UNIVERSITY OF WISCONSIN
NUCLEAR REACTOR
LICENSE NO. R-74
DOCKET NO. 50-156

LICENSE RENEWAL APPLICATION
SAFETY ANALYSIS REPORT,
AND
TECHNICAL SPECIFICATIONS
REVISION 2

REDACTED VERSION*
SECURITY-RELATED INFORMATION REMOVED
*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

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

October 17, 2008 RSC 981

Director, Nuclear Reactor Regulations
ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: License Renewal of the University of Wisconsin Nuclear Reactor; License R-74, Docket 50-156

Dear Sir:

The attached information is submitted in support of our application for license renewal dated May 9, 2000, as supplemented on May 6, 2004 and September 2, 2004. As a result of the extensive changes, a complete replacement copy of the Safety Analysis Report for Renewal of License R-74 for the University of Wisconsin Nuclear Reactor is enclosed.

The changes described in the attachment are indicated by a vertical line in the margin of the enclosed revision 2 to the Safety Analysis Report for Renewal of License R-74 for the University of Wisconsin Nuclear Reactor, dated September 2008.

In accordance with 10 CFR 50.30(b), I declare under penalty of perjury that the foregoing is true and correct.

If you have any further questions, please contact me at (608) 262-3392.

Sincerely,

Robert J. Agasie
Reactor Director

Attachment
Enclosure

cc: Daniel Hughes, NRC Project Manager

A020
NRR

Section	Rev2 Page	Change	Reference
Chapter 1			
1.2	1-1	Fixed typo.	
1.3	1-2	Updated excess reactivity (4.9 to 4.3) and shutdown margin (4 to 4.2) to reflect current core measurements.	
1.3	1-2	Changed "TRIGA-FLIP High" to "TRIGA-FLIP," fixing typo.	
1.3	1-3	Fixed typos.	
1.3	1-6	Expanded "STD fuel" to "TRIGA Standard fuel" for clarification.	
1.4	1-6	Eliminated mention of sub-critical assembly and hotcell, since the hotcell is now contained within the restricted area and the subcritical assembly has been removed.	
1.4	1-6	Corrected facilities wording since now all HVAC is non-shared use except offices.	
1.6	1-7	Removed sentence about iodine and strontium-90 inventory limits; iodine limits are covered in Technical Specification 14.3.8.2, and limits on all fission products are covered in section 11.1.1.2, page 11-3, which states that all activity must be below 10 CFR 20 limits for effluent when assuming a months dilution with the ventilation system.	
1.8	1-8	Corrected initial criticality date from January 5th to March 26th 1961. It is not clear if anything significant occurred on January 5th, but it was not the initial criticality date as reported in chapter 4 of the SAR and as printed on the dedication plague displayed in the control room.	
1.8	1-8	Clarified that the initial replacement of 9 Standard fuel bundles with FLIP was for a mixed core.	
1.8	1-8	Fixed typo by adding "was initially critical in March 1974".	
1.8	1-8	Updated current MW Hours to over 20,000 (20,030 as of 9/16/2008) and added units of MWd.	
1.8	1-8	Updated most recent amendment number to 16 dated August 30, 2006.	
1.8	1-8,9	Updated facility modification history.	

Section	Rev2 Page	Change	Reference
Chapter 2			
2.1.1.1	2-1	Updated Madison population with 2000 census data (changed from 191,000 to 208,000).	
2.1.1.1	2-1	Added clarification that 130m distance to nearest residence is 80m from building wall to avoid confusion with LEU report analysis for ground release.	
2.1.1.2	2-1	Changed reference from Figure 2-4 to 2-5, typo.	
2.1.1.2	2-1	Updated reactor lab room number from 130 to 1215, and clarified the site boundary definition.	
2.1.1.2	2-1	Added reference to ME floor maps for 4th and 5th floors. Also added reference to new building cross section figures.	
2.1.2	2-2	Updated to 2000 census data. New reference has replaced original reference 1.	Response to NRC comment #4 (1)
2.1.2	2-2	Updated ME building occupancy estimate. The initial estimate was 1182 assuming all lecture halls, computer labs, and offices were fully occupied, but this does not account for laboratories (such as engine labs), or hallways/lobbies (such as with Engineering Expo). Therefore the initial estimate was rounded up to 1500. The previous value was 300.	
2.1.2	2-2	Updated current Camp Randal capacity from 76,129 to 80,321. Source: UW Badger Athletics website (http://www.uwbadgers.com/facilities/camp_randall/index_38.html).	
Figure 2-3	2-3	Updated Madison city map using Microsoft Virtual Earth. Distance circles had been requested by NRC.	
Figure 2-4	2-4	Updated campus map using www.map.wisc.edu.	
Figure 2-5	2-5	Updated engineering campus map using www.map.wisc.edu.	
Figure 2-6	2-6	Updated ME basement map.	
Figure 2-7	2-7	Updated ME first floor map.	
Figure 2-8	2-8	Updated ME second floor map.	
Figure 2-9	2-9	Updated ME third floor map.	
Figure 2-10	2-10	Added new ME fourth floor map.	

Section	Rev2 Page	Change	Reference
Figure 2-11	2-11	Added new ME fifth floor map.	
Figure 2-12	2-12	Added new ME cross-section map looking north.	
Figure 2-13	2-13	Added new ME cross-section map looking east.	
Table 2-1	2-14	Updated with 2000 census data. Corresponding reference 1 has also been updated.	Response to NRC comment #4 (1)
2.2.1	2-14	Updated National Guard annual flight missions from 1999 figure of approximately 3000 to 2006 figure of approximately 4000 (exact number is 8130 events).	
2.2.2	2-15	Updated 1999 traffic statistics with 2006 numbers for total events and commercial/general/military percentages. Also corrected typo with runway headings (310° should have been 320°).	
Figure 2-12	2-22	Updated Madison Well figure and made it full-page to make it easier to see.	
2.6	2-26	Updated reference 1.	Response to NRC comment #4 (1)
2.6	2-26	Replaced references 3 and 4 (national guard and airport telephone interviews) with new reference 3 (email interview with airport). All later references were renumbered and the corresponding footnote labeling in the body of the chapter was also renumbered, but these renumbering changes were not redlined.	
Chapter 3			
3.1	3-1	Added history of ME construction, new ventilation system and cooling system, design criteria.	
3.2	3-1	Updated 40 year history to 50.	
Chapter 4			
4.1.1	4-1	Updated excess reactivity (4.2 to 4.3) and shutdown margin (4 to 4.2) to reflect current core measurements.	
4.1.2	4-3	Fixed typos.	
4.1.2	4-5	Updated excess reactivity (4 to 4.3) and reactivity in control blades (6.9 to 7.1) to reflect current core measurements.	

Section	Rev2 Page	Change	Reference
4.1.3.#	4-5,6	Added outline numbering.	
4.2	4-7	Added reference to 10 CFR 73.2 for definition of formula quantity HEU.	
4.2.1	4-8	Removed reference to aluminum irradiation thimble in 3-element fuel bundle assemblies, as they are no longer used nor is there any plan to use them in the future.	
4.2.1	4-8	Corrected Figure 4-2 reference, fixing typo.	Response to NRC comment #7 (1)
4.2.2.#	4-13,14,15	Added outline numbering.	
4.2.2.1	4-13	Corrected Figure 4-2 reference, fixing typo.	Response to NRC comment #7 (1)
4.2.2.2	4-14	Clarified wording of safety blade section.	
4.2.5	4-20	Fixed typo.	
4.2.5	4-23	Added reference to Table 4-1 and 4-2 to explain codes in Figure 4-15.	
4.3	4-26	Updated 40 year history to 50.	
4.4	4-26	Included reference to section 13.1.3.2 for calculation of dose rates and integrated dose for 3rd floor occupant. Stated that with evacuation, public dose would be less than 100 mrem (about 13 mrem).	
4.4.#	4-27	Added outline numbering.	
4.5.1	4-27	Sections 4.5.1 and 4.5.2 were already combined, but outline numbering was revised to indicate section 4.5.1 rather than 4.5.1/2 for clarity.	
4.5.1.#	4-28,29,32, 34,37,40, 44,47,48	Added outline numbering.	
4.5.1.2.1	4-32	Fixed typo (reference to Figure 4-15)	
4.5.1.3.1	4-37	Fixed typo.	
4.5.1.3.2	4-43	Fixed typo.	
4.5.1.3.3	4-44,46,47	Fixed typos.	
4.5.2	4-51	Updated numbering from 4.5.3 to 4.5.2 (due to combination of previous sections)	

Section	Rev2 Page	Change	Reference
Chapter 5			
5.5	5-3,4	Split sections 5.5 and 5.6 (water cleanup and makeup sections) for clarity. Also added description of waste system to section 5.5. Previous Figure 5-2 has also been updated and split into Figures 5-2, 5-3, 5-4, and 5-5 to separate pool, cleanup, makeup, and waste systems.	
5.5	5-3,4	Updated flow rate and valve numbers for new demineralizer (18 gpm, valve 10102 service out, valve 22).	
Figure 5-2	5-5	Added new pool water systems schematic.	
Figure 5-3	5-6	Added new water cleanup system schematic.	
Figure 5-4	5-7	Added new water waste system schematic.	
Figure 5-5	5-8	Added new water makeup system schematic.	
Chapter 6			
6.2.1	6-1	Updated reference to figures to account for added 1st floor plan.	
6.2.1	6-1	Changed word roof to ceiling to clarify that the confinement ceiling is no longer the building roof.	
6.2.1	6-1	Eliminated mention of the console air conditioner and updated description to reflect new ventilation system.	
6.2.1	6-1,2	Updated lab description to reflect construction changes, elimination of windows, change of doors, etc. Security details of doors and window are not specified due to SGI. Changed name of basement area from "Nuclear Engineering Laboratory" to "Reactor Laboratory auxiliary support space." Deleted paragraph about future plans for room on north of lab.	
6.2.2	6-2	Corrected typo in outline numbering; 6.2.3 should have been 6.2.2.	
Figure 6-1	6-3	Updated lab basement floor plan.	
Figure 6-2	6-4	Added new lab first floor plan.	
Figure 6-3	6-5	Updated diagram of lab facing south.	
Figure 6-4	6-6	Updated diagram of lab facing north.	
Figure 6-5	6-7	Updated diagram of lab facing east.	

Section	Rev2 Page	Change	Reference
Figure 6-6	6-8	Updated diagram of lab facing west.	
Chapter 7			
7.1	7-1	Updated to include digital fuel temp, computer pulse, all digital recorder.	
Figure 7-1	7-2	Updated figure to reflect console upgrade.	RSC 773 (2) RSC 803 (3) RSC 887 (4)
7.2.3.#	7-3,4	Added outline numbering.	
7.2.3	7-3	Corrected references to bistables which should be relays.	
7.2.3.3	7-3	Clarified pulse output on console computer to reflect new pulse channel.	
7.2.3.5	7-4	Clarified console recorder.	
7.2.4	7-4	Updated history to 50 years.	
7.2.5	7-5	Removed reference to LogN not in operate scram which no longer exists.	RSC 887 (4)
7.3.#	7-5,6,8,9	Added outline numbering.	
7.3.3	7-5	Corrected typo, “PULSE” position to “SQUARE WAVE” position.	
7.3.4	7-6	Clarified pulse output on console computer to reflect new pulse channel.	
7.4	7-10	Removed reference to LogN not in operate scram which was removed.	RSC 887 (4)
7.4	7-10	Clarified that pool level alarm is on high or low.	
Figure 7-2	7-11	Updated per console upgrade.	RSC 887 (4)
Figure 7-3	7-12	Updated per console upgrade.	RSC 887 (4)
7.6.#	7-13,14,15	Added outline numbering.	
7.6.1	7-13	Added note that area radiation level high alarms at UWPD and initiates evacuation.	
7.6.1	7-13	Added Evacuation Alarm in Local annunciator.	RSC 893 (5)
7.6.1	7-13	Updated SAM/CAM trouble annunciator.	RSC 896 (6)
7.6.1	7-13	Added Loss of Off-Site Power annunciator.	RSC 856 (7)
7.6.1	7-14	Removed pn blower annunciator.	RSC 857 (8)
7.6.1	7-14	Updated to reflect UWPD name change.	
7.6.3	7-14	Updated pneumatic system panel description.	RSC 857 (8)
7.6.4	7-15	Updated ventilation system panel description, including new BP&TC EF-17.	RSC 879 (9)

Section	Rev2 Page	Change	Reference
7.6.5	7-15	Added reference to chapter 5 for details of cooling system.	
7.6.6	7-15	Added whale system panel description for consistency.	
7.7.#	7-15,16	Added outline numbering.	
7.7.2	7-16	Updated stack air monitor description.	RSC 896 (6)
Chapter 8	All	This entire chapter has been rewritten to reflect changes due to ME building construction, going into greater detail than previously. Also added electrical drawings.	
8.2	8-3	Added paragraphs describing UPS per facility mod, but also expanded the last sentence to reiterate that the UPS is not required "for maintaining the facility in safe shutdown, even for extended periods of time." Typo in RSC document referring to 6000 kVA has been corrected to 6000 VA.	RSC 895 (10)
Chapter 9			
9.1	9-1,2,3,4	Rewrote ventilation system description. This is a complete rewrite which is based on the RSC description.	RSC 879 (9)
9.1.#	9-1,2,3,4	Added outline numbering.	
Figure 9-1	9-5	Updated for new vent system.	RSC 879 (9)
Figure 9-2	9-6	Added new figure 9-2 for cross-section ventilation layout.	
Figure 9-3	9-7	Added new figure 9-3 for cross-section ventilation layout.	
Figure 9-4	9-8	Revised fuel rack diagram to reflect actual locations, and added North legend.	
9.2.2	9-9	Deleted potentially sensitive information on un-irradiated fuel storage.	
9.2.3	9-10	Corrected typo.	
9.2.3	9-10	Added new dummy element with reduction AND increase in diameter.	RSC 869 (11) RSC 919 (12)
9.3	9-13	Updated fire systems.	RSC 894 (13)
9.3	9-13	Updated P&S to UWPD notation.	
9.4	9-13	Changed wording to auxiliary support space for consistency.	
9.4	9-13	Removed description of 2nd intercom system which was uninstalled.	

Section	Rev2 Page	Change	Reference
9.5	9-13	Removed old rooms, updated lab room number to 1215, basement area room number to B1215, and added B1135.	
	9-14	Updated Agreement State license number.	
Chapter 10			
10.2.#	10-1,3,6,8	Added outline numbering.	
10.2.2	10-3	Revised "beams" to "beams of radiation" for clarification.	
Figure 10-2	10-3	Updated Figure 10-2 to include core box and actually label TC and beam ports.	
10.2.3	10-6,7	Updated for new pneumatic system and modified to reflect the basement move.	RSC 857 (8)
10.2.3	10-6	Clarified wording of activity limits.	RSC 879 (9)
Figure 10-4	10-7	Updated for new pneumatic system.	RSC 857 (8)
10.2.4	10-8	Deleted description of old smaller hydraulic irradiation facility.	
10.3	10-11	Changed "if released with 30 days of dilution" to "when averaged over 30 days of dilution".	
10.3	10-11	Removed outdated reference to 130 vs 131 activity limits (they are now the same).	RSC 879 (9)
10.3	10-11	Changed 1 SRO to 2 SROs to reflect new commitment to RSC.	RSC Charter (14)
10.3	10-11	Fixed gender language.	
Chapter 11			
11	11-1	Updated Agreement State license number.	
11	11-1	Updated reference to radiation safety regulations to remove url (since this has changed and may change again).	
11.1.1.1.#	11-1,2	Added outline numbering.	
11.1.1.1.2	11-2	Updated Ar-41 release calculations for new vent system.	RSC 879 (9)
11.1.1.1.2	11-3	Deleted redundant sentence that volatile or powder would be of importance.	
11.1.1.1.2	11-3,4	Updated dilution assumptions for new vent system.	RSC 879 (9)
11.1.1.1.2	11-3	Updated descriptions for single fume hood.	
11.1.1.1.2	11-4	Clarified wording on RSC approval of non-routine samples.	RSC 879 (9)
11.1.2/3	11-5	Added note that these 2 sections are combined.	

Section	Rev2 Page	Change	Reference
11.1.5	11-5	Changed students to experimenters to reflect that we will no longer TLD badge all students.	RSC 922 [15]
11.1.5	11-5,6	Paragraph about high radiation levels in experiments was revised to reference access control and postings in accordance with 10 CFR 20.1601.	
11.1.5	11-6	Tour/visitor dose paragraph was revised to reflect current use of non-radiation worker classification (based on University Radiation Safety Regulations).	
11.2.2	11-7	Added radioactive sink to list of liquid wastes to holdup tank.	
11.2.2	11-7	Included parenthetical reference to short description of waste holdup tank in section 5.5.	
11.2.3	11-7	The filter size for liquid waste discharge was supposed to be 0.5 micron (0.4 micron was a typo). The UWNR has always used 0.5 micron filters, as approved by the Reactor Safety Committee. 10 CFR 20.2003 says that waste must be readily soluble. NUREG-1556 Vol. 7 provides clarification of "readily soluble" by referencing Information Notice 94-07, but there is no guidance on what size filters to use to remove any potential dissolved solids. UW Radiation Safety was consulted and stated that 0.5 microns is the common filter size used to remove any possible dissolved solids in order to comply with 10 CFR 20.2003.	
11.3	11-7	Removed website location from reference 1, since this has changed and may change again in the future.	
Chapter 12			
12.1.2.#	12-1,2,3	Added outline numbering.	
12.1.2.2	12-1	Updated department name that Radiation Safety is part of (now University Department of Environment, Health and Safety).	
12.1.2.3	12-1	Added responsibilities of Department Chair. ANS 15.4 provides very little detail for any of the levels. Level 1 just says "Individual responsible for the reactor facility's licenses or charter." In our case the chair is also responsible for appointing the reactor director.	
12.1.2.6	12-3	Fixed gender language.	

Section	Rev2 Page	Change	Reference
12.1.3	12-4	Corrected outline numbering typo.	
12.1.4	12-4,5	Revised description to reflect current Operator Candidacy Program. Also spelled out OJT as On the Job Training for clarification, and removed course url in case of future change.	
12.1.5	12-5	Updated department name that Radiation Safety is part of (now University Department of Environment, Health and Safety).	
12.2.1	12-7	Updated Radiation Safety name.	
12.2.3	12-7	Fixed gender language.	
12.2.4	12-8	Updated Radiation Safety name.	
12.3	12-9	Changed 1 SRO to 2 SROs required for temporary procedure changes, and fixed gender language.	
12.4	12-10	Fixed gender language.	
Chapter 13			
13.1	13-1,2,3,6,8-12,14-16	Sections 13.1 and 13.2 were previously combined, but the outline numbering has now been revised to eliminate 13.1/2 notation. Also added note to clarify combination of sections.	
13.1.1.#	13-2,3,6	Added outline numbering.	
13.1.1.4	13-3	Added clarification on assumed evacuation time (5 minutes to exit confinement, another 5 minutes to exit building). Also added reference to Appendix A for clarification, and added sub-section numbering.	
13.1.1.4	13-3	Revised self-contained breathing apparatus to powered air purifying respirator.	
Table 13.1	13-4	Updated H and J values for new vent system.	RSC 879 (9)
13.1.1.5	13-6,7	Updated for new vent system and made minor clarifications.	RSC 879 (9)
13.1.2.#	13-9,10	Added outline numbering.	
13.1.3.#	13-11,12,14	Added outline numbering.	

Section	Rev2 Page	Change	Reference
13.1.3.1	13-11	Updated pool drain time calculations. There is no record of how the previous calculation was performed, but current calculation uses reasonable methods and assumptions with references.	
13.1.3.2/3	13-12-14	Updated unshielded core dose calculations. The calculated 3rd floor classroom dose-rates were significant, so an additional analysis was performed to model the integrated dose received to ensure it would be less than 100mrem during the evacuation. Integrated dose was calculated to be about 13 mrem. This section was also divided into 13.1.3.2 for confinement doses, and 13.1.3.3 for unrestricted area doses.	
13.1.6	13-15	Fixed typo; missing section reference to 13.1.1	
13.1.8	13-15	Changed "incredible" to "not credible" for clarification.	
13.1.9	13-16,17	Updated reference to Figure 2-12 (was Figure 2.10) and deleted reference to drain thimble used for cooling tower blowdown (no longer exists). Also corrected typo referring to case 2 when it should be case 3.	
13.2/3	13-17	Updated section number from 13.3 to 13.2 and from 13.4 to 13.3 because of combination of sections 13.1/2 into 13.1.	
13.3	13-18	Added new reference 9 for the pool drain equation used in section 13.1.3.1	
Chapter 14	All	All outline numbering was revised to include a "TS" Prefix to avoid confusion with earlier chapters of the SAR.	
1.1	14-1	Updated summary to reflect elimination of Rev. 0 redlining.	
1.3.2	14-5	Added new definition for non-secured experiment (just opposite of secured experiment which was already defined).	
2.2	14-10,11	Clarified between first and second LSSS in basis.	
3.1.2	14-13,14	Corrected control rod to control element.	
3.2.5	14-22	Clarified wording for interlocks preventing application of air to fire transient rod.	
3.4	14-26	Renamed section heading to "Confinement."	Response to NRC comment #11 (1)

Section	Rev2 Page	Change	Reference
3.4	14-27	Updated stack height from 17m to 26.5m, and corresponding reduction fraction from 10 to 2.6.	
3.7.2	14-29	Updated release concentration and dilution fraction for new vent system.	RSC 879 (9)
3.8.1	14-30	Clarified limits on experiment reactivity are for a single experiment.	
3.8.2	14-31	Revised outline numbering for consistency.	
3.8.2	14-31	Updated dilution fraction for new vent system.	RSC 879 (9)
3.8.2	14-32	Changed formatting of basis list for clarification only.	
3.8.3	14-32	Clarified that the experimental capsule would be removed and inspected if failure occurs (fuel may be inspected if warranted, but would not be removed).	
4.2	14-36	Clarified wording.	Response to NRC comment #14 (1)
4.2	14-37	Updated 40 year history to 50 years.	
4.4	14-38	Renamed section heading to "Confinement."	Response to NRC comment #11 (1)
4.5	14-39	Revised the basis for quarterly checks on ventilation system.	
5.1	14-41	Changed stack height from 17m to 26.5m.	RSC 879 (9)
5.2	14-41	Deleted outlet pipe from 15 foot requirement (outlet pipe requirement is specified in the next sentence).	
5.3	14-42	Revised outline numbering for consistency.	
6.1.1	14-45	Added reference to new Figure 14-1 (just copy of Figure 12-1).	
6.1.2	14-45	Fixed gender language.	
Figure 14-1	14-46	Added new figure for clarification (copy of Figure 12-1).	
6.1.3	14-47	Clarified wording in sub-section 1-b.	
6.2.1	14-48	Revised list of required qualifications for RSC.	
6.3	14-49	Clarified authority of state license and updated license number.	
6.4	14-50	Fixed gender language.	
6.5(2)	14-51	Fixed gender language and clarified wording.	
6.6.2	14-52	Fixed gender language.	
6.7.1	14-53,54	Reordered sub-sections under item 1 and revised outline numbering for consistency.	

Section	Rev2 Page	Change	Reference
Appendix A		Revised headings and order of entire Appendix to make it clear which sections of the SAR the calculations are supporting.	
	A-1	Updated stack height from 17.1 to 26.5m.	RSC 879 (9)
	A-1	Updated equation 2 to reflect new stack height.	RSC 879 (9)
	A-2	Updated equation 5 to reflect new building cross-section (12,200ft^2)	RSC 879 (9)
	A-2	Updated equation 6 to reflect changes in equation 5.	RSC 879 (9)
	A-2	Added number for equation 7 (previously not numbered).	
	A-2	Added number for equation 8 (previously not numbered).	
	A-2	Added number for equation 9 (previously not numbered).	
	A-3	Added number for equation 10 (previously not numbered).	
	A-3	Added number for equation 11 (previously not numbered).	
	A-3	Renumbered equation 9 to equation 12, and updated equation to reflect flow-rate changing from 1000 scfm to 2700 scfm (also clarified the unit conversion).	RSC 879 (9)
	A-3	Renumbered equation 10 to equation 13, and updated equation to reflect changes in equations 2 and 12.	RSC 879 (9)
	A-4	Renumbered equation 11 to equation 14, and updated equation to reflect changes in equations 6 and 12.	RSC 879 (9)
	A-4	Updated factor from 10.17 to 2.6 because of new vent system.	RSC 879 (9)
	A-4	Deleted and replaced sentence referring to building wake effects vs. reactor lab wake effects being a 10 fold difference. From now on all wake effects are calculated using the new ME building dimensions (12,200 ft^2) since the reactor lab is entirely enclosed within the ME building.	RSC 879 (9)
	A-4	Clarified intermediate calculation of stack outlet concentration, and clarified that equation 15 (which uses equation 2) is assuming stack height.	
	A-4	Renumbered equation 7 to equation 15, and updated equation to reflect changes in equation 2.	RSC 879 (9)

Section	Rev2 Page	Change	Reference
	A-4	Deleted calculation of discharge concentration using only reactor lab building cross-section (in support of section 11.1.1.1), now only calculates using entire ME building cross-section which is more appropriate.	RSC 879 [9]
	A-4	Clarified that equation 16 is for zero stack height with building wake effect.	RSC 879 [9]
	A-4	Renumbered equation 9 to equation 16, and updated to reflect changes in equation 6.	RSC 879 [9]
	A-4	Updated final sentence because now only 2 calculations are done, not 3 (because the reactor-lab-only wake effect calculation was removed). Also updated reference to equation number 15 being more realistic, but added that the more conservative equation 16 was used in text.	RSC 879 [9]

References

(1) RSC 812, "NRC Request for Additional Information", May 6, 2004.
(2) RSC 773, "Reactor Pulse Channel Digital Recorder Replacement", March 3, 2003.
(3) RSC 803, "Replace Console Recorder with Honeywell Multitrend Paperless Recorder", March 12, 2004.
(4) RSC 887, "Install New Neutron Measuring Channels; LCR, LogN, and Picoammeters", May 1, 2006.
(5) RSC 893, "Installation of New Building Reactor Evacuation Alarm System", August 4, 2006.
(6) RSC 896, "Installation of New Reactor Stack Air and Continuous Air Monitors", August 4, 2006.
(7) RSC 856, "Loss of Alternating Current SCRAM", October 31, 2005.
(8) RSC 857, "Pneumatic Tube Replacement", October 31, 2005.
(9) RSC 879, "Modification to Reactor Ventilation System", March 21, 2006.
(10) RSC 895, "Installation of Uninterruptible Power Supply", August 4, 2006.
(11) RSC 869, "Report on 2005-2006 Annual Maintenance Activities", February 14, 2006.
(12) RSC 919, "Report on 2006-2007 Annual Maintenance Activities", February 1, 2007.
(13) RSC 894, "Installation of New Fire Detection and Suppression System", August 4, 2006.
(14) RSC Charter, as revised on December 6, 2006. See also RSCM-72, "Minutes of the UWNR Safety Committee Meeting, December 6, 2006".
(15) RSC 922, "Dosimetry for Students in NE 234, 427 and 428", March 14, 2007.

SAFETY ANALYSIS REPORT

FOR RENEWAL OF LICENSE R-74

FOR THE

UNIVERSITY OF WISCONSIN NUCLEAR REACTOR

Rev 2, September 2008

REVISION HISTORY

Rev 2, September 2008

This revision primarily consists of changes related to the remodeling of the Mechanical Engineering Building. New floor plans and room layouts were described. New equipment installed included the ventilation system, demineralizer, and the pneumatic tube system. The console instrumentation was significantly upgraded. Because this revision impacted so much of the document, the opportunity was taken to make many minor corrections and clarifications. As a result, this revision was issued as a complete replacement to Rev 1, since over 95% of pages were updated. The index was removed and will no longer be maintained. All changes from Rev 1 are indicated with a vertical line in the margin for each line changed.

Rev 1, August 2004

The reasons for this revision were a new reactor pool water cooling system, and a new computer-based pulse channel. Chapters 5 and 7 were completely revised, and the Table of Contents and Index were completely revised accordingly. All changes from Rev 0 were indicated with a vertical line in the margin for each line changed. Change indications in Rev 0 (Chapter 14) have been removed.

Rev 0, April 2000

This was a major revision issued as part of the application for license renewal in 2000. Sections were reorganized to conform with the format guidelines in NUREG-1537, but no substantial changes were made to the content of the SAR with the exception of the Technical Specifications (Chapter 14). These changes were indicated with a vertical line in the margin for each line changed.

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1 THE FACILITY

1.1 Introduction

This document is prepared as part of an application for renewal of License R-74.

The University of Wisconsin has operated a teaching and research reactor, licensed as R-74 under Docket 50-156 since 1961. The reactor supports teaching as a facility of the Engineering Physics Department, other departments of the university, and other educational institutions. Research use of the reactor supports the department, other University of Wisconsin departments, numerous other educational institutions, and some non-educational groups.

The reactor is located on the campus of the university in a building located at 1513 University Avenue in Madison, Wisconsin. It currently operates at a 1000 kW steady-state power with pulsing capability to 1000 MW.

The original Hazards Summary Report has been amended a number of times over the operating history, and the present version of the Safety Analysis Report has been kept up to date by issuance of replacement pages at each annual report submission. As a request for license renewal progressed, however, it was determined that the Safety Analysis Report should be replaced with this completely new version structured in accordance with the guidance in NUREG 1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" dated February 1996.

1.2 Summary and Conclusions on Principal Safety Considerations

Analysis of possible accident scenarios is included in Chapter 13. As a TRIGA-type reactor, the primary safety features stem from the use of a fuel with a strong negative prompt temperature coefficient of reactivity which limits excursions from reactivity insertions, thus preventing fuel damage from credible reactivity accidents. Ejection of the transient rod from the core when the core is operating at the power level scram point will result in no fuel damage. Since experiments are limited to the same reactivity worth as the transient rod, experiment failure cannot result in more severe transients.

In addition, the operating power level of 1000 kW results in a decay heat potential in the fuel small enough that loss of reactor coolant does not result in fuel damage or release of fission products.

In extreme accident conditions in which operation is taking place with already damaged fuel, releases to the public are shown to be nominal except for a combination of loss of pool water and loss of the ventilation system concurrent with the fuel damage. Even with these extremely conservative assumptions, analysis of this accident shows the event will result in exposure to the public that would be classified as an Alert. More realistic assumptions used in the accident analysis indicates that the maximum hypothetical accident would result in emissions and radiation exposure within those allowed by 10 CFR Part 20.

1.3 General Description of the Facility

The University of Wisconsin Nuclear Reactor is located in the Mechanical Engineering Building on the Madison campus of the University within the city of Madison, WI. The building also contains classrooms, laboratories, shops, and staff offices for the departments of Mechanical Engineering, Industrial Engineering, and Engineering Physics departments.

Figure 1-1 is a pictorial view of the reactor. The reactor is a heterogeneous pool-type nuclear reactor currently fueled with TRIGA-FLIP fuel modified to adapt to 4-element bundle assemblies. The coolant is light water which circulates through the core by natural convection. The core is reflected by water and graphite. Maximum steady-state power level is 1000 KW.

A 7 by 9 grid, surrounded by a core box, positions fuel, reflector, and control elements. Three shim-safety blades, a transient control rod, and a regulating blade control core reactivity. The transient control rod is guided by a tube replacing a fuel element in a central fuel bundle, while the control blades move vertically in two shrouds extending the length of the core. The grid box and control element drive mechanisms are supported by a suspension frame from the reactor bridge.

Cold, clean core excess reactivity is about 4.3 % $\Delta k/k$. Control elements having a scram capability provide a shut-down margin of about 4.2 % $\Delta k/k$.

Proposed technical specifications for the facility are included in this report as Chapter 14.

SUMMARY OF REACTOR DATA

Responsible Organization	The University of Wisconsin
Location	Madison, Wisconsin
Purpose	Teaching and Research
Fuel	
Type	TRIGA and TRIGA-FLIP Hydride in 4 element clusters
Number of elements in standard 1000 KW Core	[redacted]

Control

Safety elements	Three vertical blades
Regulating-servo element	One vertical blade
Transient control	One rod

Experimental Facilities

Thermal column	One, 40-inch square graphite, ϕ th = 2 x 10^8 nv
Beam Ports	Four, 6-inch diameter ϕ th = 1 - 3 x 10^{10} nv at shield side of shutter; about 8 x 10^{11} at core end of port
Pneumatic Tube	One, 2-inch (sample size 1 1/4 inch diameter by 5-1/2 inches long), ϕ th = 4 x 10^{12} nv.
Hydraulic in-pool irradiation facilities	Presently three, 2.5 inch (sample size up to 1 7/8 diameter by 4 inches long) ϕ th = 8 x 10^{12} - 2.4 x 10^{13} nv, depending upon location

Thermal neutron fluxes for isotope production include the above, plus large irradiation spaces outside the core with thermal neutron fluxes of around 1.3 x 10^{13} nv.

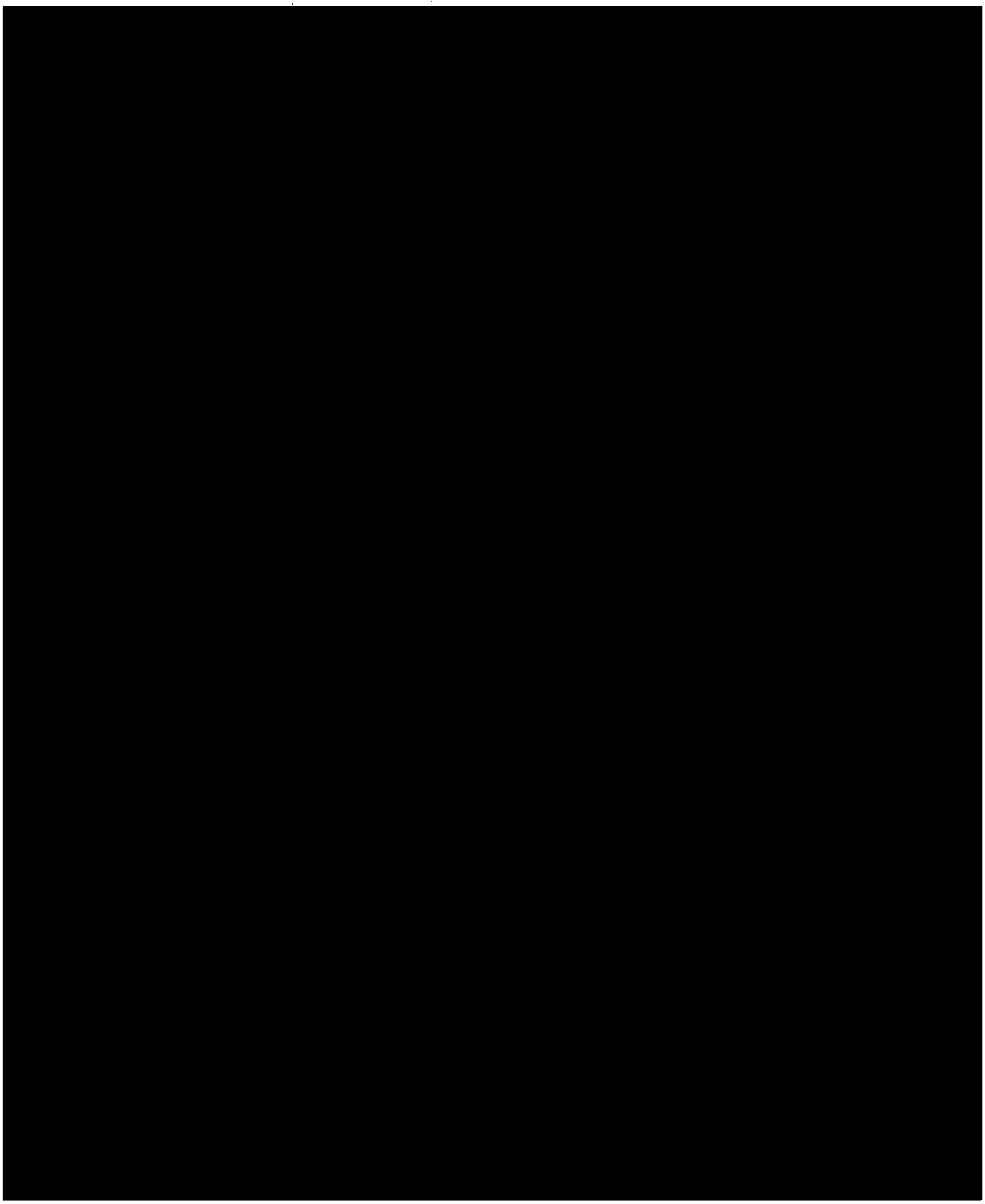

Figure 1-1 UWNR Open Pool Reactor

Dimensions

Pool	8 x 12 x 27-1/2 ft. deep
Standard 1000 KW core	15 x 17 x 15 inches high
Grid box	9 x 7 array of 3-inch modules
Control blades	10-1/2 inches wide
Fuel Element	
Diameter	[redacted]
Length	[redacted]
Fueled length	[redacted]

Nuclear Characteristics

1 MW Steady state:	
Maximum thermal neutron flux	3.2×10^{13} nv
Maximum fast neutron flux	3.0×10^{13} nv
1000 MW Pulse	
Maximum thermal neutron flux	3.2×10^{16} nv
Maximum fast neutron flux	3.0×10^{16} nv
Prompt temperature coefficient of reactivity	-1.26×10^{-4} $\Delta K/^{\circ}C$
Void coefficient of reactivity	$-.2 \times 10^{-4}$ $\Delta K/\%$ void

Prompt neutron lifetime	42 μsec TRIGA Standard fuel, 24 μsec FLIP
Effective delayed neutron fraction	0.007

1.4 Shared Facilities and Equipment

The Reactor Laboratory and supporting laboratories are an integral part of the Mechanical Engineering Building, and thus share walls, water supplies, sewage, and main electrical distribution with the remainder of the building. Heating Ventilation & Air Conditioning (HVAC) systems are dedicated to non-shared use except for those HVAC systems in office spaces. The restricted area contains only reactor-related activities.

1.5 Comparison With Similar Facilities

The best indication of reactor characteristics is the performance of the facility itself, which has been in routine operation with the present operational core since August, 1979.

The reactor at Washington State University is very similar to UWNR, having also originally been a General Electric Open Pool Reactor which was converted to TRIGA fuel, and eventually partially converted to FLIP fuel. The reactor at Texas A & M University is also a converted core, though the original reactor was not built by GE. The pool size and experimental facility configuration differs on the three reactors, but basic reactor behavior and accident analysis are quite similar. In addition, the nuclear characteristics of UWNR are quite similar to the TRIGA Mark III prototype and other FLIP fueled reactors. In chapter 4 of this report the similarity between the UWNR and the prototype is detailed.

1.6 Summary of Operations

Present plans and previous usage involve use of the reactor in performance of the following experiments:

1. Reactor Start-up and Operation;
2. Radiation Survey of the Reactor and Surroundings;
3. Control and Regulating Blade Calibration;
4. Measurement of Reactor Power and Calibration of Reactor Instruments;
5. Measurement of Shutdown Power Level;

6. Measurement of Reactor Period,

7. Measurement of Temperature Coefficient of Reactivity;

8. Measurement of Void Coefficient of Reactivity;

9. Experiments Involving the Danger Coefficient Method;

10. Experiments to Measure the Disadvantage Factor;

11. Studies of Reactor Buckling and Delta K/K.

12. Critical Mass Experiments;

13. Measurement of Thermal Neutron Cross Sections;

14. Delayed Neutron Emission;

15. Activation Analysis;

16. Experiments Utilizing Pile Oscillator Techniques;

17. Flux Distributions in Reactor and Effect of Absorbers on Flux Patterns;

18. Shielding Experiments;

19. Experiments on the Production of Radioisotopes;

20. Neutron Diffractometer Measurements.

21. Neutron Radiography

The above represents the experiments planned at present, but it is anticipated that further experiments (both for training and research) will be added.

1.7 Compliance With the Nuclear Waste Policy Act of 1982

In accordance with a letter from the U. S. Department of Energy (R. L. Morgan) to the U. S. Nuclear Regulatory Commission (H. Denton) dated May 3, 1983, it has been determined that all universities operating non-power reactors have entered into a contract with DOE that provides that DOE retain title to the fuel and DOE is obligated to take the spent fuel and/or high level waste for storage or reprocessing. Because the University of Wisconsin has entered into such a contract with DOE, the applicable requirements of the Nuclear Waste Policy Act of 1982 have been satisfied by the University of Wisconsin Nuclear Reactor Facility. A copy of the current Task Order under Master Task Agreement (which is the successor to contract DE-AC07-76ER01560) is included in Appendix B.

1.8 Facility Modifications and History

Construction permit CPRR-55, authorizing construction of the University of Wisconsin Nuclear Reactor (UWNR), was issued on June 7th, 1960. License R-74 was issued November 23rd, 1960. The expiration date of the license was set at 40 years after the issuance of the construction permit (June 6th, 2000). The University of Wisconsin Nuclear Reactor achieved initial criticality on March 26th, 1961 as a 10 KW teaching and research reactor. After a license amendment dated October 22nd, 1964, the power level was increased to 250 KW on December 7th, 1964, using the original flat-plate aluminum clad fuel. Operations with the original core ended October 13th, 1967, after 2268.5 critical hours and 105.65 megawatt hours of core exposure.

