the following does the Master I switch (Rabbit I) not do:

a. Permits fan operation by fan on switch

b. Opens electrically operated valve to supply gas to the system

c. Turns on audio to system

d. Turns on chart recorder that records output of RM-14 monitor

(3) 5. List the three types of filters in the reactor bay emergency exhaust system.

Pre (or roughing)

Absolute

Carbon

(4.5) 6. List the four ways to add water to the reactor pool.
(✓) Check the one you would use only as a last resort.

From processed water tank

Through surge tank in demineralizer room

City Water - floor drains

✓ MWHX - secondary side

(1) 7. The air line pressure to the transient rod system is usually 75-85 psig.

(1.5) 8. Match the following (Rabbit I System)

a	U-tube pressure release	a.	2 psig
c	Low gas pressure light	b.	1/4 psig
b	Pressure on containment box	c.	15 psig

(2) 9. List the types of gas used in the rabbit systems.

Rabbit I CO_2 Rabbit II Nitrogen

C. GENERAL OPERATING CHARACTERISTICS

(8) 1. Match a value from the right column with the appropriate function in the left column.

c Normal secondary inlet Temperature of the heat exchanger

b Peak flux one might expect to see during a \$2.75 pulse

e Peak fuel temperature one might expect to see during a \$2.75 pulse

a Flux one might expect to see in the central thimble

d Equilibrium design temperature for the pool with the heat exchanger running and the reactor at 1 MW

h Reactivity one might expect to use to go to 1 MW

g Reactivity one might expect to use to go from an average core temperature of 200°c to [illegible]00°c

f Reactivity one might expect to lose if the reactor rolled away from the thermal column

a. 3.2×10^{13} N/cm^{2}sec

b. 3.2×10^{16} N/cm^{2}sec

c. 55°F

d. 95°F

e. 400°C

f. \$0.70

g. \$2.00

h. \$3.50

(3) 2. Shortly after (~ 10 min) a shutdown from 1 MW operation, you set the pens on the recorder chart, you turn on the POWER ON switch, you turn the key switch to RESET and let it return to OPERATE, then you observe opposite indications from the linear power and countrate channels, i.e., linear power is decreasing and the countrate is increasing. These observations explained because:

a. Short half life precursors are gone but long half life precursors are being generated.

b. Short half life precursors are being generated but long half life precursors are gone.

c. Long half life precursors are still present and the fission chamber is driving down.

d. The core is cooling down and the fission chamber is driving down.

e. None of the above.

(2) 3. The purpose for having an Sb-Be source present in the core is:

a. To provide the first neutron to start the chain reaction as rods are withdrawn.

b. To insure that the most sensitive instrument channel always sees neutrons.

c. To make the flux higher at 1 MW operation.

d. Because Be is a good moderator.

e. None of the above.

(2) 4. The reason one goes to standby to load samples into or remove samples from the core is:

a. To allow more room in the core for samples because some rod has been removed.

b. To allow a greater shutdown margin because one now has $3.50 worth of rod to drop in the core.

c. To make the reactor system a little more sensitive to reactivity changes because it is closer to k = 1.

d. To get additional fuel into the core with the fueled followers beneath three of the control rods.

e. None of the above.

(2) 5. Which of the following cannot be done with the servo system (automatic):

a. Maintain power

b. Start-up

c. Increase power

d. Decrease power

(3) 6. The negative temperature coefficient is due in part to:

a. The $Zr\ H_{1.7}$ imparting energy to neutrons as the temperature of the core increases.

b. Higher energy neutrons being able to diffuse farther thus increasing the probability of leakage.

c. The resonance regions in U-238 broadening as the core temperature increases (Doppler).

d. All of the above.

D. INSTRUMENTATION AND CONTROL

(2) 1. On a Monday morning (lowest residual) you go to standby then quickly take the reactor slightly supercritical. With no further movement of the rods, you observe the period continuing to decrease (shorten). This observation is explained because:

a. The fission chamber is seeing an increase in the fission rate.

b. The % power G.I.C. is seeing an increase in the fission rate.

c. The log C.I.C. is overcompensated.

d. The log C.I.C. is under compensated.

e. The log C.I.C. is perfectly compensated.

(2) 2. To determine a period during a pulse one would:

a. Need to make a calculation.

b. Need to read the period meter.

c. Need to read the decade per minute meter.

d. Find it impossible to do.

e. None of the above

(2) 3. The source light comes on because:

a. The countrate is below 2 c/s

b. The countrate calibrate switch is off the operate position.

c. The Sb-Be source is out of the core on a Monday morning.

d. It could be indicating the failure of a component in the log countrate channel.

e. All of the above.

(4) 4. Fill in the following blanks with appropriate values and units:

Mode Selector Switch Setting	Indicating Pen	100% On the Recorder Chart Corresponds to
PULSE LO	Blue	400 MW
PULSE LO	Red	600 °C
PULSE HI	Blue	2000 MW
PULSE HI	Red	600 °C

(4) 5. The output of the period amplifier in the log power and period channel goes to four places. List them.

a. PERIOD METER

b. DECADE/MIN METER

c. PERIOD BISTABLE

d. AUTO-CONTROLLER

(4) 6. For the following approximate ranges, list the channel which provides the coverage:

Range	Channel
10^{-3} watts to 10^{2} watts	START-UP
10^{-2} watts to 10^{6} watts	LINEAR
0.3 watts to 10^{6} watts	LOG
10^{4} watts to 2×10^{9} watts	% POWER

(1) 7. How is the reactor kept from a period scram when the log power channel is calibrated on the HI position?

a. By turning the period bistable switch to TEST.

b. By switching to COUNTRATE RECORD.

c. By performing the calibration quickly.

d. By the period interlock on the output of the log amp.

e. None of the above.

(1) 8. How many positions on the REACTOR POWER switch?

a. 9 b. 11 c. 13 d. 15 e. None of the above.

E. Safety and Emergency Systems

(7) 1. List the instruments or devices that can cause the building evacuation horn to sound.

Bay East	Co^{60} BAY
Bay West	Reactor BHL
Air East	Evac. Button - console
Air West	

(6) 2. List the alarms sent to police services (excluding hot cell fire alarm and APB fuel storage room alarm).

UPS	Radiation
WTL	Pool Level
FPM (activity)	Intrusion

(4) 3. List the alarm points for the following:

a. Bridge east monitor 30 mR/hr

b. Bridge west monitor 200 mR/hr

c. Air east monitor 5000 CPM

d. Beam hole lab monitor 6 mR/hr

(3) 4. What is the scram indication (by name) on the reactor console for the following conditions:

External Scram a. Pushing the scram button on the reactor bay east wall

External Scram b. An alarm from the Air East monitor

High Voltage Scram c. Decrease or loss of core chamber voltage

F. Procedures, Tech Specs, and Gov't Regulations

(6) 1. Matching (definitions as per Tech Specs)

b	shutdown	a.	\$7.00
c	stuck rod criteria	b.	2% ΔK/K
h	maximum transient rod worth	c.	0.4% ΔK/K
e	enrichment of fuel	d.	17¢/sec
j	interlock functional check	e.	20%
f	wt % of fuel	f.	8.5%
K	linear power channel calibration	g.	1.5% ΔK/K
g.	semi-annual pulse insertion (ΔK/K)	h.	2.9% ΔK/K
i.	maximum pulse insertion	i.	2.38% ΔK/K
d.	maximum rate of rod reactivity insertion	j.	semi-annual
a.	maximum excess reactivity	k.	once a year
l.	maximum reactivity worth of movable samples	l.	1% ΔK/K

(3) 2. a. During operation of rabbit I, what two occurrences would cause you to scram the reactor?

system radiation alarm

stuck rabbit

b. What is the next action you should take after scramming?

turn off fan

(1) 3. What stairwell alarm forbids operating staff re-entry following a building evacuation alarm without consulting a member of the Health Physics staff?

air monitor alarm

(1) 4. Samples released in containers which are hand held (envelopes, etc.) should read < 10 mRem/hr in contact with a thin-walled detector.

(3) 5. a. Whenever the reactor is not shutdown, at least 3 persons must be present at the facility.

b. What are the license requirements for these persons?

1 SRO + 1 RO

c. The exception is the daily checkout procedure which may be done with what personnel?

1 SRO + any other person

(1) 6. In case of a fire at the facility, what phone number should you use to report the fire?

110

(3) 7. What control room indications or alarms occur if the pool level drops at least 26 cm below normal (assume drop in level is not sufficient to activate any alarms from radiation detectors).

Visicom pool level low

Visicom FPM flow

Indication on police services phone line

(2) 8. a. When starting the MWHX, which side do you bring up to operating flow rate and pressure first?

Secondary

b. When securing the MWHX, which side do you turn off first?

primary

G. Radiation Control and Safety

(2) 1. When releasing samples from the reactor pool, the health physicist uses an aluminum shield between the sample and the survey meter

a. to knock out any alphas present.

b. because the Xd^2/Γ relationship used to calculate the sample activity only holds for gammas.

c. to find out how many milli-roentgens per hour are due to betas and how many milli-roentgens per hour are due to gammas.

d. to eliminate secondary x-ray production.

(4.5) 2. Match the following:

c	Plane mono-directional source	a.	Rad
f	Point Source	b.	Quality facto
i	Unit of activity	c.	MPC
b	Unit of relative biological effectiveness	d.	Roentgen
h	Half-value thickness	e.	$I = Ioe^{-\mu x}$
a	Unit of dose	f.	$\phi = \frac{Soe^{-\mu x}}{4\pi r^2}$
d	Unit of exposure	g.	R/hr = 6CE
c	Concentration limit (μc/ml)	h.	$X_{1/2} = \frac{.693}{\mu}$
g	Relates curies to exposure rate at one foot	i.	Curie

(1) 3. This instrument should have its battery checked each week even though it is normally plugged into an AC circuit. Name it.

RM-14

(1) 4. This type of radiation interacts with matter indirectly by means of the photo-electric effect, compton effect, and pair production:

gamma

(1.5) 5. Match the following (in terms of the best shielding material):

c	neutrons (mixed energies)	a.	lead
b	betas	b.	aluminum
a	gammas	c.	poly-boron-lead

(3) 6. Match the following:

b	D_2O tank	a. argon-41
a	vertical tubes	b. tritium
c	fuel element rupture	c. iodine isoto
e	Monday morning alarm check procedure source	d. cesium-137
f	action of fast neutrons on oxygen	e. cobalt-60
d	.66 MeV check source in Fission Product Monitor	f. nitrogen-16

(2) 7. You suspect that there is transferrable contamination present on the floor in the reactor bay (assume reactor is at 1 mw). Which is the best way to check for this suspected transferrable contamination.

a. Check area with E-120 and pancake probe.

b. Smear area and hold smear under contamination detector along the bay west wall.

c. Check area with Victoreen 440.

d. Smear area and count smears in health physics lab using gas flow proportional counter.

(3) 8. Match the following:

a	During a 40-hour period, a radiation level of 2mRem/hr exists in a restricted area	a. no sign needed
b.	During a one-hour period, a radiation level of 10 mRem/hr exists in a restricted area	b. radiation area s
c.	A radiation level of 200 mRem/hr exists in a restricted area for at least one hour	c. high radiation area sign

(2) 9. a. Licensee activities cause a radiation level of > 2 mRem/hr to exist for one hour in an unrestricted area. Is this a licensee violation (Yes or No)? Yes

b. Licensee activities cause a radiation level of 0.5 mRem/hr to exist for seven consecutive days in an unrestricted area. Is this a licensee violation (Yes or No)? No

TECHNICAL SPECIFICATIONS

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TECHNICAL SPECIFICATIONS FOR THE PENN STATE BREAZEALE REACTOR (PSBR) FACILITY LICENSE NO. R-2

1.0 INTRODUCTION

Included in this document are the Technical Specifications and the bases for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, are included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1.1 DEFINITIONS

1.1.1 ALARA

The ALARA program (As Low As Reasonably Achievable) is a program for maintaining occupational exposures to radiation and release of radioactive effluents to the environs as low as reasonably achievable.

1.1.2 AUTOMATIC OPERATION

Automatic operation shall mean operation of the reactor with the mode selector switch in the automatic position. In this mode, the reactor operates under the control of the servo system.

1.1.3 CHANNEL

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

1.1.4 CHANNEL CALIBRATION

A channel calibration is an adjustment of the channel such that its output responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a Channel Test.

1.1.5 CHANNEL CHECK

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

1.1.6 CHANNEL TEST

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

1.1.7 COLD CRITICAL

The reactor is in the cold critical condition when it is critical with the fuel and bulk water temperatures both below 100°F (43.3°C).

1.1.8 CLOSE PACKED ARRAY

An arrangement of fuel elements wherein no empty grid positions are completely surrounded by fuel elements.

1.1.9 CONFINEMENT

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

1.1.10 CORE LATTICE POSITION

The core lattice position is that region in the core over a grid plate hole used to position a fuel element. It may be occupied by a fuel element, an experiment, an experimental facility, or a reflector element.

1.1.11 EXCESS REACTIVITY

Excess reactivity is that amount of reactivity that would exist if all control rods (control, regulating, etc.) were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff}=1$).

1.1.12 EXPERIMENT

Experiment shall mean (a) any apparatus, device, or material which is not a normal part of the core or experimental facilities, but which is inserted in these facilities or is in line with a beam of radiation originating from the reactor core; or (b) any operation designed to measure reactor parameters or characteristics.

1.1.13 EXPERIMENTAL FACILITY

Experimental facility shall mean beam port, including extension tube with shields, thermal column with shields, vertical tube, central thimble, in-core irradiation holder, pneumatic transfer system, and in-pool irradiation facility.

1.1.14 INSTRUMENTED ELEMENT

An instrumented element is a TRIGA fuel element in which sheathed chromel-alumel or equivalent thermocouples are embedded in the fuel.

1.1.15 LIMITING CONDITIONS FOR OPERATION

Limiting conditions for operation of the reactor are those administratively established constraints required for safe operation of the facility.

1.1.16 LIMITING SAFETY SYSTEM SETTING

A limiting safety system setting is a setting for an automatic protective device related to a variable having a significant safety function.

1.1.17 MANUAL OPERATION

Manual operation shall mean operation of the reactor with the mode selector switch in the steady state position which means that the power level is established by operator adjusting the control rod positions.

1.1.18 MEASURED VALUE

The measured value is the value of a parameter as it appears on the output of a channel.

1.1.19 MOVABLE EXPERIMENT

A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.1.20 NORMALIZED POWER (NP)

The normalized power, NP, is the ratio of the power of a fuel element to the average power per fuel element.

1.1.21 OPERABLE

Operable means a component or system is capable of performing its intended function.

1.1.22 OPERATING

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

1.1.23 PULSE MODE

Pulse mode operation shall mean operation of the reactor with the mode selector switch in a pulse position which allows the operator to insert preselected excess reactivity by the ejection of the transient rod.

1.1.24 REACTIVITY LIMITS

The reactivity limits are those limits imposed on reactor core excess reactivity. Quantities are referenced to a Reference Core Condition.

1.1.25 REACTIVITY WORTH OF AN EXPERIMENT

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter experiment position or configuration.

1.1.26 REACTOR INTERLOCK

A reactor interlock is a device which prevents some action, associated with reactor operation, until certain reactor operation conditions are satisfied.

1.1.27 REACTOR OPERATING

The reactor is operating whenever it is not secured or shutdown.

1.1.28 REACTOR SAFETY SYSTEMS

Reactor safety systems are those systems, including their associated input circuits, which are designed to initiate a reactor scram.

1.1.29 REACTIVITY LIMITS

The reactivity limits are those limits imposed on reactor core excess reactivity. Quantities are references to a Reference Core Condition.

1.1.30 REACTIVITY WORTH OF AN EXPERIMENT

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of

intended or anticipated changes or credible malfunctions that alter experiment position or configuration.

1.1.31 REACTOR SECURED

A reactor is secured when:

(1) It contains insufficient fissile material or moderator present in the reactor, adjacent experiments or control rods, to attain criticality under optimum available conditions of moderation and reflection, or
(2) A combination of the following:
   a. The minimum number of neutron absorbing control rods are fully inserted or other safety devices are in shutdown positions, as required by technical specifications, and
   b. The console key switch is in the off position and the key is removed from the lock, and
   c. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, and
   d. No experiments in or near the reactor are being moved or services that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment or one dollar whichever is smaller.

1.1.32 REACTOR SHUTDOWN

The reactor is shutdown if it is subcritical by at least one dollar in the Reference Core Condition and the reactivity worth of all experiments is accounted for.