A cooling system was installed and the reactor was converted to a 1000 KW, TRIGA reactor with pulsing capability in 1967. Construction permit CPRR-97 authorizing the changes was issued on June 7, 1967. Amendment No. 8 to the operating license for the conversion was issued on November 13, 1967. Initial criticality with the TRIGA core occurred on November 14, 1967. After over 3,000 megawatt hours of operation with the TRIGA core a partial refueling was necessary. FLIP fuel was available to afford significantly improved core lifetimes, so a new Safety Analysis Report was submitted in April, 1973 describing facility characteristics and safeguards using standard fuel, FLIP fuel, and mixtures of the two fuel types in defined compositions. License amendment No. 10 was issued in response. The initial partial refueling to a mixed core with 9 Standard fuel bundles replaced with FLIP (Fuel Life Improvement Program) fuel was initially critical in March 1974. Additional fuel replacements in January 1978 and August 1979 resulted in the present operating core, consisting entirely of FLIP fuel. The total fuel exposure since converting to TRIGA fuel is over 20,000 megawatt hours, or 833 megawatt days.

A number of other license amendments were issued during the term of the license, involving inclusion of security, training, and emergency plans. Several other amendments changed the amount and types of Special Nuclear Material used in connection with the license. None of these changes had any effect on the operating characteristics of the reactor, and are therefore not detailed here. The most recent amendment was No. 16, dated August 30, 2006.

A new Safety Analysis Report was submitted in April 2000 as part of a license extension application. Two changes were included in the new SAR: elimination of the reactor trip on short period and elimination of the electronic scram capability of the safety amplifier. These changes were approved by our Reactor Safety Committee based on a 10 CFR 50.59 analysis. Neither of these items had been required by Technical Specifications, since they did not provide protection for a pulsing TRIGA reactor. Sections 7.2.3 and 7.4 describe the instrumentation as it has been changed.

A new cooling system was installed in 2003. Revision 1 of the SAR was issued in August 2004 mainly to reflect the new cooling system, which is described in Chapter 5.

From 2004 through 2007 the Mechanical Engineering building surrounding the reactor laboratory under went construction. The center wing was completely demolished and rebuilt, and the east, west, and north wings, as well as the reactor auxiliary support spaces, were substantially renovated. Some reactor systems were also renovated, but the reactor itself remained unchanged. Revision 2 of the SAR was issued in September 2008 to address the relevant changes. The new building floor plans are shown in Chapter 2. A new ventilation system was installed, described primarily in Chapter 9 (the calculations in Chapter 11, Chapter 13, and Appendix A were also affected). Several new reactor control console components were installed as part of a console instrumentation upgrade, and Chapter 7 was updated to reflect these changes. A new pneumatic tube sample transfer system (which mates with the existing irradiation facility) was installed and is described in Chapter 10. These changes were approved by our Reactor Safety Committee based on a 10 CFR 50.59 analysis.

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2 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification and Location

The University of Wisconsin Nuclear Reactor Laboratory is located within the Mechanical Engineering Building at 1513 University Avenue on the University of Wisconsin-Madison (UW) campus. The reactor is located at [redacted] On the United States Geological Survey (USGS) Madison West, WIS 15' Quadrangle topographical map, the Universal Transverse Mercator Coordinates are; [redacted] [redacted].

The UW campus is surrounded by the city of Madison in Dane county, Wisconsin. **Figure 2-1** shows the location of Dane county within Wisconsin. Madison, a city of approximately 208,000 residents (2000 Census statistics), is built around two lakes in the center of Dane county, **Figure 2-2**. Lake Mendota (15 square miles) lies northwest of Lake Monona (5 square miles) and the two lakes are only 2/3 of a mile apart at one point. The UW campus is set on this narrow neck of land between the two lakes, known as the isthmus, and on the southern shore of Lake Mendota. The Mechanical Engineering Building is near the southwestern border of the UW campus, where the nearest non UW owned property is 425 ft (130 m) from the reactor site (approximately 80m from the Mechanical Engineering Building west wall). The reactor is 2300 ft (700 m) south of the shore of Lake Mendota.

2.1.1.2 Boundary and Zone Area Maps

A map of the City of Madison detailing the general topography and the surrounding urban and rural zones up to a distance of 8 km is reproduced in **Figure 2-3**. The UW campus, which is located on the southern shore of Lake Mendota, is shown in **Figure 2-4**. The Mechanical Engineering Building is located on the engineering campus which is the southwest corner of the UW campus as shown in **Figure 2-5**.

The operations boundary is defined as the Reactor Laboratory, Room 1215, of the Mechanical Engineering Building. The site boundary is defined as the center and east wings of the Mechanical Engineering Building, but not including the north or west wings, plus the portion of Engineering Drive (formally Johnson Drive) south of the designated areas of the building. **Figure 2-6**, **Figure 2-7**, **Figure 2-8**, **Figure 2-9**, **Figure 2-10**, and **Figure 2-11** depict the floor plans of the Mechanical Engineering Building's basement, first floor, second floor, third floor, fourth floor, and fifth floor respectively. **Figure 2-12** and **Figure 2-13** depict cross sections of the building through the core centerline looking north and east, respectively. The emergency preparedness zone is entirely within the operations boundary, as defined above.

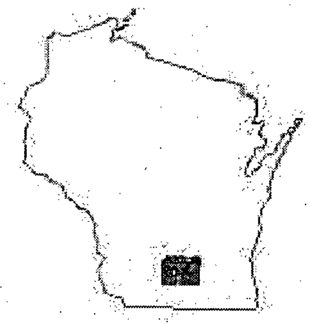

Figure 2-1 Wisconsin Map

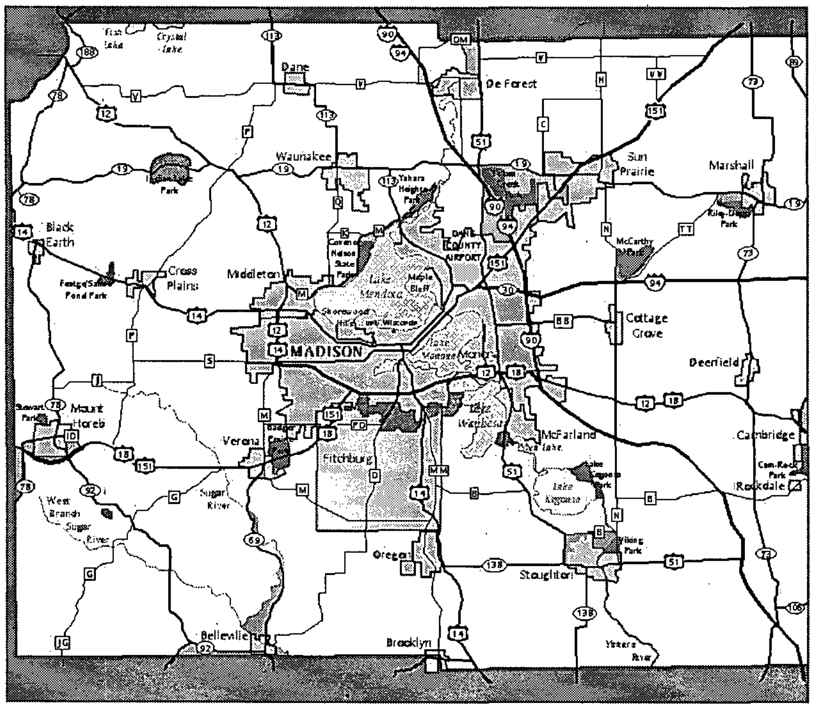

Figure 2-2 Dane County, Wisconsin

2.1.2 Population Distribution

Population distributions estimated by the uniform density model and the 2000 Census[1] are shown in Table 2-1. The area around the campus is mature residential, and the central business district is only a short distance away. Population is therefore quite stable in the immediate surrounding areas of the UW campus. The 8 kilometer radius includes much more sparsely-settled regions to the north of Lake Mendota, and population in this region is likely to increase markedly in the future. The nearest permanent residence is approximately 150 m west of the reactor site.

Transient population around the reactor include students present in classrooms and offices in the Mechanical Engineering Building during the months of September through May as well as spectators attending sporting events at Camp Randall Stadium which is approximately 250 m due south of the reactor. The maximum number of students present in the east and center wings of the Mechanical Engineering Building at any one time is estimated at 1500. The maximum number of spectators contained by Camp Randall Stadium is 80,321.

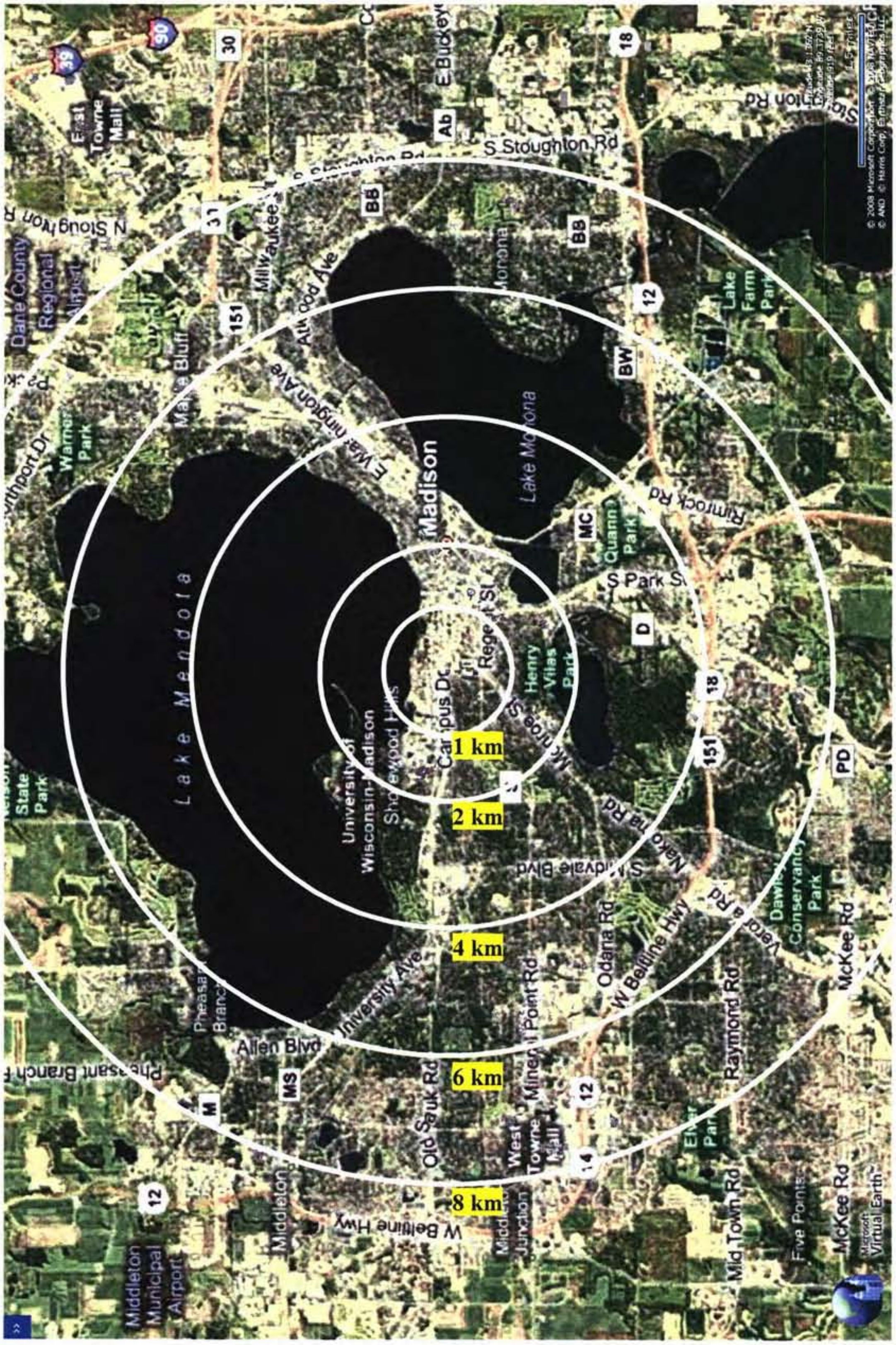

Figure 2-3 City of Madison, Wisconsin

Figure 2-4 University of Wisconsin Campus

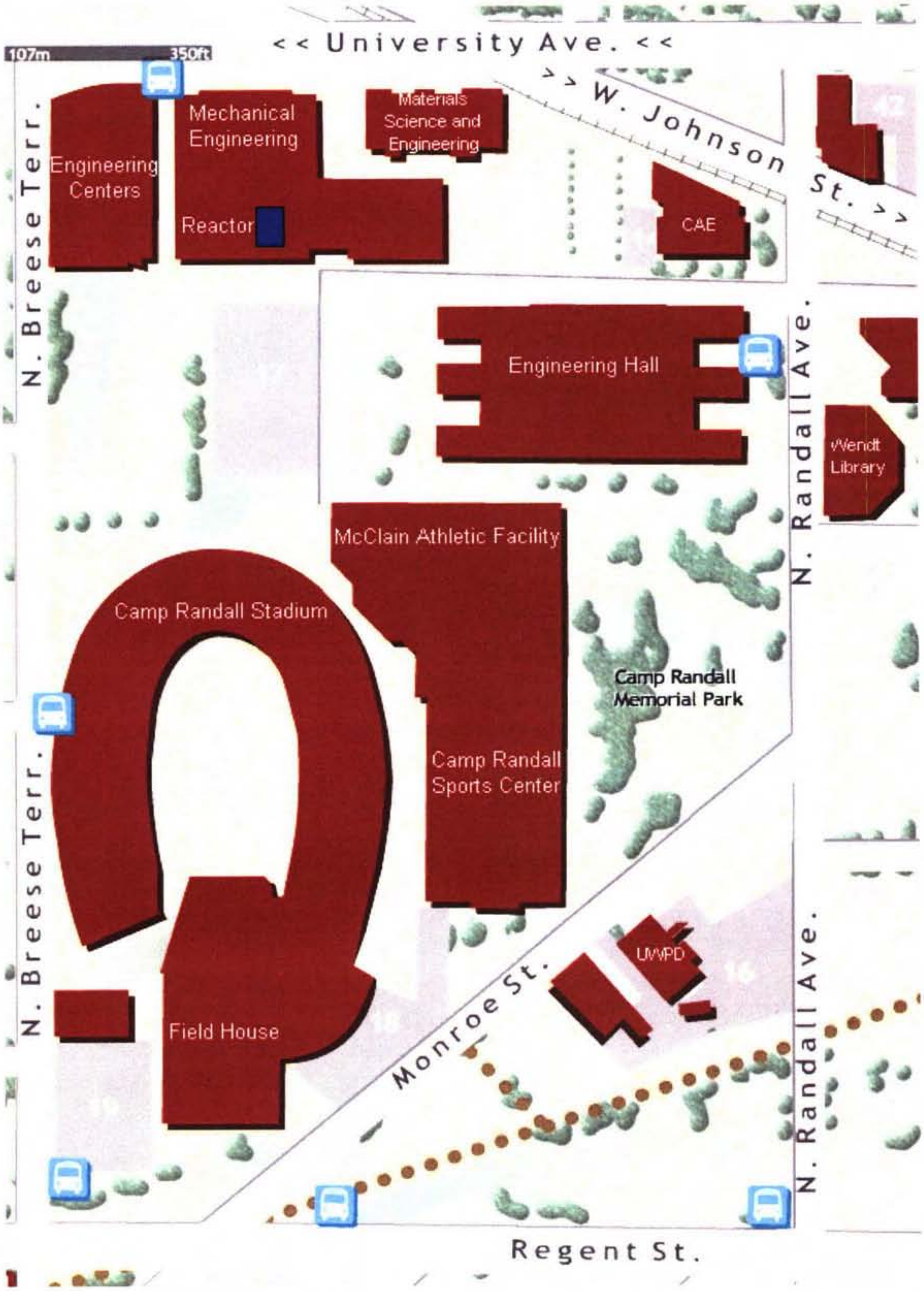

Figure 2-5 UW Engineering Campus

[redacted]

Figure 2-6 Mechanical Engineering - Basement

[redacted]

Figure 2-7 Mechanical Engineering - First Floor

[redacted]

Figure 2-8 Mechanical Engineering - Second Floor

[redacted]

Figure 2-9 Mechanical Engineering - Third Floor

[redacted]

Figure 2-10 Mechanical Engineering - Fourth Floor

[redacted]

Figure 2-11 Mechanical Engineering - Fifth Floor

[redacted]

Figure 2-12 Mechanical Engineering Cross Section through Core Centerline, looking North

[redacted]

Figure 2-13 Mechanical Engineering Cross Section through Core Centerline, looking East

TABLE 2-1
Population Distribution

Distance from Facility(kilometers)	Estimated 2000 Population[1]
1	12,595
2	37,814
4	71,780
6	118,919
8	192,575

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Locations and Routes

The UW campus is surrounded, mainly, by a residential district to the south and west, while to the east is primarily a commercial business district with city and state government office buildings. No industrial facilities are in the vicinity of the reactor.

A railroad spur of the Wisconsin & Southern Railroad Company runs through campus and is 100m to the Reactor Laboratory at its closest approach. A rail car holding yard which is a part of this spur is approximately 300 m northwest. The primary commodity transported over this spur or resident at the rail car holding yard is coal[2].

The reactor is located approximately 4.5 km north of a bypass highway, known as the Beltline, for US highways 12, 14, 18 and 151. Interstates 90 and 94 are located approximately 10 km to the east and northeast of the reactor site.

There are no military facilities in the Madison area with the exception of the Wisconsin Air National Guard which is located on a military ramp of the Dane County Regional Airport. While the Wisconsin Air National Guard flies approximately 4000 missions annually, the flight pattern of these missions typically are north of Lake Mendota and the City of Madison. At no time do any of these mission flights carry live ammunition[3]. More information about the Dane County Regional Airport facility is reported in section 2.2.2 Air Traffic.

2.2.2 Air Traffic

The Dane County Regional Airport is approximately 8 km to the north east of the reactor site. This is the only commercial airport near the reactor. While there are three smaller air fields within 16 km from the reactor in the communities surrounding Madison, these air fields are for general aviation only.

The Dane County Regional Airport has 3 runways with the following outbound headings; 360°(north)/180°, 320°/140°, and 210°/30°. None of these headings have trajectories that take commercial traffic directly over the reactor immediately before arrival or after departure. The airport is serviced by several commercial express carriers and is utilized by the Wisconsin Air National Guard as well as general aviation aircraft[3]. Of the 115,613 events (arrival and departure are counted as two separate events) in 2006, 59% were classified as general aviation, 34% commercial carriers/taxi, and 7% military. Due to the infrequent arrival of commercial traffic, the air traffic control tower does not place inbound traffic in holding patterns around the city of Madison[4].

2.2.3 Analysis of Potential Accidents at Facilities

There are no industrial, transportation or military facilities within the vicinity of the reactor site that have the potential for accidents with consequences significant to impact the Reactor Laboratory. While a railroad spur passes within 100 m of the reactor facility, this spur transports non hazardous cargo and other major ground transportation routes are located at great distances from the Reactor Laboratory. Due to the frequency and flight paths of commercial and military air traffic the probability of occurrence of an accident is considered extremely low.

2.3 Meteorology

2.3.1 General and Local Climate

Madison has the typical continental climate of interior North America with a large annual temperature range and with frequent short period temperature changes. The range of extreme temperatures is from about 110 to -40 °F. Winter temperatures (December - February) average near 20 °F and the summer (June - August) average temperature is in the upper 60s. Daily temperatures average below 32 °F about 120 days and above 40 °F for about 210 days of the year.

Madison lies in the path of the frequent cyclones and anticyclones which move eastward over this area during fall, winter and spring. In summer, the cyclones have diminished intensity and tend to pass farther north. The most frequent air masses are of polar origin. Occasional outbreaks of arctic air affect this area during the winter months. Although northward moving tropical air masses contribute considerable cloudiness and precipitation, the true Gulf air mass does not reach this area in winter, and only occasionally at other seasons. Summers are pleasant, with only occasional periods of extreme heat or high humidity.

There are no dry and wet seasons, but about 60 percent of the annual precipitation falls in the five months of May through September. Cold season precipitation is lighter, but lasts longer. During July, August, and September rainfall is mostly from thunderstorms and tends to be erratic and variable. Average occurrence of thunderstorms is just under 7 days per month during this period. Tornadoes are infrequent. Dane County has about one tornado in every three to five years.

The ground is covered with 1 inch or more of snow about 60 percent of the time from about December 10 to near February 25 in an average winter. The soil is usually frozen from the first of December through most of March with an average frost penetration of 25 to 30 inches."[5]

2.3.2 Site Meteorology

The summary of meteorological conditions for Madison is based on the records obtained from the International Station Meteorological Climate Summary[6] jointly produced by the National Oceanic and Atmospheric Administration (NOAA), the United States Air Force (USAF) and United States Navy. The data specifically compiled for Madison was obtained from the National Weather Service and unless specifically noted, is for the period of record from 1948 to 1995.

The Reactor Laboratory does not have a continuing onsite meteorological data measurements program. All future meteorological data will be obtained from the National Weather Service station in Madison.

2.3.2.1 Temperature

The monthly average and daily average extreme temperatures for the Madison area are shown in Table 2-2. The record extreme temperatures in the Madison area, as reported by the National Weather Service, have ranged from a low of -37 °F in January 1951 to a high of 107 °F in July 1936.

2.3.2.2 Precipitation

The Madison area normally receives an annual average of 31.6 inches of precipitation. The monthly average precipitation data is reported in Table 2-3. The record maximum level of precipitation to fall in Madison in one year, is reported by the National Weather Service to be 52.9 inches in 1881. The greatest 24-hour rain fall total for Madison was 5.25 inches on July 15-16, 1950. The 48-hour, 100-yr. return period rainfall for south central Wisconsin is estimated to be 7.82 inches[7].

The annual average snow fall for Dane County is 43.7 inches. The monthly average snow fall data is reported in Table 2-4. The record maximum snow to fall during the winter season is reported by the National Weather Service as 76.1 inches during the winter of 1978-79. The 24 hours state record for heaviest snow occurred December 27-28, 1904 in Neillsville, Wisconsin in which 26 inches of snow fell.

TABLE 2-2
Average Temperatures for the Madison Area

	Average Monthly Temperature (°F)	Average Daily Maximum Temperature (°F)	Average Daily Minimum Temperature (°F)
January	16.8	25.7	8.0
February	21.3	30.6	12.0
March	32.3	41.9	22.7
April	46.1	57.6	34.7
May	57.5	70.0	44.9
June	67.0	79.4	54.5
July	71.4	83.3	59.5
August	69.2	81.0	57.4
September	60.4	72.3	48.6
October	49.5	60.9	38.2
November	35.3	44.0	26.5
December	22.6	30.7	14.4
Year	45.8	56.5	35.1

TABLE 2-3
Monthly Precipitation Data for the Madison Area

	Average Monthly Total Precipitation (inches)	Average Monthly Maximum Precipitation (inches)
January	1.14	2.45
February	1.14	2.77
March	2.17	5.04
April	3.02	7.11
May	3.14	6.26
June	3.83	9.95
July	4.05	10.93
August	3.90	9.49
September	3.12	9.22
October	2.29	5.63
November	2.15	5.13
December	1.66	4.09

TABLE 2-4
Monthly Snowfall Data for the Madison Area

	Monthly Average Total Snowfall (inches)	Monthly Maximum Snowfall (inches)	Year
January	10.3	31.8	1929
February	7.7	37.0	1994
March	8.5	28.4	1923
April	2.3	17.4	1973
May	0.1	5.0	1935
June	0		
July	0		
August	0		
September	Trace		
October	0.2	5.0	1917
November	3.8	18.3	1985
December	10.9	32.8	1987

2.3.2.3 Winds

The average annual wind speed in the Madison area is 9.8 mph. The prevailing winds during the months of November through March are from the west-northwest direction, the remaining months of April though October the prevailing winds are from the south[8]. Table 2-5 reports the frequency of surface wind direction versus wind speed. The record gust in the Madison area occurred in June 1975 when wind gusts were reported at 83 mph from the west.

TABLE 2-5
Frequency of Surface Wind Direction versus Wind Speed

	Speed (knots)											
Direction	1 - 3	4 - 6	7-10	11-16	17-21	22-27	28-33	34-40	41-47	>=48	Percent	Wind Speed (knots)
N	0.5	1.4	1.6	1.3	0.2	*	*	*	0	0	4.7	8.2
NNE	0.3	0.7	1.1	0.8	0.2	*	*	0	0	0	3.3	9.3
NE	0.6	1.1	1.4	1.1	0.2	*	*	0	0	0	4.6	9
ENE	0.6	1.7	1.7	1.2	0.2	*	*	0	0	0	5.5	8.2
E	0.5	1.1	1.4	0.9	0.1	*	*	*	0	0	3.8	7.8
ESE	0.3	0.9	1.2	0.7	0.1	*	0	0	0	0	3.2	8.5
SE	0.4	1.1	1.5	0.7	0.1	*	0	0	*	0	3.8	8.5
SSE	0.4	1.4	1.7	1	0.1	*	0	0	0	0	4.9	8.4
S	0.6	2.8	4.1	2.9	0.5	0.1	*	*	*	0	10.4	8.8
SSW	0.3	1.1	2.4	2.3	0.4	0.1	*	*	0	0	6.8	10.1
SW	0.3	1.1	2.4	2	0.4	0.1	*	*	*	0	6.5	10.3
WSW	0.3	1.1	2	1.5	0.3	0.1	*	*	*	0	5.5	10
W	0.4	1.6	2.5	1.9	0.5	0.1	*	*	0	0	6.7	9.5
WNW	0.4	1.9	2.9	2.6	0.6	0.1	*	*	0	0	8.6	9.7
NW	0.6	1.6	2.4	2.5	0.6	0.1	*	*	*	0	8.3	10.1
NNW	0.5	1.6	1.6	1.6	0.3	0.1	*	*	0	0	5.8	9
VAR	0	0	0	0	0	0	0	0	0	0	0	0
CLM	0	0	0	0	0	0	0	0	0	0	7.6	0
ALL	6.9	22.2	32.1	25.2	4.7	1	0.1	*	*	0	100	8.5

* = PERCENT < .05

2.4 Hydrology

Madison is located just northeast of the "Driftless Area" of southwestern Wisconsin. The glacially shaped topography of Madison and the surrounding area in central Dane County is irregular, ranging from flat or gently rolling to hilly. The most prominent geomorphic features were glacially formed. Among these are Lakes Mendota, Monona, Waubesa, Kegonsa and Wingra. Glacial drift covers the entire area except for local areas of bedrock outcrop[9].

In the vicinity of the reactor, a glacial deposit exists which contains clay and large boulders. Although this deposit may be as much as 100 feet thick, it is probably less than twenty. The reason for the variation in thickness is that the bed-rock sandstone which underlies the deposit is very uneven. The bed-rock consists of Cambrian sandstones which are 700 to 800 feet thick and which are permeable to water.

The Yahara River is the main drainage feature in the Madison area. The Yahara River gradient is essentially flat where it flows through the lakes in Madison. Water surface elevations are controlled mostly by dams at the outlets of lakes Mendota and Waubesa. The ground water flow from the reactor site is generally toward the Lake Mendota - Yahara River - Lake Monona system. Thus, the general flow is toward the east and south. Historical data obtained from the USGS Wisconsin Geological and Natural History Survey[10] indicated the annual average water table in the vicinity of the reactor is approximately 60 feet below the surface.

Madison obtains its drinking water supply from several deep well aquifers drilled, typically several hundred feet, into the Cambrian sandstone described above. The location of these wells is shown on **Figure 2-12**, and they supply the University as well as the city. All of these wells are cased from ground level into the sandstone so as to keep out ground water from the glacial deposit. The closest well to the reactor is Madison City Well 27 located 2,000 feet southeast.

Due to the large drainage capacity of the Yahara river and the outlet dams of lakes Mendota and Wausbesa flooding is not a serious problem in most of Madison. The 100 year flood is estimated to increase the surface level of Lake Mendota approximately 3 feet causing the lake to over flow its existing banks by about 30 feet[11].

[redacted]

Figure 2-12 City of Madison Well Water Supply

2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Regional Geology

The Midwestern United States does not lie on, or anywhere near, a tectonic plate boundary. The region is in the middle of the North American Plate, hundreds of miles from both the eastern and western edges. The Midwest, however, has a series of faults around the Mississippi Valley, **Figure 2-13**, the most active of which is the New Madrid Fault System. These faults were formed by the tearing open of the ancient continental crust almost 5 million years ago. This region is known as the New Madrid Seismic Zone which includes the states of Missouri, Arkansas, Louisiana, Illinois, Indiana and parts of Kentucky and Tennessee[12]. Wisconsin is not associated with this region. The New Madrid Fault System, located near New Madrid, Missouri is greater than 500 miles from Madison, Wisconsin and poses little or no seismic hazard.

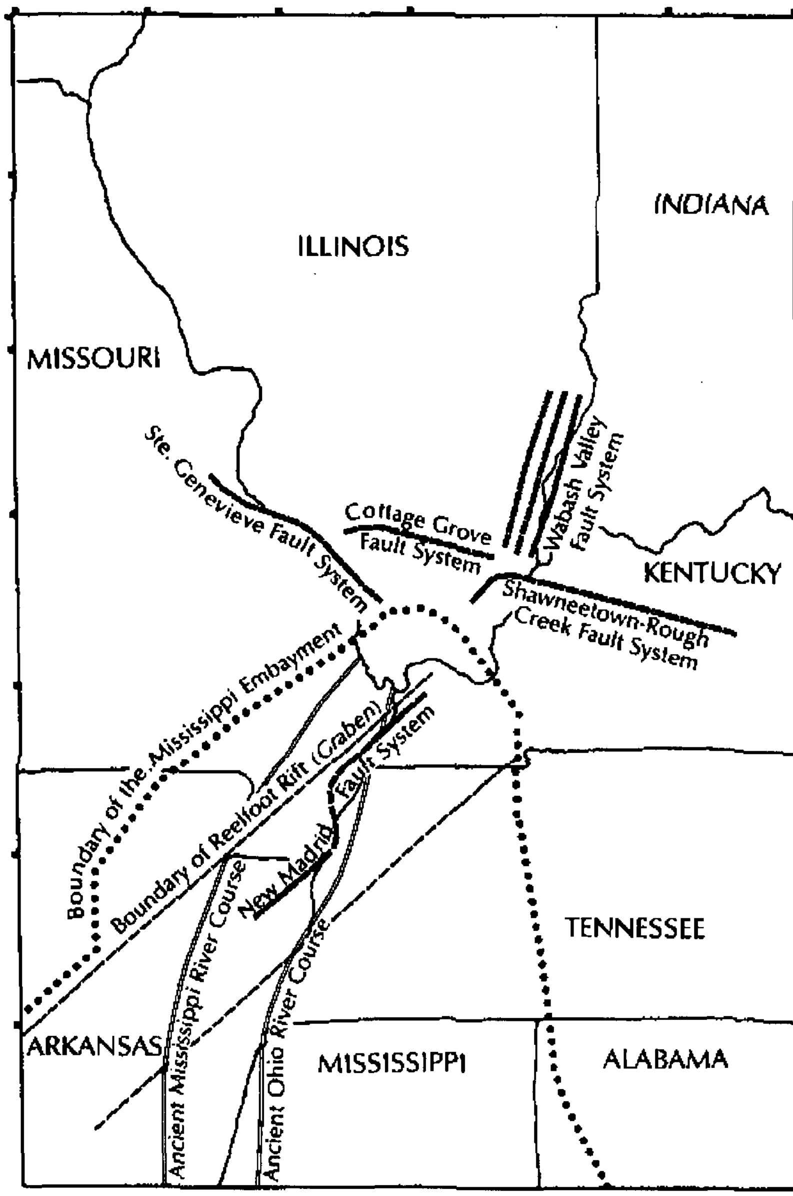

Figure 2-13 New Madrid Seismic Zone

2.5.2 Site Geology

As discussed in section 2.4 Hydrology, the local geology of the reactor site includes a layer of glacial deposits which rests on the Cambrian sandstone bedrock. The layer of the glacial deposit is variable, however, soil bores at the reactor site indicate this layer is approximately 16 feet deep, Table 2-6[13,14,15]. The Cambrian sandstone layer below the glacial drift is approximately 700 to 800 feet thick. Below the sandstone is impermeable basement rock. There are no geological structures of consequence in the vicinity of the reactor site.

TABLE 2-6
Depth of Glacial Deposit to Cambrian Sandstone

Distance from Reactor Site (Feet)	Depth of Glacial Deposit (Feet)
0	16
150	16.5
175	4
200	14
250	13.25
300	9.5
350	17.5
400	15
450	9.5

2.5.3 Seismicity

As discussed in section 2.5.1, Regional Geology, Wisconsin is located in a geologically stable region of the United States. The closest active fault is greater than 500 miles in the New Madrid Seismic Zone. While no seismic events have occurred at the reactor site, there are records of earthquakes occurring within 200 km of the reactor site. A review of the USGS earthquake database[16] for all earthquakes of modified Mercalli intensity greater then IV or magnitude greater than 3.0 which is within 200 km of the reactor site resulted in the data reported in Table 2-7.

TABLE 2-7
History of Seismic Events Within 200 km of Madison, Wisconsin.

Date	Magnitude (Richter)	Intensity (Modified Mercalli)	Distance (km)
August 20, 1804	4.4	VI	178
May 27, 1881	4.6	VI	198
May 26, 1909	5.1	VII	195
January 2, 1912	4.5	VI	190
November 12, 1934	4.0	VI	196
September 15, 1972	4.5	VI	158
September 9, 1985	3.0	V	178

2.5.4 Maximum Earthquake Potential

The determination of earthquake potential and frequency is based on data from previous events. Because there are few historic moderate to large earthquakes in the vicinity of the reactor, analysis is difficult. The maximum earthquake potential may be inferred from data supplied by the USGS for the 50 year peak ground acceleration estimate[17]. The estimated 50 year peak ground acceleration due to a seismic event in the vicinity at the reactor is less than 0.01 g.

2.5.5 Vibratory Ground Motion

Due to insufficient data from previous seismic events the vibratory ground motion can only be inferred from the peak ground acceleration data of section 2.5.4 of less than 0.01 g.

2.5.6 Surface Faulting

Based on the distance to any known active faults and the stable site geology, surface faulting is not considered to be a credible event in the vicinity of the reactor site.

2.5.7 Liquefaction Potential

Based on the distance to any known active faults and the stable site geology, the liquefaction potential is considered to be insignificant.

2.6 References

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3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

When the University of Wisconsin Nuclear Reactor was to be upgraded by increasing authorized power to 1000 kW the principal design criterion was to assure the facility could withstand loss of pool water and any other credible accident with no hazard to the public, without reliance on engineered safety features. This criterion was met by selecting stainless-steel-clad TRIGA fuel due to the well-documented characteristics of this fuel type[1]. Details of the physical mechanisms and characteristics that cause TRIGA and FLIP fuel to exhibit the prompt negative temperature coefficient responsible for the fuel characteristics are given in the reference and a number of other documents and are not included here. Detailed analysis for this facility (Chapter 13 of this report) agree with the conclusions in the reference. The design criteria that result in this negligible safety risk are the result of the fuel composition and cladding, not of specific features provided in the equipment and building that surrounds the reactor.

The Mechanical Engineering Building which houses the reactor laboratory was completed in 1930. Extensive remodeling of the north, east, and west wings, and construction of a new center wing was completed in 2007. Extensive remodeling of the room that became the Reactor Laboratory took place in 1960 when the reactor was installed. The original reactor installation used fuel and components manufactured by General Electric, and the specifications to which structures were built were those stated by General Electric. Specific design criteria were not stated. The original ventilation system was designed in 1962 and installed in 1963. The design specifications stated only the desired flow rates and stack height. The current ventilation system was installed in 2006. The design specifications included the desired flow rates and stack height, and are detailed in section 9.1. The original cooling system was designed in 1966 and installed in 1967. The only design specification was the heat removal rate required. The current cooling system was installed in 2002. The design specifications included the heat removal rate required, and are detailed in chapter 5. During the 1966 upgrade, the N^{16} diffuser system was also installed. This system was designed and fabricated by General Atomic to their design specifications. Conversion to TRIGA fuel took place in 1969, and the auxiliaries for pulsing operation (transient rod and drive) were designed and built by General Atomic to their specifications. All building modifications and equipment additions were in conformance with the building codes in existence at that time.

3.2 Meteorological Damage

There are no design criteria for the protection of facility structures from meteorological conditions except that all facility structures were constructed to applicable building codes in existence at the time. The Reactor Laboratory has endured approximately 50 years of Wisconsin weather with no meteorological damage. Furthermore, no facility structures are assumed to be operable in this SAR for the mitigation of any accident (see Chapter 13, Accident Analysis).

3.3 Water Damage

There are no design criteria for the protection of facility structures, systems and components from water damage. The possibility of flooding due to the lake system of Dane County is considered insignificant due to the distance of the Reactor Laboratory to the Lake Mendota flood plain as described in Chapter 2, section 2.4, Hydrology. Furthermore, no facility structures, systems and components susceptible to water damage are assumed to be operable in this SAR for the mitigation of any accident (see Chapter 13, Accident Analysis).

3.4 Seismic Damage

There are no design criteria for the protection of facility structures, systems and components from seismic damage except that all facility structures were constructed to applicable building codes in existence at the time. The probability of a seismic event in the vicinity of the reactor site is considered insignificant due to the stable regional geology (see Chapter 2, section 2.5, Geology, Seismology and Geotechnical Engineering). Furthermore, no facility structures, systems and components, including the reactor pool, susceptible to seismic damage are assumed to be operable in this SAR for the mitigation of any accident (see Chapter 13, Accident Analysis).

3.5 Systems and Components

At the time of original construction of the Reactor Laboratory, design bases were not provided by General Electric for facility systems and components. With the upgrade to TRIGA and TRIGA-FLIP fuel, accident analyses, including NUREG/CR 2387 and Chapter 13, show that by the design of TRIGA fuel, reliance upon other systems, structures and components are not necessary to ensure safety of the general public. Therefore, with the exception of the fuel, no other facility structure, system or component is assumed to be operable in this SAR for the mitigation of any accident.

Nevertheless, experience gained on facility systems and components over many years of operation have shown these systems to be highly reliable. Descriptions of system design and operation of these systems are discussed in the succeeding chapters of this SAR.

3.6 References

1. NUREG/CR2387, Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors, Hawley and Kathren, Pacific Northwest Laboratory, April 1982

4 REACTOR DESCRIPTION

4.1 Summary Description

4.1.1 Introduction

The reactor was constructed and installed by the Atomic Power Equipment Department of the General Electric Company. The present modification employs a core composed of TRIGA-FLIP fuel supplied by the General Atomic Company.

Initial criticality was achieved on March 26th 1961. The original maximum steady state power level was 10 kW. Power was increased to 250 kW on December 7th 1964 and again increased to the present maximum steady state power level of 1,000 kW on November 14th 1967. Operation with FLIP fuel began in March 1974 with a mixed core containing 9 FLIP bundles. In January 1978 an additional 6 FLIP bundles were added. In August 1979 the conversion to FLIP fuel core was completed.

Figure 4-1 is a pictorial cutaway view of the reactor. The reactor is a heterogeneous pool-type, fueled with TRIGA or TRIGA-FLIP fuel which is cooled by natural convection. The fuel is currently all 70% enriched in Uranium U^{235}, although 20% enriched fuel can also be used. Light water acts as both coolant and moderator as well as being a biological shield. The core is reflected on two sides by graphite and on two sides by water, the water-reflected areas being utilized as irradiation facility locations. The top and bottom reflector region is partially graphite and partially water.

A 7-by-9 grid, surrounded by a core box, positions fuel, reflectors, control elements, and irradiation facilities. Core reactivity is changed and controlled by three shim safety blades, a regulating blade, and a transient control rod. All control elements move vertically in shrouds positioned in the core box or inside a fixed tube as is the case of the transient rod. A suspension frame supports the grid box and control element drive mechanisms. The suspension frame, in turn, is supported by the reactor bridge.

Cold, clean core excess reactivity in the present operational core is about 4.3 % $\Delta k/k$. Control elements (control blades and the transient rod) provide a shutdown margin of about 4.2 % $\Delta k/k$.

[redacted]

Figure 4-1 University of Wisconsin Nuclear Reactor (UWNR)

4.1.2 Summary of Reactor Data

Responsible Organization	The University of Wisconsin
Location	Madison, Wisconsin
Purpose	Teaching and Research
Fuel	
Type	TRIGA Hydride in 4 element clusters
Number of elements in standard 1000 kW core	[redacted]
Control	
Safety elements	Three vertical blades
Regulating-servo element	One vertical blade
Transient control	One rod
Experimental Facilities	
Thermal Column	One, 40-inch square graphite, ϕ th = 2 x 10^8 nv
Beam Ports	Four, 6-inch diameter ϕ th = 1 - 3 x 10^{10} nv at shield side of shutter; about 8 x 10^{11} nv at core end of port
Pneumatic tube	One, 2-inch (sample size 1-1/4 inch diameter by 5-1/2 inches long), ϕ th = 5 x 10^{12} nv

Thermal neutron fluxes for isotope production include the above, plus large irradiation spaces outside the core with thermal neutron fluxes of around 1.3 x 10^{13} nv.

Reactor Materials	
Fuel - moderator material	U-Zr $H_{1.6}$

U^{235} enrichment	70%
U^{235} content/element (average)	[redacted]
Burnable poison	1.5wt.% Natural Erbium
Cladding	20 mil stainless steel
Reflector	Water and graphite
Coolant	Light water
Control	Boral & stainless steel; borated graphite for transient rod
Structural material	Aluminum
Shield	Concrete and water

Dimensions

Pool	8 x 12 x 27-1/2 ft. deep
Standard 1000 kW Core	15 x 17 x 15 inches high
Grid box	9 x 7 array of 3 inch modules
Fuel element	
Diameter	[redacted]
Length	[redacted]
Active Length	[redacted]

Nuclear characteristics

1 MW Steady State:

Maximum thermal neutron flux	3.2×10^{13} nv
Maximum fast neutron flux	3.0×10^{13} nv

1000 MW Pulse

Maximum thermal neutron flux	3.2×10^{16} nv
Maximum fast neutron flux	3.0×10^{16} nv
Core Loading (Standard 1000 kW core)	[redacted]
Operating excess reactivity	~4.3% ΔK_{eff}
Reactivity in control blades	~7.1% ΔK_{eff}
Average prompt temperature coefficient of reactivity	-1.26×10^{-4} ΔK/°C
Void coefficient of reactivity	$-.2 \times 10^{-4}$ ΔK/% void
Prompt neutron lifetime	2.4×10^{-5} second
Effective delayed neutron fraction	0.007

4.1.3 Experimental Facilities

Facilities are provided to permit use of radiations from the reactor in experimental work without endangering personnel. These facilities include three hydraulic irradiation facilities ("whales"), four beam ports, one thermal column, and a pneumatic transfer system ("rabbit").

4.1.3.1 Hydraulic Irradiation Facility (Whale)

Aluminum pipes of 2-7/16" internal diameter extend from approximately 18" below the pool surface to grid box positions on the periphery of the core. These pipes draw sample bottles made of polyethylene down and position them approximately at the center line of the fuel. Two sample containers can be loaded in each tube. The addition of a second sample bottle, however, causes the natural rotation of the first bottle to stop. Thermal Neutron Fluxes in these positions are approximately 10^{13} nv.

Irradiations are also conducted in the other reflector regions surrounding the core. Irradiation baskets may be inserted in any vacant grid position, and irradiations can also be conducted outside the grid box in specially fabricated enclosures.

4.1.3.2 Thermal Column

The thermal column is a graphite-filled horizontal penetration through the biological shield which provides neutrons in the thermal energy range (about 0.025 eV) for irradiation experiments. The column, which is about 8 feet long, is filled with about 6 feet of graphite. A small experimental air chamber (40" x 40" x 24") between the face of the graphite and the thermal column door has conduits for service connections (air, water, electricity) to the biological shield face. The compensated ion chambers for the safety and logN instrumentation channels are located in the thermal column.

Personnel in the building are protected against gamma radiation from the column by a dense concrete door which closes the column at the biological shield. The door moves on tracks set into the concrete floor perpendicular to the shield face.

4.1.3.3 Beam Ports

Four 6-inch beam ports penetrate the shield and provide fluxes of both fast and thermal neutrons for experimental use. The ports are air filled tubes, welded shut at the core ends and provided with water-tight covers on the outer ends. The portions of the ports within the pool are made of aluminum, while the portions within the shield are steel.

A shutter assembly, made of lead encased in aluminum, is opened for irradiations by a lifting device. When closed, the shutter shields against gamma rays from the shut-down core, allowing experiments to be loaded and unloaded without excessive radiation exposure to personnel.

Shielding plugs are installed in the outer end of each port. The plugs, made of dense concrete in aluminum casings, have spiral conduits for passage of instrument leads.

4.1.3.4 Pneumatic Tube

A pneumatic tube system conveys samples from a basement room to an irradiation position beside the core. The "rabbits" used in the system will convey samples up to 1-1/4 inches diameter and 5-1/2 inches long. The system operates as a closed loop with carbon dioxide cover gas limiting generation of Ar^{41} activity.

The reactivity effect from a sample in the pneumatic tube is restricted to less than 0.2% ρ. Tests run with water and cadmium samples indicate that sample reactivity effects will normally be less than 0.01% ρ. Static reactivity measurements will be run for samples of fissionable material or particularly strong absorbers such as some of the rare earths.

4.2 Reactor Core

The reactor may be operated with either standard (20% enriched) or FLIP (70% enriched) fuel as described in section 4.2.1. In addition, mixed cores containing fuel of both types may be loaded.

Since funding is not presently available for replacing the FLIP fuel with LEU with burnable poison, the core that will usually be operated is composed only of FLIP fuel in order to maintain the radiation levels of this fuel above the point at which it is self-protecting. The timing of funding for LEU fuel is unknown at present. If funding for converting the entire core is received it may be necessary to revert to cores of mixed standard and FLIP cores before loading a new core in order to assure that less than a formula quantity of HEU (as defined in 10 CFR 73.2) becomes non-self-protecting. If funding for LEU replacement is received a few fuel bundles at a time, reversion to a mixed standard/FLIP core may not be required, but further analysis of cores of mixtures of FLIP and the new LEU fuel would be required. Such cores are not considered in this SAR.

The use of the reactor as a training and research tool requires flexibility of core arrangement. These arrangements are subject, however, to the following criteria:

a. A mixed core must contain at least 9 FLIP bundles.
b. Any FLIP fuel must be located in a central contiguous region.
c. The core must be a close packed array except for single fuel element (not fuel bundle) positions or grid positions on the core periphery.
d. Calculations indicate that operation will be within safety limits on power generation per element and fuel temperature.

4.2.1 Reactor Fuel

The fuel is of the TRIGA four-element bundle type developed to provide a simple means of converting reactors using flat-plate fuel to TRIGA reactors. A variant bundle, called a three-element bundle, has only three fuel elements installed; the fourth space is used for an aluminum control rod guide tube or an instrumented fuel element. **Figure 4-2** shows a four-element bundle, a three-element bundle containing a control rod guide tube, and a three-element bundle containing an instrumented fuel element (the conduit for the thermocouple leads is shown cut short)..

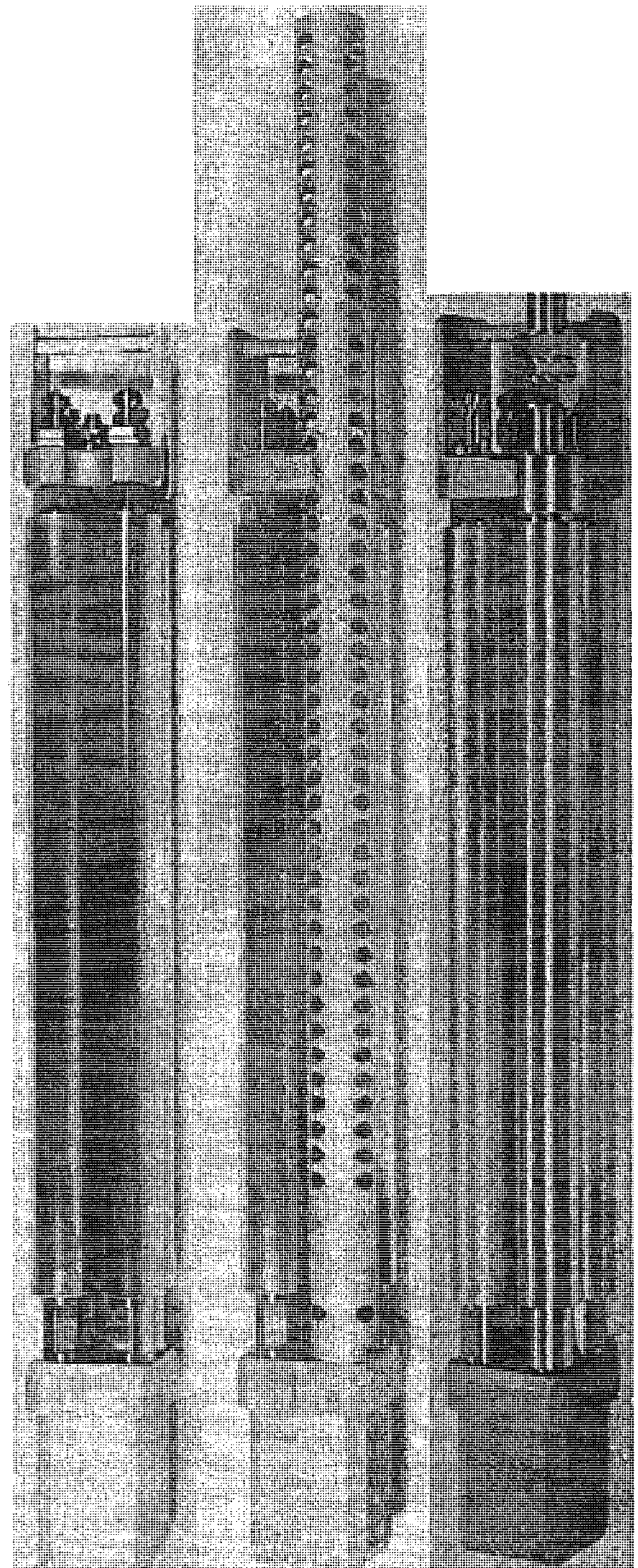

Figure 4-2 Fuel Bundles

The four-element bundle (**Figure 4-3**) consists of bottom adapter, top adapter, and four TRIGA elements. The bottom adapter of the bundle fits the existing grid plate as did the original flat-plate fuel elements. The end fittings on individual TRIGA elements are threaded into the bottom adapter until a flange on the element seats firmly against the adapter, providing rigid cantilever-type support. The top adapter serves both as a handling fitting and as a spacer for the upper ends of the fuel elements. A sliding fit between this adapter and the fuel element end fittings allows for differential expansion of the elements. This top fitting can be removed with remote handling tools to disassemble the bundles for replacement of individual fuel elements or for shipping spent elements for reprocessing.

The individual fuel elements (**Figure 4-4**) are quite similar to the TRIGA elements used on TRIGA reactors using the standard TRIGA grid plates. The differences are (1) reduction of diameter from [redacted] inches to maintain the proper metal-to-water ratio in this core; (2) the bottom end fixture is threaded; (3) flats on the stainless steel bottom end fixture provide wrench surfaces for disassembly without stressing the cladding; and (4) the top end fixture is modified to allow the top end fitting to be locked in place.

The TRIGA elements used at UWNR are of two types, standard and FLIP. Both have outside dimensions, clad thickness, and construction as shown in **Figure 4-4**. The two types differ as shown in the following table:

Design Parameter	Standard Fuel	FLIP Fuel
Fuel moderator material	U-Zr $H_{1.7}$	U-Zr $H_{1.6}$
U^{235} enrichment	20%	70%
U^{235} content/element (average)	[redacted]	[redacted]
Burnable Poison	None	Natural erbium
Erbium content	---	1.5 wt %

The FLIP fuel was designed to extend the lifetime of TRIGA fuel, and was used in step-wise additions of fuel to the University of Wisconsin Nuclear Reactor. The reactor is currently operating with a core consisting entirely of FLIP fuel. Fuel bundles contain only one type of fuel, and the top adapters for FLIP fuel bundles are marked (by notches machined into the top of the top adapter) to facilitate identification during underwater fuel handling.

[redacted]

Figure 4-3 Four-element Bundle Assembly

[redacted]

Figure 4-4 Fuel Element Construction

Figure 4-5 shows an instrumented element. This element is fitted with three thermocouples. At least one such element (inserted into the vacant position of a three-element bundle) is included in every core. The sensing tips in the thermocouples are located at the vertical centerline of the fuel section and one inch above and below the centerline. The thermocouple leads pass through a seal in a stainless steel tube which provides a water-tight conduit carrying the lead-out wires above the surface of the pool water.

[redacted]

Figure 4-5 Instrumented Fuel Element

4.2.2 Control Elements

Both blade and rod shaped control elements are used.

4.2.2.1 Control Blade Shrouds and Guide Tubes

Each blade type control element, both safety and regulating, is guided throughout its travel by a shroud as shown in **Figure 4-6**. The shroud consists of two thin aluminum plates 38 inches high, separated by aluminum spacers to provide a 1/8-inch water annulus. The shrouds can be removed, if necessary, by use of a grapple hook. Small flow holes at the bottom of the shroud minimize the effect of viscous damping on the scram time.

Rod shaped control elements are guided by a guide tube as shown in one of the three-element bundles of **Figure 4-2**. Holes drilled in the sides of the guide tube allow for water displacement when the control rod is fired out during pulsing operation or dropped in response to a scram condition.

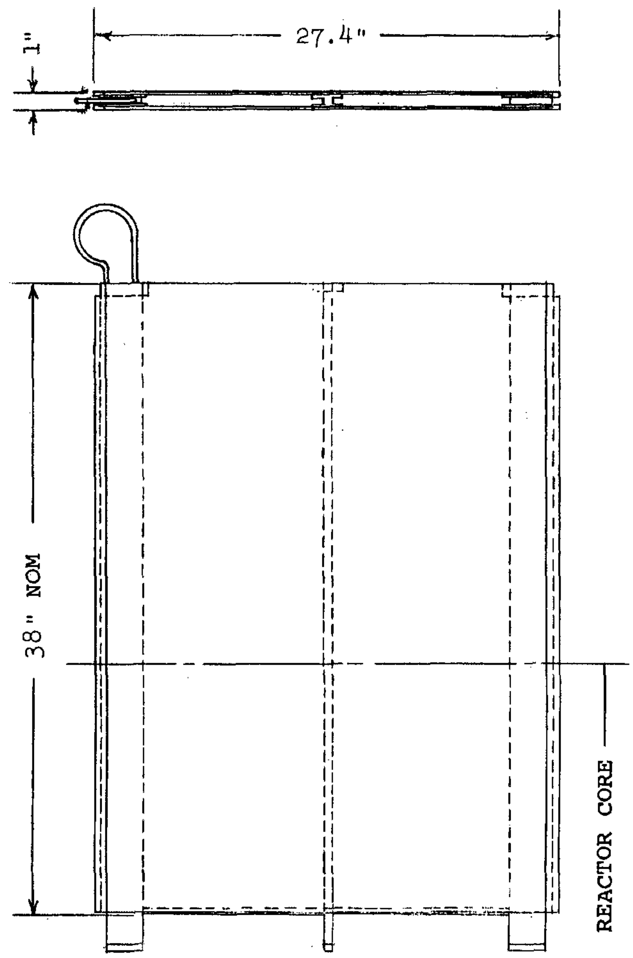

Figure 4-6 Control Blade Shroud

4.2.2.2 Safety Blades

Reactor control for startup and shutdown is accomplished by three blade-type control elements, **Figure 4-7**, with a total shutdown worth between 6.9 and 11 per cent ΔK_{eff}. The poison section is boral sheet (boron carbide and aluminum sandwiched between aluminum side plates). Each safety blade is 40.5 inches long. When a blade is full in the bottom of the blade overlaps the bottom of the active fuel by 1.5 inches.

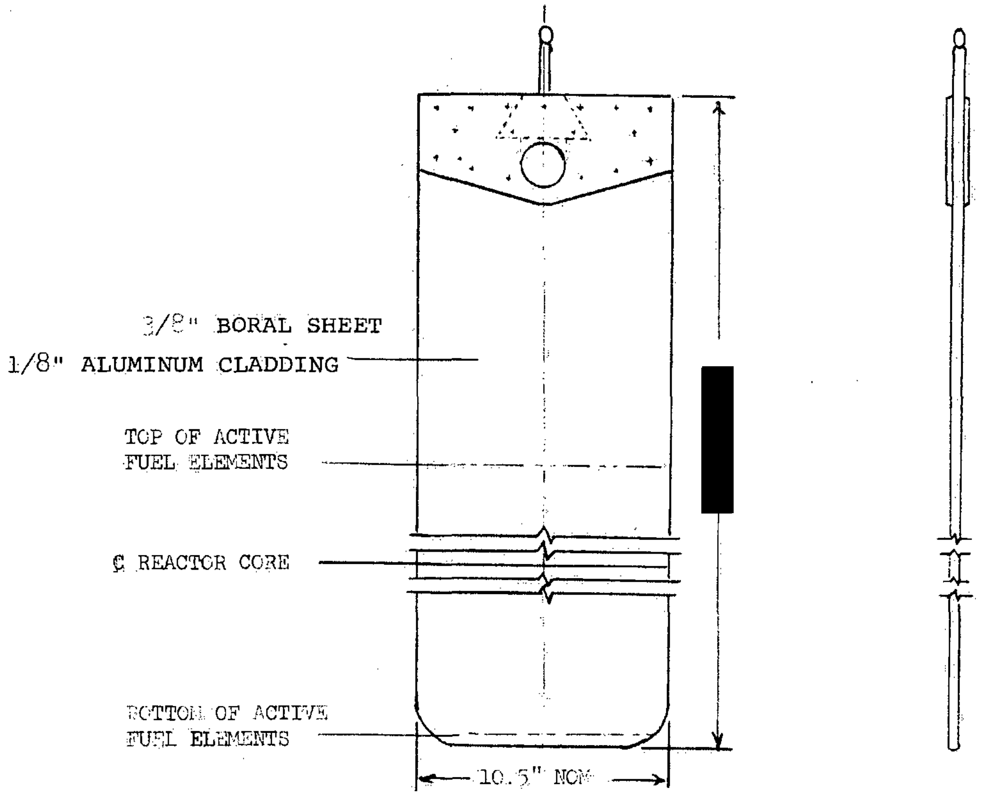

Figure 4-7 Shim/Safety Blade

4.2.2.3 Regulating Blade

The regulating blade, **Figure 4-8** (shown upside down for ease in reading the dimensions), is a stainless-steel sheet with curls on the vertical edges, about 11 inches wide and 40 inches long, supported and guided in the same manner as the safety blades. It is used to compensate for small changes of reactivity during normal reactor operation and may be actuated by a servo-control channel.

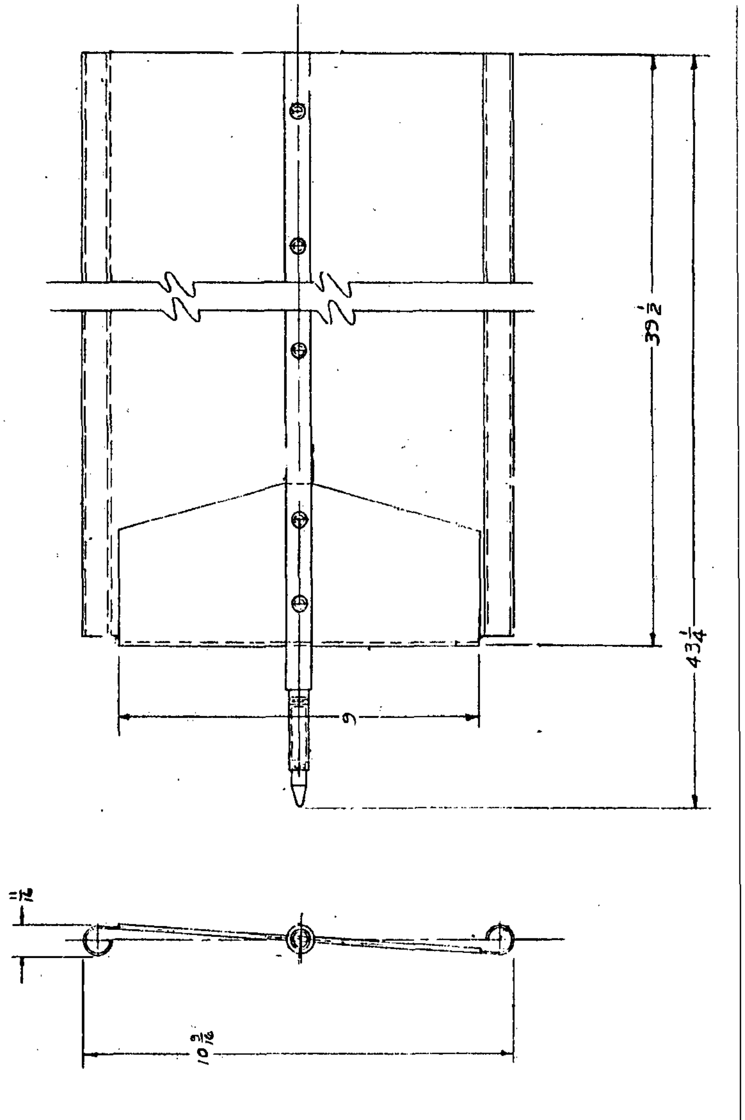

Figure 4-8 Regulating Blade

4.2.2.4 Transient Control Rod

The transient control rod is boron carbide or borated graphite contained in a 1.25 inch diameter stainless steel or aluminum tube (**Figure 4-9**). The poison section is approximately 15 inches long. This rod is guided laterally by the aluminum guide tube in a special three-element fuel bundle. A hold-down tube extends from this guide tube to the top of the reactor structure and holds the three-element bundle in place during transient rod movement.

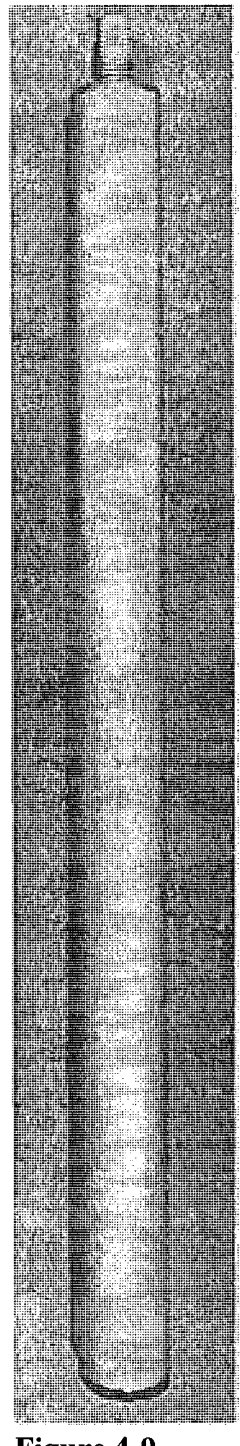

Figure 4-9 Transient Control Rod

4.2.3 Neutron Moderator and Reflector

Pool water serves as moderator for the core and as reflector above, below, and on those sides of the core not provided with graphite reflectors or special reflector elements designed to condition the quality of a beam being extracted from the core. (Individual fuel elements contain an internal 3.5 inch long graphite end reflector above and below the fueled portion, so the core top and bottom are actually reflected by a mixture of graphite and water.)

The reflectors are standard General Electric Company reflectors furnished at initial startup of UWNR. The nominal 3-inch square reflector elements are made of AGOT grade graphite clad with aluminum (**Figure 4-10**). Reflector element lifting handles are diagonal to facilitate identification when viewing the core and storage racks.

Special reflectors are the same size as the graphite reflectors; but may consist of solid aluminum, hollow aluminum, or combinations of graphite sections with the center portion replaced with solid aluminum, voids, or gamma absorbers such as lead or bismuth. Such special reflectors are used for irradiation facilities or to adjust the mix of thermal, epithermal, and fast neutrons transmitted to experimental facilities.

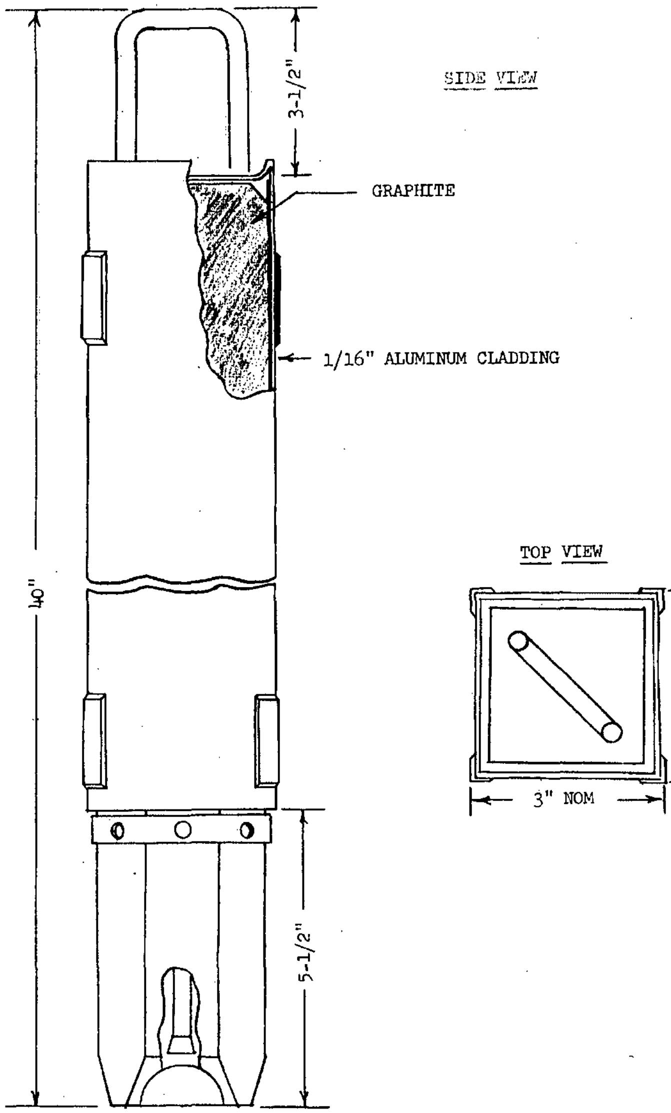

Figure 4-10 Graphite Reflector Element

4.2.4 Neutron Startup Source

The neutron source is a [redacted] radium-beryllium source irradiated to give an output greater than 10^7 neutrons/second. It is encapsulated in a 0.515 inch diameter by 3.10 inch long stainless steel welded cylindrical capsule, which in turn contains two 1.25 inch long welded stainless steel capsules.

The source fits into a source holder (**Figure 4-11**).

The source holder, in turn, fits into an irradiation basket (**Figure 4-12**) occupying one grid module adjacent to the active core. The source is usually left in for full power operation, and will, with the normal operating cycle, maintain its output of about 10^7 neutrons/second. If the source is not left in during full power operation, neutron emission rate will decrease over a long time period to approximately 10^6 neutrons/second.

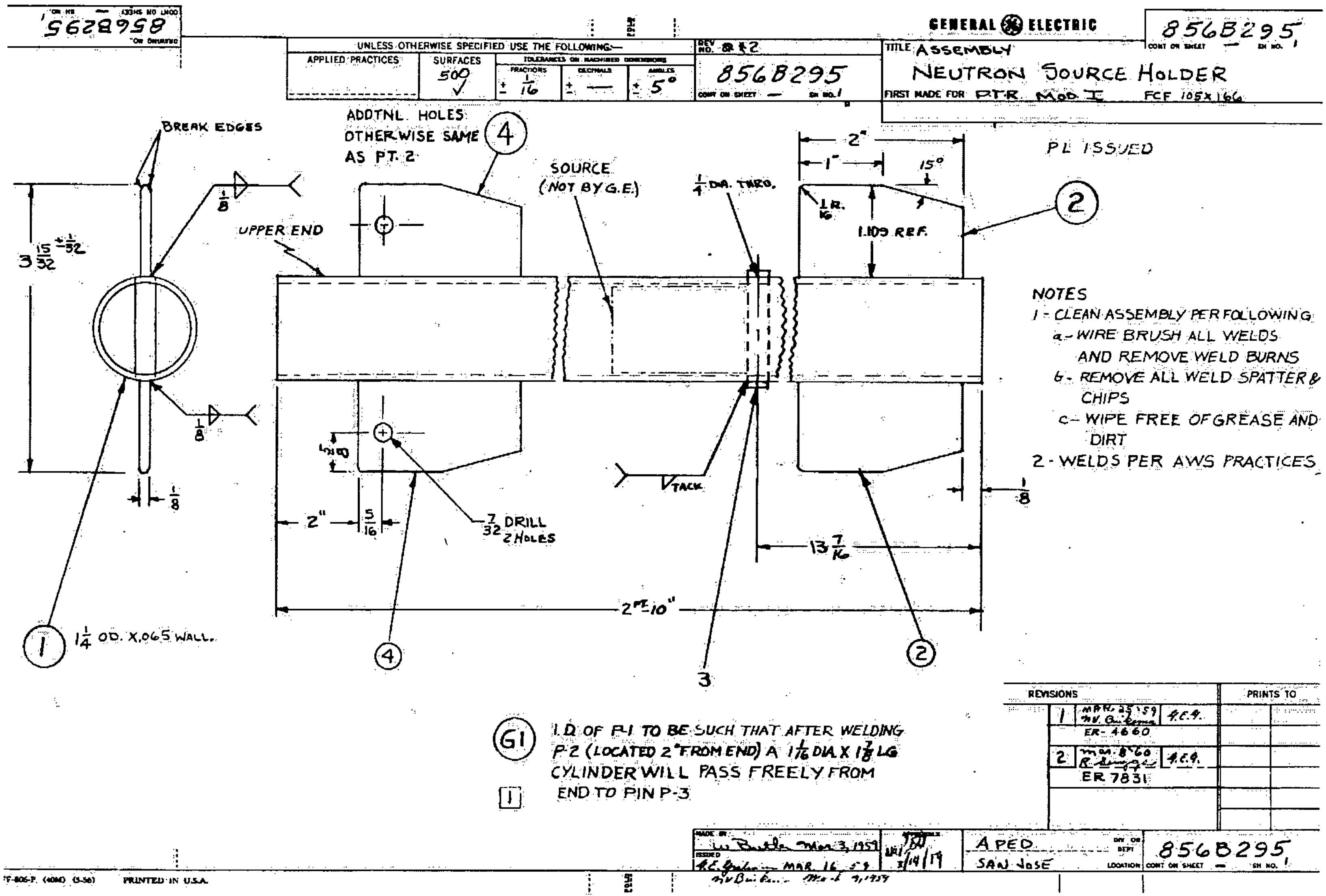

Figure 4-11 Neutron Source Holder

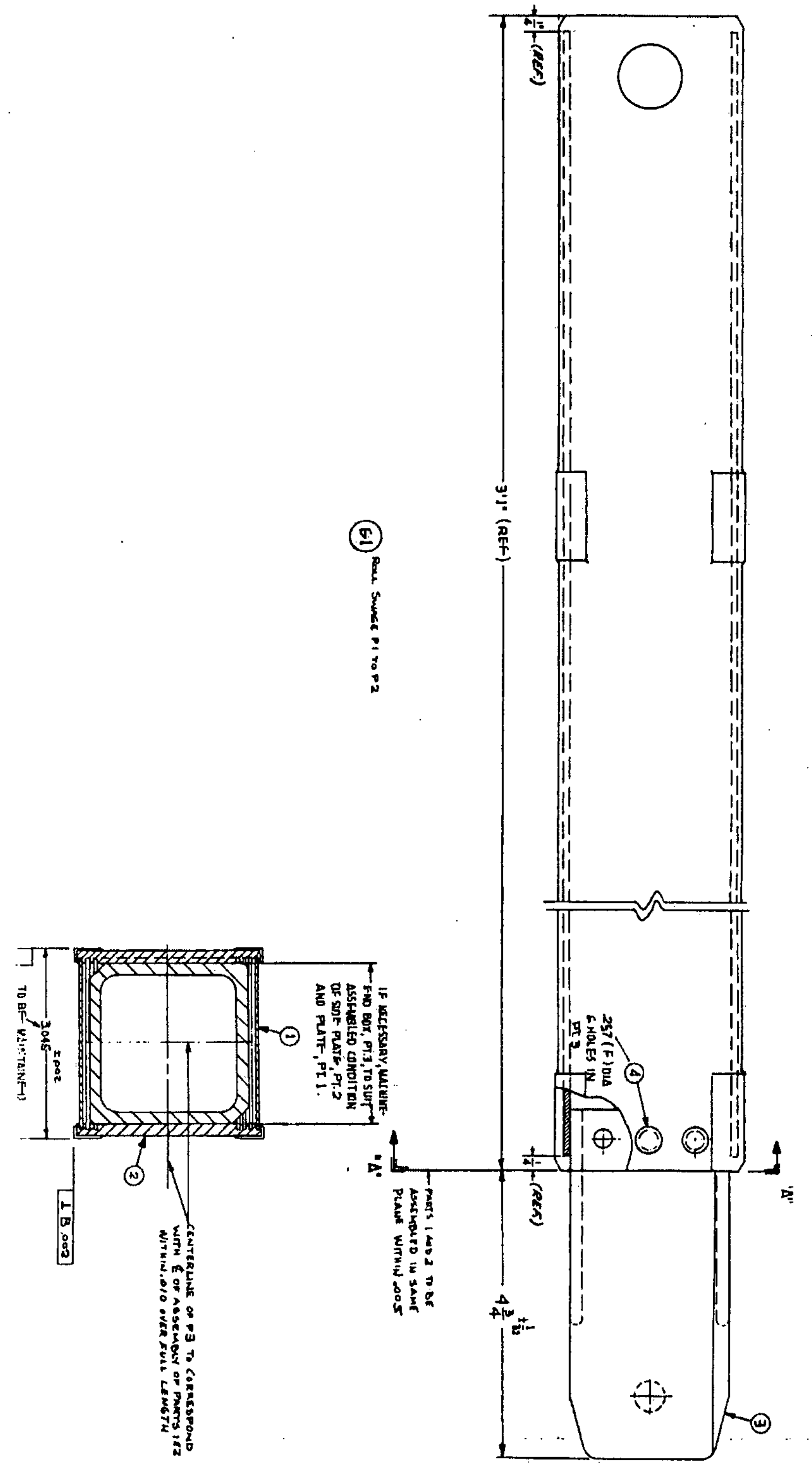

Figure 4-12 Irradiation Basket

4.2.5 Core Support Structure

The core is suspended from an all-aluminum frame, **Figure 4-13**, which extends from the grid box to a height of about one foot above the pool surface. One of the hollow corner posts of the suspension frame serves as a guide for the gamma chamber used in pulsing operation. The other three corner posts may also be used to position detectors in positions above the core.

The reactor bridge (mounted over the pool) supports the core suspension frame. The all-steel, prefabricated bridge was bolted together in the field and aligned with shims.

A locating plate, made of 0.5-inch steel, spans the upper end of the suspension frame. It is bolted to the bridge and aligns the four control blade drive mechanisms and the transient rod drive with the core. The five mechanisms work through individual clearance holes, each mechanism being secured to the locating plate. The plate and mechanisms are not removable as a unit to prevent accidental withdrawal of the control elements. The fission counter drive is mounted on a portion of the hand railing support structure.

Four 4-inch square 6061 aluminum suspension tubes (0.25 inch wall thickness) extend from the bridge to the grid box, and support the grid box by bolted connections. The aluminum grid box, **Figure 4-14**, encloses and supports the 6-inch thick cast aluminum grid plate which is machined to locate and support the control element shrouds and bottom end fittings of fuel bundles, reflectors and in-core experimental facilities such as hydraulic irradiation positions and irradiation baskets. **Figure 4-15** shows the grid position designation system, location of experimental facilities and radiation detectors relative to the grid box, and letter and number codes used in later descriptions to identify fuel and reflector reactivity worths.

An aluminum coolant header (not shown in the figures) mates with the bottom of the grid box and forms a transition to the coolant piping originally provided (but not used) for future use with a forced convection cooling system. An opening in the side of the header, 24 inches wide by 12 inches high, allows cooling water flow for natural convection.

A diffuser pump and jet above the core deflects the cooling water streaming from the core to reduce N^{16} activity above the core.

[redacted]

Figure 4-13 Core Suspension

[redacted]

Figure 4-14 Grid Box and Grid Plate

[redacted]

Figure 4-15 Grid Arrangement With Fuel and Reflector Codes

See Table 4-1 and 4-2 for an explanation of coding in the figure above.

4.3 Reactor Pool

The aluminum-lined concrete pool, **Figure 4-16**, is 8 feet wide, 12 feet long, and 27 1/2 feet deep. The reinforced concrete pool walls also serve as the biological shield (further details on the pool walls are found in the next section). The pool is penetrated by experimental ports. Piping systems connecting with the pool are discussed in detail in Chapter 5. In summary, all piping connections are built to preclude accidental loss of pool water by failure of components located outside of the pool. When possible, pipes enter and leave the pool above the water surface (primary cooling system, diffuser system, and hydraulic irradiation tube water system) and are equipped with passive siphon breakers that prevent loss of more than a few inches of water even in the event of a pipe break or system mis-operation.

[redacted]

Figure 4-16 Reactor Pool and Shield

Two 8 inch aluminum pipes intended for use in a forced-convection cooling system were imbedded in the concrete at initial construction. (See **Figure 5-2**, especially the note concerning anti-siphon loop). One of these pipes penetrates the pool wall about 14 feet below the pool curb, but is closed with flanges on both the inside and outside ends, preventing loss of pool water unless both flanges fail. The other 8 inch pipe was looped inside the concrete and equipped with a siphon breaker that extends from the top of the loop to above the pool curb. One end of the loop is flanged closed outside the shield, while the other end vertically penetrates the bottom of the pool within the coolant header (below the grid box). If the outer flange of this pipe were to fail, this pipe could drain the pool to 14 feet below the pool curb. At [redacted]

[redacted]

The pool makeup and purification system supply pipe penetrates the pool 38 inches below the pool surface, thus limiting water loss to that level reduction upon failure. The discharge pipe from this system is discussed in the paragraph immediately above.

The 4 beam ports penetrate the pool wall at mid-core level. The beam ports are operated with a watertight flange on the outside end which will prevent leakage should the in-pool portion of the beam port fail. The in-pool portion of the tubes are aluminum pipes with welded end closure and bolted flanged connection to the beam port shutter assembly. The four beam ports have a common drain system, but the discharge valve for the drain system is maintained closed during operation. The beam ports also have vent connections which connect to the Beam Port and Thermal Column Ventilation system (see Chapter 9, Section 9.1). The vent connections are equipped with valves (normally kept open for ventilation flow), but which may be closed if a beam port leak occurs. Finally, each beam port vent has a check valve which allows air flow only out of the vent, thus preventing pressure differences between the beam ports from causing circulation between beam ports.

The thermal column case also penetrates the pool wall. It is of welded aluminum construction and has no valves or flanges which could be opened to drain the pool. [redacted]

[redacted]

The pool water is kept within the following limits:

Temperature (at Core cooling water inlet)	<130°F
Resistivity	>2 x 10^5 Ohm - cm
Radioactivity	<10 CFR Part 20 Appendix B Table 3 values for radioisotopes with >24 hour half-life

The reactor core is cooled by natural convection of pool water through the core. The 130°F temperature limit is imposed by demineralizer resin tolerance and by humidity control considerations.

The resistivity limit is set to reduce corrosion effects, extending the expected lifetime of the fuel elements and controlling water radioactivity. Routine checks of resistivity are made to determine the necessity of regenerating the demineralizer.

The radioactivity of the pool water is continuously monitored by an area monitor station located near the demineralizer. Should the pool water reach the activity limit above, the reading on this area monitor will increase during periods when the reactor is not operating. In addition, water samples are routinely analyzed for activity by other methods which give a more exact identification of quantity and type of activity present.

No problem has been experienced in maintaining pool water radioactivity below the indicated limits in nearly 50 years of operation.

4.4 Biological Shield

The reactor is shielded by concrete and water (See **Figure 4-16**). At normal pool level the core is covered by 20 feet of water. The shield at core level consists of about 3 feet of water plus 8 feet (9 feet on thermal column side) of ordinary concrete. Denser concrete is used in the thermal column door and beam port plugs. Calculations and measurements of radiation levels for 1000 kW operation are (excepting N^{16} activity) discussed below:

Surface of shield, excepting beam port and thermal column openings - less than 1.5 mrem/hr. Pool surface (leakage radiation) (No N^{16}) less than 15 mrem/hr.

"Hot spots" - measurements have shown that higher radiation levels exist around the beam ports and thermal column. Measurements of the maximum radiation levels at these "hot spots" at 1000 kW are about 10 mrem/hr around the beam ports and 40 mrem/hr at the hottest spot around the thermal column door. The dose one foot away from the hot spots is about 5 mrem/hr.

Since the third and fourth-floor classrooms and offices, and fifth-floor mechanical room, are above the level of the pool curb, an analysis of the effect of complete water loss on persons in these areas was performed and results included in the updated Emergency Plan. The computer code MCNP5 was used to model the dose rate, as described in section 13.1.3.2. Since the biological shield does not offer any shielding to the central wing third floor classrooms, the dose in these classrooms would be greater than any location on the fourth or fifth floors. The building evacuation alarm would evacuate people from these areas before the pool was completely drained, such that the total integrated dose received by evacuating members of the public would be about 13 mrem.

4.4.1 Pool Surface Radiation Levels - N^{16} Activity

The radiation level due to N^{16} activity at the pool surface directly above the core when operating at 1000 kW would range from 80 to 220 mrem/hr if no N-16 control system were in operation (variability is due to changes in surface flow patterns). The N-16 diffuser system is normally in operation, however, reducing the dose rates at the pool surface to 2-4 mrem/hour at the pool surface. These radiation levels are low enough that no hazard will exist to personnel outside the Reactor Laboratory or in normally occupied levels within the Reactor Laboratory. Radiation levels on the walkway surrounding the pool are around 20 mrem/hr while the reactor is operating at 1000 kW without the diffuser operating and <0.5 mrem/hour with the diffuser operating.

All of the Reactor Laboratory outside of the console area is posted as a radiation area and a radioactive materials area. A cable and switch arrangement is positioned on the north stairway to the pool surface so that an alarm will be sounded should entry to that area be made while the reactor is operating, thus assuring that personnel will not enter the area without knowledge of the reactor operator.

The south stairway, leading from the console area to the pool surface does not have a cable and switch arrangement as does the north stairway. Access to these stairs is gained only through the console area and is well monitored. No difficulty has been experienced in maintaining radiation doses to individuals well below those doses permitted in 10 CFR 20.