1.1.33 REACTOR SAFETY SYSTEMS

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

1.1.34 REFERENCE CORE CONDITION

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<.30 dollars).

1.1.35 RESEARCH REACTOR

A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research, development, educational, training, or experimental purposes, and which may have provisions for the production of radioisotopes.

1.1.36 REPORTABLE OCCURRENCE

A reportable occurrence is any of the following which occurs during reactor operation:

a. Operation with the safety system setting less conservative than specified in Section 2.2, limiting safety system setting.

b. Operation in violation of a limiting condition for operation.

c. Failure of a required reactor safety system component which could render the system incapable of performing its intended safety function.

d. Any unanticipated or uncontrolled change in reactivity greater than one dollar.

e. An observed inadequacy in the implementation of either administrative or procedural controls which could result in operation of the reactor outside the limiting conditions for operation.

f. Release of fission products from a fuel element.

1.1.37 ROD-TRANSIENT

The transient rod is a control rod with scram capabilities that is capable of providing rapid excess reactivity insertion to produce a pulse or square wave.

1.1.38 SAFETY LIMIT

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. The principal physical barrier is the fuel element cladding.

1.1.39 SCRAM TIME

Scram time is the elapsed time between reaching a limiting safety system set point and a specified control rod movement.

1.1.40 SECURED EXPERIMENT

A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restaning forces must be substantially greater than those to which the experiment might be subjected by hydraulic, penumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

1.1.41 SECURED EXPERIMENT WITH MOVABLE PARTS

A secured experiment with movable parts is one that contains parts that are intended to be moved while the reactor is operating.

1.1.42 SHALL, SHOULD AND MAY

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

1.1.43 SHIM, REGULATING, SAFETY RODS

A shim, regulating, or safety rod is a control rod having an electric motor drive and scram capabilities. It has a fueled follower section.

1.1.44 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 operation condition although the most reactive rod is in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.1.45 SQUARE WAVE OPERATION

Square wave operation shall mean operation of the reactor with the mode selector switch in the square wave position which allows the operator to insert preselected excess reactivity by the ejection of the transient rod, and which results in a maximum power of 1 MW or less.

1.1.46 TRIGA FUEL ELEMENT

A TRIGA fuel element is a single TRIGA fuel rod of standard type, either 8.5 wt% U-ZrH in stainless steel cladding or 12 wt% U-ZrH in stainless steel cladding enriched to less than 20% uranium-235.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1 SAFETY LIMIT-FUEL ELEMENT TEMPERATURE

Applicability

The safety limit specification applies to the maximum temperature in the reactor fuel.

Objective

The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element and/or cladding will result.

Specifications

The temperature in a TRIGA fuel element shall not exceed 1150°C under any operating condition.

Basis

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured at a point within the fuel element. The measured fuel temperature is directly related to the maximum fuel temperature of the region. A loss in the integrity of the fuel element cladding could arise from a build-up of excessive pressure between the fuel-moderator and the cladding if the maximum fuel temperature exceeds 1150°C. The pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel-moderator. The magnitude of this pressure is determined by the fuel-moderator temperature, the ratio of hydrogen to zirconium in the alloy, and the rate change in the pressure.

The safety limit for the standard TRIGA fuel is based on data, including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding due to the increase in the hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided that the temperature of the fuel does not exceed 1150°C (2102°F) and the fuel cladding is water cooled. See Safety Analysis Report, Ref. 13 in section IX and Simnad, M.T., F.C. Faushie, and G.B. West, "Fuel Elements for Pulsed Reactors," Nucl. Technology, Vol.28, p. 31-56 (January 1976).

2.2 LIMITING SAFETY SYSTEM SETTING (LSSS)

Applicability

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

Objective

The objective is to prevent the safety limit (1150°C) from being reached.

Specifications

The limiting safety system setting shall be a maximum of 700°C as measured with a 12 wt% U-ZrH instrumented fuel element. The instrumented fuel element shall be located in the B-ring and adjacent to an empty fuel position when an empty fuel position exists in the B-ring.

Basis

The limiting safety system setting is a temperature which, if reached shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. Experiments and analyses described in the Safety Analysis Report, Chapter IX - Safety Evaluation, show that the measured fuel temperature at steady state power has a simple linear relationship to the normalized power or power of the highest powered fuel element in the core. Maximum fuel temperature occurs when a new 12 wt% U-ZrH fuel element is placed in the B-ring of the core. The measured fuel temperature during steady state operation is close to the maximum fuel temperature. Thus, 450°C of safety margin exists before the 1150°C safety limit is reached. This safety margin provides adequate compensation for using a depleted instrumented 12 wt% U-ZrH fuel element instead of unirradiated one to measure the fuel temperature. See Safety Analysis Report, section IX.

In the pulse mode of operation, the same limiting safety system setting shall apply. However, the temperature channel will have no effect on limiting the peak power generated, because of its relatively long time constant (seconds), 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 the "tail" of the power transient if the pulse rod remains stuck in the fully withdrawn position with enough reactivity to exceed the temperature-limiting safety system setting.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR CORE PARAMETERS

3.1.1 CONSTANT POWER OPERATION

Applicability

This specification applies to the maximum power generated during manual and automatic operation.

Objective

The objective is to assure that the safety limit (fuel temperature) will not be reached during manual and automatic operation by providing a set point to automatically limit the maximum fuel temperature produced in the core and to limit the energy produced in any seven (7) consecutive days to that used in the LOCA analysis in the Safety Analysis Report.

Specification

a. The operating power level of the reactor shall be limited to one megawatt.
b. The reactor shall not be operated to produce more than 70 megawatt hours of energy in any seven (7) consecutive days.

Basis

Thermal and hydraulic calculations and operational experience indicate that TRIGA fuel can be safely operated up to power levels of at least 1.15 megawatts with natural convective cooling. Power operation at 1.15 megawatts will not produce fuel temperatures which exceed 600°C using any allowed core configuration giving a large safety measure when the power of operation is limited to 1 MW. See Safety Analysis Report, section IX.

This specification limits the energy output of the PSBR to that used for the Safety Analysis Report analysis of a maximum Hypothetical Accident.

3.1.2 REACTIVITY LIMITATION

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worth of control rods, experiments, and experimental facilities. They apply for all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times and to assure that the safety limit will not be exceeded.

Specifications

1. The maximum excess reactivity above cold, clean, critical plus samarium poison of the core configuration with experiments and experimental facilities in place shall be 4.9% $\Delta k/k$ ($7.00).

Basis

Limiting the excess reactivity of the core to $7.00 prevents the fuel temperature in the core from exceeding 1150°C under any assumed

accident condition as described in the Safety Analysis Report, Chapter IX - Safety Evaluation.

3.1.3 SHUTDOWN MARGIN

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worth of control rods, experiments, and experimental facilities. They apply for all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times and to assure that the safety limit will not be exceeded.

Specifications

1. The reactor shall not be operated unless the shutdown margin provided by control rods is greater than 0.25 dollar with:

    a. All movable experiments, experiments with movable parts, and experimental facilities in their most reactive state.

    b. The highest reactivity worth control rod fully withdrawn.

    c. The reactor in the reference core condition.

Basis

The value of the shutdown margin assures that the reactor can be made subcritical from any operating condition even if the highest worth control rod should remain in the fully withdrawn position.

3.1.4 PULSE MODE OPERATION

Applicability

This specification applies to the energy generated in the reactor as a result of a pulse insertion of reactivity.

Objective

The objective is to assure that the safety limit will not be exceeded during pulse mode operation.

Specification

The stepped reactivity insertion for pulse operation shall not exceed 2.31% $\Delta k/k$ (\$3.30) and the maximum worth of the poison section of the transient rod shall be limited to 2.59% $\Delta k/k$ (\$3.70). When the core excess reactivity is less than 2.59% $\Delta k/k$ (\$3.70) the maximum worth of the poison section of the transient rod can be greater than 2.59% $\Delta k/k$ (\$3.70).

Basis

Experiments and analyses described in the SAR show that the peak pulse temperatures can be predicted for new 12 wt% fuel placed in the B-ring. These experiments and analyses show that the maximum allowed pulse reactivity of $3.30 prevents the maximum measured fuel temperature from reaching 700°C for any allowed core configuration.

The maximum worth of the pulse rod is limited to $3.70 to prevent exceeding the safety limit (1150°C). Accidental insertion of the transient rod during cold clean conditions will limit the maximum measured temperature to 720°C and the maximum fuel temperature to 1150°C. See Safety Analysis Report, section IX.