4.4.2 Heating Effects in Shield and Thermal Column

Heating effects caused by absorption of gamma radiation and fast neutrons are within allowable limits. For all calculations, it was assumed that the pool water was at the 130° temperature limit, and the reactor was operated continuously at 1.5 MW.

The heating in the concrete shield is approximately 20% of the maximum suggested by Rockwell[1]. Analysis of the heating rate in the lead shield for the thermal column indicates that the maximum temperature of the lead will be less than 217°F. Calculation of the graphite temperature in the thermal column indicates a maximum of 244°F.

4.5 Nuclear Design

4.5.1 Normal Operating Conditions and Reactor Core Physics Paremeters

Note: NUREG-1537 specifies separate sections for "Normal Operating Conditions" and "Reactor Core Physics Parameters." These two sections are combined to enable concise inclusion of the measured core parameters of the several cores which have been operated under the license.

4.5.1.1 Core Arrangements

The use of the reactor as a training and research tool requires flexibility of core arrangement. Permitted arrangements are subject, however, to the following criteria:

a. A mixed core must contain at least 9 FLIP fuel bundles (clusters)
b. FLIP fuel must be located in a central contiguous region
c. The core must be a close packed array except for single fuel element positions or fuel bundle positions on the core periphery
d. Calculations indicate that operation of a specific core will be within technical specification limits on power generation per element and fuel temperature.

When the Safety Analysis Report for converting UWNR from flat-plate to TRIGA fuel was written, expected performance was based on computations and on the behavior of a "prototype" TRIGA Mark III reactor, the Torrey Pines Reactor at General Atomics. The prototype reactor used individual TRIGA fuel elements in a right circular cylindrical array typical of TRIGA reactors; UWNR uses four-element bundles in a rectangular arrangement in the grid box provided for the original flat-plate fuel. The uranium loading in the prototype was 8 wt% uranium, while UWNR has a uranium loading of 8.5 wt%. Both the prototype and initial UWNR TRIGA cores had stainless steel clad, and both used 20% enriched uranium. The heat transfer characteristics were quite similar, although the diameter of the clad for UWNR was slightly smaller to fit to the grid box array spacing. UWNR also differed by having shrouds dividing the core box into three regions. These shrouds guide control blades, but also introduce water gaps within the core lattice.

The prototype was operated for many years at steady-state power levels up to 1500 kW and thousands of pulses up to 6000 MW. In this report, although the prototype performance characteristics are indicated and sometimes compared to UWNR, most of the information is based on the measured performance of the cores which are currently operable under the present license and technical specifications.

The current core is an all-FLIP (stainless steel clad) configuration consisting of 23 FLIP fuel bundles. Cores with 9 and 15 FLIP fuel bundles also have been operated for significant times, and have been thoroughly tested for conformance to technical specifications and the predictions and descriptions in this and the previous Safety Analysis Report. It is planned that the all-FLIP core will continue to be the operating core until funding (and analysis) for refueling with LEU is completed, at which time this SAR will be amended. It may become necessary to revert to the 15- or 9-bundle FLIP cores before refueling, however, in order to maintain less than a formula quantity of HEU fuel at non-self protecting levels during the return of the HEU to DOE.

4.5.1.2 Standard TRIGA fuel cores[2]

This core, shown in **Figure 4-17**, and a succeeding core enlarged to 30 fuel bundles because of fuel burnup was operated from November 1967 until March 1974 when it was shut down. Measured core parameters for the initial 25 bundle version of the core are presented below . Several variations in the peripheral reflector configuration were included in this series of cores, including up to 18 reflector elements around the periphery, and some cores with special voided-center or Bismuth-center reflectors.

Core Designation	A25-R10
Excess reactivity	4.82 % ρ
Shutdown Margin	5.17 % ρ
Transient Rod worth	2.10 % ρ
FP reactivity defect	2.95 % ρ
Peak pulse power	1930 MW
Prompt neutron lifetime	42E-6 sec

[redacted]

Figure 4-17 Standard TRIGA Fuel Core

The measured fuel temperatures (**Figure 4-18**) in the UWNR standard TRIGA core almost matched those in the prototype, although this could have differed significantly, depending upon instrumented element placement in the two cores. The UWNR instrumented element was located near the core center in grid position D4NW as indicated in **Figure 4-17**.

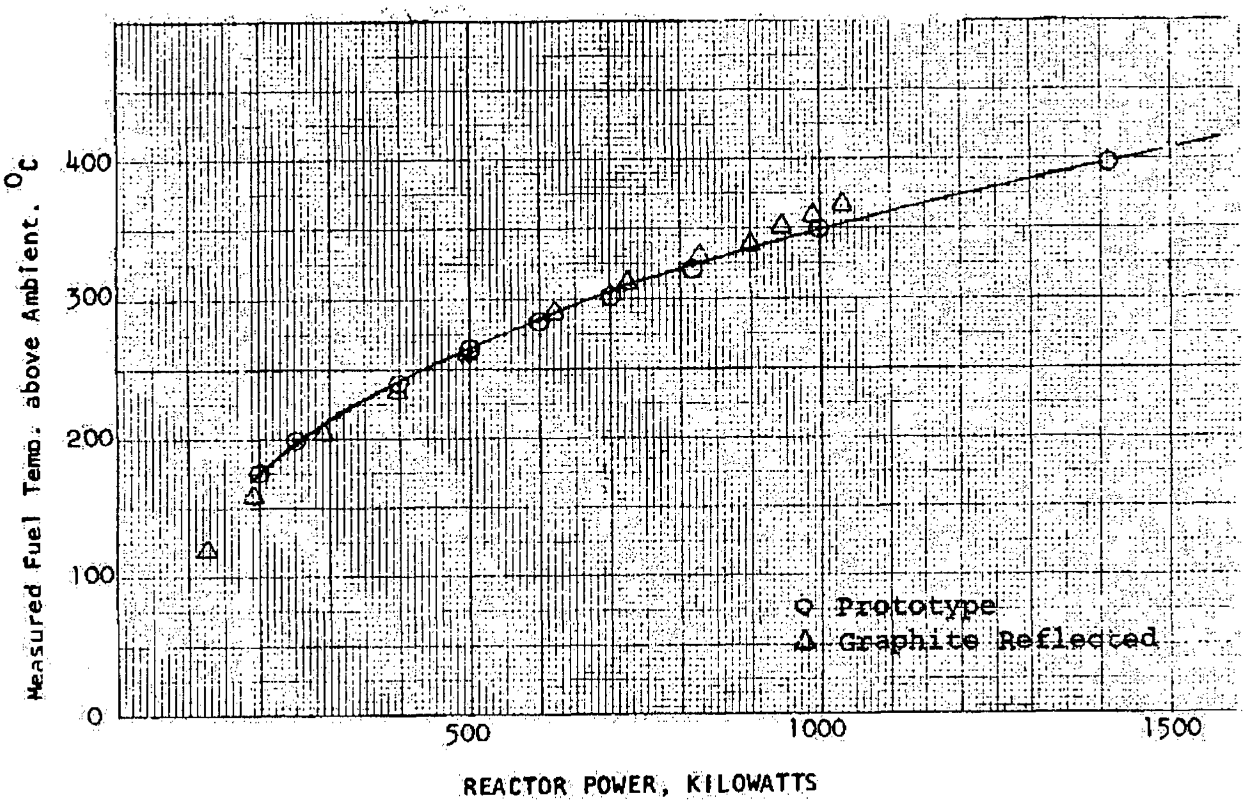

Figure 4-18 Fuel Temperature- Standard Fuel

When the reactivity loss from power operation for UWNR with the original TRIGA core is compared to the prototype (**Figure 4-19**) it is apparent that the power defect for UWNR is significantly larger than for the prototype. In both cases, the core had been pulsed a significant number of times before the temperature measurements were made, so the difference is not from clad stretching in pulsing. The large loss was considered to be a function of the different core geometry and reflection making the leakage change more drastically with temperature in UWNR.

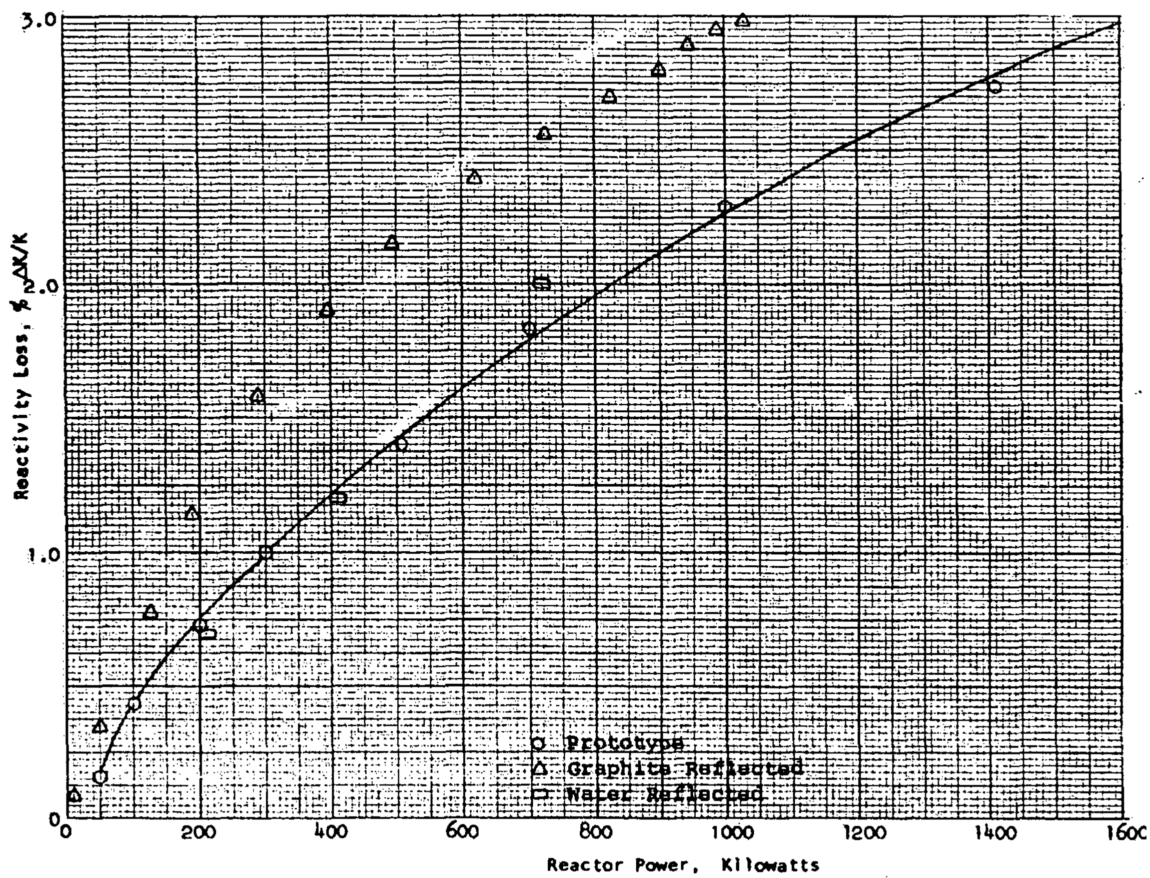

Figure 4-19 Power Defect vs. Power- Standard TRIGA Core

When pulsing behavior of the two cores was compared, other differences were expected and found. The pulsing behavior differed from the typical TRIGA core primarily due to the water gaps in the control blade shrouds and the graphite reflector, both of which increased neutron lifetime, resulting in longer periods for the same pulsed reactivity insertion, and thus broader pulses. Later graphs, **Figure 4-35** through **Figure 4-37**, compare the pulsing behavior of this and all of the other pulsing cores with that of the prototype TRIGA reactor. Note that the pulsed reactivity addition limit for the standard TRIGA fuel was 2.1 % ρ , instead of the 1.4 % ρ for cores containing FLIP fuel. Pulsing behavior differences will be discussed later in this report.

4.5.1.2.1 Reactivity Effects In Standard Fuel Cores

The reactivity effect of fuel bundles in different lattice positions have been estimated from measured and calculated values. The position codes in Table 4-1 are those shown in **Figure 4-15**. The reactivity value given is the worth of adding or removing a fuel bundle while the remainder of a [redacted]-bundle core is already present. The worth of the bundles when added in an approach to critical would be radically different, since cores are loaded as compact cores during the loading sequence; that is, the fuel loading plan is planned to assure that the next fuel bundle loaded will have a smaller reactivity effect than the bundles previously loaded.

Table 4-1 Fuel Bundle Worths in UWNR Cores-all in % ρ

Position	Water Reflected	Graphite Reflected
A	2.60	4.0
B	1.95	3.46
C	1.22	2.18
C'	1.16	1.15
D	0.77	-
E	1.76	2.76
F	1.85	1.55

The reactivity effects of reflector variations also have been measured and/or estimated and are indicated in Tables 4-2 and 4-3. First, the worth of both a graphite reflector element and a voided reflector element are indicated relative to a water reflector, with the position codes being those shown in **Figure 4-15** .

Table 4-2 Reflector Element Reactivity Worths

Position	Replace water with graphite % ρ	Replace water with air % ρ
1	1.228	-0.239
2	0.180	-0.198
3	0.058	-0.064
4	0.076	-0.083
5	0.115	-0.126
6	0.157	-0.172
7	0.029	-------

Next, the effect of other changes that affect reflection are indicated. Most of these were measured in a core of standard TRIGA fuel.

Table 4-3 Reactivity Effect of Reflector Region Changes

Condition	Result-%ρ
Flooding all 4 beam ports	+0.0005
Flooding pneumatic tube	+0.002
Pneumatic tube samples water filled Cadmium filled	-0.0003
Dropping fuel bundle on top of core	+0.5
Adding fuel bundle on side of core	+0.77

4.5.1.3 Cores containing FLIP fuel

The longer operating lifetime for FLIP fuel was the major reason for selecting this fuel type for refueling the University of Wisconsin Nuclear Reactor. The higher enrichment of FLIP fuel coupled with erbium poisoning provides the longer operating lifetime, but it also causes changes in operating characteristics relative to standard fuel. The prototype FLIP core was also the Torrey Pines TRIGA Mark III fueled with FLIP fuel. The most marked changes from use of FLIP fuel are a reduction of prompt neutron cycle time to about 10E-6 seconds at beginning of core life (20E-6 at end of core life) and a temperature coefficient that is strongly temperature dependent. (Figure 3-16, page 3 of reference)[3]. These data are for the prototype reactor; values in UWNR were expected to and do differ because of the water gap in the control blade shrouds and the graphite reflector, making the neutron lifetime considerably longer in UWNR FLIP cores than it was in the standard TRIGA core.

In addition, the harder spectrum in a FLIP core leads to power peaking in regions near water gaps. This leads, in a compact core, to a peaking factor within a FLIP element of 1.43. If a large water-filled flux trap is located adjacent to an element, the peaking factor in the element can increase to 2.65 peak/average within the cell.

Thermal and hydraulic parameters of FLIP fuel remain the same as standard fuel.

FLIP fuel elements are not mixed with standard elements in the same fuel bundle at Wisconsin. Thus, the smallest increment of FLIP fuel addition possible will be three FLIP elements (in a bundle containing the transient rod guide tube). Placing such a bundle in the center of a 5 x 5 array of standard TRIGA fuel leads to the highest value of power peaking possible, with resultant power generation of 31.2 kW in each element. Although no operation with this core is anticipated or desired, other TRIGA reactors have operated with power generation rates at least as high as 32kW per element.

Addition of less than five FLIP fuel bundles (24 FLIP elements) was not considered useful for a full power operating core, since it would not provide sufficient additional reactivity to compensate for burnup in the standard elements.

Calculations were performed for cores with 1, 2, 5, 9, 15 and 25 FLIP -bundles in central contiguous regions of the core. All calculations were for a 5 x 5 array of fuel bundles with the transient rod guide tube in the fuel bundle at grid-position D5. Calculations were performed with a two-dimensional diffusion theory code (Exterminator 2). Standard seven group cross sections obtained from Gulf-General Atomic were used in the calculations. The accuracy of the calculations was checked by analysis of cores with known values of K_{eff} and power density. Results on calculation of mixed cores and FLIP cores were found to be consistent with similar calculations performed elsewhere[4]. Subsequent computations using a 3-dimensional diffusion theory code (DIF-3D) in support of use of LEU fuel agreed well with the results for Exterminator-2 for the all-FLIP core.

FLIP fueled cores can experience significant power peaking which must be considered in permissible fuel arrangements and setting of limiting safety system settings. The power produced in individual fuel elements was predicted from the computations done for safety analyses. The following table shows both the power density in individual fuel elements and the worth of a fuel bundle loaded in a particular location (if applicable to the condition) for several different core arrangements analyzed. See **Figure 4-15** for position descriptions.

Table 4-4 Maximum Power Density and Reactivity Worth of Fuel Bundles- Cores Containing FLIP FUEL

Core arrangement (keyed to fuel bundle locations in **Figure 4-15**) NOTE: D5 is a 3-element bundle with transient rod guide tube	Power in Maximum Element- kW	Reactivity Effect of Removing Fuel Bundle in Indicated Position-%ρ
5 FLIP +20 Standard fuel bundles (FLIP in Positions A & B)	21.4	----
Replace FLIP in E5 with H_2O	28.3	2.83
9 FLIP +16 Standard fuel bundles (FLIP in Positions A, B, and E)	18.1	----
Replace FLIP in D5 with H_2O	20.0	0.93
Replace FLIP in C5 or E5 with H_2O	25.9	1.69
Replace FLIP in D4 or D6 with H_2O	23.2	0.98
Replace FLIP in E4, C4, C6, or E6 with H_2O	22.3	1.49
15 FLIP +10 Standard fuel bundles (FLIP in Positions A, B, C, E, &F)	17.2	----
Replace FLIP in D5 with H_2O	19.0	0.87
Replace FLIP in C5 or E5 with H_2O	24.6	1.65
Full FLIP- 25 FLIP fuel bundles (FLIP in all positions except D)	15.5	----
Replace FLIP in D5 with H_2O	17.2	0.79
Replace FLIP in C6 or D6 with H_2O	20.0	0.51
Replace FLIP in C5 or E5 with H_2O	22.1	1.42

Reference to the table above (Table 4-4) shows that power generation in any individual element is well below 23 KW in all compact FLIP fuel arrangements. Further, the presence of a 3-inch square water gap in the FLIP fuel region will result in power generation rates below 23 KW/element in most of the cores.

The initial operational mixed core contained nine FLIP fuel bundles (35 elements), and the calculations indicate that flux traps could not be permitted for full power operation in this arrangement for locations C5, E5, D4, and D6 if the maximum kW/element is to be kept below

23 kW. The combination of fuel bundle proximity to control blade shrouds and the transient rod guide tube causes the greatest power peaking in any of these cores.

It is also apparent from comparing Tables 4-1 and 4-4 that the reactivity worth of an individual FLIP bundle is lower than that of a standard fuel bundle, even in mixed cores.

The Technical Specifications under which the facility has operated since conversion to FLIP fuel required at least nine (9) FLIP fuel bundles (35 elements) in a central contiguous location with no water gaps larger than a single element except on the core periphery. As a result, the maximum power density in any fuel element at 1,000 kW was limited to 18.1 kW for any of the cores considered. This is approximately 11% higher than the maximum in an all standard fuel core.

Three different FLIP-containing cores have been operated at UWNR. Characteristics of each core, measured during the startup and acceptance testing of each core, are shown in the following sections. During core test programs, one quadrant of each core containing FLIP fuel was mapped for temperature by moving an instrumented fuel element into each unique core position. Interpretation of the fuel measurements was complicated by instrumented element failures so that measurements were made with different instrumented elements in different cores as explained below. The standard fuel temperature measurements were made using two different instrumented elements, since the original standard instrumented element failed before the 15-bundle FLIP core was tested. There was, however, only one standard instrumented element in the core at a time, so no comparisons between the indication of the elements in the same core position are available. Two instrumented FLIP elements were available, and both were used in the temperature mapping. The individual FLIP instrumented elements were both placed in at least one common position to enable comparison between the indication of the different elements in the same core position. However, because one instrumented element had all thermocouples fail, three different instrumented FLIP elements have been used in the tested cores, and widely varying temperatures (as much as 130° C) were measured when these different elements were placed in the same core position. This makes interpretation of the predicted and measured fuel temperatures more difficult, but the conclusion reached during the test programs was that the results were reasonably consistent with the predicted values, considering that the non-instrumented fuel assemblies probably have as large a range of heat transfer characteristics as the instrumented elements do. Data tables from these core test programs include fuel temperatures as predicted and measured.

4.5.1.3.1 First mixed core- 9 FLIP bundles and 16 standard bundles[5]

This initial mixed core, **Table 4-4, Figure 4-20** and **Figure 4-21**, was operated from March 1974 through December 1977.

Some parameters of this core were:

Core Designation	F25-R10
Excess reactivity	4.05 % ρ
Shutdown margin	3.60 % ρ
Transient Rod worth	1.37 % ρ
FP reactivity defect	1.92 % ρ
Peak pulse power	805 MW
Prompt neutron lifetime	29.7E-6 sec

[redacted]

Figure 4-20 9-Bundle FLIP Core

Core Position		KW in Element	Predicted Temp. °C	Measured Temperatures (°C) Bdle 41 FLIP	Bdle 42 FLIP	Bdle 22 STD
D 5	NE	18.1	372	Can't Measure		
	NW	16.1	360	Can't Measure		
	SW	18.1	372	Can't Measure		
E 5	NE	17.4	363	402	365	
	NW	16.1	350	395	356	
	SW	16.9	360	402	365	
	SE	17.0	361	397	361	
E 4	NE	15.4	342	382	343	
	NW	15.4	342	383	346	
	SW	16.1	350	384	348	
	SE	17.0	361	380	343	
D 4	NE	16.1	350	397	359	
	NW	16.1	350	399	353	
	SW	17.4	363	400	363	
	SE	16.2	350	398	342	
F 5	NE	9.3	272			290
	SE	6.7	239			238
F 4	NE	8.0	257			260
	NW	8.8	266			280
	SW	6.3	238			233
	SE	5.7	223			216
F 3	NE	5.8	230			200
	NW	6.8	242			222
	SW	4.9	210			185
	SE	4.6	204			175
E 3	NE	7.4	250			240
	NW	6.6	237			269
	SW	6.9	242			275
	SE	7.4	250			246
D 3	SW	6.9	242			280
	SE	8.3	261			260

Figure 4-21 Power/element and Temperature -9 FLIP Bundle Core

Fuel temperatures in the 9-Bundle FLIP core are shown in **Figure 4-22**. Instrumented elements were located in grid position F5NE for bundle 22 (standard fuel), grid position C4SW for bundle 41, and grid position E5NE for bundle 42. The instrumented element in bundle 42 was located immediately adjacent to the transient rod guide tube. Although the power density was much higher in the FLIP fuel in this partial FLIP core, the fuel temperatures were reasonable.

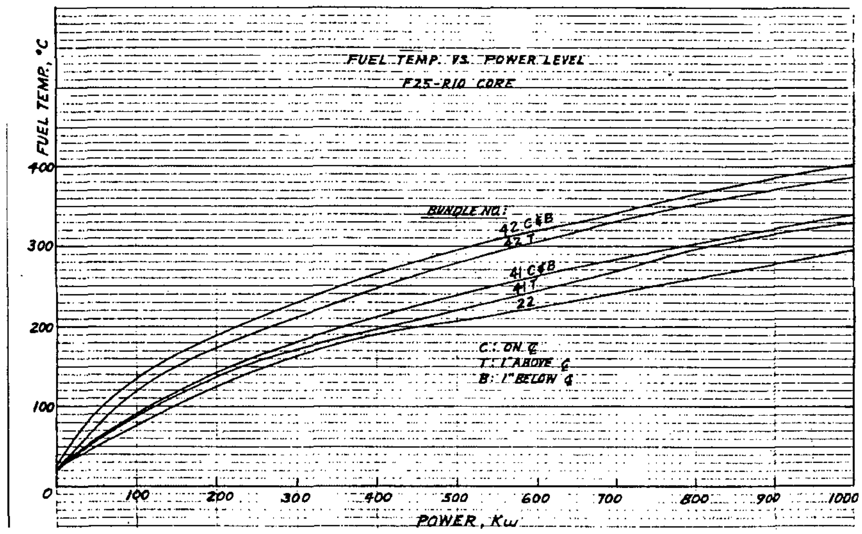

Figure 4-22 Fuel Temperatures vs. Power- 9 Bundle FLIP

Power Defect vs. Power - 9 Bundle FLIP

The power defect was strongly affected by the FLIP fuel. See **Figure 4-23**. Since the temperature coefficient in FLIP fuel becomes more negative as temperature rises, the total power defect during normal full power operation is smaller, while accident response remains essentially the same as in standard TRIGA fuel.

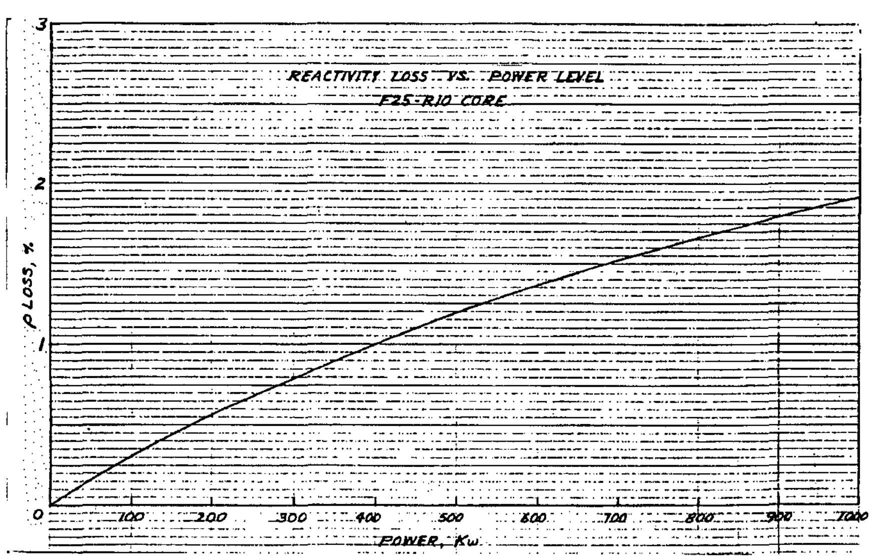

Figure 4-23 Power Defect vs. Power - 9 Bundle FLIP

4.5.1.3.2 Second mixed core - 15 FLIP fuel bundles and 10 standard bundles[6]

This core, shown in **Figure 4-24** and **Figure 4-25**, was operated from January 1978 until June 1979, although some initial high power operation with a core intermediate between this and the all FLIP core was done to make the initial shipment self-protecting before the final batch of fuel was received.

Some parameters for this core were:

Core Designation	G25-R10
Excess reactivity	3.87 % ρ
Shutdown Margin	3.90 % ρ
Transient Rod worth	1.38 % ρ
FP reactivity defect	1.75 % ρ
Peak pulse power	930 MW
Prompt neutron lifetime	24E-6 sec

[redacted]

Figure 4-24 15 Bundle FLIP core

			Measured Temperatures (°C)		
Core Position	KW in Element	Predicted Temp. (26 or 41)°C	Bdle 41 Center TC	Bdle 42 Center TC	Bdle 26 Center TC
D5 NE	17.2	363	CAN'T MEASURE		
NW	16.0	349	CAN'T MEASURE		
SW	17.2	363	CAN'T MEASURE		
E5 NE	16.5	355		435	
NW	15.3	342		---	
SW	15.8	347		361	
SE	15.8	347		361	
E4 NE	11.9	304		340	
NW	14.6	334		355	
SW	15.0	338		345	
SE	13.3	318		322	
D4 NE	12.5	309			
NW	15.2	340			
SW	16.4	354	269	400	
SE	12.6	310		359	
E3 NE	9.7	277	237	415	
NW	9.9	280	290		
SW	11.2	294	247		
SE	10.6	288	294		
D3 NE	11.0	293			
NW	10.4	285			
SW	10.4	285	301		
SE	11.0	293	250		
F5 NE	8.8	267			271
NW	8.8	267			
SW	6.4	237			
SE	6.4	237			237
F4 NE	7.5	252			246
NW	8.4	262			263
SW	6.1	231			220
SE	5.5	222			213
F3 NE	5.4	220			203
NW	6.3	236			220
SW	4.8	208			177
SE	4.5	200			163

Figure 4-25 Power/element and Temperature - 15 FLIP Bundle Core

With the [redacted]-bundle core, FLIP fuel characteristics become more pronounced . The standard fuel instrumented element in standard fuel bundle 25 was located in grid position F5NE, and the measured temperatures are shown in **Figure 4-26**.

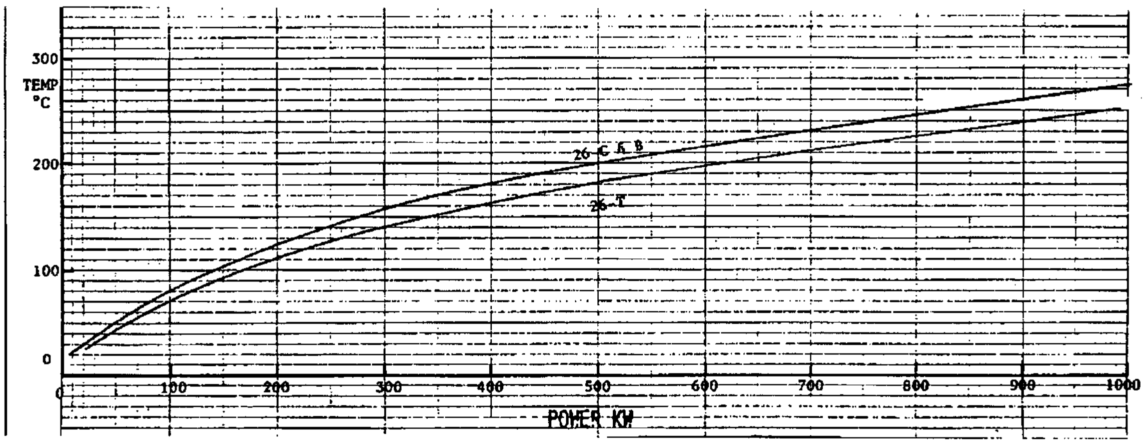

Figure 4-26 Standard Fuel Temperature vs. Power -15 Bundle FLIP

Figure 4-27 shows the temperatures for the instrumented element in fuel bundle F41, located in grid position E3NE. The unusual behavior of the bottom thermocouple (41-B in the figure) was due to the beginnings of failure of the thermocouple due to internal shorting.

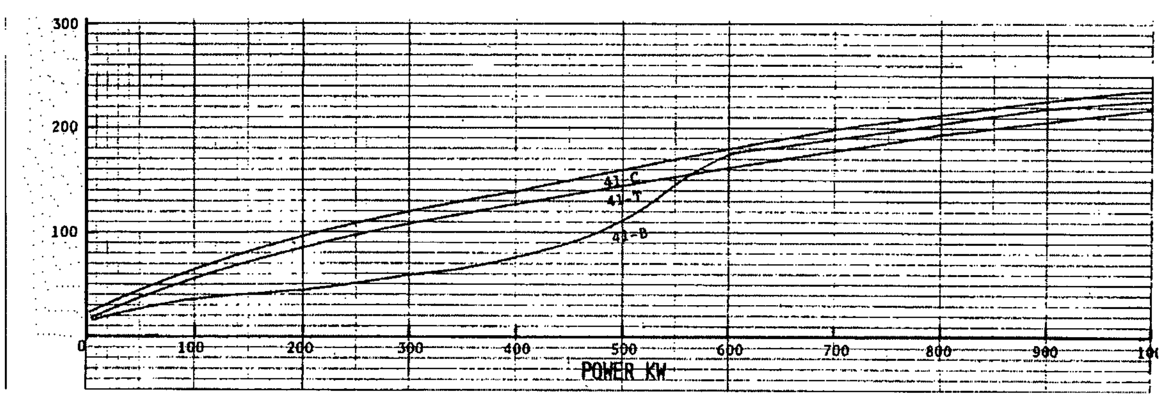

Figure 4-27 Bundle 41 Fuel Temperature vs. Power -15 Bundle FLIP

Figure 4-28 shows the temperature for the fuel bundle F42 instrumented element vs. reactor power.

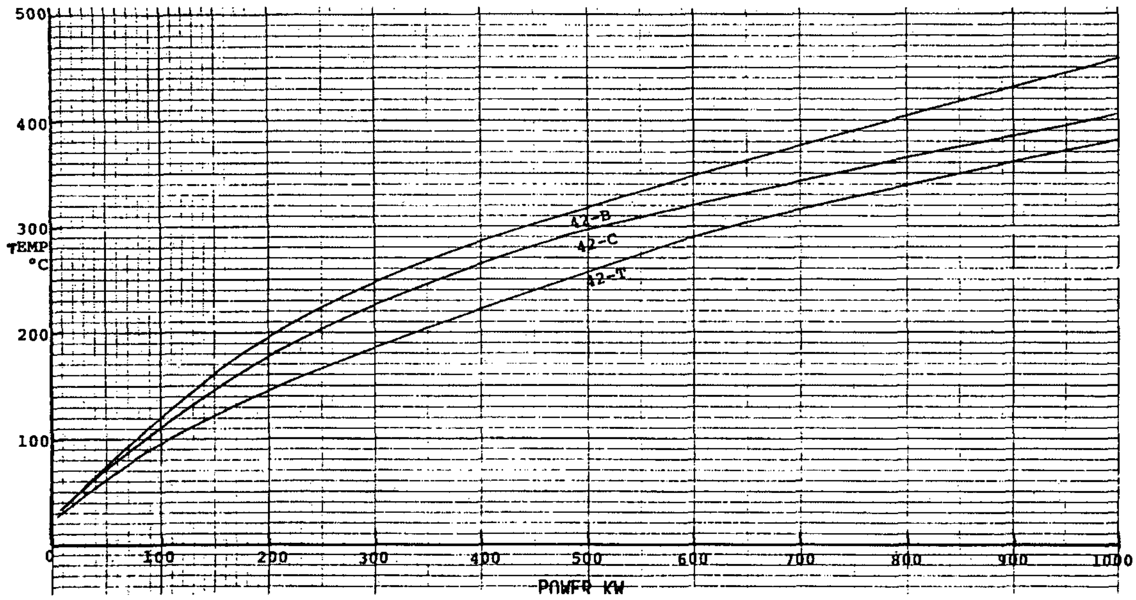

Figure 4-28 Bundle 42 Fuel Temperature vs. Power - 15 Bundle FLIP

The power defect for this core is shown in **Figure 4-29**. Again, the effect of the FLIP fuel results in a still lower power defect with the higher amount of power generation in the FLIP portion of this core. Pulsing behavior also showed a further reduction in the prompt neutron lifetime and thus faster periods for the same reactivity input in a pulse.

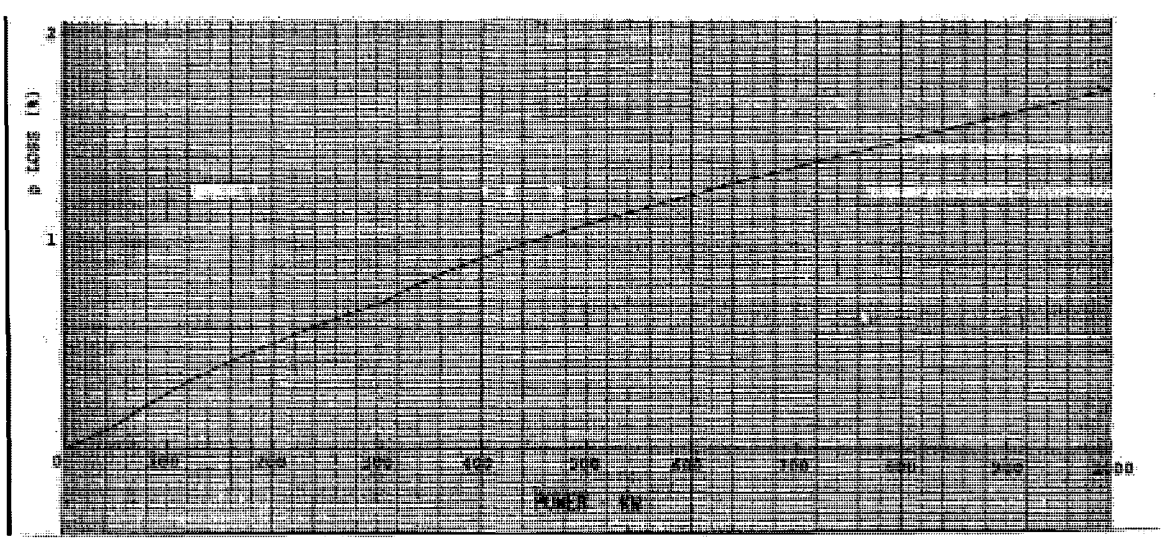

Figure 4-29 Power Defect vs. Power - 15 Bundle FLIP

4.5.1.3.3 All-FLIP core[7]

The reactor has been operated with cores containing only FLIP fuel since July 1979. Two variants of this core have been commonly used, differing only in the number of graphite reflectors used and location of irradiation facilities. The arrangement shown in **Figure 4-30** with the characteristics indicated below and in **Figure 4-31** is the most-used variant.

Some measured parameters for this core were:

Core Designation	I23-R10
Excess reactivity	
4.23 % ρ	
Shutdown Margin	3.80 % ρ
Transient Rod worth	1.395% ρ
FP reactivity defect	1.53 % ρ
Peak pulse power	950 MW
Prompt neutron lifetime	23E-6 sec

[redacted]

				TEMPERATURES AT FULL POWER					
		KW in	Predicted Temp.		#41			#42	
Core Position		Element	(#42)°C	Bot	Ctr	Top	Bot	Ctr	Top
D5	NE	16.0	490		Can't Measure				
	NW	14.9	470		Can't Measure				
	SW	16.0	490		Can't Measure				
E5	NE	15.5	480	365	382	353			
	NW	14.4	460						
	SW	14.6	465						
	SE	14.6	465	340	362	338			
E4	NE	11.9	415	300	315	295			
	NW	13.6	442	334	350	328			
	SW	13.7	450	330	350	326			
	SE	12.1	420	305	325	305			
D4	SW	15.2	475	390	405	380	492	432	407
	SE	15.1	470				426	380	343
E3	NE	14.2	455		277	259	375	337	300
	NW	12.2	420	275	290	270			
	SW	9.9	385	275	285	265			
	SE	10.6	395		241	224			
F5	NE	11.6	415				450	400	350
	SE	7.9	345				341	301	268
F4	NE	9.8	380				410	362	325
	NW	11.0	405				442	395	352
	SW	7.5	340				340	300	270
	SE	6.8	325				305	270	240
F3	NE	8.7	360				318	282	241
	NW	8.1	345				362	320	280
	SW	5.9	300				220	245	280
	SE	6.7	325				242	220	198

Figure 4-31 Power/element and Temperature - All FLIP Core

| Fuel temperatures in the all-FLIP core are shown in **Figure 4-32** and **Figure 4-33**. The bottom thermocouple in fuel bundle 41 had developed a short from one side of the couple to ground, and thus reads well below the actual temperature in this graph. The instrumented element in bundle F41 was in grid position E3NE, while that for bundle F42 was in grid position D4SW next to the transient rod guide tube.

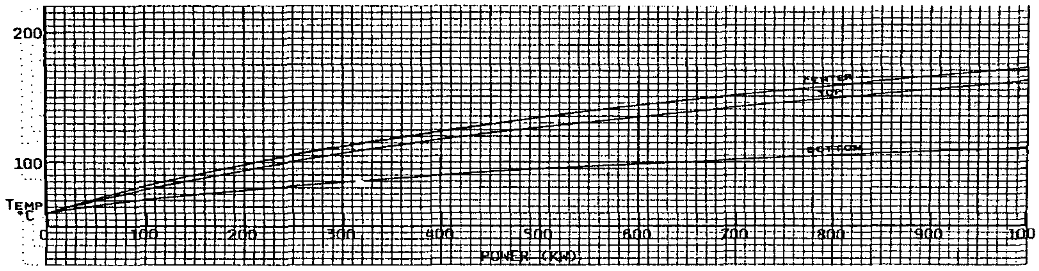

Figure 4-32 Bundle 41 Fuel Temperature vs. Power - All FLIP

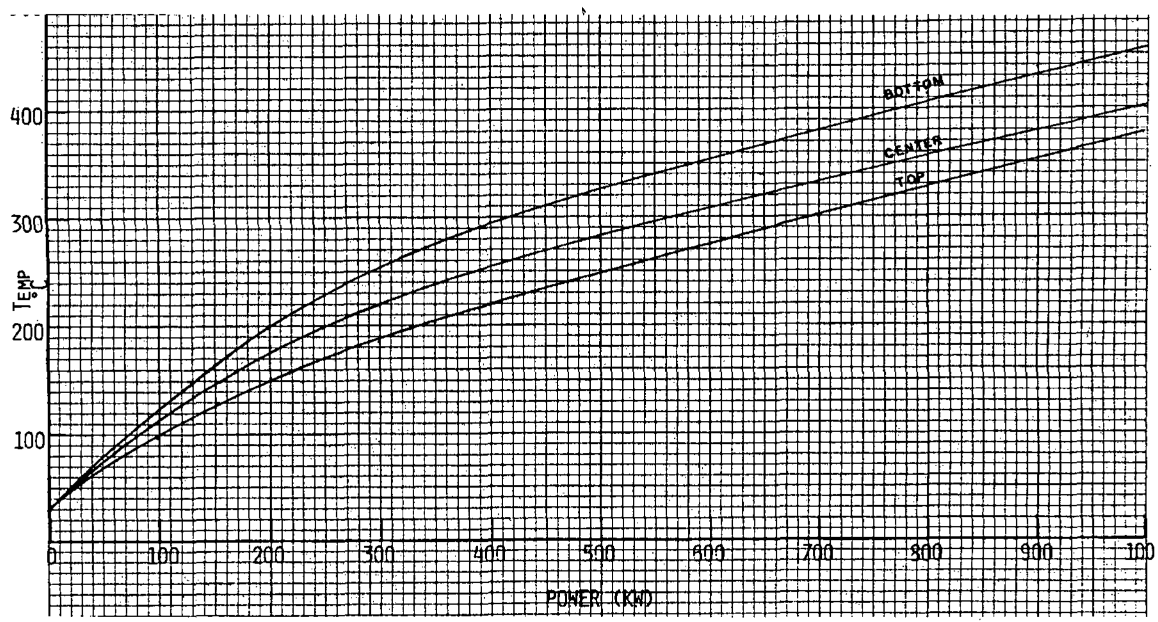

Figure 4-33 Bundle 42 Fuel Temperature vs. Power - All FLIP

Figure 4-34 shows the power defect versus power for this core. In this all-FLIP core the power defect for licensed full power decreased slightly from the 15-bundle FLIP core value. At the end of 1999, after more than 477 MWd operation, the power defect at licensed full power remains at 1.51 % $\Delta K/K$.

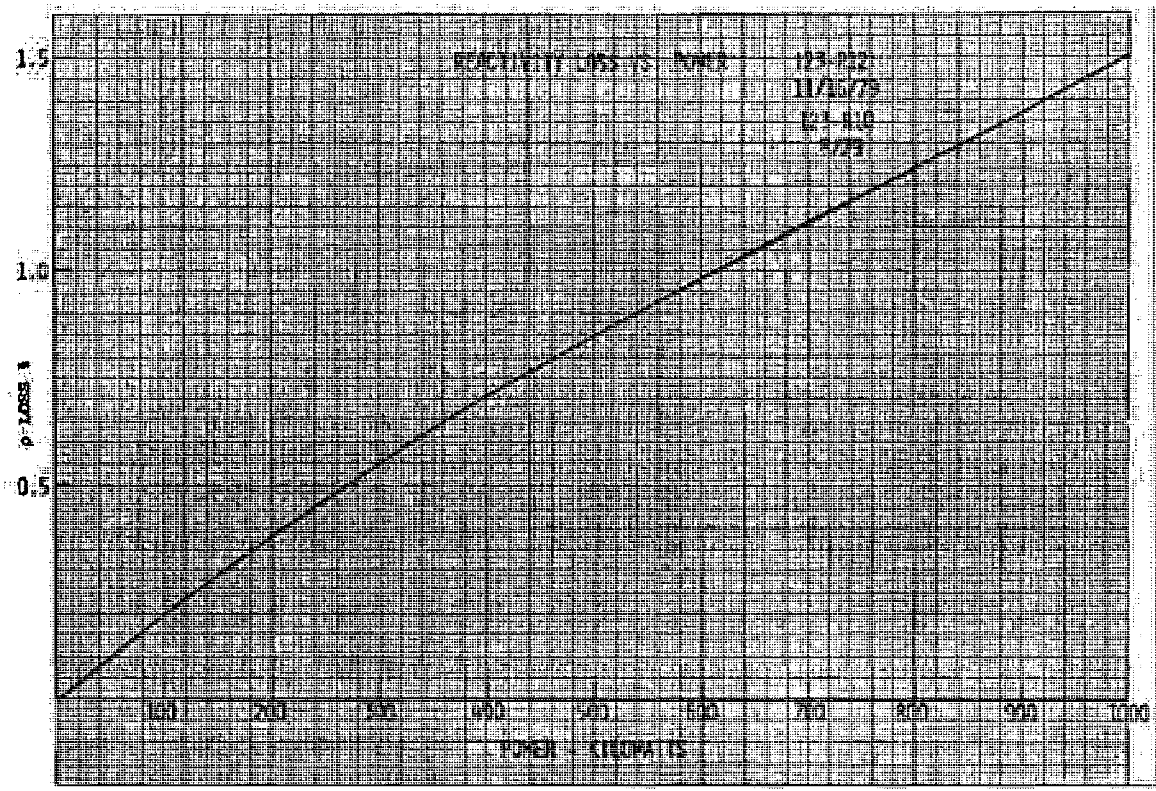

Figure 4-34 Power Defect vs. Power - All FLIP

4.5.1.4 Isothermal Temperature Coefficient

The coolant water temperature in the prototype was varied over wide ranges (20° to 60°C) to measure the resulting reactivity change. The measurements were made at power levels of less than 10 watts. The coefficient is slightly positive with a net gain in available reactivity of 0.077% over the range indicated. The average coefficient,0.0019%/°C, is small enough that it is essentially negligible for normal operating conditions.

The effect of the water gap left in the shrouds when the control blades are withdrawn was expected to increase the temperature coefficient by about 20% in the UWNR, giving a temperature coefficient estimated at 0.0024%/°C. This value was small enough to be considered negligible for normal operating conditions. Values measured during startup testing were 0 and 0.0042, but with vary large uncertainty in the values because of other possible reactivity variations that might occur (bubbles, variation of rest position of control blades and the transient rod, and other extremely small variations). Considering these other variations it is not possible to see any change in reactivity in the UWNR cores as the bulk water temperature changes for either the standard, mixed, or all-FLIP cores.

4.5.1.5 Pulse Parameters

Measurements were made of the various parameters relating to pulsing operation of the prototype and of the UWNR cores. The most important of these are given below for step insertions of reactivity up to 2.1% ΔK/K for the standard-fueled cores and 1.4% ΔK/K for the cores that contain FLIP fuel. The data for the prototype TRIGA Mark III core, the UWNR standard TRIGA core (both water and graphite reflected), and the mixed cores are indicated in the figures referenced below.

Period and Pulse Width

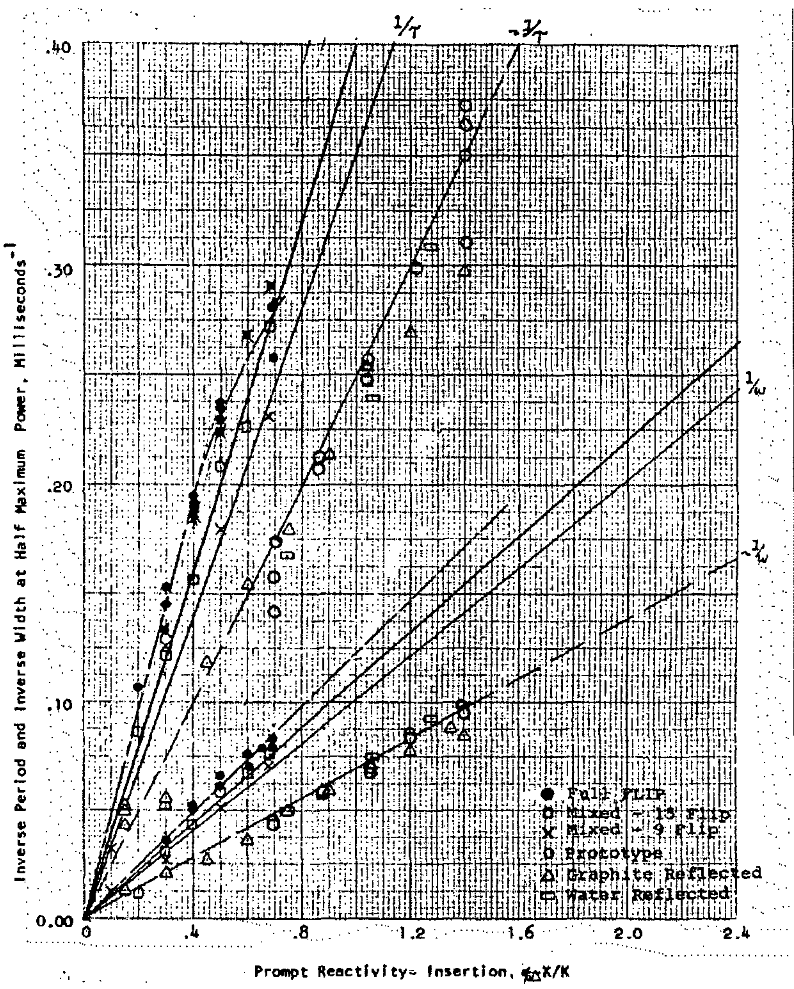

Figure 4-35 Inverse Period and Inverse Width at Half-Max vs. Prompt Reactivity Insertion

During pulsing operation the reactor is placed in a super-prompt-critical condition in which the asymptotic period is related to the prompt reactivity insertion divided into the prompt neutron cycle time. The pulse width is inversely related to the prompt reactivity insertion. Behavior of the different cores and the prototype is indicated in **Figure 4-35** with points of inverse period and FWHM shown on the same graph. The plots show the results of plotting the reciprocal of the measured period versus the prompt reactivity insertion. Since the period data were obtained from an oscillographic recording of the reactor power versus time at a portion of the pulse before fuel temperature limiting effects have begun, the accuracy of the measurements is not as good as for other parameters, and some scatter in the data are expected. As can be seen, the minimum period in standard fuel obtained for reactivity insertions of 2.1% ΔK/K is about 2.6 msec while that of the FLIP core is 3.4 msec for a 1.4% ΔK/K insertion.

As FLIP fuel replaces standard fuel in the mixed cores, the decrease in prompt neutron cycle time results in a different straight line plot for each core. Further, it becomes apparent on the all-FLIP core that the faster cycle time, even for the smaller prompt reactivity insertion allowed for the mixed cores, causes the data for larger reactivity insertions to depart from a straight line. This is because the transient rod has not completed its travel before the reactor reaches a substantial power level, thus resulting in a longer period as negative reactivity is inserted by the fuel temperature coefficient.

Pulse Width

The width of the power pulse is most conveniently described as the time interval between half-power points. Also shown in **Figure 4-35** are plots of the reciprocal of the measured width versus prompt reactivity insertion. Each of the cores has a different straight line plot, again due to the increasingly short prompt neutron cycle time as the amount of FLIP fuel increases.[2]

When the pulse widths (Full Width at Half Maximum power) of the various cores are compared to their prompt period as shown in **Figure 4-36**, all of the cores conform fairly well to the same straight line because the difference in prompt neutron cycle time is not a factor in the relationship between these two variables as it is in the reactivity-period and reactivity-FWHM relationships.

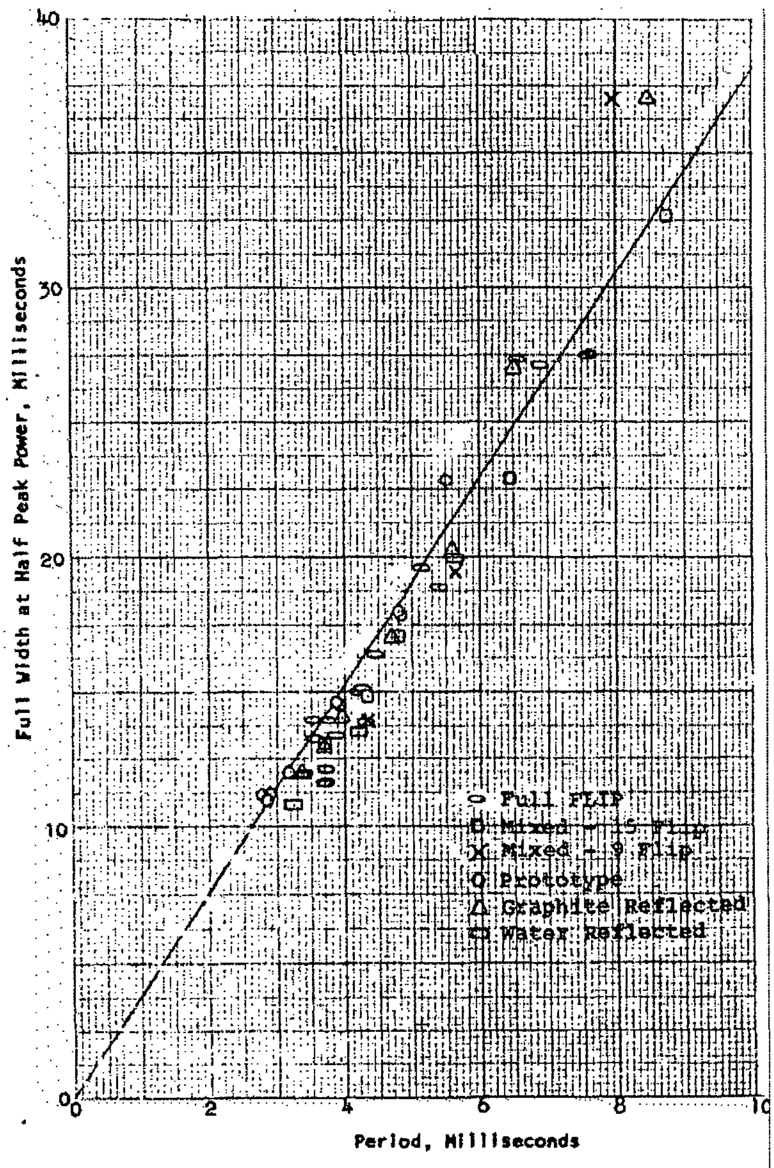

Figure 4-36 FWHM vs Period

Peak Power

Peak power in a pulse is proportional to the square of the prompt reactivity insertion, while energy generated in a pulse is directly related to the prompt reactivity insertion. **Figure 4-37** shows the inter-relationship between maximum transient power and pulse width. Peak power is plotted against the square of inverse FWHM in order to get a straight line plot. The standard and FLIP fueled cores show differences due to the shorter prompt neutron cycle time.

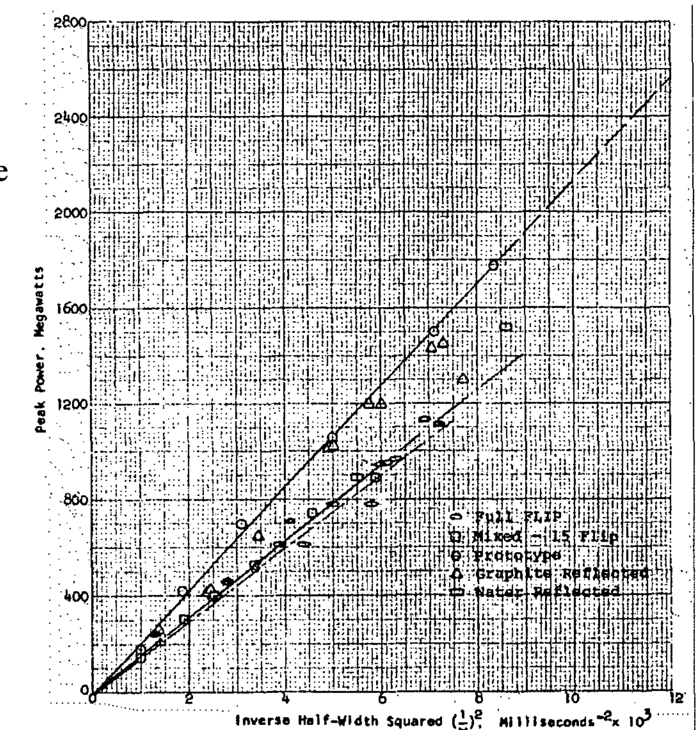

Figure 4-37 Peak Power vs Inverse Half-width Squared

Figure 4-38 shows the relationship between initial period and peak power and seems to be fit fairly well by a single straight line. Note that the prompt reactivity insertion is limited to 1.4 % ΔK/K for the FLIP cores.

For a given core configuration, the peak power, integral power in the prompt burst, and width of the pulse are determined by the reactivity insertion made. It can be seen from the plots that the peak power is controllable over a rather wide range since this parameter is very nearly proportional to $(\Delta K/K - 0.7\%)^2$. Pulse width and integral powers, on the other hand, are approximately linear functions of reactivity insertions above prompt critical so that their range is more limited.

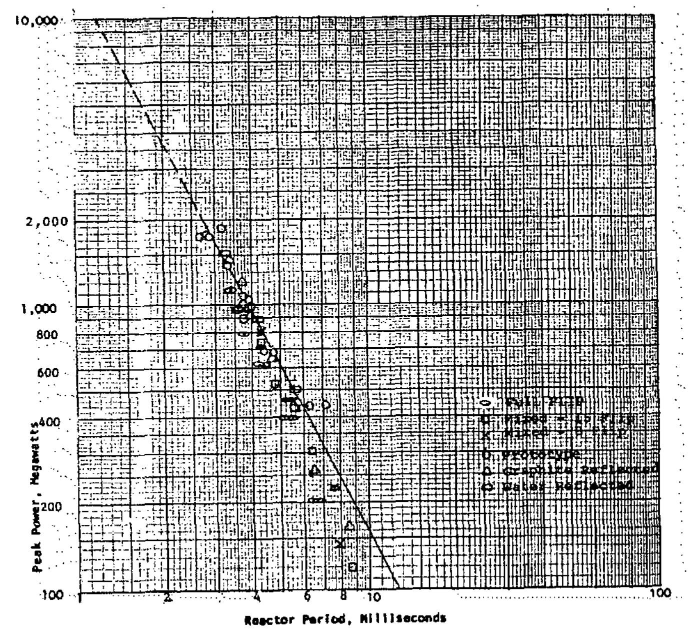

Figure 4-38 Peak Power vs. Reactor Period

4.5.2 Operating Limits

For previous operation of the reactor at 1 MW, the reactivity has been less than 4.9 % ΔK/K above clean cold critical. The reactivity is allocated approximately as indicated below:

Power Coefficient	1.75% ΔK/K
Xenon Poisoning	1.75% ΔK/K
Control & Flux Balancing	1.40% ΔK/K

No specific amount of negative reactivity available from control element action is specified, since the requirement on minimum shutdown margin assures safe shutdown from any operating condition.

The minimum shutdown margin with the most reactive control element and any un-scrammable control elements full out will not be less than 0.2% ΔK/K. Shutdown margin is verified by calibrated control element positions and by rod-drop measurements.

The limitations on cores containing FLIP fuel will maintain power density to levels capable of natural convection cooling during power operation up to 1.5 MW power. This limitation will also assure that power density in any fuel element will be below that at which loss of reactor coolant will result in fuel damage.

In addition, the maximum reactivity for an experiment is limited to 1.4 % ΔK/K All in-pool experiments will be constrained at least as well as the fuel bundles. In-core experiments are designed so they are constrained by the grid or grid box structure, although part of their support may be from other pool structure.

Should an experiment having the maximum reactivity worth allowed for all experiments (1.4% ΔK/K) fail, the resulting step change in reactivity worth would be less than that deliberately inserted during pulsing operation.

Should the beam ports and pneumatic tube flood while the reactor is operating at full power, a step reactivity addition of 0.07% ΔK/K would result. This reactivity change is so small that it would not cause any disruption of normal operation.

If a gross departure from procedure were to be made and a fuel element bundle were added to the outside of the core while operating at full power, the maximum reactivity that would result would be about 0.7% ΔK//K. This is a reactivity smaller than that routinely inserted during pulsing operation.

Despite the built-in safeguards and inherent safety of the reactor and its fuel, great attention is paid to proper supervision of operation and adherence to procedures approved by competent authority. It is the policy of the University of Wisconsin that standard operating procedures are carefully prepared and reviewed, strictly followed, and kept current. Likewise, competent supervision assures that operation is kept within the limits set by licenses, technical specifications, existing procedures, and general good practice.

4.6 Thermal-Hydraulic Design

General Atomics has done extensive thermal-hydraulic computations over the years, including one study specifically directed at the use of four-element bundles of TRIGA fuel as a replacement fuel for reactors which were originally fueled with flat-plate fuel elements and operated at power levels up to 2,000 kW with natural convection cooling[8]. The Puerto Rico Nuclear Center TRIGA-FLIP reactor used 4-element fuel bundles and operated at power levels much higher than the 1,000 kW steady-state power level of UWNR with natural convection cooling[9]. The thermal and hydraulic design for operation of the Torrey Pines Thermionic Reactor, section 3.3 of the reference, describes the core parameters for a very similar core[10]. The conclusions of these references were that four element bundles of TRIGA fuel could be used for power levels up to 2000 kW with natural convection cooling. The University of Wisconsin Nuclear Reactor has been operated with natural convection cooling at steady-state power levels up to 1,000 kW for many years with no cooling problems and no fuel damage. Many other TRIGA reactors have been operated at power levels up to 1.5 MW with natural convection cooling and no cooling problems.

4.7 References

1. Reactor Shielding Design Manual, Theodore Rockwell III, Editor, McGraw Hill Book Company, 1956, page 178.

2. Memo No. 4, Report on Refueling the University of Wisconsin Nuclear Reactor, R. J. Cashwell, Nuclear Engineering Department, University of Wisconsin, March 1968.

3. GA-9064, Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, Section 3.2 and Figure 3-16, General Atomics, Jan. 5, 1970.

4. Same as 2.

5. Core Test Program UWNR Mixed TRIGA-FLIP Core (9 FLIP Bundles),R. J. Cashwell, Nuclear Engineering Department, University of Wisconsin, July 1974.

6. Core Test Program UWNR Mixed TRIGA-FLIP Core (15 FLIP Bundles),R. J. Cashwell, Nuclear Engineering Department, University of Wisconsin, February 1978.

7. Core Test Program All FLIP Core) ,R. J. Cashwell, Nuclear Engineering Department, University of Wisconsin, January 1980.

8. Steady State Thermal Analysis for the Proposed Use of TRIGA Fuel Elements in MTR Reactors, GA-5708, General Atomics, 1965.

9. Safeguards Summary Report for the TRIGA-FLIP Reactor at the Puerto Rico Nuclear Center, PRNC 123, Puerto Rico Nuclear Center, November 11, 1969.

10. Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, GA-9064, Gulf General Atomics, January 5, 1970.

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5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The pool water is cooled by the system shown schematically in **Figure 5-1.** The design basis of the reactor coolant system is to dissipate 1.0 MW with primary temperatures approximately 80 °F and prevent the inadvertent loss of pool water. The system, however, performs no safety function.

The system consists of three loops; the closed-loop primary coolant system, the closed-loop intermediate coolant system and the closed-loop campus chilled water system. Heat from the primary coolant system is transferred to the intermediate coolant system through the primary heat exchanger. Heat from the intermediate coolant system is then rejected to the campus chilled water system through the intermediate heat exchanger. The system is designed to maintain a pressure gradient towards the pool in order to prevent the inadvertent loss of pool water.

5.2 Primary Coolant System

The primary coolant system is composed of a pump, isolation valves and various devices used to extract flow rate, temperatures and pressures. Stainless steel components and piping are used in the system in order to maintain primary water quality more easily. The primary system continuously circulates pool water through the primary heat exchanger. The intake and outlet diffusers include siphon breaker holes to preclude draining more than 1 foot of water even in the case of a pipe rupture. This will maintain at least 19 feet of water above the active core.

5.3 Intermediate Coolant System

The intermediate coolant system consists of a pump, isolation valves and various devices used to extract temperatures and pressures. Stainless steel components and piping are used in the system to maintain reactor grade quality water in the intermediate coolant system. Circulation in this system is maintained by the intermediate pump, discharging through the intermediate heat exchanger, where it will reject heat to the campus chilled water system. The cold water will then circulate through the primary heat exchanger to cool the primary water and return to the pump suction.

The intermediate coolant system is equipped with a pressurized expansion tank and a make-up water system. The expansion tank will accommodate volumetric changes in the intermediate system process fluid and maintain the intermediate system pressure above the primary coolant system pressure under both static and operational conditions. By maintaining the intermediate system pressure higher than the primary system, should a leak occur, it would result in intermediate water entering the primary system thereby protecting against the inadvertent loss of pool water. A pressure sensor provides indication at the control console and an interlock prevents starting the primary pump unless the intermediate loop pump is running.

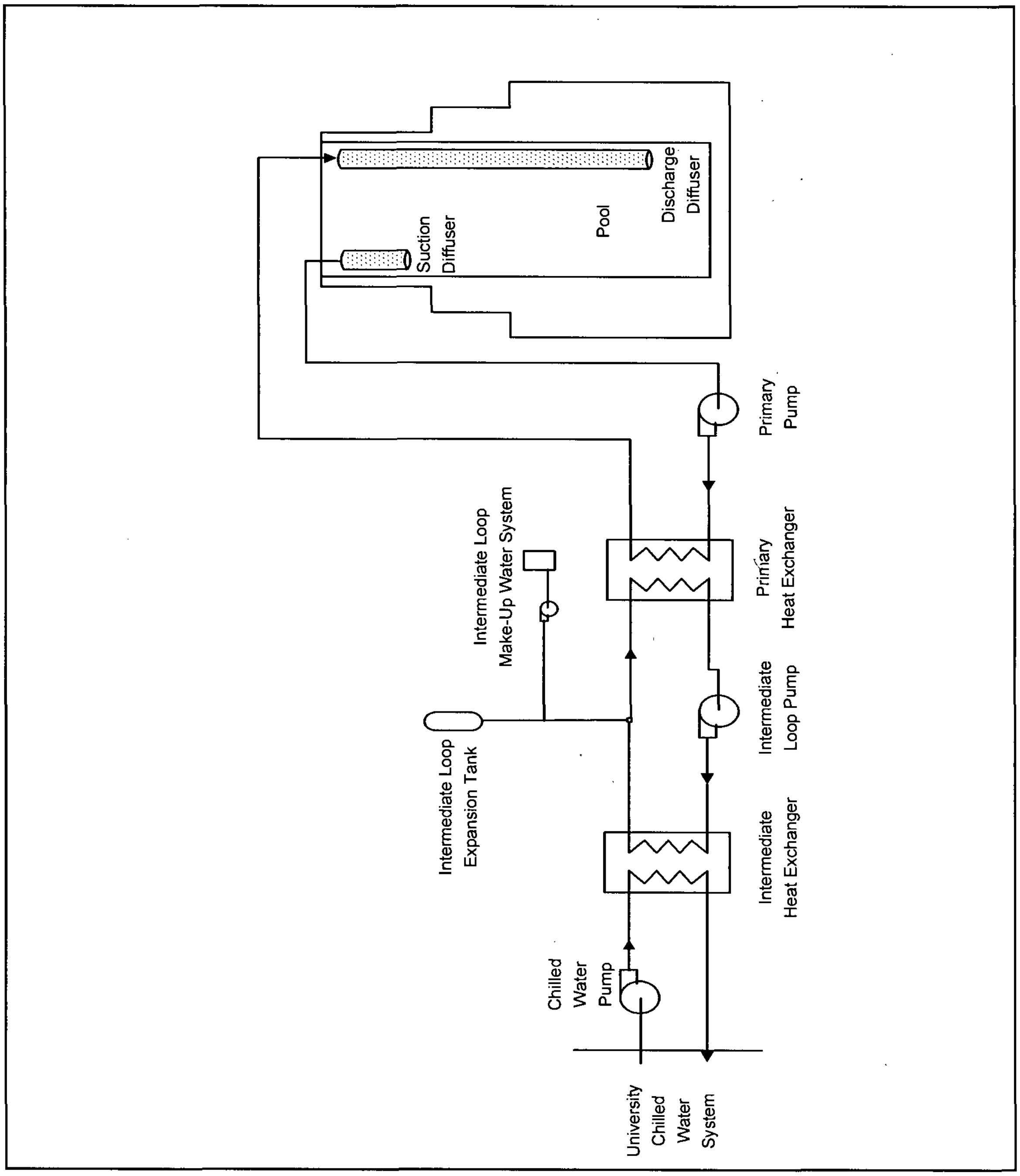

Figure 5-1 Cooling System Schematic

Should leakage occur, it could be detected in three ways. First, the intermediate loop will be unable to maintain pressure and a low pressure annunciator will alarm at the reactor control console. Second, the pool level float switch will be actuated by high pool water level should as much as 150 gallons of intermediate water enter the pool system. Finally, if the integrity of the intermediate water heat exchanger is also compromised, an influx of degraded quality intermediate water will increase conductivity in the pool water.

5.4 Campus Chilled Water System

The campus chilled water system consists of carbon steel piping, pump, isolation valves and various devices used to extract temperatures and pressures. Circulation in this system is maintained by the chilled water pump, taking a suction on the main campus chilled water system. The pump discharges through a filter and into the intermediate heat exchanger, where it cools the intermediate loop water and returns to the main campus chilled water system.

The campus chilled water loop is maintained at a higher pressure than the intermediate system. This pressure gradient will insure that in the extremely unlikely event of leaks in both the primary coolant system and intermediate coolant system heat exchangers and loss of intermediate system pressure that inadvertent loss of pool water will be physically impossible.

5.5 Primary Coolant Cleanup System

Water connections through the biological shield are shown in **Figure 5-2**. The pool clean-up system is shown schematically in **Figure 5-3**. Water is circulated from the pool surface, through the pump, through the demineralizer, and then into the pool under the core box and coolant header. The pump maintains about 18 gallons/minute flow through the demineralizer. The demineralizer is a mixed-bed type with provisions for regeneration of resins or discharge of spent resin and loading with new resin. A water softener supplies softened water for regeneration of the demineralizer.

Flow from the demineralizer to the pool is through valve 10102, check valve 22 which prevents back flow, and valve 719 into the 8 inch pipe loop and into the bottom of the grid box. The 8 inch line is equipped with a siphon breaker at the top of the pool so that rupture of the line at the demineralizer outlet or of the 8 inch line outside the shield cannot drain the pool to a level that will uncover the core. A second 8 inch line is flanged off on both ends. The 8 inch lines were originally installed to allow a forced-convection cooling mode, but the lines are used only as indicated above.

A two inch line whose rupture could have caused loss of pool water has been permanently plugged inside the concrete shield and is presently sealed off outside the shield. A pool drain line and valve have been eliminated. There are no valves in the system that, if opened, can drain the pool.

Should valve number 5 (shown in **Figure 5-3**) be left open upon placing the system in its normal operating condition, as much as 400 gallons of pool water could be pumped to the holdup tank. No further loss of water would then occur, since check valve 22 will prevent reverse flow from the 8 inch pipe loop to the demineralizer and the siphon breaker at the top of the loop will prevent additional water loss.

Wastes from demineralizer regeneration and waste poured down the reactor laboratory floor drain or radioactive sink are collected in the waste system holdup tank. The waste system consists of a 2000 gallon holdup tank, pump, and filter, and is shown schematically in **Figure 5-4**. The holdup tank is periodically sampled, analyzed, and then pumped out into the sanitary sewer through 0.5 micron filters to preclude any particulate activity from being discharged.

All operations involving the cleanup system are performed by written checklist-type procedures designed to prevent draining of the pool.

5.6 Primary Coolant Makeup Water System

The pool makeup water system is shown schematically in **Figure 5-5**. Normally, makeup water is supplied by the still. The still delivers water to a system of storage tanks from which it is pumped (by the pool recirculating pump) into the pool to maintain pool water level. Although distilled water is normally used for makeup, alternate flow paths allow softened or city water to be fed through the demineralizer into the pool. In either case, impurities in make-up water are reduced to less than 1 ppm before going into the pool.

All operations involving the makeup system are performed by written checklist-type procedures designed to prevent draining of the pool.

5.7 Nitrogen-16 Control System

Nitrogen-16 suppression is accomplished by a jet-type diffuser system. This system pumps about 80 gallons of water per minute from near the pool surface through a single nozzle having a 0.75 inch wide, 6.5 inch long opening. The nozzle is located 5.5 feet above and 0.5 feet east of the core, with the diffusing stream directed downward at a 45 degree angle toward the west end of the pool.

The pump for the N-16 suppression system is located on the outside east face of the reactor's concrete pool shield structure, about 8 feet below the pool surface. The system is constructed with siphon breaker holes which preclude draining more than one foot of water from the pool in the event of a pipe rupture.

5.8 Auxiliary Systems Using Primary Coolant

There are none.

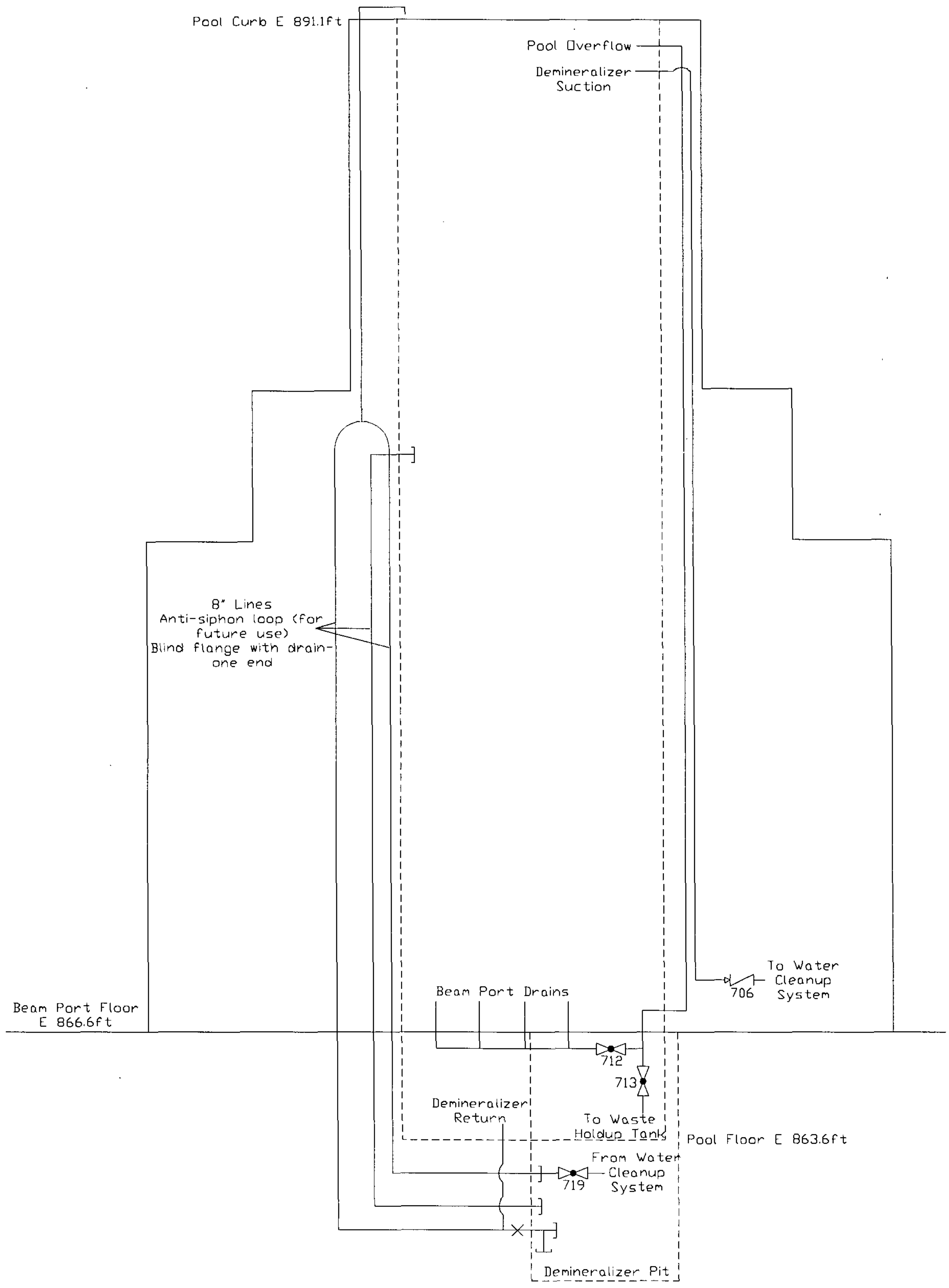

Figure 5-2 Pool Water Systems

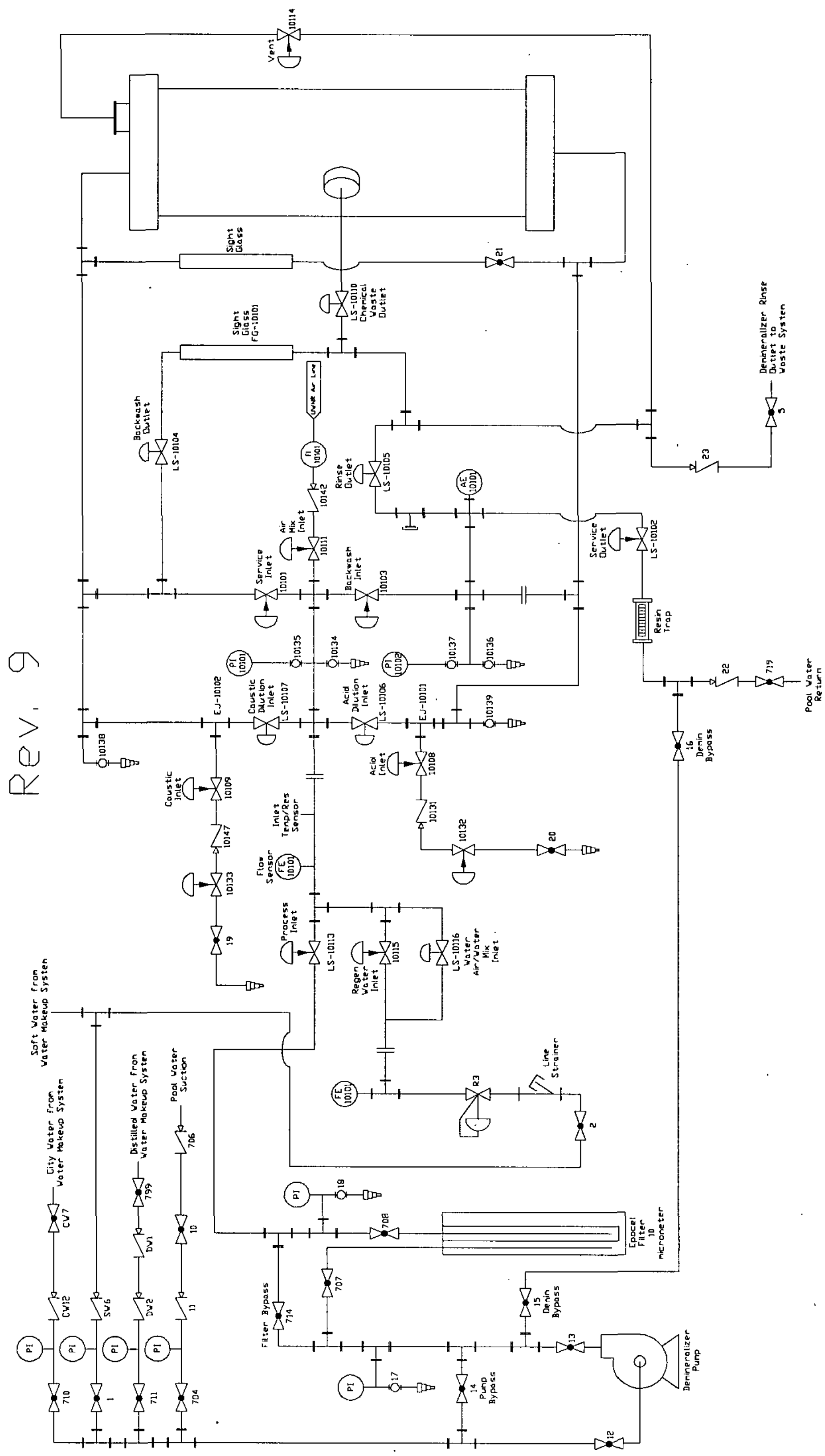

Figure 5-3 Water Cleanup System Schematic

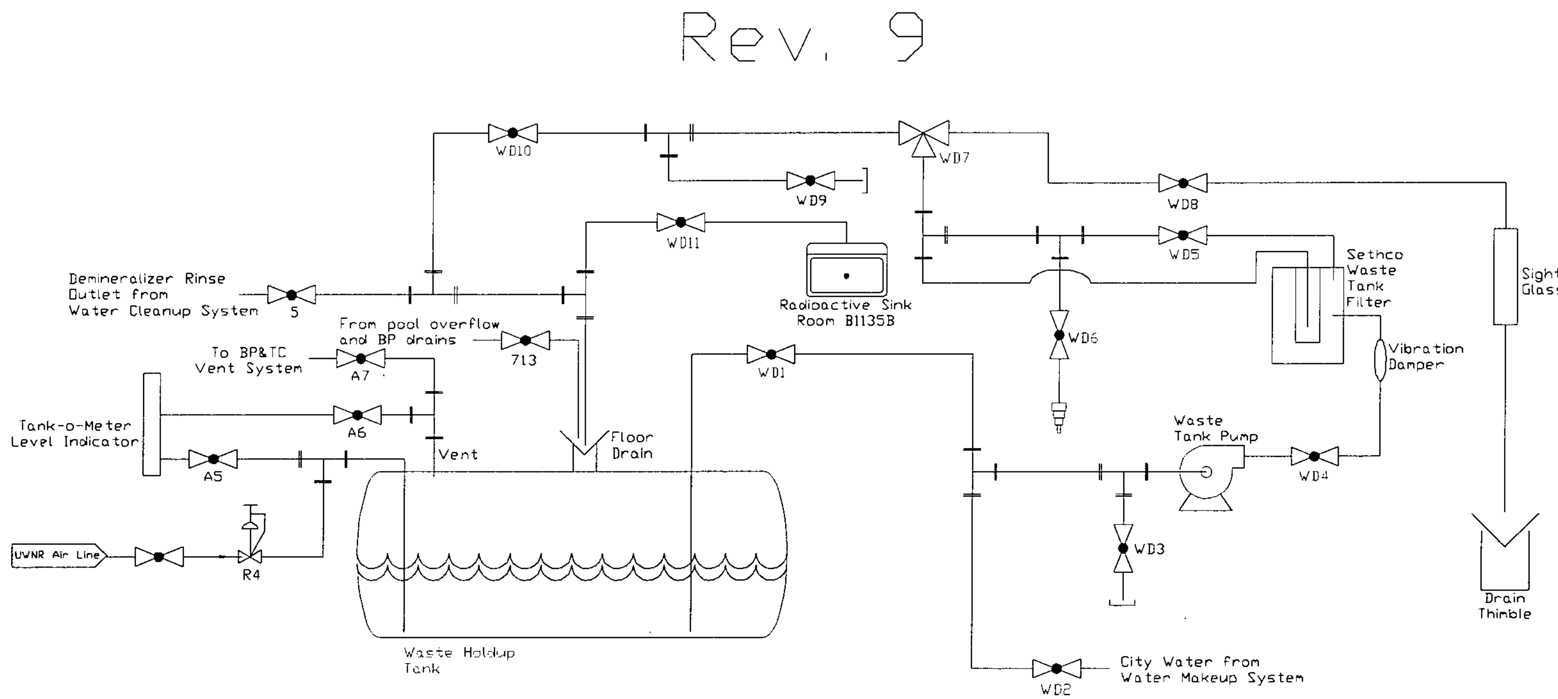

Figure 5-4 Water Waste System Schematic

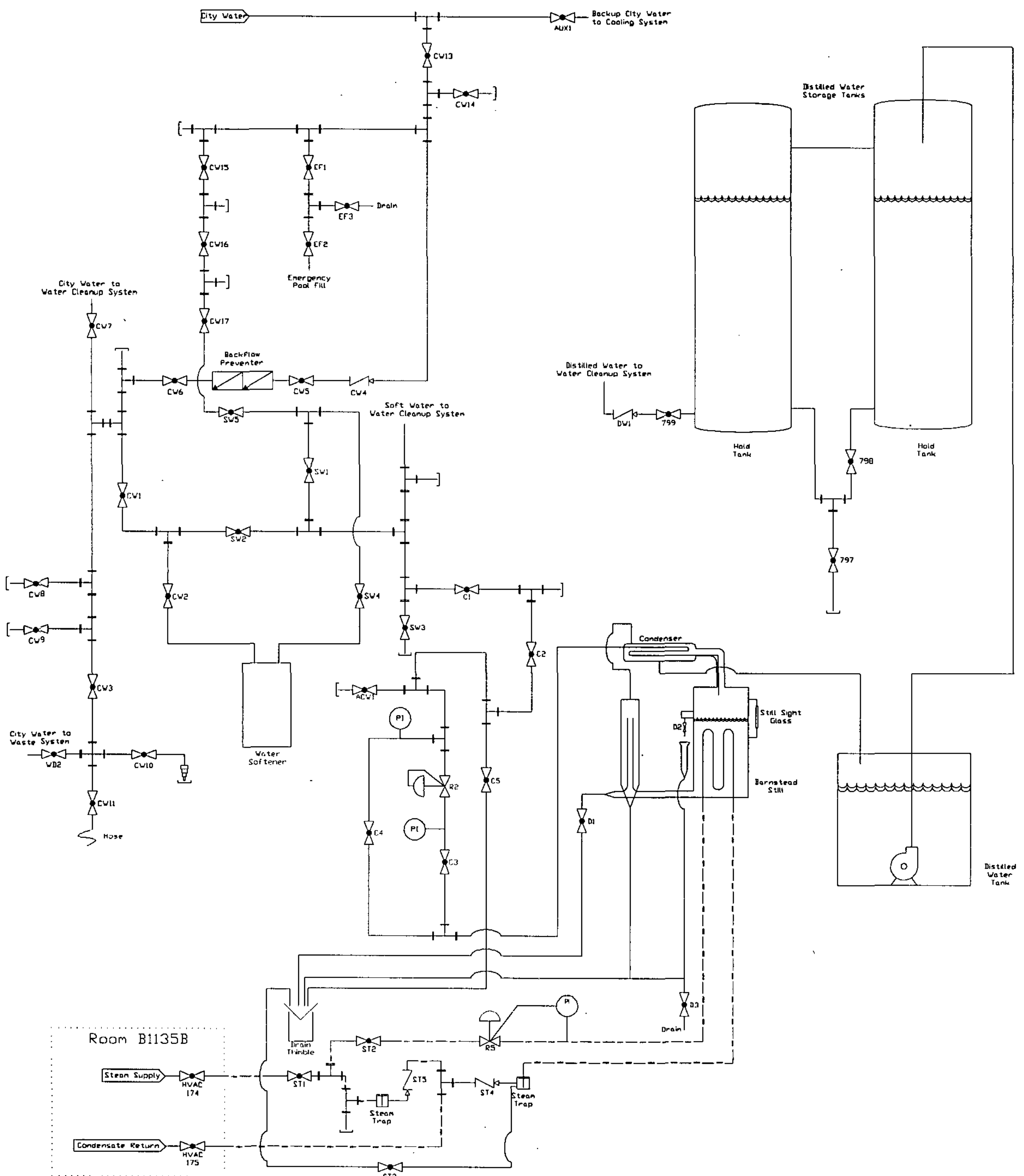

Figure 5-5 Water Makeup System Schematic

5.9 References

There are none.