When the core excess reactivity is less than 2.59% $\Delta k/k$ ($3.70), a pulse of any magnitude will not produce temperatures that exceed the safety limit.

3.1.5 CORE CONFIGURATION LIMITATION

Applicability

This specification applies to a core configuration with water holes producing large power peaking in some fuel elements.

Objective

The objective is to assure that the safety limit will not be exceeded due to power peaking effects in the various core geometries.

Specifications

a. The critical core shall be an assembly of either 8.5 wt% stainless steel clad or a mixture of 8.5 wt% and 12 wt% stainless steel clad TRIGA fuel-moderator elements placed in water with a 1.7 inch center line grid spacing.

b. The fuel and fueled follower control rods shall be arranged in a close packed array except for single positions and core lattice positions the centers of which are greater than 4 inches from the center of the core where flux peaking and corresponding power densities produce fuel temperatures less than in the B-ring.

c. When the k_{eff} of the core is less than or equal to 0.99 with all control rods at their upper limit, the fuel need not be arranged in a close packed array. The source and detector shall be arranged such that the k_{eff} of the subcritical assembly shall always be measured and monitored to assure compliance with $k_{eff} \leq 0.99$ when all control rods are fully withdrawn from the core.

Basis

Calculations and experiments performed with the PSBR have shown that with only one empty fuel position in the central region of the core defined as lattice positions with centers less than 4 inches from the core center, the power peaking for any mixture of 12.0 wt% and 8.5 wt% fuel remains less than 23.2 kw per fuel element, i.e., an NP < 2.2 (see Safety Analysis Report, Chapter IX - Safety Evaluation). This automatically limits the maximum fuel temperature, which always occurs in the B-ring, to well below 700°C for any mode of operation. The maximum fuel temperature always occurs in a new 12 wt% fuel element in the B-ring adjacent to a water filled fuel position also in the B-ring.

When the k_{eff} of the core is less than 0.99 with all control rods at their upper limit, the core can not be taken critical. Hence, the requirement for close packed arrays is not necessary to prevent the core from attaining high fuel temperatures.

3.1.6 TRIGA FUEL ELEMENTS

Applicability

This specification applies to the mechanical condition of the fuel.

Objective

The objective is to assure that the reactor is not operated with damaged fuel.

Specification

The reactor shall not be operated with damaged fuel. A TRIGA fuel element shall be removed from the core if:

a. In measuring the transverse bend, the bend exceeds 0.125 inch over the length of the cladding.

b. In measuring the elongation, its length exceeds its original length by 0.125 inch.

1c. A clad defect exists as indicated by release of fission products.

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. This is because (1) during steady state operation, the

maximum fuel temperatures are several hundred degrees Centigrade below 1150°C (the safety limit), and (2) during a pulse, the cladding temperatures remain well below their stress limit due to the adiabatic nature of the fuel temperature rise. 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.

3.2 CONTROL AND SAFETY SYSTEM

3.2.1 REACTOR CONTROL RODS

Applicability

This specification applies to the reactor control rods.

Objective

The objective is to assure that sufficient control rods are operable to maintain the reactor subcritical.

Specification

There shall be a minimum of three operable control rods in the reactor core.

Basis

The shutdown margin and excess reactivity specifications assure that the reactor can be made subcritical with the most reactive control rod withdrawn.

3.2.2 MANUAL AND AUTOMATIC CONTROL

Applicability

This specification applies to the maximum reactivity insertion rate associated with movement of a standard control rod.

Objective

The objective is to assure that adequate control of the reactor can be maintained during manual and automatic operation.

Specification

The maximum rate of excess reactivity insertion associated with movement of either the regulating, shim, or safety control rod shall be no greater than 0.12% $\Delta k/k$ (17¢) per second.

Basis

This limits the insertion of excess reactivity to a rate much less than that during a pulse insertion of \$3.30 excess reactivity. At a maximum insertion rate of 17¢/sec it takes almost 6 seconds to insert \$1 of excess reactivity; the large negative temperature coefficient of the core (see page IX-29 of the SAR) limits the increase of the average core fuel temperature to 70°C. Thus, the core temperature will compensate for the rate of insertion of reactivity. In addition, the maximum fuel temperature will be the same as that during constant power operation, i.e., near the position of the thermocouple. Hence, when either the linear, percent power, or temperature scram occurs, the maximum fuel temperature will be far below the 1150°C safety limit.

3.2.3 REACTOR CONTROL SYSTEM

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 the measuring channels listed in Table 1 are operable.

Table 1
Measuring Channels

Measuring Channel	Min. No. Operable	Effective Mode SS	Pulse	SW
Fuel Element Temperature	1	X	X	X
Linear Power	1	X		X
Percent Power	1	X		X
Pulse Peak Power	1		X	
Count Rate	1	X		
Log Power	1	X		X

Basis

Fuel temperature displayed at the control console gives continuous information on this parameter which has a specified safety limit. The power level monitors assure that the reactor power level is adequately monitored for both steady state and pulsing modes of operation. The specifications on reactor power level indication are

included in this section since the maximum fuel temperature is related to the power level.

3.2.4 REACTOR SAFETY SYSTEM AND INTERLOCKS

Applicability

This specification applies to both the reactor safety system channels and the interlocks.

Objective

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

Specification

The reactor shall not be operated unless all of the channels and interlocks described in Table 2a and Table 2b are operable.

Table 2a

Minimum PSBR Safety Channels

Safety Channel	Number Operable	Function	Effective Mode SS	Pulse	SW
Fuel Temperature	1	SCRAM $\leq$ 700°C	X	X	X
High Power	2	SCRAM $\leq$ 115% of 1 MW	X		X
Detector Power Supply	1	SCRAM on failure of supply voltage	X		X
Scram Bar on Console	1	Manual scram	X	X	X
Preset Timer	1	Transient rod scram 15 seconds or less after pulse		X	

Table 2b

Minimum PSBR Safety Interlocks

Safety Interlocks	Number Operable	Function	Effective Mode SS	Pulse	SW
Source Level	1	Prevent rod withdrawal with less than two neutron induced counts per second on the startup channel	X		
Log Power	1	Prevent pulsing from levels above 1 kW		X	
Transient Rod	1	Prevent applications of air unless cylinder is fully inserted	X		
Shim, Safety, and Regulating Rod	1	Movement of any rod except transient rod		X	
Simultaneous Rod Withdrawal	1	Prevents simultaneous manual withdrawal of two rods	X		X

Basis

A temperature scram and two power level scrams provide protection to assure that the reactor can be shut down before the safety limit on the fuel element temperature will be exceeded. The manual scram allows the operator to shut down the system in any mode of operation if an unsafe or abnormal condition occurs. In the event of failure of the power supply for the safety chambers, operation of the reactor without adequate instrumentation is prevented. The preset timer insures that the reactor power level will reduce to a low level after pulsing.

In the pulse mode, movement of any rod except the transient rod is prevented by an interlock. This interlock action prevents addition of excess reactivity over that in the transient rod. The interlock to prevent startup of the reactor with less than 2 cps assures that sufficient neutrons are available for proper startup. The interlock to prevent the initiation of a pulse above 1 kW is to assure that the magnitude of the pulse will not cause the safety limit to be exceeded. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing the reactor in the steady state mode.

3.2.5 CORE LOADING AND UNLOADING OPERATION

Applicability

This specification applies to the period scram and the low count rate interlock.

Objective

The objective of this specification is to eliminate interference with fuel loading procedures.

Specification

During core loading and unloading operations when the reactor is subcritical, the period safety circuit may be momentarily defeated using a spring loading switch and/or the low count rate interlock may be defeated in accordance with the fuel loading procedure.

Basis

During core loading and unloading, the reactor is subcritical. Hence, failure of the period scram and low count rate interlock to perform will have no affect on the core behavior. Should the core become inadvertantly supercritical, the accidental insertion of excess reactivity will not allow fuel temeprature to exceed the 1150°C safety limit because no single TRIGA fuel element is worth more than 1% $\delta k/k$ in the most reactive core position.

3.2.6 SCRAM TIME

Applicability

This specification applies to the time required to fully insert any control rod to a full down position from a full up position.

Objective

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

Specification

The time from scram initiation to the full insertion of any control rod from a full up position shall be less than 1 second.

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.

3.3 COOLANT SYSTEM

3.3.1 Coolant Level Limits

Applicability

This specification applies to operation of the reactor with respect to depth of water above the bottom of the reactor core.

Objective

The objective is to assure that adequate water is present to provide adequate personnel shielding and core cooling when the reactor is operated.

Specification

The reactor will not be operated with less than 20 ft. of water above the bottom of the core.

Basis

Tests and calculations show that during a LOCA (see Chapter IX of the SAR), 20 ft. of water above the bottom of the core allows sufficient time (greater than 1000 sec) before the pool water level drops to the bottom of the core to prevent the fuel temperature from reaching 900°C and rupturing the cladding.

3.3.2 LEAK OR LOSS OF COOLANT DETECTION

Applicability

This specification applies to detecting a pool water loss.

Objective

The objective is to detect the loss of a significant amount of pool water.

Specification

An alarm will be activated and corrective action taken when the pool level drops 26cm from the normal operating position.

Basis

This alarm level provides time to initiate corrective action before the radiation from the core poses a serious hazard.

3.3.3 FISSION PRODUCT ACTIVITY

Applicability

This specification applies to the detection of fission product activity.

Objective

The objective is to assure that fission products from a leaking fuel element are detected to enhance safe operation of the reactor.

Specification

An air particulate monitor shall be operating in the reactor bay whenever the reactor is operating. An alarm on this unit will activate a building evacuation alarm.

Basis

This unit will be sensitive to particulate matter from decayed fission products and fission gases.

3.3.4 EMERGENCY POOL WATER SUPPLY

Applicability

This specification applies to emergency water supplies for the reactor pool.

Objective

The objective is to assure that a supply of water is available to replenish reactor pool water in the event of a pool water loss.

Specification

An emergency source of water of at least 100 GPM must be available either from the University water supply by diverting the heat exchanger secondary flow to the pool.

Basis

Provisions for both of these supplies are in place and will supply more than the specified flow rate.

3.3.5 COOLANT CONDUCTIVITY LIMITS

Applicability

This specification applies to the conductivity of the water in the pool.

Objective

The objective is:

a. To prevent activated contaminants from becoming a radiological hazard.

b. To help preclude corrosion of fuel cladding and other primary components.

Specification

The reactor shall not be operated if the conductivity of the water is greater than 5 micromhos/cm at the output of the purification system, averaged over one week.

Basis

Experience indicates that 5 micromhos/cm is an acceptable level of water contaminants in an aluminum/stainless steel system such as that at the PSBR. Based on experience activation at this level does not pose a significant radiological hazard.

3.3.6 COOLANT TEMPERATURE LIMITS

Applicability

This specification applies to the pool water temperature.

Objective

The objective is to maintain the pool water temperature at an acceptable level.

Specifications

An alarm shall annunciate and corrective action shall be taken if during operation the bulk pool water temperature reaches 100°F.

Basis

This specification is primarily to preserve demineralizer resins. Information available indicates that temperature damage will be minimal up to this temperature.

3.4 CONFINEMENT

Applicability

This specification applies to reactor bay doors.

Objective

To assure that no large air passages exist to the reactor bay during reactor operation.

Specification

The reactor bay truck door will be closed when the reactor is critical. Personnel doors to the reactor bay will not be blocked open when the reactor is critical.

Basis

This specification helps to assure that the air pressure in the reactor bay is lower than the remainder of the building and the outside air pressure.

3.5 ENGINEERED SAFETY FEATURE - FACILITY EXHAUST SYSTEM

Applicability

This specification applies to the operation of the facility exhaust system.

Objective

The objective is to mitigate the consequences of the release of radioactive materials resulting from reactor operation.

Specification

The facility exhaust system shall be maintained in an operable condition except for periods of time less than 48 hrs. when it is necessary to permit maintenance and repairs.

Basis

During normal operation, the concentration of airborne radioactivity in unrestricted areas is below MPC as described in the Safety Analysis Report, Section IX - Safety Evaluation. In the event of a substantial release of airborne radioactivity, an air radiation monitor and/or an area radiation monitor will sound a building evacuation alarm which will automatically cause the exhausted air to be passed through the system of filters before release. This reduces the radiation within the building. The filters will reduce by 90% all of the fission products except noble gases that escape to the atmosphere. Radiation monitors within the building independent of the facility exhaust system will give warning of high levels of radiation that might occur during operation with the facility exhaust system out of service.

3.6 EMERGENCY POWER

Not Applicable

3.7 RADIATION MONITORING SYSTEM

3.7.1 RADIATION MONITORING INFORMATION

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 are operable.

Table 3

Radiation Monitoring Channels

Radiation Monitoring Channels	Function	Number
Area Radiation Monitor	Monitor radiation levels in the reactor bay	1
Air Radiation Monitor	"	1

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.

3.7.2 EVACUATION ALARM

Applicability

This specification applies to the evacuation alarm which must be audible to personnel within the PSBR building when activated by the radiation monitoring channels in table 3 or a manual switch.

Objective

The objective is to assure that all personnel are alerted to evacuate the PSBR building when a potential radiation hazard exists within this building.

Specification

The reactor shall not be operated unless the evacuation alarm is operable.

Basis

The evacuation alarm produces a loud pulsating sound throughout the PSBR building when there is any impending or existing danger from radiation. The sound notifies all personnel within the PSBR building to evacuate the building as prescribed by the PSBR emergency procedures.

3.7.3 ARGON-41 DISCHARGE LIMIT

Applicability

This specification applies to the concentration of argon-41 that may be discharged from the PSBR.

Objective

To insure that the health and safety of the public is not endangered by the discharge of argon-41 from the PSBR.

Specification

The concentration in the unrestricted area of argon-41 in the effluent gas from the facility as diluted by atmospheric air due to the turbulent wake effect shall not exceed 4.0×10^{-8} μCi/ml averaged over one year in the unrestricted area.

Basis

The maximum allowable concentration of argon-41 in air in unrestricted areas as specified in Appendix B, Table II of 10 CFR 20 is 4.0×10^{-8} μCi/ml.

3.7.4 ALARA

Applicability

This specification applies to all reactor operations that could result in significant personnel exposures.

Objective

To maintain all exposures to ionizing radiation to the staff and the general public as low as reasonably achievable.

Specification

As part of the review of all operations, consideration shall be given to alternative operational profiles that might reduce staff exposures, release of radioactive materials to the environment, or both.

Basis

Experience has shown that experiments and operational requirements can, in many cases, be satisfied with a variety of combinations of facility options, core positions, power levels, time delays, and other modifying factors. Many of these can reduce radioactive effluents or staff radiation exposures. Similarly, overall reactor scheduling achieves significant reductions in staff exposures. Consequently, ALARA must be a part of both the overall reactor scheduling and the detailed experiment planning.

3.8 LIMITATIONS OF EXPERIMENTS

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities.

Objective

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

Specifications

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

a. The excess reactivity of a movable experiment and/or movable portions of a secured experiment plus the maximum allowed pulse excess reactivity shall be less than 2.59% $\delta k/k$ ($3.70). However, the reactivity of a movable experiment and/or movable portions of a secured experiment shall have a reactivity worth less than 2.1% $\Delta k/k$ ($3.00).

b. A single secured experiment will be limited to a maximum of 2.31% $\Delta k/k$ ($3.30). The sum of the reactivity worth of all experiments shall be less than 2.59% $\Delta k/k$ ($3.70).

c. When the k_{eff} of the core is less than 1 with all control rods at their upper limit and no experiments in or near the core,

then secured negative reactivity experiments can be added without limit.

d. Explosive materials in quantities greater than 250 milligrams shall not be allowed within the PSBR facility. Irradiation of explosive materials shall be restricted as follows:

   Explosive materials in quantities greater than 25 milligrams shall not be irradiated. 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.

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

   When calculating activity limits, the following assumptions shall be used:

   (1) If an experiment fails and releases radioactive gases or aerosols to the reactor bay or atmosphere, 100% of the gases or aerosols escapes.

   (2) 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.

   (3) 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 vapors can escape.

   (4) 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.

f. 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.

g. If a capsule fails and releases material which could damage the reactor fuel or structure by corrosion or other means, physical inspection 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 Director or

a designated alternate and determined to be satisfactory before operation of the reactor is resumed.