6 ENGINEERED SAFETY FEATURES

6.1 Summary Description

Engineered safety features are not required for this reactor due to low operating power and good fission product retention in the fuel. A confinement with a controlled ventilation system is provided, however, to reduce the consequence of fission product release from fuel or experiment malfunctions to even lower levels

6.2 Detailed Descriptions

6.2.1 Confinement

The Reactor Laboratory is a 43 by 70 foot room of conventional construction within the Mechanical Engineering Building ,with a ceiling height of approximately 36 feet in most of the room. The portion of the ceiling above the console area is at a height of 22 feet. **Figure 6-1** through **Figure 6-6** show the outlines of the room and location of major reactor components.

The floor of the room is concrete laid on the ground. The walls are concrete and brick. The ceiling is a 1-1/2 inch steel deck with 2 inches of rigid insulation and a 4-ply, built-up surface.

The console area is located in the southwest corner of the Reactor Laboratory. It is separated on the north and east sides from the laboratory proper by wire reinforced glass provided to reduce noise originating from the cooling system and other pumps and equipment. Two doors, one on the east and one on the north of the console area open into the remainder of the Reactor Laboratory. The non-shared use Reactor Laboratory ventilation system provides both supply and exhaust air to the console area. The console is therefore within the confinement system of the Reactor Laboratory

The Reactor Laboratory has no exterior windows, but the control room does have a single borrowed-light window on the south side looking into the visitor's center, room 1101, which has two large windows facing the parking lot and stadium. The control room window does not open. There are three single doors; one opening at ground level on the control room south wall into the visitor's center (which connects to the Mechanical Engineering lobby through a locked door), one opening at ground level on the east wall into a locked vestibule (which connects to a public hallway), and one opening at basement level on the west wall into the Reactor Laboratory auxiliary support space. One double door opens at basement level on the east wall into the Reactor Laboratory auxiliary support space. All doors have narrow viewing windows, are interior doors which do not open to public, un-restricted areas, and are fire doors and are not weather-stripped. No special seals are provided for lines that penetrate the walls. All doors are normally closed and locked for security and air flow control considerations.

The Reactor Laboratory auxiliary support space surrounds the Reactor Laboratory on the west, north, and east basement level (See **Figure 6-1**). This auxiliary support space contains small rooms having concrete walls on the order of 8 inches thick. The small rooms house the pneumatic tube equipment (B1135C) and dispatch station (B1135B), a sample preparation room (B1135D), air activity monitor equipment (B1135E), and both general storage (B1215D) and radioactive storage areas (B1135A and B1215C). A small hot cell is located on the east end (B1215C), an instrumentation shop is located in the north-west corner, while the west end of the room is a counting laboratory with HPGe, NaI, and proportional swipe counters. The east end is used for teaching nuclear engineering laboratory classes (NE 427 and 428).

The ground-floor level of the Mechanical Engineering Building to the east of the Nuclear Reactor Laboratory houses an office area for the Reactor Laboratory.

The Reactor Laboratory is a restricted area. All doors are kept locked at all times except when authorized personnel are in the room. Keys are issued to a small number of authorized personnel.

6.2.2 Containment

No containment is needed or provided.

6.2.3 Emergency Core Cooling System

No emergency core cooling system is required due to the low operating power.

6.3 References

There are no references for this chapter.

[redacted]

Figure 6-1 Reactor Laboratory Basement Floor Plan

[redacted]

Figure 6-2 Reactor Laboratory First Floor Plan

[redacted]

Figure 6-3 Reactor Confinement Cross Section Through Core Centerline, Facing South

[redacted]

Figure 6-4 Reactor Confinement Cross Section Through Core Centerline, Facing North

[redacted]

Figure 6-5 Reactor Confinement Cross Section Through Core Centerline, Facing East

[redacted]

Figure 6-6 Reactor Confinement Cross Section Through Core Centerline, Facing West

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The reactor operates in three standard modes:

Mode 1 Manual or automatic operation at power levels up to 1,000 KW.

Mode 2 Square-wave operation (reactivity insertions to reach a desired steady state power level essentially instantaneously) at power levels between 100 and 1,000 KW.

Mode 3 Pulsed operation produced by rapid transient rod withdrawal that results in a step insertion of reactivity up to the reactivity limit established in the Technical Specifications.

A selector switch is provided to select manual, automatic, square-wave, or pulsing modes of operation.

Operation is from a console displaying all pertinent reactor operation conditions. Instrumentation is entirely analog, except for a digital chart recorder, digital fuel temperature indication, and a computer based pulse recorder used to display the power trace during pulsing operation.

7.2 Design of Instrumentation and Control Systems

Four ranges of radiation-based power instrumentation, with significant overlap, are provided to cover the operating range from source level to the maximum permitted pulse power. Fuel temperature is also measured and used by the reactor protection system. In addition, other process variables are measured, but not used in the reactor protection system. **Figure 7-1** shows the instrumentation and control system for the UWNR.

7.2.1 Design Criteria

The instrumentation and control system provides the following functions:

- Provides the operator with information on the status of the reactor
- Provides the means for insertion and withdrawal of control elements
- Provides for automatic control of reactor power level
- Provides the means for detecting over-power or fuel over-temperature and automatically scram the control elements to terminate the condition
- Provides auxiliary trip functions based on possible loss of operability of the channels providing the overpower protection
- Provides a record of operation and radioactivity discharged from the stack

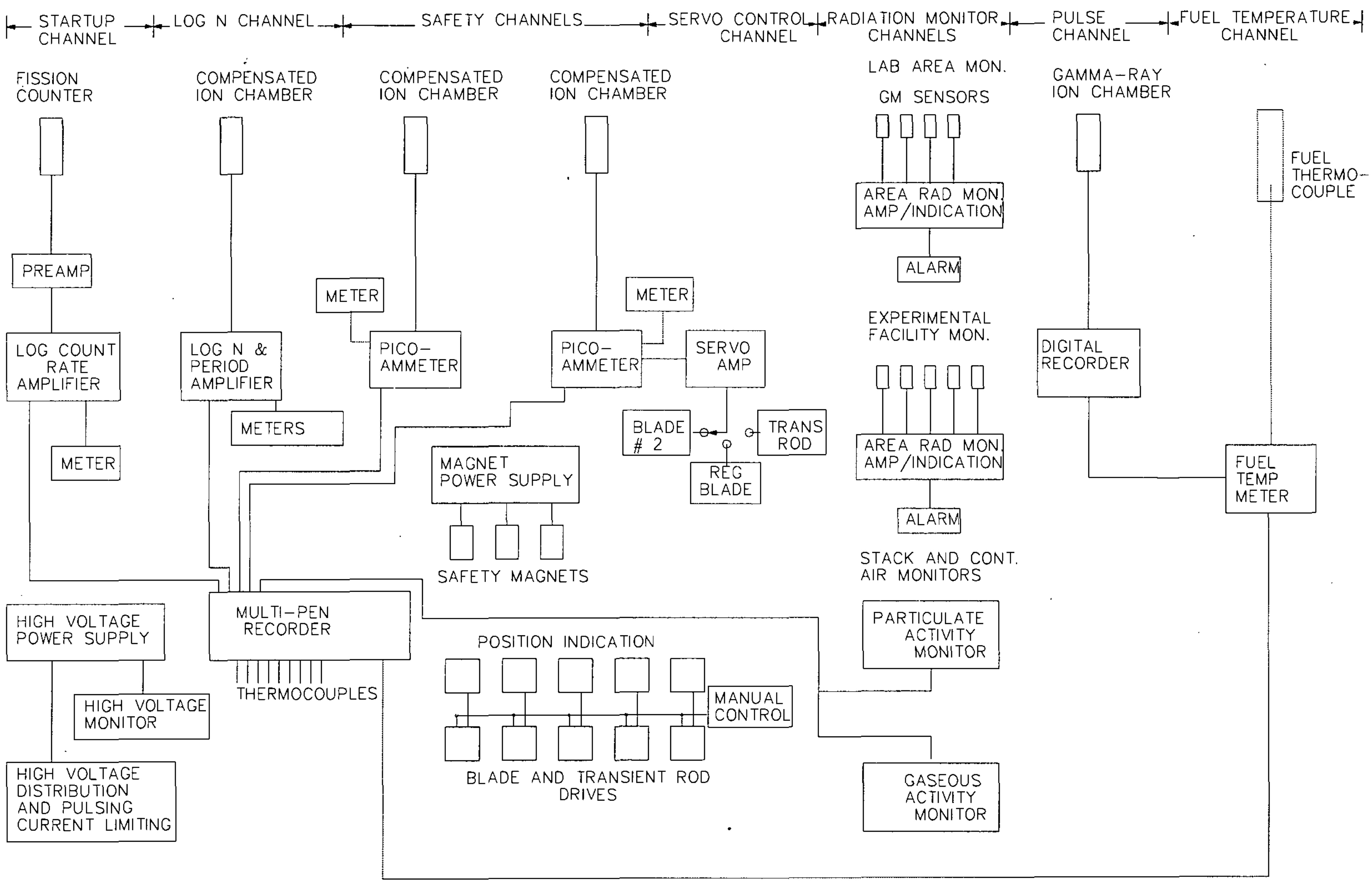

Figure 7-1 Instrumentation and Control Block Diagram

7.2.2 Design-Basis Requirements

The primary design basis for TRIGA reactor safety is the safety limit on fuel temperature. A trip on high fuel temperature is set to assure that the fuel temperature will not be exceeded. Since fuel temperature measurement includes time lag due to thermocouple response time, a reactor trip based on reactor power level as measured by a neutron sensing system is provided.

7.2.3 System Description

7.2.3.1 Start-up Channel

As shown in **Figure 7-1** , the sensing element for this channel is a fission counter with a drive which can be positioned by the console operator. The counter has a range from 2 nv to 10^6 nv. Since the counter is moveable, its effective range is thus from about 2 micro-watts to 2 MW. The pulses from the startup counter are amplified and converted to a logarithmic count-rate displayed on a meter and recorded. The amplified pulses may also be sent to a scaler that is used for subcritical measurements. The amplifier includes a normally open relay which allows control element withdrawal only if the count rate is greater than 2 counts/second. Another normally closed relay provides protection to the fission counter by preventing insertion of the fission counter drive when the count rate is too high. The start-up channel, in the full in position, overlaps the low end of the safety channel range instruments.

7.2.3.2 Log N - Period Channel

This channel monitors the power level of the reactor over the range from 0.1 watt to full power. The Log N - period amplifier detects the signal from a compensated ionization chamber and amplifies the signal to provide a 7-decade logarithmic display proportional to power level. The amplifier also extracts period (startup rate) information. The Log N signal is recorded and operates a normally open relay used in pulse and square wave modes to prevent firing the transient rod when above 1 kW. The period signal is recorded, displayed on a meter on the console, and fed to the automatic control channel when the mode switch is in AUTO mode.

7.2.3.3 Pulse Power Channel

Current from a gamma ionization chamber (or an uncompensated neutron ionization chamber) is fed to a digital data acquisition channel. In "PULSE" mode, a signal concurrent with firing the transient rod causes data to be recorded and displayed on the console computer. Information on peak power and integrated power in the pulse is automatically computed and displayed. Peak fuel temperature after the pulse is also recorded and displayed on the console computer.

7.2.3.4 Safety Channels

Two safety channels monitor reactor power level from about 0.1 watt to full power. The signal from each channel originates in a compensated ionization chamber. The chamber signal is fed into a solid state picoammeter. The picoammeter includes normally open relay contacts which open on an overpower condition to cause a reactor scram. Should either or both safety channel scram signals be present, the reactor shuts down. The power level scram trip point is set to 1.25 times the maximum operating level. The safety channels provide an additional interlock signal to pulse and square wave logic to prevent firing the transient rod unless the picoammeters are on the full-power range.

7.2.3.5 Temperature Measurements

Fuel element internal temperature is indicated at the console. It causes an alarm and scram at the limiting safety system setting.

The temperature of the bulk pool water is measured at the core inlet by a thermocouple. This temperature is indicated on the console recorder and causes an alarm and a scram on high temperature.

Primary cooling, intermediate cooling, and campus chilled water systems inlet and outlet temperatures, and demineralizer inlet temperature are indicated on the console recorder. An alarm on this recorder indicates excessive temperature at any of these points.

7.2.4 System Performance Analysis

The instrumentation and control systems have been in routine operation for the almost 50 years of operating history. All of the measuring instruments have been replaced over the years with instruments incorporating advances in electronics but meeting or exceeding the original design criteria. The switch to entirely solid-state electronics has resulted in marked improvement in stability and reliability.

Limiting safety system settings, limiting conditions for operation, surveillance requirements, and action statements concerning the control and instrumentation systems are detailed in the proposed Technical Specifications in Chapter 14.

7.2.5 Conclusion

Operation during the term of the license has shown the instrumentation and control system to be capable of performing all intended functions with excellent stability and reliability. The system is expected to continue to perform the intended functions.

The safety limits of fuel temperature and reactor power are adequately protected by the safety channels and the fuel temperature channels, each of which will cause a reactor shutdown if the limiting safety system settings are exceeded. Additional components which cause trips on loss of conditions necessary for continued power operation (loss of high voltage to detectors, loss of water from the pool, pool water temperature above the temperature used in calculation of event consequences, loss of electrical power) provide assurance that the primary protection equipment will operate as planned.

7.3 Reactor Control System

7.3.1 Mode Switch

The mode switch and associated logic circuits provide the following capabilities and operating restrictions for the different positions. The switch is a cam operated switch which is rotated through consecutive positions. From " MANUAL", rotating clockwise places the switch in "AUTO" mode; rotating counter-clockwise places the switch first in "SQUARE WAVE" mode, then in "PULSE" mode. Returning from "PULSE" to "MANUAL" requires going through the "SQUARE WAVE" position. **Table 7-1** details the conditions and restrictions invoked in the different mode switch positions.

7.3.2 Manual Operation

For manual operation the control elements are slowly withdrawn to obtain the desired power level. At this level the reactor may continue to be operated manually or it may be switched to automatic control. The automatic control channel maintains power level by servo control of the regulating blade, transient rod, or #2 control blade. **Figure 7-1** shows a block diagram of the control system.

7.3.3 Square Wave Operation

This mode is provided for those applications which require that the power level be brought rapidly to some high level, held there for a period of time, and then reduced rapidly producing a square wave of power.

In the square wave mode the reactor is brought to a level of less than 1000 watts in the manual mode. The mode switch is then changed to the square wave position. Changing of the mode switch to the "SQUARE WAVE" position removes an interlock that prevents application of air to the transient rod unless the transient rod is in the full "IN" position. A preadjusted step reactivity change is then made to bring the reactor to preset power levels between 100 and 1000 kW. The reactivity step change is made with the transient rod. The automatic control system inserts additional reactivity required to maintain the preset power level as the fuel heats up. The operator must manually augment the reactivity inserted by the servo because the transient rod does not have sufficient worth to overcome the power defect at high power levels. The linear power level scram is maintained at 1.25 P max. and an interlock prevents initiation of this mode if the range switch is not on the full power range setting.

7.3.4 Pulsing Operation

The reactor is brought to a power level of less than 1000 watts in steady state mode. The mode switch is then changed to pulsing mode. When the switch is in pulsing mode the normal neutron channels are disconnected and a high level pulsing chamber is connected to read out the peak power of the pulse on the console computer. Changing of the mode switch to the “PULSE” position removes an interlock that prevents application of air to the transient rod unless the transient rod is in the full "IN" position. Fuel temperature is recorded during pulsing operation. The pulse channels are also indicated on **Figure 7-1** .

Table 7-1 Mode Switch Functions

Mode	Switch Position	Conditions/restrictions
Manual	MANUAL	Transient rod may be fired only if scram is reset and transient rod drive is at the "IN" position.
Automatic	AUTO	Same as Manual, except automatic control system can control power, subject to period limit
Square-Wave	SQUARE WAVE	Transient rod can be fired from other than full in position only if scram is reset, both safety channels are on top range, and power level does not exceed 1 kW as indicated by the LogN channel. The period channel in the LogN amplifier is defeated (restored after a short time delay when the switch is returned to "MANUAL" position). Automatic control system can control power if actual power is within ±5% of scheduled power.
Pulse	PULSE	Period channel remains defeated. Prohibits control blade withdrawal. If the scram is reset, both safety channels are on top range, and power level does not exceed 1 kW as indicated by the LogN channel; (a) Transient rod can be fired from other than full in position, (b) High voltage is removed from the fission counter, safety channel CICs , and the LogN CIC . (c) Signals from all CICs are directed to ground rather than to instrument input. (d) The transient rod drops automatically 15 seconds or less after it is fired (e) The pulse power level channel is sent a signal causing it to record the pulse power trace. When returning to the "SQUARE/WAVE" position, high voltage is restored to the detectors immediately and the signals to the neutron measuring instruments are restored after a short time delay (to prevent damage to instrument inputs from the transients resulting from high voltage restoration).

7.3.5 Control Element Operation

There are five control elements; three shim-safety blades, a transient control rod, and a regulating blade. The shim-safety blades and the transient control rod have scram capability.
The following conditions must be met before any control element drive can be withdrawn (raised), either manually or by the automatic control system:

1. No scram conditions present and scram relays reset;

2. Count-rate on startup channel greater than 2 counts per second;

3. Fission Counter not in motion;

4. Console key switch set to "ON" position;

There are no interlocks or permissives which restrict insertion (lowering) of control element drives. Insertion is accomplished by placing the individual momentary-contact control switches in the “IN” position, or by a maintained-contact “RUNDOWN” switch which inserts all control element drives to the “IN” limit. The three shim/safety blade drives also automatically run to the “IN” limit when a SCRAM has occurred.

7.3.6 Safety Blade Control

The three safety blades are manually controlled by individual pistol-grip switches with LOWER, OFF, and RAISE positions, with spring return to OFF. One safety blade may be selected to be controlled by the automatic level control system. The position of each safety blade is indicated by separate digital read-outs, and the indicator lights on the console show when each drive is at its "IN" or “OUT" limit and when the blade magnets are engaged with the armatures . Position indication is accurate to ±0.02 inches.

The safety blades will scram from any position when stationary or during withdrawal and insertion. In the event of a scram the safety blade drives automatically run in to their "IN" limits.

7.3.7 Regulating Blade Control

The regulating blade has identical position indication and "IN" and "OUT" limit indication. It is manually controlled by a separate pistol-grip switch and may be controlled by the automatic level control system .

7.3.8 Transient Rod Control

Manual movement of the transient rod drive is controlled by the console-mounted, switch/light, push buttons which not only control movement, but also indicate in and out limits. Position indication is accurate to 0.02 inches. The transient rod drive may be selected to be controlled by the automatic level control system.

Air pressure is used to fire the transient rod to the selected position in pulse and square-wave modes and to engage and hold the transient rod at the drive position in manual and automatic modes. Additional lighted push-button switches are installed to support these functions. Illumination of the “READY/FIRE T ROD” indicator/switch indicates that the permissives for firing are met and the rod has not been fired. Illumination of the “ENG’D/AIR” indicator/switch indicates movement of the transient rod from the full-in rest position as a result of air having been applied. (Applying air while the transient rod drive is in the full in position causes the piston in the drive to move upward by compressing the spring inside the shock absorber). Pressing the “ENG’D/AIR” switch removes the air from the drive, causing the transient rod to drop to the full-in rest position.

Since the transient rod control is capable of introducing step changes in reactivity up to the Technical Specification limit, logic circuits are provided to assure the transient control rod is fired only under the appropriate conditions.

7.3.9 Automatic Level Control System

The servo amplifier (level controller) controls reactor power level in automatic and square-wave modes. The servo amplifier output drives either the regulating blade, transient rod, or a safety blade as selected by a servo element selector switch on the console. Only one element at a time may be selected; selecting another replaces the previously selected element. The servo amplifier responds to a power level signal from one of the safety channel picoammeters and controls speed and direction of the servo element through a servo motor.

In automatic mode, the automatic level control channel uses period information from the Log N - period channel to limit control element withdrawal to maintain a period longer than a pre-selected level. In square-wave mode a servo error circuit is employed. This circuit allows servo operation only when the servo error is less than 5%. Servo error (the difference between scheduled power in % and the picoammeter indication in % of full scale) is indicated at the console in both "square-wave" and "automatic" modes.

Additional indicators on the reactor control console are provided to indicate "AUTO ON" and scheduled power.

7.4 Reactor Protection System

Scram Circuits

The scram circuit, which initiates shutdown by dropping the shim-safety blades and the transient rod, is shown in **Figure 7-2** . Scram is accomplished by de-energizing the scram relays under one of the following conditions:

1. Manual scram;
2. Fuel temperature above LSSS;
3. Power level greater than 1.25 P max;
4. High voltage failure in control console;
5. Loss of control power;
6. Coolant temperature at core coolant entrance above 130°F;
7. Pool water level high or low.

In addition to these scram functions, the transient control rod logic includes a timer which causes the transient rod to drop to the full in position within 15 seconds after the transient rod has been fired in pulse mode only.

The key-operated console MASTER switch, designated 4S2 on the figures, is a cam-operated switch with a large number of contacts which are selectively operated in the three switch positions. (OFF, ON, and TEST). Six different contact sets must be closed (and are closed only in the "ON" position of the switch) in order to reset the scram relays and apply power to the trip amplifier which supplies DC voltage to the shim-safety blade magnets.

A normally-open contact of Relay 6K1A in the transient rod logic circuit opens when the relay is de-energized, resulting in the drop of the transient rod.

Normally-open contacts of both 6K1A and 6K1B are in series with the alternating current power supply to the trip amplifier as shown in **Figure 7-3** . Magnet power is turned off if either or both of the scram relays de-energizes.

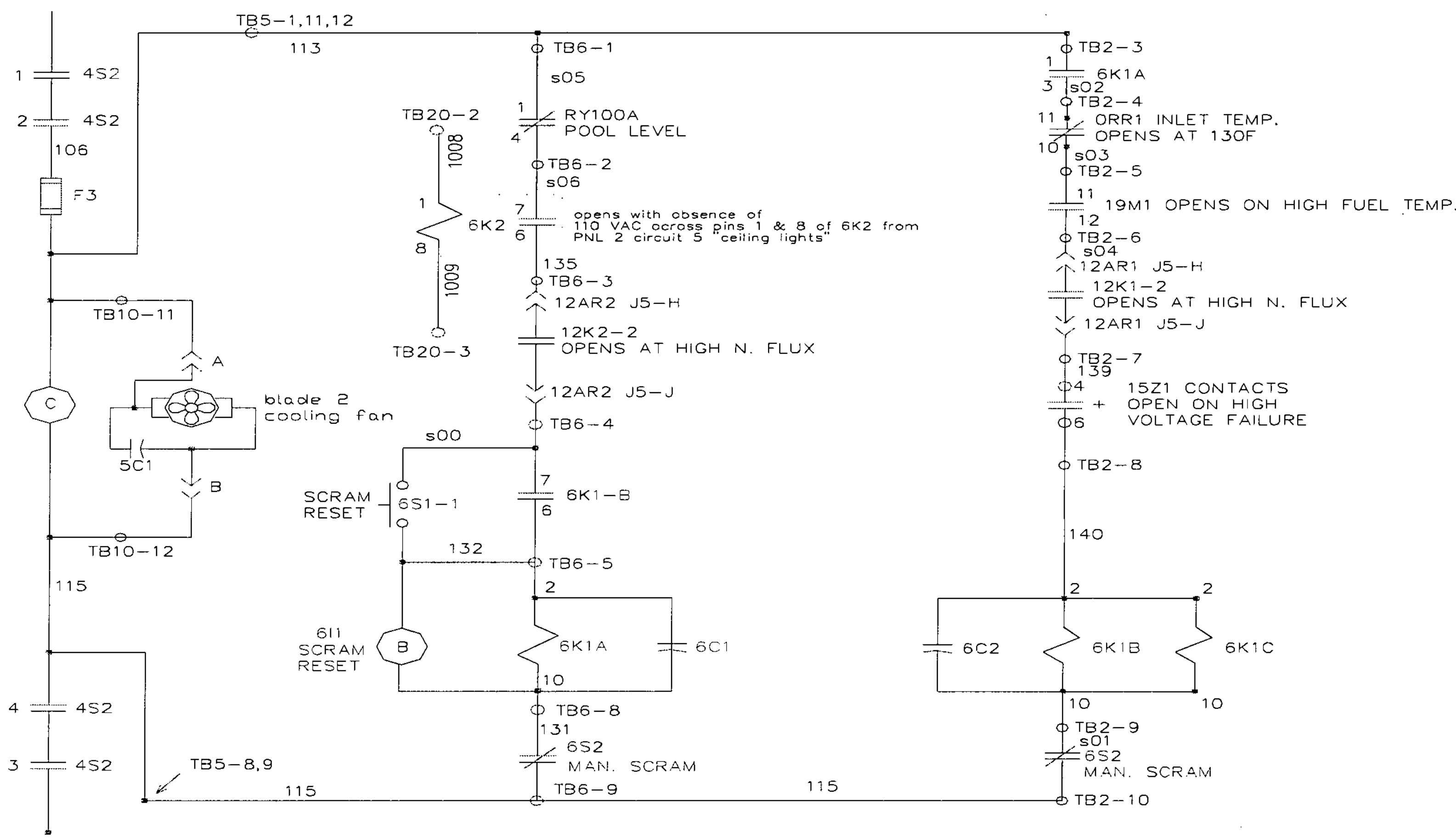

Figure 7-2 Scram Logic Elementary Diagram

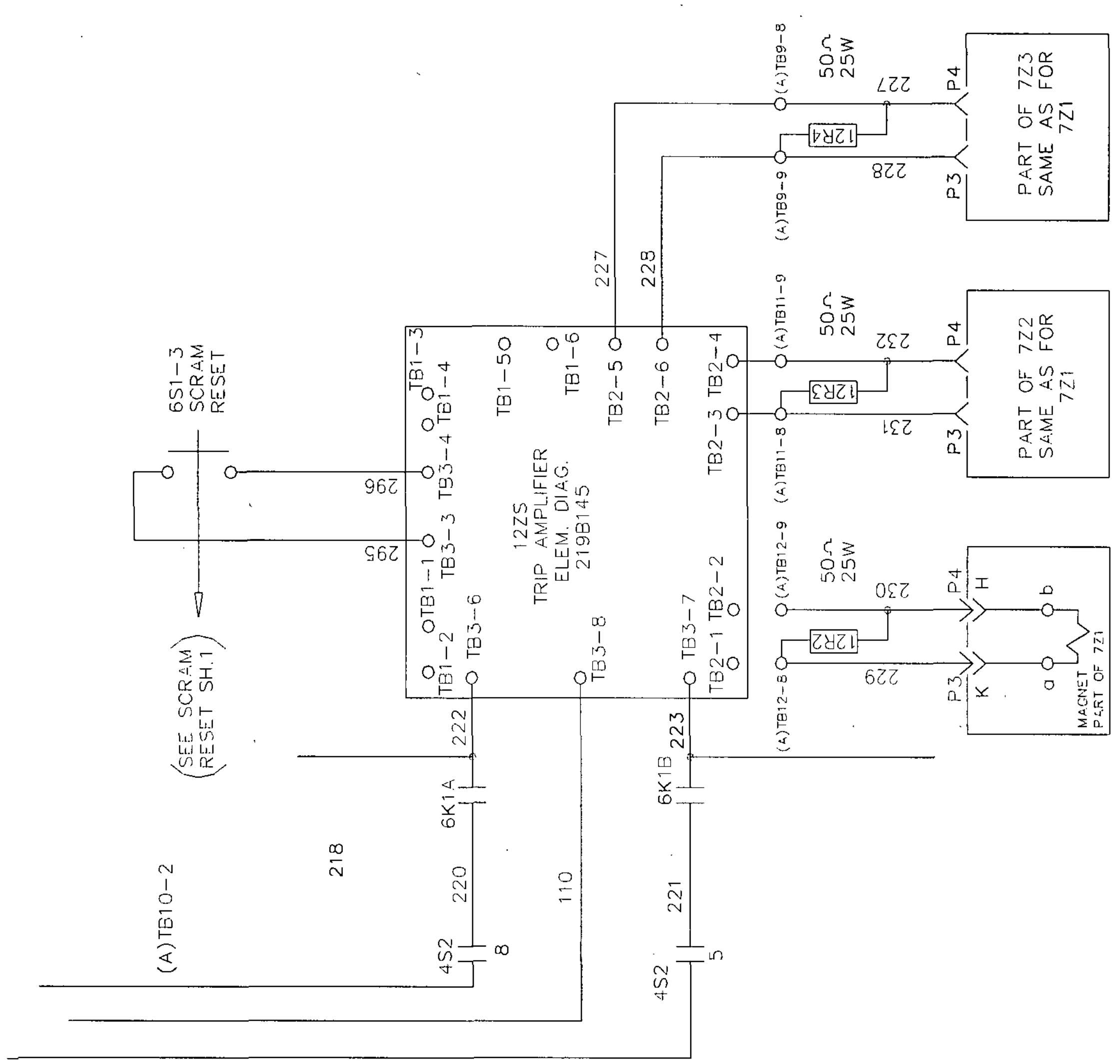

Figure 7-3 Magnet Power Supply

7.5 Engineered Safety Features Actuation Systems

There are no engineered safety features actuation systems.

7.6 Control Console and Display Instruments

7.6.1 Alarm and Indicator System

When an abnormal condition develops, an audible signal sounds and a lighted annunciator begins to flash rapidly. The operator may press the acknowledge button to silence the audible signal, at which time the audible signal stops and the lighted annunciator goes to steady illumination. When the condition is corrected, the lighted annunciator goes to slow flash, and the light is extinguished when the operator presses the Reset button.

The following conditions will actuate the alarm system:

1. High area radiation level (also gives an alarm at UW Police Department and initiates building evacuation);
2. High experimental facility radiation level;
3. Radiation monitor failed low;
4. Evacuation Alarm in Local;
5. Stack Air particulate or gaseous activity above normal level;
6. CAM Air particulate or gaseous activity above normal level;
7. Trouble in stack or continuous air monitors;
8. Neutron flux exceeding 1.15 times the normal value;
9. Reactor period less than a preset level;
10. Count rate on startup counter approaching saturation level;
11. Any scram;
12. Safety blade disengaged from magnet;
13. Failure of high voltage power supply;
14. Loss of Off-Site Power;

15. Fuel element temperature high;

16. Core inlet temperature above preset level;

17. Cooling system temperatures above preset level;

18. Intermediate coolant system low pressure.

19. Thermal Column door open;

20. Chain switch across stair actuated or entry to High Radiation Area;

21. Hold tank full;

22. Water level in pool two or more inches above or below normal (also gives an alarm at UW Police Department);

7.6.2 Indicator Lights

To provide operating information for the reactor operator, the following indicator lights are provided:

1. Scram reset;

2. Safety blade magnet engaged;

3. Power on;

4. Control elements in (distinct light for each);

5. Control elements out (distinct light for each);

6. Automatic control on;

7.6.3 Pneumatic Tube System Panel

The pneumatic system control panel provides indication lights for system on, system purge in progress, isolation valves open, and rabbit in reactor. Buttons allow the console operator to start and stop the system, and emergency return a rabbit sample. The pneumatic system is described in section 10.2.3.

7.6.4 Ventilation System Panel

The ventilation system control panel provides indication for both of the main exhaust fans (EF-7 and EF-8), the air handling unit (AHU-5), the fume hood exhaust fan (EF-13), and the Beam Port & Thermal Column exhaust fan (EF-17). Each fan has separate lights indicating whether the fan is running and whether the fan has power available to run if needed. All fans except the fume hood exhaust fan (EF-13) include switches for operation. Digital indications are provided for stack exhaust flow-rate (scfm), differential pressure (inches of water column) across the main exhaust filter bank, and static duct pressure (inches of water column) in the Beam Port & Thermal Column exhaust duct before the exhaust fan EF-17. The ventilation system is described in section 9.1.

7.6.5 Cooling System Panel

Operating controls and indicators at the control console for the cooling system include switches to operate primary, intermediate, chilled water, and diffuser pumps (the hydraulic irradiation facility pump starts when the diffuser pump starts) along with indicators which light when the pumps have discharge pressure. In addition, a pressure switch on the intermediate coolant system provides indication that the system is pressurized. The cooling system is described in chapter 5.

7.6.6 Whale System Panel

The hydraulic irradiation facility (whale system) includes indicator lights for flow direction, sample in, and buttons to reverse flow direction, as well as indication that the whale pump has discharge pressure. The whale system is described in section 10.2.4.

7.7 Radiation Monitoring Systems

7.7.1 Area Radiation Monitors

The radiation monitors are arranged into three systems; the primary area monitors, experimental facility area monitors, and air activity monitors.

The primary area monitors are located as follows:

1. Demineralizer area;
2. On the reactor bridge about one foot above the water surface;
3. Beside the thermal column door;
4. In the control console area.

All Area Radiation monitor units have ranges from 0.1 to 10000 mr/hr.

Unit 1 supplies information on radiation level from the demineralizer. It is set to alarm at a radiation level just above that expected in a normal run. Unit 2, located just above the pool water level, alarms at a radiation level just above that reached during full power operation. Unit 3 is located beside the thermal column. It too is set to alarm just above normal operating level. This unit will give an alarm if the thermal column door is left open when the reactor is operated at any substantial power. Unit 4 indicates the dose rate in the console area. The 4 units indicated above are connected to the Reactor Laboratory evacuation alarm. An alarm from one of these units will sound the evacuation alarm if it is not corrected by the operator within 30 seconds (See procedure UWNR 150).

The Experimental Facility Area Radiation Monitor is an area radiation monitor system installed to preclude the possibility of unknowingly generating high radiation levels by operating the reactor at high power levels with the beam ports open, or by return of an intensely radioactive pneumatic tube sample. The sensors for this system are installed on the walls of the Reactor Laboratory in direct line with the beam ports and at the pneumatic tube send-receive station. The system gives visual and audible alarms at the console if the radiation level exceeds a preset value. The pneumatic tube monitor also provides local alarm and indication. The monitors are normally set to alarm at a radiation level equivalent to a dose rate of 50-100 mrem/hr at the beam port flange (10 mrem/hr at the detector location). The pneumatic tube monitor also is normally set to 10 mrem/hr, but the pneumatic tube operating procedure states that it may be set to a higher level if calculated sample activity is expected to result in a higher reading.

7.7.2 Stack Air Monitor

The stack air monitor measures both particulate and gaseous activity of the air discharged from the stack. Particulate activity is collected on filter paper and counted with a thin end-window sealed gas proportional tube. Gaseous activity is also measured with a gas proportional tube. The system operates by detecting β activity. Both particulate and gaseous activity levels are recorded, and provide annunciation should preset levels be exceeded. In addition, gaseous activity levels are integrated to provide a record of total gaseous activity discharged from the stack.

The sensitivity of the particulate activity monitor allows detection of concentrations of about 1.0E-10 μC/ml of a material with a single β particle emitted per disintegration. The efficiency is higher if more than one β particle is emitted per disintegration. The sensitivity of the gaseous activity monitor is such that a concentration of about 1.0E-6 μCi/ml of Ar^{41} at the stack discharge can be detected by the instrument. The efficiency varies with the number of β particles emitted by the isotope being detected. The primary activity expected to be present in the stack discharge is Ar^{41} activity.

An identical instrument, provided as a backup for the Stack Air Monitor, is operated as a Continuous Air Monitor. It samples the atmosphere immediately above the surface of the reactor pool, although it can be made to sample other locations when desired. The backup air activity monitor can be connected to the stack monitor flow path, should the stack monitor fail.

7.8 References

There are no references for this chapter.

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8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

There are no electrical power supplies that are critical for maintaining the facility in safe shutdown, even for extended periods of time.

The Reactor Laboratory's electrical power originates in room B1101 of the Mechanical Engineering Building. The entire system is shown in **Figure 8-1**. Utility power to the building is supplied by circuits 1421 and 1422. Below is a description of each electrical circuit used in the Reactor Laboratory.

8.1.1 277/480 VAC 3 Phase Electrical Power

277/480 VAC is provided to the Reactor Laboratory by a 3000 kVA transformer and through a ground fault interrupted 3000 A circuit breaker. The Laboratory itself has its own 225 A ground fault interrupter and circuit breaker (#8). This circuit connects to the Reactor Laboratory's 277/480 panel (4RI) which is located behind the Reactor Console just outside of the Reactor Control Room.

This panel supplies 480 VAC to the Reactor Laboratory's N-16 Diffuser and Whale Pumps. Although both of these pumps are on the same circuit, each has its own motor starter. The Diffuser Pump starter is located immediately behind the Reactor Console in the Reactor Control Room and the Whale Pump starter is located to the right of the Diffuser Pump starter. The Diffuser Pump also has a local disconnect which is located next to the Diffuser Pump on the Reactor's Shield Step. The Diffuser and Whale Pumps can be turned on and off using a single shared toggle switch located on a graphic mimic panel in the Reactor Control Console.

The 277/480 panel also provides power to the 3 loops of the Reactor Laboratory's Cooling System. Each loop (Primary, Intermediate and Chilled) has its own local disconnect switch located just below the Reactor's Control Room and each can be turned on and off using individual toggle switches located on a graphic mimic panel in the Reactor Control Console.

Finally, the 277/480 panel also supplies power to the Reactor Laboratory's overhead crane and to the primary windings of the Reactor Laboratory's two power transformers, T1 and T2. These transformers are described in detail below.

8.1.2 277/480 VAC 3 Phase Backed Up Electrical Power

The Reactor Laboratory's 277/480 VAC backed up electrical power is supplied by the same 277/480 VAC 3000 kVA transformer and through the same 3000 A circuit breaker as the Laboratory's 277/480 VAC 3 phase normal electrical power. This bus has its own 1600 A breaker and passes through an Automatic Transfer Switch (ATS). This ATS automatically switches the bus's source from utility to the Mechanical Engineering emergency diesel generator. After the ATS, this circuit passes though another 1600 A breaker and a 100 A breaker to a circuit within the Mechanical Engineering Building Panel 4CR1. This panel is located in room 1133 Mechanical Engineering. A circuit in 4CR1 powers the Reactor Laboratory's backup/makeup air compressor. When the Reactor Laboratory's compressed air system, provided by the Mechanical Engineering Building, falls below 75 psi, this local air compressor activates.

The Reactor Laboratory's ventilation system is also powered by this bus. Bus EMCC6 is connected to the Reactor Laboratory's 277/480 VAC backed up electrical power through two 400 A fuses. Bus EMCC6 powers the Reactor Laboratory's two ventilation fans, EF-7 and EF-8, as well as the Reactor Laboratory's air handling unit, AHU-5.

Finally, the Reactor Laboratory's Support Space basement emergency lights are also powered by the 277/480 VAC backed up electrical bus through panel 4CR2.

8.1.3 240 4 wire VAC Electrical Power, Transformer T1

240 4 wire VAC is provided by transformer T1. This transformer's primary winding is connected to 480 VAC from panel 4RI and its secondary winding feeds panel R. Both panel R and T1 are located behind the Reactor Console just outside of the Reactor Control Room.

Panel R supplies power to the Thermal Column Door motor which allows the Thermal Column to be opened and closed by pressing a pair of buttons located on the Thermal Column Door itself. In addition, Panel R supplies 240 VAC power to several NEMA "twist-lock" type outlets located on the Reactor Shield at each beam port.

8.1.4 208/120 3 wire VAC Electrical Power, Transformer T2

208/120 VAC is provided by transformer T2. This transformer's primary winding is connected to 480 VAC from Panel 4RI and its secondary winding feeds Panel 2. Both Panel 2 and T2 are located behind the Reactor Console just outside of the Reactor Control Room.

Panel 2 supplies power to a pressure makeup pump in the Intermediate Loop of the Reactor's Cooling System. It also supplies power to the Demineralizer Pump in the Reactor's Primary Coolant Makeup System through an outlet located on the west wall by the Demineralizer. In addition, Panel 2 powers the circulating pump of the Waste Holdup System.

Various 120 VAC outlets throughout the lab as well as a set of 120 VAC and 208 VAC outlets at each beam port are also powered by Panel 2.

Finally, though the use of an uninterruptible power supply (UPS), the Reactor's Control Console is powered through Panel 2. See section 8.2 for details of this UPS. All power to the Control Console is 120 VAC.

8.1.5 277 VAC Lighting Electrical Power

Most Reactor Laboratory lighting is supplied by the same 277/480 VAC 3000 kVA transformer and through the same 3000 A circuit breaker as the Laboratory's 277/480 3 phase electrical power. The lighting circuit has its own 200 A breaker and ground fault interrupter circuit. This circuit is connected through a cutoff switch (located in room B1140 Mechanical Engineering) to Panel B1135 (located in room B1135 Mechanical Engineering). Panel B1135 has various breakered circuits which then power the various lights within the Reactor Laboratory and the Reactor Laboratory's support space.

8.1.6 277 VAC Backed Up Lighting Electrical Power

A few selected lights within the Reactor Laboratory and the support space are powered by a separate 277 VAC circuit. These lights are always on and are connected through an Automatic Transfer Switch (ATS) which transfers the circuit from utility power to an emergency generator in the event of a power failure.

8.2 Emergency Electrical Power Systems

Neither of the systems described in this section are required or necessary for safe facility operation or shutdown.

There are several Emergency Electrical Power Systems, shown in **Figure 8-2**, which the Reactor Laboratory utilizes. Those which are a part of the Mechanical Engineering building are powered by a diesel 800 kW 277/480 generator located in room B1002A Mechanical Engineering.

The other Emergency Electrical Power Systems are two battery based Uninterruptible Power Supplies located within the Laboratory in rooms 1215 and B1135E Mechanical Engineering.

8.2.1 Reactor Laboratory Uninterruptible Power Supply

A 6000 VA Uninterruptible Power Supply (UPS) is connected between utility power and the reactor's control console. This system is installed solely to protect the reactors instrumentation from sudden power losses as well as dirty power. This UPS provides protection from surges, brownouts, noise, spikes, frequency variations, transients and harmonic distortion. In the event of a loss of utility, the UPS is capable of providing up to 76 minutes of power to the reactor's control console and instrumentation. This allows for a monitored shutdown and facilitates uninterrupted recording and monitoring. A failsafe SCRAM and alarm connected to a non backed up power circuit ensures that control blades drop and the operator is informed in the event of a loss of utility.

Even with a UPS failure the reactor is designed to SCRAM in the event of a loss of power. Therefore, this UPS is not required for a failsafe shutdown when power is lost and in the event of a UPS failure and loss of utility, the reactor will shutdown. This UPS is not required for maintaining the facility in a safe shutdown condition, even for extended periods of time.

8.2.2 Reactor Laboratory Support Space Uninterruptible Power Supply

A 6000 VA Uninterruptible Power Supply (UPS) identical to the Reactor Laboratory's is located in the Laboratory's support space. This UPS serves as both an installed spare for the Reactor Laboratory UPS and as a support space power backup. In the support space, this UPS powers several outlets, allowing the support space's counting computers, sample changing equipment and, most importantly, the Stack Air Monitor (SAM) and Continuous Air Monitor (CAM) to continue operating or be shut down gracefully in the event of a utility power failure.

8.2.3 Mechanical Engineering Building Emergency Power

As mentioned in section 8.1 above, the Mechanical Engineering building has an emergency backup generator and several emergency power systems. These systems include the backed up emergency lights panel, which powers selected lights throughout the Reactor Laboratory and the Mechanical Engineering Building, the backed up equipment panel, which powers the Reactor Laboratory's air compressor, smoke detector/fire alarm and ventilation. As mentioned in Section 8.1, these systems are connected through automatic transfer switches which switch them to generator power in the event of a loss of utility power.

In addition, the Mechanical Engineering Buildings fire suppression system pump also has its own automatic transfer switch and emergency power. This system includes the fire suppression system in the Reactor Laboratory

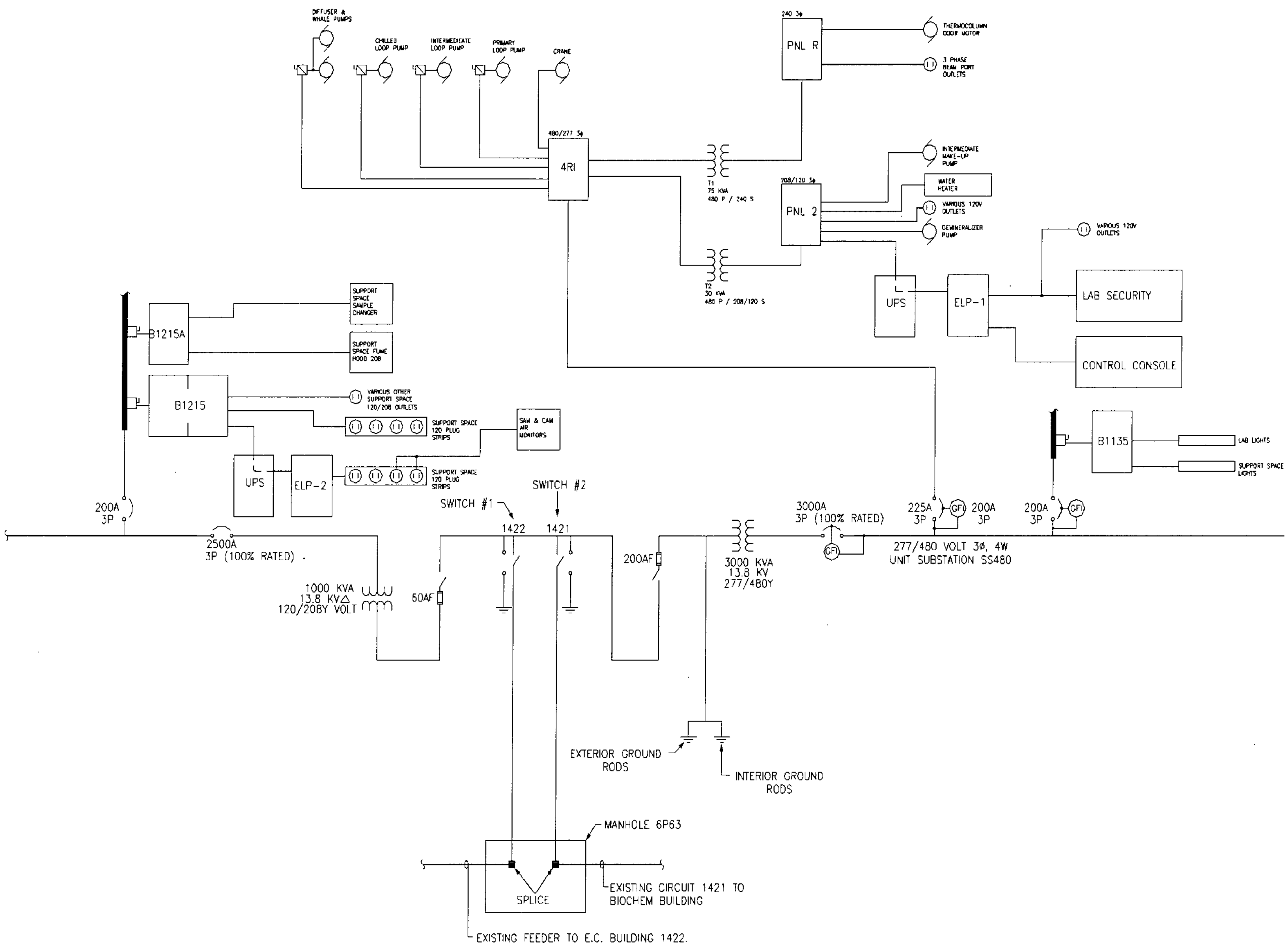


Figure 8-1 Normal Electrical System

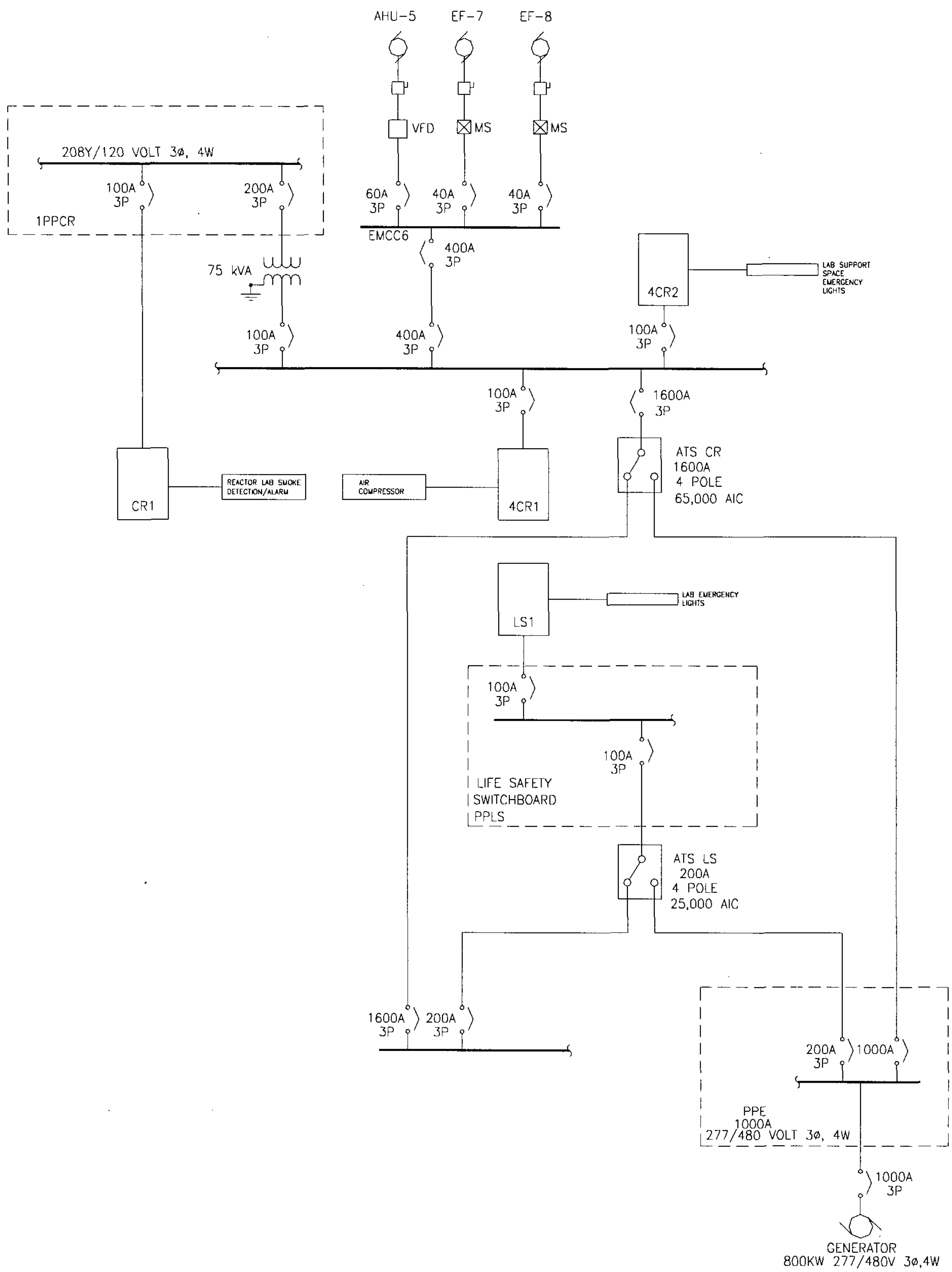


Figure 8-2 Backup Electrical System

9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air Conditioning Systems

The heating, ventilation, and air condition system (hereafter called the ventilation system), is a dedicated reactor laboratory system with no interaction with the rest of the building. The system incorporates a fresh air supply air handling unit, two exhaust fans, filter banks, variable supply and exhaust air mixing boxes, and duct work. The system provides HVAC service to both the reactor laboratory and auxiliary support spaces. **Figure 9-1** shows the ventilation system flow schematic, while **Figure 9-2** and **Figure 9-3** show the physical location of ventilation system components.

The ventilation system is designed to prevent the spread of airborne particulate radioactive material into occupied areas outside the Reactor Laboratory. It removes particulates with high-efficiency filtration and assures that all releases of either gaseous and particulate activity are monitored and discharged at an elevated release point. Accidents which might result in discharge of radioactive material from the stack are discussed elsewhere in this report, and remarks may be found there indicating the concentrations which might be expected. In addition, a portion of the ventilation system vents the beam ports, thermal column, and liquid waste holdup tank to assure that air flow is from the Reactor Laboratory into these facilities.

9.1.1 Air Handling Unit

The ventilation system consists of a dedicated air handling unit (AHU-5) mounted in the fifth floor of the Mechanical Engineering building behind a security fence which only reactor staff can access. The unit supplies 9150 scfm 70°F fresh air at a relative humidity of 30% to the reactor laboratory and auxiliary support space through 7 variable air volume boxes; 5 in the auxiliary support space, 1 in the reactor laboratory confinement, and 1 in the reactor control room. The unit is powered from a variable frequency drive which maintains duct static pressure at 1.0 inches of water gauge. The unit is interlocked with the exhaust fans to trip if neither exhaust fan is running.

9.1.2 Exhaust Fans

The ventilation system incorporates two roof mounted exhaust fans (EF-7 and EF-8), each individually capable of exhausting 9600 scfm air. Normally only one fan is running, but in emergency venting mode both fans may be run for increased exhaust dilution with 19200 scfm air. The reactor laboratory exhaust duct pressure is maintained at -1.5 inches of water gauge to maintain air flow from surrounding areas into the reactor laboratory. If one exhaust fan fails, the other fan attempts to pickup. If both exhaust fans fail, an interlock will trip the air handling unit to prevent having a positive pressure in the reactor laboratory.

Each fan has isolation dampers that close when the fan is not running to prevent short cycling the exhaust flow path. Both fans take a suction on a common header (located below the roof on the fifth floor Mechanical Engineering) to which all exhaust variable air volume boxes discharge to through the filter bank. This provides diluting air from the auxiliary support space prior to discharging reactor confinement air. In addition, the common header is equipped with duct work which draws a suction on outside air through the reactor attic for further dilution. This outside air is not filtered. This duct work includes an automatically controlled damper which is adjusted to maintain the exhaust duct pressure at -1.5 inches of water gauge. This is necessary because the exhaust fans do not have variable frequency drives and so must always run at full capacity. An additional manually controlled damper is available for bringing in additional outside air into the common header, but this damper is normally locked closed. The air from this common header is continuously monitored by the stack air monitor to maintain a record of radioactivity discharged. Flow rate indication is also displayed in the control room.

The exhaust fans are at a height of 26.5 meters above grade. These fans, manufactured by Strobic, incorporate a unique nozzle design which operates on a principle of internal and external exhaust stream dilution. The nozzle entrains outside air with the primary exhaust stream to produce a substantially diluted exhaust stream. This enhanced flow stream then undergoes a pressure increase to increase stack outlet velocities which increases the effective stack height. However, the analysis in Appendix A neglects the increase in stack height for a bounding analysis.

9.1.3 Filters

All ventilation exhaust is filtered just prior to entering the common header (before any outside air dilution). This main filter bank (F-21) consists of a 4x4 array consisting of 16 pleated pre-filters followed by 16 nuclear grade HEPA filters which are rated at 99.97% efficiency for particles of 0.1 micron and larger. The pneumatic system fume hood also has its own basement filter bank (F-22), even though the fume hood exhaust is sent to the main ventilation exhaust system. This filter bank has 1 pre-filter followed by 1 HEPA filter located upstream of the fume hood exhaust booster fan (EF-13). The main filter bank (F-21) is located with the common header in the fifth floor Mechanical Engineering, [redacted]. The fume hood exhaust basement filter bank (F-22) is located in the auxiliary support space in room B1135A. The pressure drop for the main filter bank is indicated in the control room. Local indications for pressure drop across the pre-filters and across the HEPA filters are on the fifth floor Mechanical Engineering and the basement.

9.1.4 Exhaust Variable Air Volume Boxes

As shown in Figure 9-1, the exhaust system incorporates 10 variable air volume boxes. Four boxes are located in the auxiliary support space, three are located in the reactor laboratory for normal venting, and three are located in the reactor laboratory for emergency venting. Of the four boxes in the auxiliary support space, one is designated for the pneumatic system fume hood.

Each box is adjustable from 0% to 100% design flow. The normal confinement boxes have a combined rating of 2700scfm. With this flow rate, assuming a confinement volume of 2000m^3, it would take 26.16 minutes (1569s) to completely exhaust confinement.

9.1.5 Emergency Venting Mode

Emergency venting mode is for use when it is desirable to rapidly change the air in the reactor laboratory to prevent spread of contamination to adjacent occupied areas. Use of emergency venting mode is governed by the Emergency Plan, which states that the decision to operate in emergency venting mode should be reached by common consent of the Emergency Coordinator and the campus Health Physics organization. Emergency venting is initiated by one of two large red push-buttons (with switch covers) located in the control room by the south door and in the first floor vestibule, room 1200J, by the east catwalk door. When either of these buttons are pressed, the second exhaust fan activates to double the flow-rate. The three variable air volume boxes designated for emergency venting mode are normally closed until emergency venting mode is activated by either of the switches. Because the negative pressure in reactor confinement would be extreme with both fans running, a makeup exhaust damper in the confinement roof opens in emergency venting mode to allow outside air to enter confinement through the reactor attic. This makeup exhaust damper is normally closed with a weather-proof seal.

9.1.6 Beam Port and Thermal Column Ventilation System

The Beam Port and Thermal Column Ventilation system is designed to sweep out the Ar-41 activity present in an experimental facility when the facility is opened. During ordinary operation the experimental facilities are closed and there is an essentially zero rate of discharge. When a beam port flange or the thermal column door is opened there is a slug of activity discharged. The average concentration discharged will, therefore, be extremely low due to dilution by the rest of the ventilation system and the fact that no activity is discharged most of the time. Section 11.1.1.1 of this report discusses the levels of activity discharged.

The Beam Port & Thermal Column ventilation system consists of a booster exhaust fan (EF-17) mounted in the Reactor Attic (room 3110), which discharges into the main ventilation system exhaust duct work (before the main filter bank F-21), and a makeup damper located in the Reactor Laboratory. The booster exhaust fan is needed to provide sufficient suction on the Beam Port & Thermal Column ventilation system. The makeup damper is adjusted to maintain approximately 960 cfm at 1.5 inches suction pressure and an air velocity of about 40 feet per minute into all beam ports and the thermal column, should all be opened simultaneously. Normal flow rate with the system sealed is about 450 cfm. The thermal column shielding door is weather-stripped to maintain a nearly airtight seal. An air-operated flapper valve at the end of the duct connected to the thermal column and beam port vents is normally open. This maintains a slight negative pressure within the thermal column when the door is closed, but prevents the full static suction of the system from forcing air in through the thermal column door seals. When the thermal column door is opened, however, the flapper valve closes and full system suction is impressed on the thermal column to cause the desired in-flow of air. In addition, a ball check valve within the thermal column vent prevents mixing of the air in the thermal column with the ventilation flow except when the door is not sealed. The facility ALARA program identified this feature as one which resulted in a significant reduction in activity discharged through the stack without increasing the Ar-41 dose within the laboratory. Also incorporated into the system for ALARA considerations are ball-check valves in the vent line from each beam port. Because the pressure drop in the BP&TC Ventilation System duct causes lower pressures in beam ports closer to the exhaust fan, the common drain connection for the four beam ports allows air flow through the drain connections from higher to lower pressure beam ports. The check valve prevents flow into the beam port vents, thus preventing circulation through the drain system and substantially reducing the amount of Ar-41 discharged to the atmosphere.

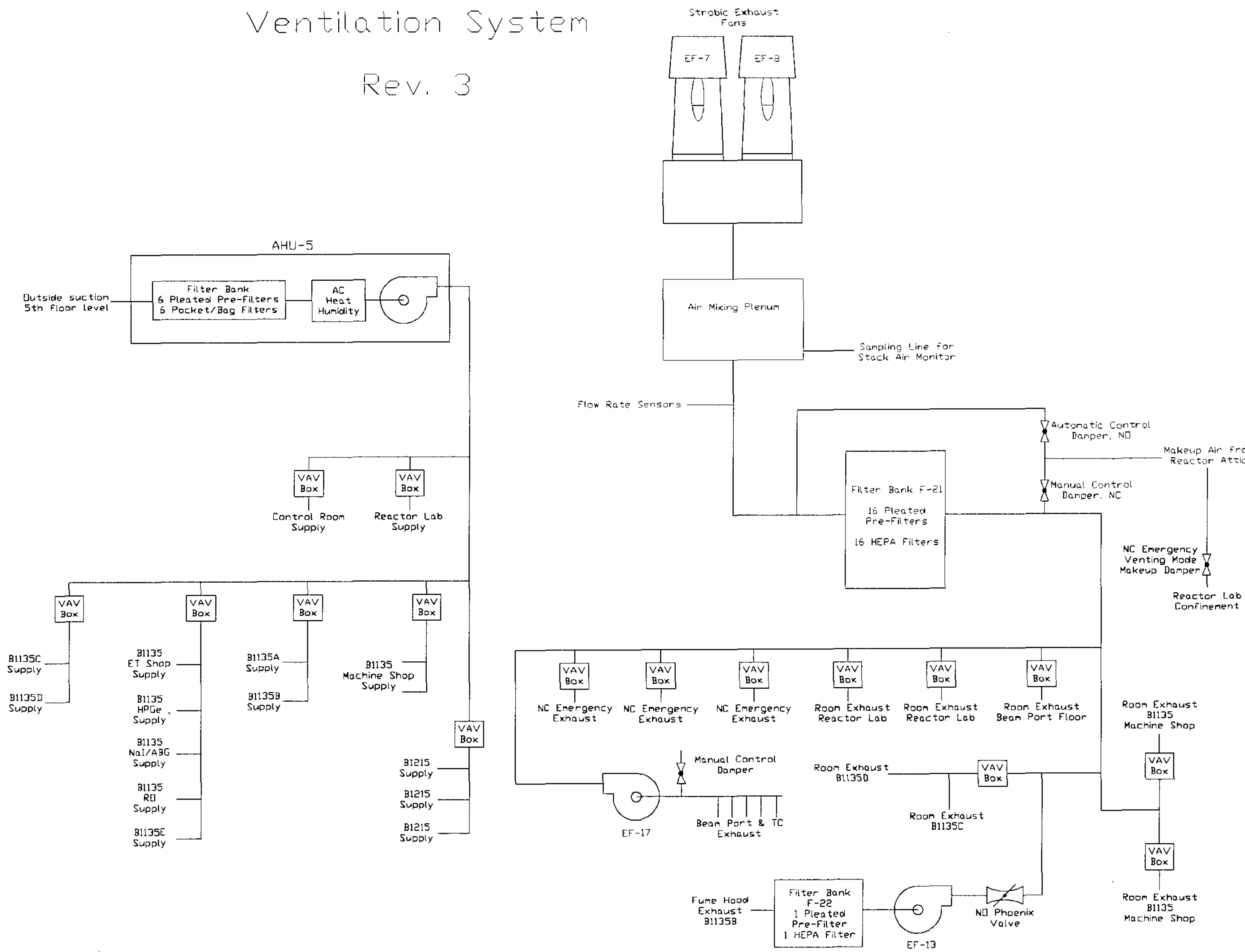
Should a beam port rupture and fill with water, water would leak out of the flapper valve at the end of the Beam Port & Thermal Column ventilation system on the shield step, which would still leave at least 11 ft of water covering the core. The water would eventually spill onto the reactor laboratory floor and into the holdup tank; no water would leak out of the ventilation system outside of the reactor laboratory because of the height of the filter bank and exhaust fans.

The vent of the waste holdup tank is also connected to the Beam Port and Thermal Column Ventilation system in order to assure any gaseous effluent from the holdup tank is discharged through a monitored release path. In addition, the stack air activity monitor discharges the sample stream extracted from the stack into this ventilation system.

9.1.7 Pneumatic System Fume Hood Exhaust

The pneumatic system fume hood includes its own exhaust booster fan (EF-13) and filter bank (F-22) to help ensure fume hood air velocity is sufficient to protect the operator in the event of a sample breakage. This booster fan exhausts into the common header of the main ventilation system. The fume hood exhaust fan is automatically activated upon starting the pneumatic system, or it may be manually run independent of the pneumatic system if desired. Operation of the booster fan has a negligible impact on the main ventilation system exhaust flow-rates due to the small size of the booster fan relative to the main exhaust fans. The control room ventilation system panel includes indication that the booster fan is running and that it has power available to run if needed, as well as the pressure drop across the fume hood filter bank (F-22).

Figure 9-1 Ventilation System Schematic



[redacted]

[redacted]

Figure 9-3 Ventilation System Cross Section Looking East

[redacted]

Figure 9-4 Fuel Storage Positions

9.2 Handling and Storage of Reactor Fuel

9.2.1 Fuel Handling

[redacted]

9.2.2 Fuel Storage

New (unirradiated) reactor fuel can be handled manually and is stored in a steel safe. Those needing further details on new fuel storage should consult the facility security plan.

Sufficient storage room is provided for all on-site irradiated fuel. The fuel storage locations are indicated in **Figure 9-4**. All fuel storage facilities are designed to allow sufficient convective water flow to remove decay heat.

[redacted]

9.2.3 Fuel Bundle Maintenance and Measurements

Fuel bundle disassembly, assembly, and fuel element bow and elongation measurements are conducted underwater using a special tool (shown in **Figure 9-5** with a dummy element installed) designed for both these purposes. This tool is used under about [redacted] The fuel-element handling tool also operates the socket and crowsfoot wrenches. For disassembly or assembly of fuel bundles, the maintenance and measuring tool holds the bundle, provides a reference plate for a crows-foot wrench, provides storage space for four individual fuel elements, and restrains the individual elements after they have been screwed out of the bottom end box. An air-operated clamping device reproducibly positions the bottom end box of the fuel bundle. The crows-foot wrench must be used for the initial loosening of each element (and for tightening each element to the specified torque upon reassembly). Once the elements are loose enough to turn freely the top end fitting is removed so a socket wrench can be used on a hexagonal portion of the top fitting to completely unscrew the individual element from the bottom end box. The fuel element handling-tool then may be used to remove individual elements and place them in the storage positions. While it is possible to disassemble and re-assemble the fuel bundles, it is tedious even with use of the written procedure and practice. Disassembly or assembly of an element takes about 30 minutes.

Required measurements of fuel element bow and elongation also are made with the maintenance and measurement tool. Because of the excessive time and added handling required to disassemble the bundle and measure each element in a separate measuring tool, the tool was designed to make the measurements without disassembly. **Figure 9-6** shows the three sensors employed (a portion of the housing is removed in this view.) Each uses a differential transformer as a transducer to give a remote electrical output proportional to displacement of the sensors.

The X and Y sensors employ spring-loaded aluminum wheels attached to the differential transformer cores. When the bundle is lowered into the tool the wheels are forced back and they then ride on the fuel element clad surface. These sensors are adjusted to give a zero signal for a standard fuel element dummy.

The length sensor differential transformer is actuated by one lobe of a cam. A second lobe of this cam is rotated into contact with the top edge of the fuel element cladding by a leaf spring attached to the operating rod. The cam pivots out into the measuring position only when the operating rod is fully withdrawn. The length sensor is also adjusted to zero output for the standard fuel element dummy.

A readout device is positioned on the pool curb or reactor bridge and connected to the underwater portion of the tool. Differential transformer core position is indicated by a meter for length measurements, and a recorder output is provided to the horizontal axis of an X-Y recorder for the bow measurements. Polarity is set so an increase in element length or a bow away from the center-line of the bundle gives a positive meter indication or recorder readout.

The dummy fuel bundle has one dummy element exactly 0.100 inches longer than the other three elements. This element also has a section in which the radius has been reduced by 0.060 inch, and a section in which the radius has been increased by 0.060 inch. By using this element the attenuation and zero controls on the readout box may be adjusted to give a calibrated readout of bow and length.

After calibration of the tool, measurement can be made on standard fuel elements. The standard setup used gives a 1 cm horizontal displacement for 0.060 inch transverse bend (bow) and a meter reading of length in thousandths of an inch deviation from the dummy element reference length. A trace is drawn for both the X and Y sensors while the length measurement meter reading is manually recorded on the form. A complete set of measurements for all four elements in a bundle can be completed in about twenty minutes.

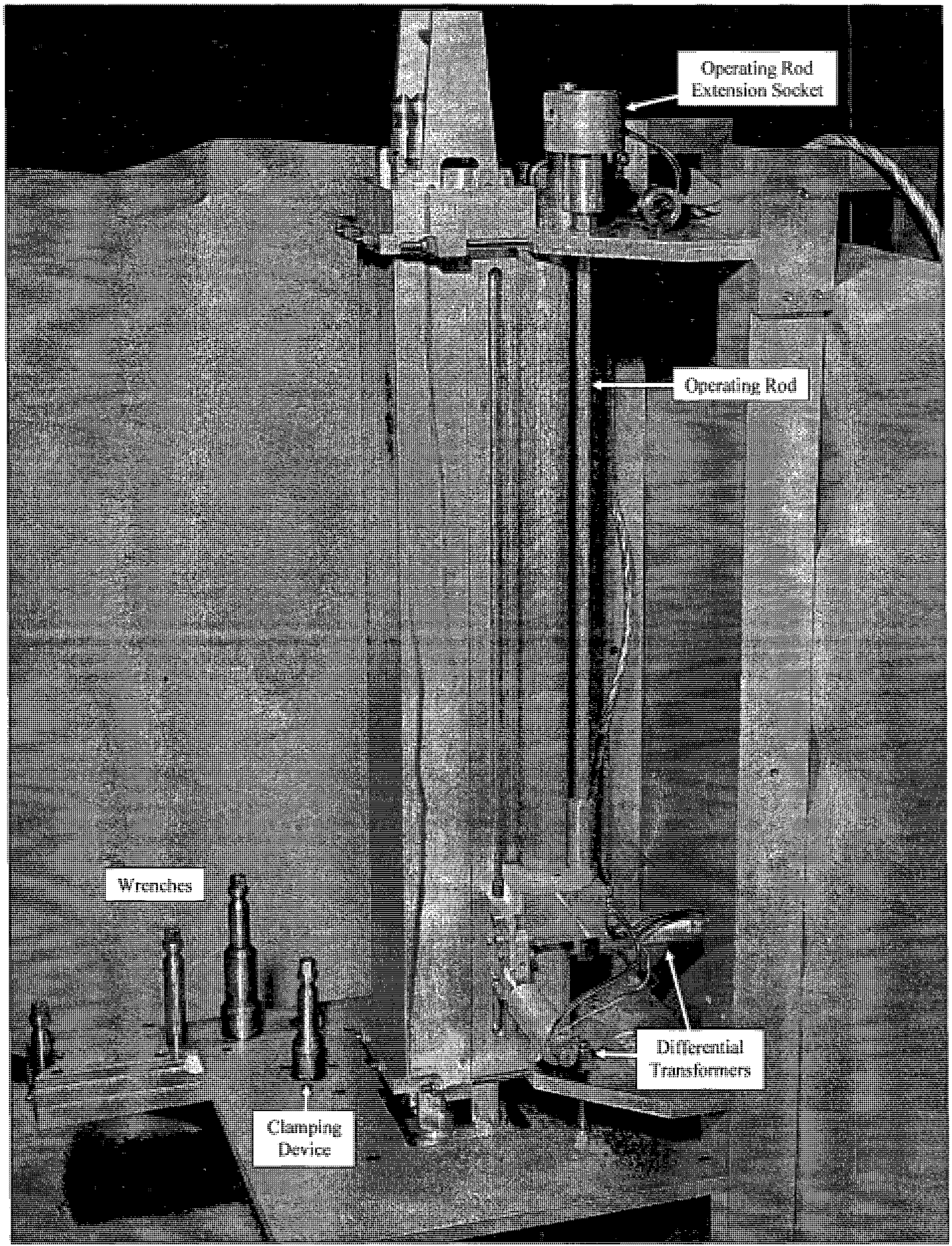


Figure 9-5 Fuel Maintenance and Measuring Tool

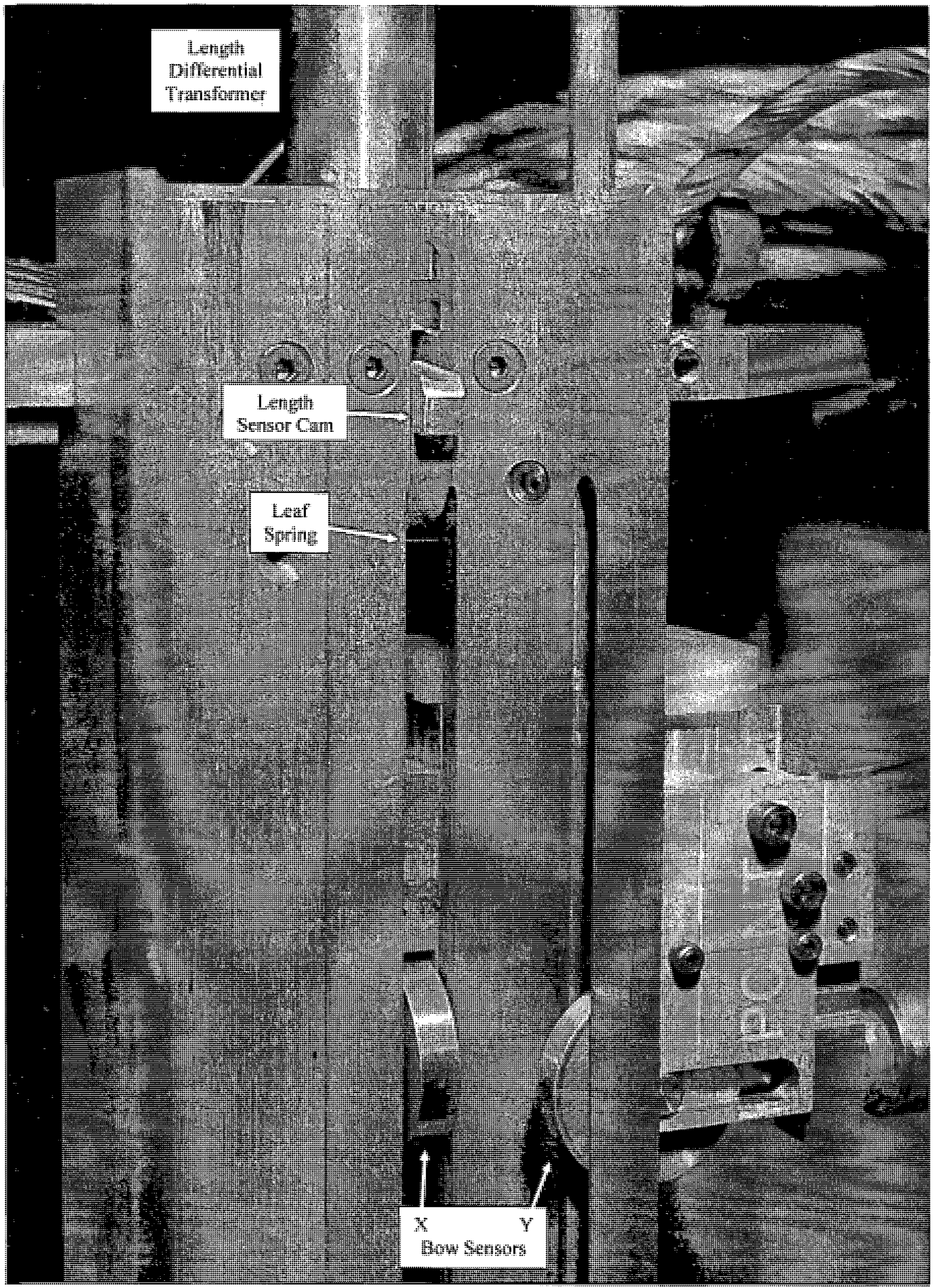


Figure 9-6 Bow and Elongation Sensors

Since the X and Y readouts are 90° apart, the maximum possible bow will be the square root of the sum of the squares of the bows indicated by direct measurement. As long as neither measured bow exceeds 0.088 inch, no calculations or other measurements are necessary. If either bow measurement exceeds 0.088 inch, then the square root of the sum of the squares of the measured bows must be calculated to determine whether or not this resultant is less than 1/8 inch. If the calculated number is less than 1/8 inch, the element is within technical specifications. Should the calculated bow exceed 1/8 inch, the crowsfoot wrench may be used to rotate the element being measured so that the reading of one bow sensor is maximized and the true bow may be determined directly to see whether it exceeds technical specification limits.

9.3 Fire Protection Systems and Programs

The fire detection and protection systems in the laboratory meet the local and state requirements. Interior doors from the Reactor Laboratory to the remaining parts of the building are also fire doors that meet local codes. All walls between the reactor laboratory and the remainder of the building are masonry. An early warning smoke detection system within the laboratory is connected to the building fire alarm system. The fire alarm system alarms locally [redacted].

The laboratory is equipped with a sprinkler system and portable fire extinguishers which are regularly inspected and serviced by the University Safety Department.

9.4 Communication Systems

The Reactor Laboratory and auxiliary support space and offices are equipped with commercial telephones. Two lines are available at the reactor control center. A cellular phone is also kept in the control room except when being used for maintaining contact with the control room.

An intercom system is installed in the Reactor Laboratory and auxiliary support space and offices. This system provides two-way communications between stations and an all-call capability for paging.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

All activities using radioactive and special nuclear materials covered under the reactor license take place within the Mechanical Engineering Building and a small portion of the Engineering Research Building in the rooms and areas indicated below:

- Reactor Laboratory (Room 1215);
- Auxiliary Support Space (Rooms B1135 and B1215, including sub-rooms);

[redacted]

[redacted]

Radioactive and special nuclear material use outside these areas is conducted under Wisconsin Department of Health and Family Services license 25-1323-01, the University of Wisconsin Radioactive Materials License.