Basis

a. This specification limits the sum of the excess reactivities of a pulse and a movable experiment to the specified maximum excess reactivity of the transient rod. This limits the effects of a pulse simultaneous with the failure of the movable experiment to the effects analyzed for a 2.59% $\Delta k/k$ ($3.70) pulse.

The limit on a single movable experiment is specified to be less than the specified excess reactivity for a pulse, thus, limiting the effects of a failure of such an experiment to less than the effects analyzed for a permitted pulse.

b. The maximum worth of all experiments is limited to 2.59% $\Delta k/k$ ($3.70) so that their inadvertant suddent 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. The worth of a single secured experiment is limited to the allowed pulse excess reactivity insertion as an increased measure of safety.

c. Since the initial core is subcritical, adding and then inadvertantly removing all negative reactivity experiments leaves the core in its initial subcritical condition.

d. The failure of an experiment involving the irradiation of up to 25 milligrams of properly contained explosive material in a reactor irradiation facility will not result in damage to the reactor or the reactor pool containment structure.

This specification is also intended to prevent damage to vital equipment by restricting the quantity of explosive materials to less than 250 milligrams within the reactor building.

e. This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B Table II of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary.

f. 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 Appendix B of 10 CFR Part 20 for an unrestricted area.

g. Operation of the reactor with the reactor fuel or structure damaged is prohibited to avoid release of fission products.

4.0 SURVEILLANCE REQUIREMENTS

4.1.1 REACTOR POWER CALIBRATION

Applicability

This specification applies to the surveillance of the reactor power calibration.

Objective

The objective is to verify the performance and operability of the power measuring channel.

Specifications

A thermal power channel calibration shall be made on the linear power level monitoring channel annually.

Basis

The thermal power level channel calibration will assure that the reactor is to be operated at the authorized power levels.

4.1.2 REACTOR EXCESS REACTIVITY

Applicability

This specification applies to surveillance of core excess reactivity.

Objective

To assure that the reactor can be made subcritical at all times.

Specification

The excess reactivity of the core shall be measured annually and following significant core or control rod changes.

Basis

Excess reactivity measurements assure that the core configuration is the same with no unexpected reactivity changes.

4.1.3 TRIGA FUEL ELEMENTS

Applicability

This specification applies to the surveillance requirements for the TRIGA fuel elements.

Objective

The objective is to verify the continuing integrity of the fuel element cladding.

Specifications

All fuel elements and control rods with fuel followers shall be inspected visually for damage or deterioration and measured for length and bend before being placed in the core and at intervals not to exceed the sum of 3,500 dollars in pulse reactivity or two years, whichever comes first.

Basis

The frequency of inspection and measurement schedule is based on the parameters most likely to affect the fuel cladding of a pulsing reactor operated at moderate pulsing levels and utilizing fuel elements whose characteristics are well known.

4.2 REACTOR CONTROL AND SAFETY SYSTEM

4.2.1 REACTVITY WORTH

Applicability

This specification applies to the reactivity worth of the control rods.

Objective

To assure that the control rods are capable of maintaining the reactor subcritical.

Specification

The reactivity worth of each control rod and the shutdown margin for the core loading in use shall be determined annually and following significant core or control rod changes.

Basis

The reactivity worth of the control rod is measured to assure that the required shutdown margin is available and to provide an accurate means for determining the reactivity worth of experiments inserted in the core.

4.2.2 REACTIVITY INSERTION RATE

Applicability

This specification applies to control rod movement speed.

Objective

To assure that reactivity addition rate specification is not violated and that control rod drives are functioning.

Specification

The rod drive speed both up and donw and the time from scram initiation to the full insertion of any control rod from the full up position shall be measured annually and when any significant work is done on the rod drive or the rod.

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. It also assures that the maximum reactivity addition rate specification will not be exceeded.

4.2.3 REACTOR SAFETY SYSTEMS

Applicability

These specifications apply to the surveillance requirements for measurements, tests, and calibrations of the reactor safety systems.

Objective

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

Specifications

a. A channel test of the scram function of the high-flux safety channels shall be made on each day that the reactor is to be operated, or prior to each operation that extends more than one day.

b. Operability checks shall be performed daily on measuring channels.

Basis

TRIGA system components have proven operational reliability. Daily channel tests insure accurate scram functions and insure the detection of possible channel drift or other possible deterioration of operating characteristics. The channel checks will assure that the safety system channel scrams are operable on a daily basis or prior to an extended run.

4.2.4 REACTOR INTERLOCKS

Applicability

This specification applies to the surveillance requirements for the reactor control system interlocks.

Objective

To insure performance and operability of the reactor control system interlocks.

Specifications

a. A channel check of the source interlock shall be performed each day that the reactor is operated or prior to each operation that extends more than one day.

b. A channel test shall be performed semi-annually on the log power interlock which prevents pulsing from power levels higher than one kilowatt.

c. A channel test shall be performed semi-annually on the transient rod interlock which prevents application of air to the transient rod unless the cylinder is fully down.

d. A channel test shall be performed semi-annually on the rod drive interlock which prevents movement of any rod except the transient rod in pulse mode.

e. A channel test shall be performed semi-annually on the rod drive interlock which prevents simultaneous manual withdrawal of more than one rod.

Basis

These channel tests and checks will verify operation of the reactor interlock system. Experience at the PSBR indicates that the prescribed frequency is adequate to insure operability.

4.2.5 Overpower Scram

Applicability

This specification applies to the over power scram channels.

Objective

To verify that over power scram channels are operable.

Specification

The over power scrams shall be calibrated annually.

Basis

Experience indicates that this interval is adequate to assure operability.

4.2.6 Transient Rod Test

Applicability

This specification applies to surveillance of the transient rod mechanism.

Objective

To assure that the transient rod drive mechanism is maintained in an operable condition.

Specification

On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient (pulse) rod system shall be performed. The transient (pulse) rod drive cylinder and the associated air supply system shall be inspected, cleaned, and lubricated as necessary annually.

The reactor shall be pulsed annually to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value or the reactor shall not be pulsed until such comparative pulse measurements are performed.

Basis

Functional checks along with periodic maintenance assure repeatable performance. The reactor is pulsed at suitable intervals and a comparison made with previous similar pulses to determine if changes in fuel or core characteristics are taking place.

4.3 COOLANT SYSTEM

4.3.1 Fire Hose Inspection

Applicability

This specification applies to the dedicated fire hose used to supply water to the pool in an emergency.

Objective

To assure that these hoses are operable.

Specification

The two (2) dedicated fire hoses provides to supply water to the pool in an emergency will be visually inspected for damage and wear annually.

Basis

This frequency is adequate to assure that significant degradation has not occurred since the previous inspection.

4.3.2 Pool Water Temperature

Applicability

This specification applies to pool water temperature.

Objective

To limit pool water temperature.

Specification

The pool temperature alarm shall be calibrated annually.

Basis

Experience has shown this instrument to be drift-free and that this interval is adequate to assure operability.

4.3.3 Pool Water Conductivity

Applicability

This specification applies to surveillance of pool water conductivity.

Objective

To assure that pool water mineral content is maintained at an acceptable level.

Specification

Pool Water conductivity shall be measured and recorded daily.

Basis

Based on experience, observation at these intervals provides acceptable surveillance of limits that assure that fuel clad corrosion and neutron activation of dissolved materials will not occur.

4.3.4 Pool Water Level Alarm

Applicability

This specification applies to the surveillance requirements for the pool level alarm.

Objective

The objective is to verify the operability of the pool-water level alarm.

Specification

The pool-water level channel shall be channel checked weekly to assure its operability.

Basis

Experience has shown that weekly checks of the pool-water level alarm assures operability of the system during the week.

4.4 CONFINEMENT

4.4.1 REACTOR BAY DOORS

Applicability

This specification applies to reactor bay doors.

Objective

To assure that reactor bay doors are kept closed as per specification 3.4.1.

Specification

Doors to the reactor bay shall be locked or located within views of the PSBR staff.

Basis

Experience has indicated that with the current degree of surveillance, the doors in question are maintained such as to meet specification 3.4.1.

4.5 VENTILATION SYSTEM

4.5.1 EMERGENCY EXHAUST

Applicability

This specification applies to the building 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 weekly whenever operation is planned that the ventilation system is operable within its design specification.

Basis

Experience accumulated over a few years of operation has demonstrated that a test of the ventilation system on a weekly basis is sufficient to assure the proper operation of the system. This provides reasonable assurance on the control of the release of radioactive material.

4.6 EMERGENCY POWER

4.6.1 Inapplicable

4.7 RADIATION MONITORING SYSTEM AND EFFLUENTS

4.7.1 RADIATION MONITORING SYSTEM AND EVACUATION ALARM

Applicability

This specification applies to surveillance requirements for the area radiation monitor, the air radiation monitor, and the evacuation alarm.

Objective

The objective is to assure that the radiation monitors and evacuation alarm are operating and to verify the appropriate alarm settings.

Specification

The area radiation monitor, the air radiation monitor, and the evacuation alarm system shall be channel tested monthly. They shall be verified to be operable by a channel check daily when the reactor is to be operated, and shall be calibrated annually. Definition ANS 15.1.

Basis

Experience has shown this frequency of verification of area radiation monitor, air radiation monitor set points and operability and the evacuation alarm operability is adequate to correct for any variation in the system due to a change of operating characteristics. Annual channel calibration insures that units are within the specifications demanded by extent of use.

4.8 EXPERIMENTS

Applicability

This specification applies to surveillance requirements for experiments.

Objective

To assure that the conditions and restrictions of Specification 3.8 are met.

Specification

Those conditions and restrictions listed in specification 3.8 shall be considered by the PSBR authorized reviewer before signing the irradiation authorization for each experiment.

Basis

This specification has proven to prevent performance of unacceptable experiments in the past. The authorized reviewer is appointed by the facility director.

4.9 FACILITY SPECIFIC SURVEILLANCE

Inapplicable

5.0 DESIGN FEATURES

5.1 REACTOR FUEL

Specifications

The individual unirradiated TRIGA fuel elements shall have the following characteristics:

(1) The uranium content shall be a maximum of either 9.0 wt% or 12.7 wt% enriched to less than 20% uranium-235.

(2) The hydrogen-to-zirconium atom ratio (in the ZrH_x) shall be a nominal 1.65 H atoms to 1.0 Zr atom.

(3) The cladding shall be 304 stainless steel with a nominal 0.020 inch thickness.

5.2 REACTOR CORE

Specifications

a. The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator elements positioned in the reactor grid plate.

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

5.3 CONTROL RODS

Specification

a. The shim, safety, and regulating control rods shall have scram capability and contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding. These rods may incorporate fueled followers which have the same characteristics as the fuel region in which they are used.

b. The transient control rod shall have scram capability and contain borated graphite, B_4C powder, or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. When used as a transient rod, it shall have an adjustable upper limit to allow a variation of reactivity insertions. This rod may incorporate a voided or an aluminum follower.

5.4 FUEL STORAGE

Specifications

a. All fuel elements shall be stored in a geometrical array where the k_{eff} is less than 0.9 for all conditions of moderation.

b. 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 reach the safety limit as defined in Section 2.1 of the Technical Specifications.

5.5 REACTOR BAY AND EXHAUST SYSTEM

Specifications

a. The reactor shall be housed in a room (reactor bay) designed to restrict leakage. The minimum free volume in the reactor bay shall be 2500 m^3.

b. The reactor bay shall be equipped with two exhaust systems. Under normal operating conditions, one of these systems exhausts unfiltered reactor bay air to the environment releasing it at a point at least 30 feet above ground level. Upon initiation of a building evacuation alarm, the previously mentioned system is automatically secured and an emergency exhaust system automatically starts. The emergency exhaust system is designed to discharge reactor bay air at a point at least 30 feet above ground level after passing it through a 3-stage filter system.

5.6 REACTOR POOL WATER SYSTEMS

Specifications

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

b. A pool level alarm shall be activated if the pool level drops approximately 26 cm below operating level.

6.0 ADMINISTRATIVE CONTROLS

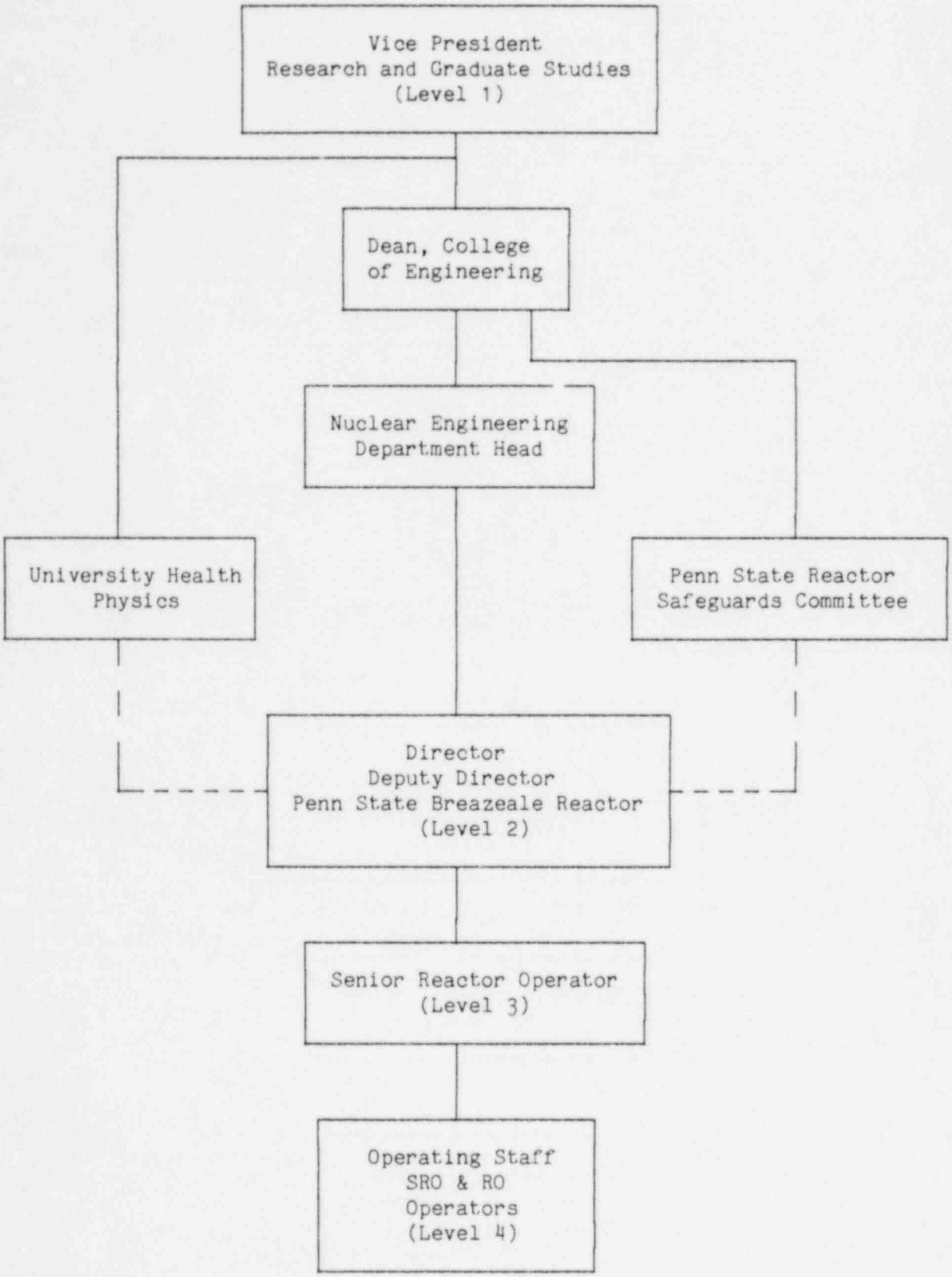
6.1 ORGANIZATION

6.1.1 STRUCTURE

The University Vice President of Research and Graduate Studies (level 1) has the rsponsibilities for the reactor facility license. The management of the facility is the responsibility of the Director and the Deputy Director (level 2), who report to the Vice President of Research and Graduate Studies through the Head of the Nuclear Engineering Department and the Dean of the College of Engineering. Administrative and fiscal responsibility is within the offices of the Department Head and the Dean.

The Director can at any time temporarily delegate his responsibility to the Deputy Director who can in-turn further delegate his responsibility to a qualified Senior Reactor Operator (level 3).

The reactor operators (level 4) report to the Senior Reactor Operator (level 5) for operational matters.



ORGANIZATION CHART

The University Health Physics reports directly to the Vice President of Research and Graduate Studies. The qualifications for the University Health Physicist position is the equivalent of a graduate degree in radiation protection, 3 to 5 years experience with a broad byproduct material license, and certification by The American Board of Health Physics or eligibility for certification.

6.1.2 RESPONSIBILITY

Responsibility for the safe operations of the reactor facility shall be within the chain of command shown in the organization chart. Individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license and technical specifications.

In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

6.1.3 STAFFING

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

a. A certified person present at the facility complex able to carry out prescribed written instructions.

b. A second person present at the facility complex able to carry out prescribed written instructions.

c. If a senior reactor operator is not present at the facility, one shall be "readily available on call".

(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. The list shall include:

a. Management personnel.

b. Radiation safety personnel.

c. Other operations personnel.

(3) Events requiring the direction of a Senior Reactor Operator:

a. All fuel or control-rod relocations within the reactor core region.

b. Relocation of any in-core experiment with a reactivity worth greater than one dollar.

c. Recovery from unplanned or unscheduled shutdown (in this instance, documented verbal concurrence from a Senior Reactor Operator is required).

6.1.4 SELECTION AND TRAINING OF PERSONNEL

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of the American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1977, Sections 4-6.

6.2 REVIEW AND AUDIT

6.2.1 SAFEGUARDS COMMITTEE COMPOSITION

A Penn State Reactor Safeguards Committee (PSRSC) shall exist to provide an independent review and audit of the safety aspects of reactor facility operations. The committee shall have a minimum of 5 members and shall collectively represent a broad spectrum of expertise in reactor technology and other science and engineering fields. The committee shall have at least one member with health physics expertise. The committee shall be appointed by and report to the Dean of the College of Engineering. The PSBR Director shall be an ex-officio member of the PSRSC.

6.2.2 CHARTER AND RULES

The operations of the PSRSC shall be in accordance with a written charter, including provisions for:

(1) Meeting frequency - not less than once per calendar year

(2) Quorums - at least one-half of the voting membership shall be present (the Director who is ex-officio shall not vote) and no more than one-half of the voting members present shall be members of the reactor staff.

(3) Use of Subgroups - the committee chairman can appoint ad-Hoc committees as deemed necessary.

(4) Minutes of the meetings - shall be recorded, disseminated, reviewed, and approved in a timely manner.

6.2.3 REVIEW FUNCTION

The following items shall be reviewed:

(1) Determinations that proposed changes in equipment, systems, test, experiments, or procedures do not involve an unreviewed safety question.

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

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

(4) Proposed changes in technical specifications, license, or charter.

(5) Violations of technical specifications, license, or charter. Violations of internal procedures or instructions having safety significance.

(6) Operating abnormalities having safety significance.

(7) Reportable occurrences listed in 6.6.2.

(8) Audit reports.

6.2.4 AUDIT

The audit function shall include selective (but comprehensive) examinations of operating records, logs, and other documents. Discussions with operating personnel and observation of operations should also be used as appropriate. Deficiencies uncovered that affect reactor safety shall immediately be reported to the Dean of the College of Engineering. The following items shall be audited:

(1) Facility operations for conformance to Technical Specifications license and procedures (at least once per calendar year with interval not to exceed 15 months).

(2) The requalification program for the operating staff (at least once every other calendar years with the interval not to exceed 30 months).

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

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

6.3 OPERATING PROCEDURES

Written procedures shall be reviewed and approved prior to the initiation of activities covered by them. The procedures shall be reviewed by the PSRSC and approved by level 2 management or designated alternates. Written procedures shall be adequate to assure the safe 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:

a. Startup, operation, and shutdown of the reactor.

b. Core loading, unloading, and movement withing the reactor.

c. Routine maintenance of major components of systems that could have an effect on reactor safety.

d. Surveillance tests and calibrations required by the technical specifications (including daily checkout procedure) or those that may have an effect on reactor safety.

e. Personnel radiation protection, consistent with applicable regulations (including radiation, evacuation, and alarm checks procedure and evacuation, loss of pool water, and gaseous release procedures)

f. Experiment evaluation and authorization.

g. Implementation of emergency and security plans.

6.4 EXPERIMENTS REVIEW AND APPROVAL

(1) All experiments which present an unreviewed safety questions shall be reviewed by the PSRSC and approved in writing by level 2 management or designated alternate prior to initiation.

(2) Substantive changes to previously approved experiments shall be made only after review and approval in writing by level 2 management or designated alternate.

6.5 REQUIRED ACTION

6.5.1 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

In the event the safety limit (1150°C) is exceeded:

a. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission.

b. The safety limit violation shall be promptly reported to Level 2 or designated alternates.

c. An immediate report of the occurrence shall be made to the Chairman, PSRSC and reports shall be made to the USNRC in accordance with Section 6.6 of these specifications.

d. 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 PSRSC for review.

6.5.2 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE

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

a. Reactor shall be returned to normal or shutdown. If it is necessary to shutdown the reactor to correct the occurence, operations shall not be resumed unless authorized by Level 2 or designated alternates.

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

c. The Director or a designated alternate shall notify the Nuclear Engineering Department Head who, in turn, will notify the office of the Dean of the College of Engineering and the office of the Vice President for Research and Graduate Studies.

d. The Director or a designated alternate shall notify the Chairman of the PSRSC.

e. A report shall be made to the PSRSC 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. This report shall be reviewed by the PSRSC at their next meeting.

f. A report shall be made to the USNRC in accordance with Section 6.5.1 of these specifications.

6.6 REPORTS

6.6.1 OPERATING REPORTS

An annual report shall be submitted to the USNRC including at least the following items:

(1) A narrative summary of reactor operating experience including the energy produced by the reactor and the hours the reactor was critical.

(2) The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.

(3) Tabulation of major preventive and corrective maintenance operations having safety significance.

(4) Tabulation of major changes in the reactor facility and procedures, and tabulation of new tests and experiments, that are significantly different from those performed previously and are not described in the Safety Analysis Report, including conclusions that no unreviewed safety questions were involved.

(5) A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the owner-operator as determined at or before the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25 percent of the concentration allowed or recommended, only a statement to this effect will be presented.

(6) A summarized result of environmental surveys performed outside the facility.

(7) A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed or recommended.

6.6.2 SPECIAL REPORTS

Special reports are used to report unplanned events as well as planned major facility and administrative changes. These special reports will contain and will be communicated as follows:

(1) There shall be a report no later than the following working day by telephone and confirmed in writing by telegraph or similar conveyance to USNRC to be followed by a written report that described the circumstances of the event within 14 days of any of the following:

a. Violation of safety limits (See 6.5.1)

b. Release of radioactivity from the site above allowed limits (See 6.5.2)

c. Special events including:

(i) Operation with actual safety-system settings for required systems less conservative than the limiting safety-system settings specified in the technical specifications.

(ii) Operation in violation of limiting conditions for operation established in the technical specifications unless prompt remedial action is taken.

(iii) A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns. (NOTE: Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

(iv) An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded.

(v) Abnormal and significant degradation in reactor fuel, cladding, coolant boundary, or containment boundary (excluding minor leaks), which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

(2) A written report within 30 days to the USNRC:

a. Permanent changes in the facility organization involving Level 1-2 personnel.

b. Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

6.7 RECORDS

To fulfill the requirements of applicable regulations, records and logs shall be prepared of at least the following items and retained:

6.7.1 RECORDS TO BE RETAINED FOR AT LEAST FIVE YEARS

a. Log of reactor operation, and summary of energy produced or hours the reactor was critical.

b. Checks and calibrations procedure file.

c. Preventive and corrective electronic maintenance log.

d. Major changes in the reactor facility and procedures.

e. Experiment authorization file including conclusions that no unreviewed safety question were involved for new tests or experiments.

f. Event evaluation forms (including unscheduled shutdowns), and reportable occurrence reports.

g. Preventive and corrective maintenance records of associated reactor equipment.

h. Facility radiation and contamination surveys.

i. Fuel inventories and transfers.

j. Surveillance activities as required by Tech Specs.

k. Records of PSRSC reviews and audits.

6.7.2 RECORDS TO BE RETAINED FOR AT LEAST ONE TRAINING CYCLE

a. Requalification records for licensed reactor operators and senior reactor operators.

6.7.3 RECORDS TO BE RETAINED FOR THE LIFE OF THE REACTOR FACILITY

a. Radiation exposure for all facility personnel and visitors.

b. Environmental surveys performed outside the facility.

c. Radioactive effluents released to the environs.

d. Drawings of the reactor facility including changes.