9.6 Cover Gas Control in closed Primary Coolant Systems

There is no cover gas control in the primary coolant system.

9.7 Other Auxiliary Systems

There are no other auxiliary systems required for safe reactor operation.

9.8 References

There are no references.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

Facilities are provided to permit use of radiation from the reactor in experimental work without endangering personnel. Facilities provided with this reactor include four beam ports, a thermal column, and pneumatic and hydraulic irradiation transfer systems. All systems are designed to control radiation exposure to personnel using the facility as well as members of the public. Consideration is given to controlling the Ar-41 effluent from the experimental facilities in order to meet both the limits on releases to the environment and exposure of personnel within the laboratory.

10.2 Experimental Facilities

10.2.1 Thermal Column

The thermal column, **Figure 10-1**, is a graphite-filled, horizontal penetration through the biological shield which provides neutrons in the thermal energy range (about 0.025 eV) for irradiation experiments. The column, which is about 8 feet long, is filled with about 6 feet of graphite. A small experimental air chamber between the face of the graphite and the thermal column door has conduits for service connections (air, water, electricity) to the biological shield face. Detectors for the safety channels and the LogN channel are located within the thermal column. The location of the thermal column is indicated in **Figure 10-2**.

Personnel in the building are protected against gamma radiation from the column by a dense concrete door which closes the column at the biological shield. The door moves on tracks set into the concrete floor perpendicular to the shield face.

A ventilation system maintains a low pressure within the thermal column so that air flow is into the column when the door is open. The door is gasketed so that air flow is very small when the door is closed. When the door is opened, however, an air velocity of about 40 feet per minute into the column prevents the Ar-41 activity from diffusing into the Reactor Laboratory. Section 9.1 contains further information on the ventilation system for the thermal column and beam ports.

An annunciator is activated whenever the thermal column door is not fully closed. In addition, an area radiation monitor beside the thermal column door will give an alarm should the reactor be operated at a substantial power with the door open.

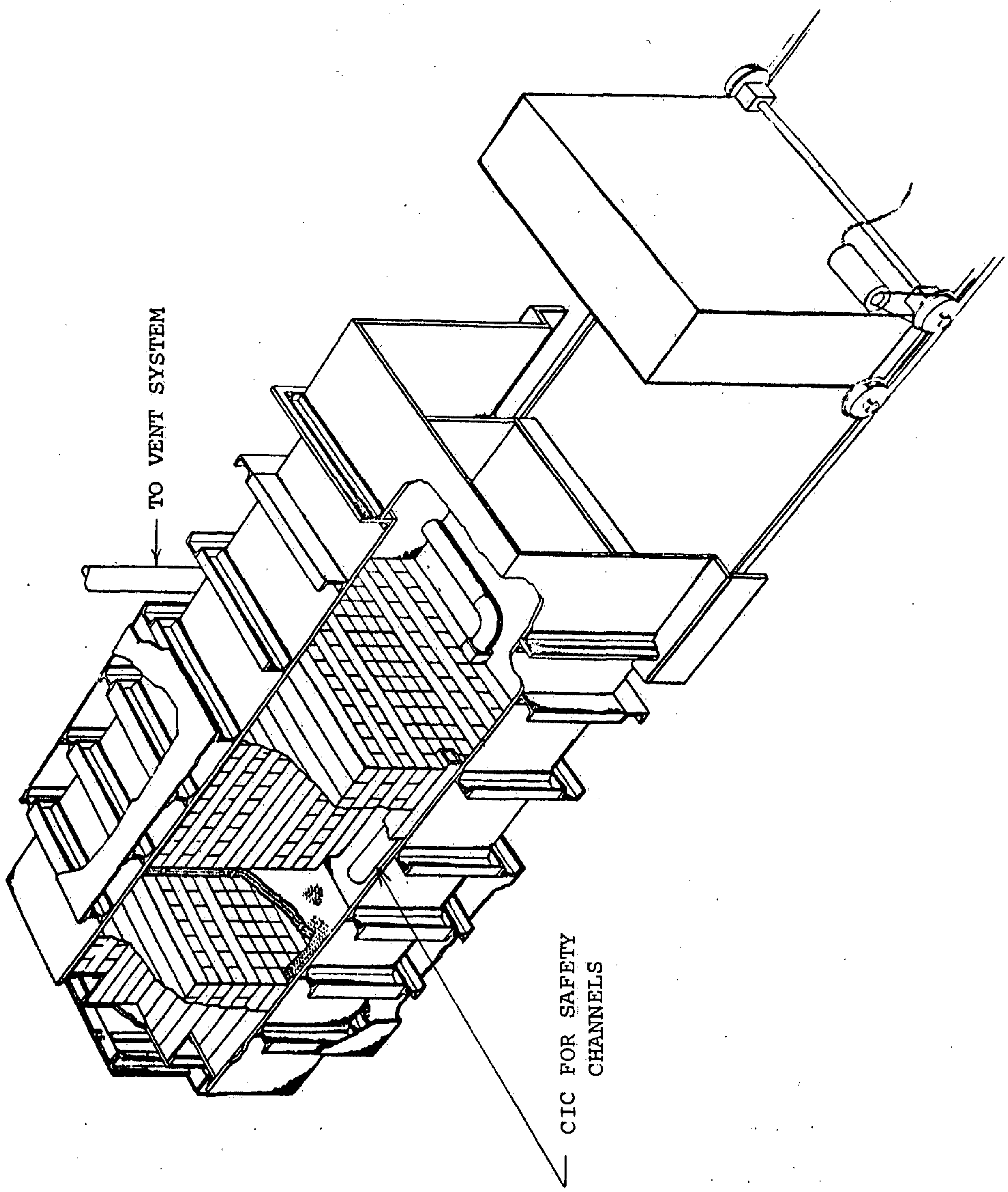


Figure 10-1 Thermal Column

10.2.2 Beam Ports

Four 6-inch beam ports penetrate the shield and provide fluxes of both fast and thermal neutrons for experimental use. **Figure 10-2** indicates the positions of the beam ports with respect to the grid box and shield while **Figure 10-3** shows the construction of the beam ports .

The ports are air-filled tubes, welded shut at the core ends and provided with water-tight covers on the outer ends. The portions of the ports within the pool are made of aluminum, while the portions within the shield are steel.

A shutter assembly, made of lead encased in aluminum, is opened for irradiations by a cable lifting device that extends to the pool curb. When closed, the shutter shields against gamma rays from the shut-down core, allowing experiments to be loaded and unloaded without excessive radiation exposure to personnel. A drain line is attached to the bottom of the shutter housing, while a vent line attaches to the top of the shutter housing. All beam port drains combine before exiting the concrete shield, where a stop valve is provided.

[redacted]

Figure 10-2 Location of Beam Ports & Thermal Column

When beams of radiation are not being extracted, shielding plugs are installed in the outer end of each port, filling almost all of the volume within. These plugs, made of dense concrete in aluminum casings, have spiral conduits for passage of instrument leads. These plugs completely stop the beam of radiation and minimize the production of Ar-41 in the beam ports.

Sealed aluminum cans are installed in the in-pool portion of the beam ports unless the particular beam port experiment requires installation of a collimator or filter in that location. These cans contain the Ar-41 produced and further minimize the release of Ar-41 activity. Additional control of Ar-41 activity released is accomplished by bellows seals on the lifting cables and maintaining the valve on the common drain header closed.

Since extremely high radiation levels could exist should the reactor be operated at substantial power levels with the shielding plugs removed, a beam port monitoring system is provided. The system consists of radiation detectors mounted on the walls in line with each beam port and a read-out device at the console which gives an audible and visual alarm should a preset radiation level be exceeded. The system is set to alarm at a radiation level equivalent to a dose rate of about 60 mrem/hour at the beam port openings.

The beam port & thermal column ventilation system (Section 9.1) exhausts the beam ports through the vent pipes shown in **Figure 10-3**. Vent pipes are connected to the ventilation system through a check valve which prevents back-flow into the vent and an isolation valve which may be closed should the beam port fill with water. With the beam port open, a linear flow velocity of about 40 feet per minute is maintained into the port opening, preventing diffusion of the airborne activity into the laboratory. With the beam port closed Ar-41 is almost entirely contained within the beam port.

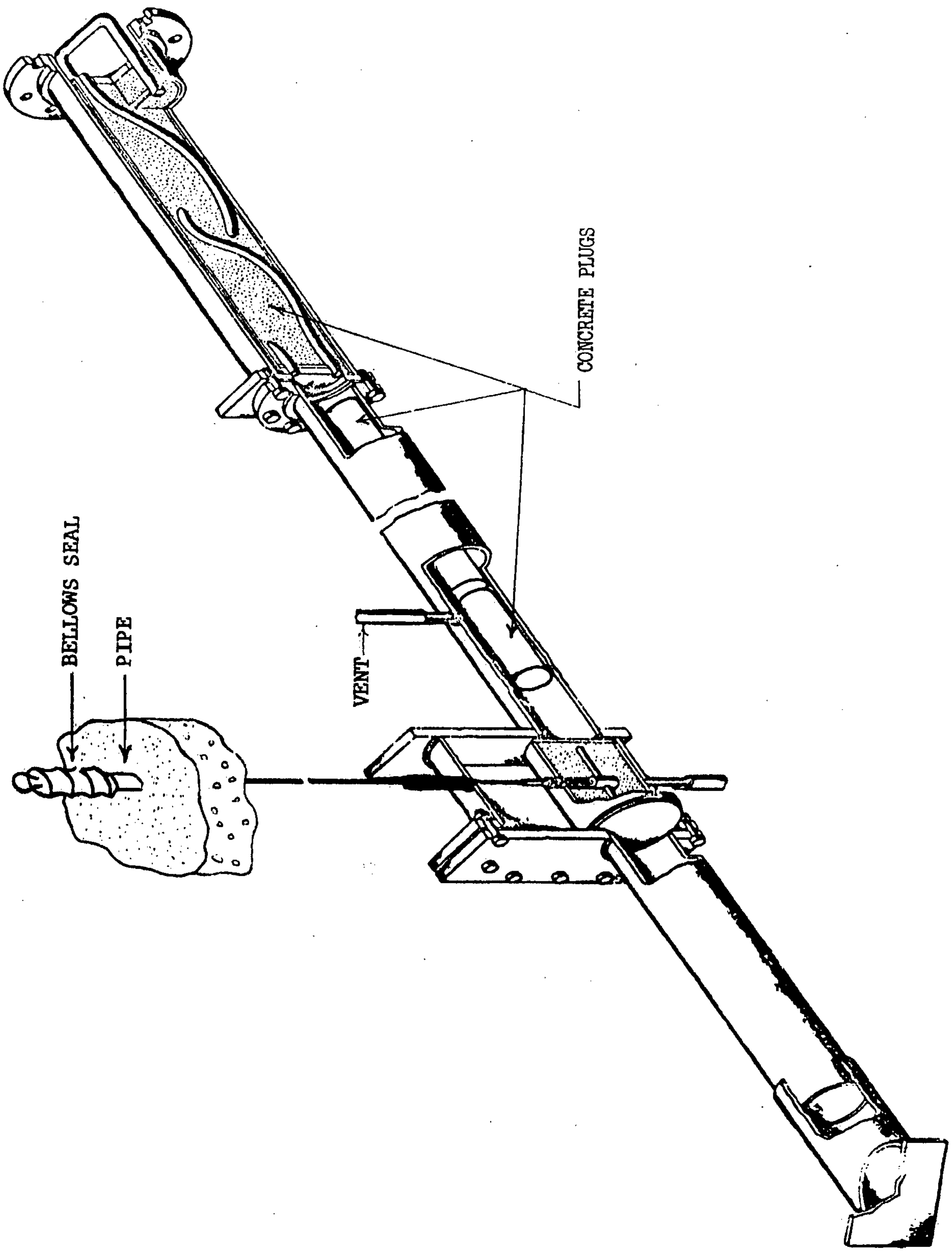


Figure 10-3 Beam Port

10.2.3 Pneumatic Tube

A pneumatic tube is used to irradiate samples for a short time and when the sample must be processed immediately after irradiation, as in neutron activation analysis of short-lived radioisotopes. The currently installed pneumatic tube system conveys samples from the basement auxiliary support space to an irradiation position beside the core (**Figure 10-4**). The "rabbits" used in the system will convey samples up to 1-1/4 inches diameter and 5-1/2 inches long, although the gross weight of a sample is kept below 12 ounces. Although the polyethylene rabbits used in the system can withstand longer irradiations, this facility is usually used for shorter irradiations of small objects. The system operates as a closed loop with CO_2 cover gas controlling generation and discharge of Ar-41 activity. The system is purged with CO_2 upon startup to remove any air which may have leaked in. An in-line CO_2 detector is used to monitor concentrations during the purge. Two isolation ball-valves are installed just outside of the biological shield which are closed unless the system is running. These valves serve as a barrier against air leaking into the reactor portion of piping when not in use, as well as preventing loss of pool water if the internal piping should rupture.

The reactor control room pneumatic system panel includes pneumatic system "System Start" and "System Stop" buttons, an "Emergency Return" button (which allows the reactor operator to return rabbits to the fume hood if desired), and indicator lights for "System On," "System Purge In Progress" (for CO_2 purge), "Reactor Isolation Valves Open," and "Rabbit In Reactor." All indications and controls except for the system start capability are duplicated at the pneumatic tube control center in the basement auxiliary support space immediately west of the Reactor Laboratory. Automatic timing of irradiations is done at this control center, and rabbits are inserted, dispatched, and removed at this location. An area monitor indicates radiation level and gives a visible and audible alarm should the radiation level exceed a preset level at the station. The preset level is selected according to the computed activity of the sample being irradiated.

The pneumatic tube station is installed in a fume hood with a high efficiency filter to control any releases from sample failures. Sample activity is limited to a level which, should the sample rupture upon discharge from the system, will result in keeping the concentration exhausted below 10 CFR Part 20 limits for unrestricted areas when averaged over a period of 24 hours for routine samples, or 30 days for non-routine samples (requiring RSC approval).

The reactivity effect from a sample is restricted to less than 0.2% ρ. Tests run with water and cadmium samples indicate that sample reactivity effects will normally be less than 0.01% ρ. Static reactivity measurements will be run for samples of fissionable material or particularly strong absorbers such as some of the rare earths.

Since the pneumatic tube penetrates the shield below water level, a leak in the tubing could drain the pool. Spring-close air-open automatic ball valves located in the tubing just outside the shield automatically close when the pneumatic system is turned off. Should these valves fail to close, water will not be lost from the system from a break inside the pool unless another break in the tubing occurs outside the pool. Further, to drain more than 8 feet of water from the pool a siphon action would have to be set up. A siphon action is prevented by a solenoid valve controlled siphon breaker at the highest point in the system. The solenoid valves close when the pneumatic system is started. When the pneumatic system is off, the solenoid valves open and check valves will then allow air to enter the system if a siphon action starts. Normally these check valves prevent loss of cover gas from the system.

The system is operated using a written check-list type procedure to assure that the built-in safeguards remain effective.

[redacted]

Figure 10-4 Pneumatic Tube System Layout

10.2.4 Grid Box Irradiation Facilities

Irradiation of larger samples and most irradiations of more than twenty minutes duration are performed in irradiation facilities on the core periphery inside the grid box.

Radiation baskets (**Figure 10-5**) are 3-inch square aluminum containers which fit into the grid plate and may contain one or more samples. The bottom end boxes are similar to those of reflector elements, thus positioning the devices in fixed positions relative to the core. These devices may contain internal shelves or other positioning devices to position samples in fixed positions.

Figure 10-6 shows a hydraulic irradiation tube (called a whale tube at the facility). In this facility sample movement is powered by a separate pump located beneath the north side of the reactor bridge. The bottom ends of these tubes fit into the grid plate, and the top of the tube is fastened to the bridge structure to provide further support and prevent inadvertent movement. The motor for the pump is electrically paralleled to the diffuser pump and thus runs when the diffuser pump is in operation. The pump takes its suction just below the pool surface and directs its flow to a jet pump near the bottom end of each tube, causing sufficient induced flow down the tube to move samples to the irradiation position and hold them in place. Samples which float return to the top of the tube where they are retained until removal by operating personnel. Non-floating samples can be removed with a retriever tool, or they may be installed with a retrieving string or wire attached. Flow direction and "sample in" indicators and controls are located at the pool top and control console.

Rupture of piping connected to the hydraulic irradiation facility will not result in loss of pool water due to its location within and immediately above the pool. Reactivity effects of samples are much smaller than those associated with installation and removal of conventional irradiation baskets with samples in them. The remarks regarding reactivity effects for samples in the pneumatic tube apply to this facility.

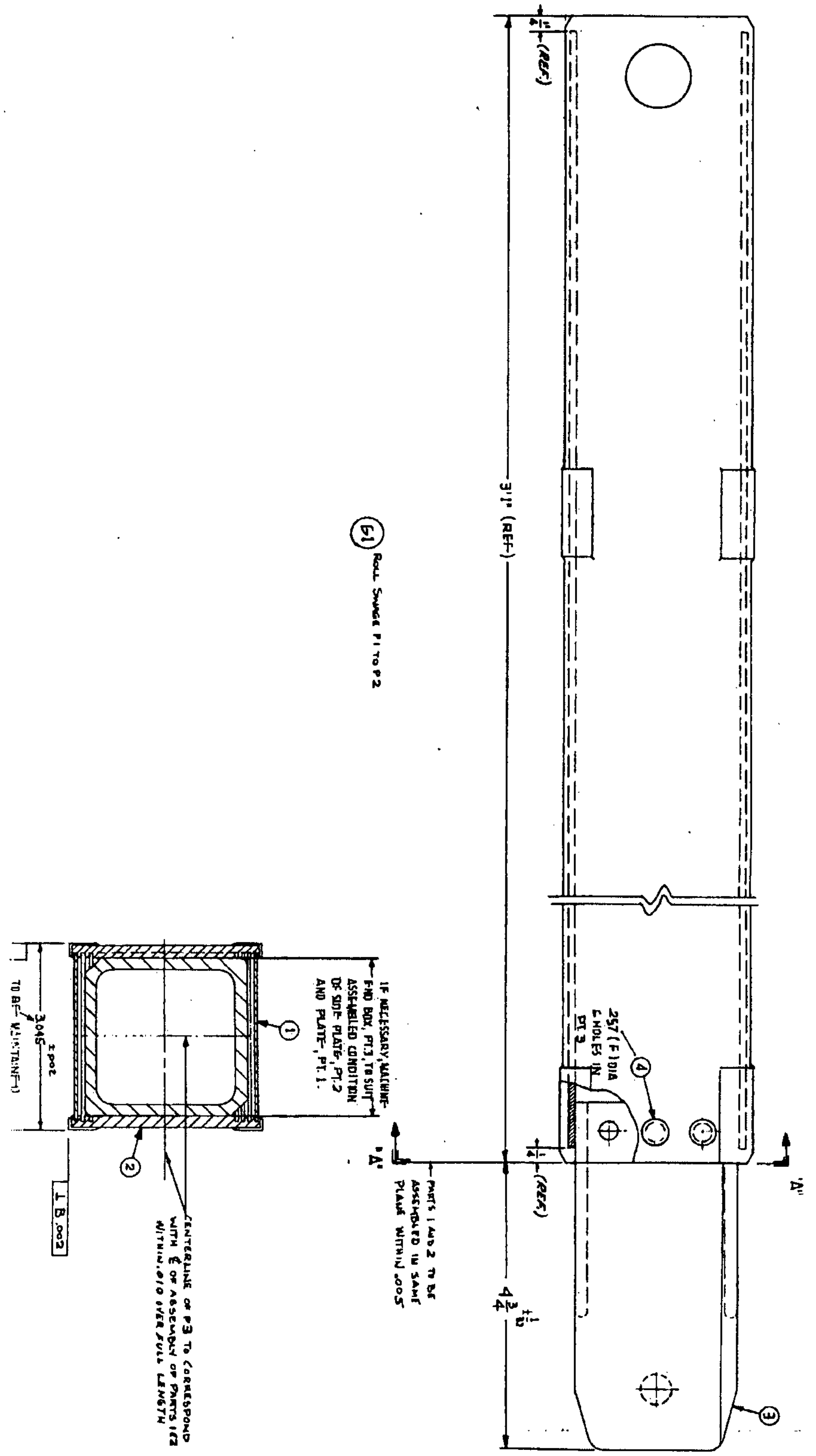


Figure 10-5 Radiation Basket

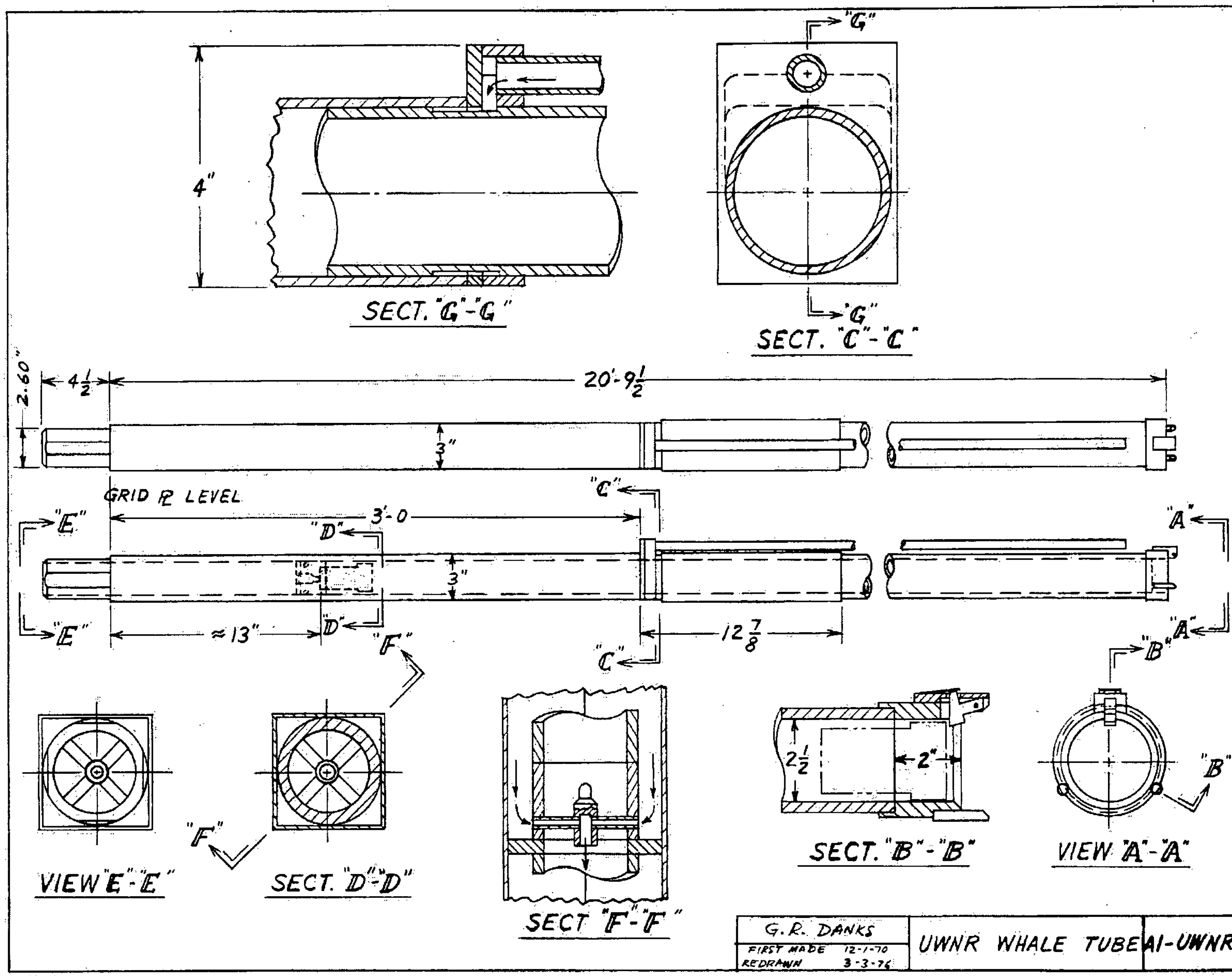


Figure 10-6 Whale Tube

10.3 Experiment Review

A body of operating procedures is in place to assure that experiments are conducted in a manner that will ensure the protection of the public. Experiment review meets the requirements of Regulatory Guide 2.2[1] and standard ANSI N401-1974/ANS-15.6 [2] as modified by Regulatory Guide 2.4 [3].

UWNR 002, Experiment Standing Operating Instructions, defines several classes of experiments that are routinely conducted and states the limitations and precautions to be observed as well as the methodology to be used. Control Element Calibrations, reactivity coefficient measurements, in-core neutron flux distribution measurements and sample irradiation/isotope production experiments are specifically defined in these instructions.

Since sample irradiation and isotope production are major experimental activities at UWNR, several additional standard operating procedures and limitations are in effect for these activities. Limits on potential airborne radioactivity produced in the event of sample breakage are included in the procedures to assure releases will not exceed those considered in the safety analysis report or those permitted under technical specifications. Irradiation of fueled experiments is controlled so that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 Curies. UWNR 002 allows SRO approval of irradiations meeting the requirements of UWNR 131 up to the limits for routine approval of gas, dust, highly volatile material, and fissionable material stated on UWNR 130, Request For Isotope Production. Approval for limits above those stated in UWNR 130 but below 10 CFR Part 20 limits when averaged over 30 days of dilution require approval by the Reactor Safety Committee. Other written procedures in the UWNR 130 series, including sample packaging requirements for the different irradiation facilities and approvals, are in effect for operation of all experimental facilities. Irradiation of material that is to be transferred to the campus broad radioisotope license requires both written and telephone approvals to assure that the recipient of the material is permitted possession and use of the material under that license.

For other experiments the senior reactor operator (SRO) responsible for operation when the experiment is performed classifies the experiment as routine (previously approved and performed), modified routine (determined not to be significantly different from previously performed experiment), special (not previously approved, but within constraints of technical specifications), or special requiring NRC approval (involving technical specification changes or unreviewed safety questions) .

Routine experiments may be approved by the SRO without further evaluation. For other experiments, the SRO evaluates the experiment in terms of its effect on reactor operation and the possibility and consequences of experiment failure, including consideration of chemical reactions, physical integrity, design life, proper cooling, reactivity effects, and interaction with core components. If the experiment is classified as modified routine then two SROs may approve operation of the experiment if it is determined and specified in a written record that the hazards associated with the modified routine experiment are not significantly greater or different from those involved with the corresponding routine experiment.

If the experiment is determined to be a special experiment, an experiment review questionnaire (UWNR 030) which includes description of the experiment including materials inserted, thermodynamics, reactivity effects, radioactivity, shielding, instrumentation related to control panel instruments, administration, and procedures, as well as a safety analysis must be completed and reviewed. Special experiments must be reviewed and approved by the Reactor Director and the Reactor Safety Committee. Favorable evaluation of an experiment shall conclude that failure of the experiment will not lead directly to damage of reactor fuel or interference with movement of a control element. Special requiring NRC approval experiments require local approvals as well as approval by NRC.

10.4 References

1. Regulatory Guide 2.2, Development of Technical Specifications for Experiments in Research Reactors, US Nuclear Regulatory Commission, November 1973

2. American National Standard ANSI N401-1974/ANS 15-6, Review of Experiments for Research Reactors, American Nuclear Society, November 19, 1974

3. Regulatory Guide 2.4, Review of Experiments for Research Reactors, U. S. Nuclear Regulatory Commission, May 1977

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

This chapter deals with the program and procedures for dealing with radioactive materials, radiation, and radioactive waste management. Since the Nuclear Reactor Laboratory is a part of the University of Wisconsin-Madison the campus radiation safety regulations govern activity under the reactor license. Information on these campus regulations was promulgated under Wisconsin Department of Health and Family Services license 25-1323-01. The Radiation Safety Regulations[1] are also available on the UW Radiation Safety website. This information is incorporated by reference as part of this Safety Analysis Report.

The intent of the campus radiation safety program is to maintain radiation exposure to experimenters, students, and the general public as low as reasonably achievable as well as below regulatory limits while using radiation and radioactivity for teaching and research purposes. The implementation of the campus program within the activities of the Nuclear Reactor Laboratory has the same intent.

11.1 Radiation Protection

The radiation protection program at the reactor facility, while conforming to the campus program, has some specific aspects that apply only to the reactor facility. For instance, the design of the experimental facilities, the reactor pool, and the reactor shield includes protective measures and devices which limit radiation exposures and release of radioactive material to the environment. Information on these aspects of the radiation control program is included in the sections of this report that describe that equipment. General requirements, such as dosimeter use and records, certification of training, survey frequency, leak testing of sources, and overall ALARA program are discussed in the campus documentation. The remaining portions of this chapter will deal with the issues specific to the reactor.

11.1.1 Radiation Sources

11.1.1.1 Airborne Radiation Sources

11.1.1.1.1 Releases from abnormal reactor operations

The fuel retains the fission products, with releases to the environment only if the fuel clad is breached. This possibility is one of the accidents considered in Chapter 13 of this report in the analysis of the maximum hypothetical accident. This event would result in maximum dose to personnel within the Reactor Laboratory and the maximum dose released to the environment. The maximum occupational dose calculated is 10 millirem whole body, 1 rad to the lung, and 18.9 rads to the thyroid, while the maximum dose to persons in unrestricted areas will be less than 0.153 rem whole body and 1.019 rad to the thyroid.

11.1.1.1.2 Releases from normal reactor operations

Argon-41 is the only activity released in significant quantities during normal operations. Calculations and measurements have been performed to determine production and release rates of the various activities that might be discharged due to normal operation. The calculation method used for Ar-41 release is shown in Appendix A, Sections A and B.

Due to the operation of the beam port and thermal column ventilating system and the laboratory exhaust fan, the airborne activity levels in the laboratory are low. Some Ar-41 is produced in the dissolved air in the pool water as it passes through the reactor core and is released as the water is warmed while passing through the core. Some of the resulting activity eventually reaches the pool surface where it is released to the laboratory atmosphere. The concentration of Ar-41 in the air immediately above the pool surface during full-power operation reaches about one-third of the DAC for occupational exposure; as this air diffuses throughout the laboratory, the activity in the laboratory as a whole is at least a factor of 6 below the DAC. Therefore, further discussion will be concerned with the activity released to the atmosphere.

The maximum release rate of Ar-41 would occur with the reactor operating continuously at 1 MW and all four beam ports and the thermal column open. Such operation is not reasonable, but it does establish an upper limit to the activity that might be discharged. This maximum release rate is 13.3 μCi/sec, giving an Ar-41 concentration at the stack outlet of 2.94×10^{-6} μCi/ml. The EPA COMPLY program[2] indicates that the maximally exposed receptor would receive a dose of 0.6 mrem/year if all activity generated were discharged continuously.

The maximum concentration to which the public would be exposed (using Gifford's model as discussed in Appendix A and assuming a zero stack height) in this case would be about 3.31×10^{-9} μCi/ml.

As previously indicated, the above maximum value is for a situation not likely to occur during operation. The usual procedure is to have the experimental facilities in a no-flow condition if possible. Under no-flow conditions the beam port and thermal column ventilation system keeps the pressure in the experimental facilities lower than room pressure, and the activity produced in the facilities remains there and decays. The ALARA measures taken on the experimental facilities limits the typical release rate to about 10% of the production rate. Historically, in the year in which the maximum recorded Ar-41 release to the environment occurred (1999-2000 fiscal year), the COMPLY program indicated a resulting dose of 0.004 mrem/year.

One theoretically important consideration in the analysis of a reactor location is the effect on surrounding unrestricted areas of a spillage of radioactive materials. A release of radioactive material might occur, for example, if a highly-volatile liquid were irradiated in the reactor for the production of isotopes. If, while it was being transferred from the reactor to a cask, it were dropped and its container broken, the atmosphere within the Reactor Laboratory could become conceivably contaminated; further, this atmosphere could conceivably be released to the surroundings in such a fashion as to present an exposure in unrestricted areas.

For a typical solid or liquid spill, no special problems exist other than the direct radiation from the sample and cleaning up contamination. Since the level of radiation will be known for each sample, adequate equipment for handling the sample will be available when the material is discharged from the reactor. Equipment adequate for cleanup of spills will be kept available so that spills can be dealt with immediately, lessening the possibility of spreading contamination to adjacent areas.

The remainder of this section will deal with gases, highly volatile liquids, or powdered samples which might cause air-borne activity in the event of a spill. This problem is handled at Wisconsin by a combination of administrative and operational procedures. For routine operations, a concerted effort will be made to keep the concentration of contaminants in the atmosphere released from the Reactor Laboratory well below the limits as stated in Table 2, Appendix B, 10 CFR Part 20, "Standards for Protection Against Radiation." Among the procedures which will be followed to achieve this goal will be the double-encapsulating of materials to be exposed in the reactor in aluminum containers (for long exposure) or sealed polyethylene containers for exposures of less than 4×10^{17} thermal neutrons/cm^2 with accompanying gamma ray and fast neutron fluxes. Only members of the reactor staff (or selected and trained individuals working under their supervision) will be permitted to handle these capsules within the Reactor Laboratory and the capsules will normally be opened only at appropriate locations outside the laboratory. Further, a log book will be maintained of all material exposures. However, because accidents can occur, the amount of radioactivity which will be generated in any one sample of material will be limited. Specifically, this amount of radioactivity will be limited for routine samples such that, should a container be broken and its contents disperse in the air within the Reactor Laboratory, the concentrations discharged through the stack when averaged over 24 hours will not exceed concentrations of 10 CFR Part 20 Appendix B Table 2. In normal operation (single fan running), the ventilation system has a capacity of 9,600 scfm through its filters, therefore 24 hours of dilution is 3.9×10^{11} ml. For non-routine samples, RSC approval will be required, but the above limits will still apply assuming a 30 day dilution instead of 24 hours. 30 days of dilution is 1.2×10^{13} ml. These approvals will consider all other activity discharged, and will insure that the total stack discharge lies within permissible limits should the sample rupture.

The pneumatic tube station is located in the Reactor Laboratory auxiliary support space and thus it is subject to the laboratory ventilation system. The station is installed within a fume hood having a face velocity of > 100 lfpm to protect the system operator in case of sample breakage. . The air discharged from the hood is passed through a dedicated high efficiency filter before connecting to the main laboratory ventilation exhaust.

Although special packaging requirements are enforced to prevent breakage of pneumatic tube samples, such breakage may occur. Sample activity is limited as discussed above, assuming 24 hours of normal ventilation dilution for routine samples, and 30 days of dilution for non-routine samples requiring RSC approval. The fume-hood blower operates automatically whenever the pneumatic tube system is used. As with the other samples, the maximum activities generated for non-routine samples must have RSC approval, and only quantities considerably smaller are routinely approved.

11.1.1.2 Liquid Radioactive Sources

The only activity produced in liquid form in amounts sufficient to be a personnel exposure hazard is Nitrogen-16, which is produced in the reactor coolant as it passes through the reactor core when operating at power levels above 100kW. N-16 is controlled by use of the diffuser system (discussed in Section 5.6), which reduces the dose rate at the pool surface to 2 to 3 mrem/hour during full power operation. If the diffuser system fails during full power operation the dose rate at the pool surface is less than 100 mrem/hour.

Small quantities of liquid radioactive waste are generated by regeneration of the demineralizer and from liquids irradiated as part of sample irradiation. The radiation level from such liquids is extremely low and does not produce radiation exposure hazards. Disposal of this material is addressed in section 11.2.3. Releases are made to the sewer system within 10 CFR Part 20 Appendix B Table 3 limits. Annual liquid releases have ranged from 0 to 10,000 gallons, with 3000 gallons being typical.

11.1.1.3 Solid Radioactive Sources

The major source of radiation and radioactivity is the fission product generation in the reactor fuel. Typical four-element fuel bundles will generate fields of 100 to more than 1000 R/hour in air at 3 feet if removed from the reactor pool. [redacted] As long as the fuel is contained within the pool filled with water this source of radiation dose presents no personnel hazard. Loss of pool water is considered in Chapter 13, with the conclusion that the dose rates from pool water loss after long periods of operation could result in high radiation levels at the pool top (1200 R/hour one day after shutdown), but not so high that persons could not perform corrective actions to restore enough pool level to reduce the dose rate to tolerable levels (dose rate at the pool top level when shielded by the pool curb would be about 240 mrem/hour at the same decay time). Further, the pool is designed to preclude loss of pool water, and operation would not take place if there were any difficulty in maintaining pool level.

Other possibilities of significant radiation exposure from solid radioactive material are the standard 20% enriched TRIGA core, samples irradiated for isotope production, reactor components which have spent a long time near the core, and the reactor startup source. All of these are small sources compared to fuel fission product activity in the operating core. Dose rates from the old fuel are several orders of magnitude lower than those from the operating core. Sample handling equipment and procedures and use of aluminum for almost all structure near the core reduce exposure rates from samples and activated materials to levels which generate no significant personnel hazard during operation or maintenance of the reactor. For example, the shim-safety blades, reflector elements, and transient control rod have maximum radiation levels of a few R/hour at contact after a week of reactor shutdown. Activity produced during irradiations is calculated before the irradiations are performed and equipment and procedures are in place to deal with the activity after the irradiation is completed.

11.1.2 Radiation Protection Program
11.1.3 ALARA Program

Note: These two sections are combined.

The University Radiation Safety Regulations[1] are written to incorporate ALARA principles and practices. The Nuclear Reactor Laboratory policies and procedures reflect the commitment to ALARA principles. An annual ALARA review is conducted jointly by campus Safety Department health physics staff and the Reactor Laboratory staff with a report of the results of the review being submitted to the Reactor Director and the Reactor Safety Committee.

11.1.4 Radiation Monitoring and Surveying

The campus regulations[1] specify requirements on monitoring and surveying. Procedures for reactor operation reflect these requirements. Installed radiation and air activity monitors are described in Section 7.7 of this report. Area radiation surveys are conducted each month, including checks for contamination and particulate air activity. Sample irradiation procedures and forms require checks of radiation level each time a sample is removed from an irradiation facility. Experiment reviews and approvals require radiation surveys for new experiments and modifications of experiments.

11.1.5 Radiation Exposure Control and Dosimetry

The campus regulations[1] specify requirements on radiation control and dosimetry, and the Safety Department administers the dosimetry program. TLD dosimeters are used for operating personnel and experimenters using the laboratory on a regular basis, and electronic dosimeters are used and records are maintained for tour groups and visitors.

Experiment approval requires that no Very High Radiation Areas are created external to the experiment shielding. Some experiments have shield cavities large enough for personnel entry, however, and higher radiation levels can exist inside the shield. Should an experiment design be approved with a Very High Radiation Level within the experiment shield, protective measures will be in place that will reduce radiation levels to no more than a high radiation area if access is attempted. If a High Radiation Area is created inside or outside of the shielding, access will be controlled and posted in accordance with 10 CFR 20.1601.

Radiation doses received by visitors and tour groups are so low that they routinely cannot be measured; the maximum dose rate allowed for any tours is 0.5 mrem/hour and for any non-radiation workers is 2.0 mrem/hour. Visitors, who are radiation workers not part of the campus dosimetry program such as visiting researchers, are allowed higher dose rates, but rarely do they exceed 2.0 mrem/hour due to ALARA practices. No student dosimeter has ever received a measurable exposure from reactor operation. Occupational exposures of operations and maintenance personnel have historically been very low, seldom exceeding 0.5 Rem TEDE in a year and usually below 100 mrem/year.

11.1.6 Contamination Control

The campus regulations[1] specify requirements on Contamination Control. As noted in section 11.1.4, monthly contamination surveys are conducted. Laboratory policy is that no detectable removable contamination is allowed; any contamination discovered is immediately decontaminated.

For routine cleaning, the laboratory has cleaning equipment which is dedicated to use in the laboratory area, and custodial personnel use this equipment in order to prevent the possibility of spreading unidentified contamination. Floor sweepings are surveyed for radioactivity before disposal.

11.1.7 Environmental Monitoring

Environmental TLD monitors are used and evaluated on a quarterly basis. The dosimeters are distributed around the engineering campus so that they surround the Reactor Laboratory. At the present time more than 25 points are monitored. Effluent concentrations are measured at the point of release.

11.2 Radioactive Waste Management

The campus regulations[1] specify requirements for dealing with radioactive waste on campus. The Reactor Laboratory follows the campus regulations.

11.2.1 Radioactive Waste Management Program

This is a campus program.

11.2.2 Radioactive Waste Control

This is a campus-wide program. Liquid waste from beam port drains, pool overflow, laboratory floor drains, radioactive sink, and demineralizer regeneration is stored in a 2000-gallon holdup tank (see Chapter 5, section 5.5), and other liquid radioactive wastes generated in the laboratory are collected in local containers. Filled local containers may be dumped into the holdup tank.

11.2.3 Release of Radioactive Waste

Solid radioactive waste is transferred to the Safety Department for disposal.

Liquid wastes can be transferred to the Safety Department, but most are placed into the holdup tank. The Reactor Laboratory occasionally discharges liquid waste from the holdup tank to the sewer system. All discharges are filtered so that no particulate activity above 0.5 micron size is discharged. Sampling, analysis, and release of the holdup tank contents are governed by a written procedure that assures releases are within 10 CFR Part 20 Appendix B Table 3 limits and that the pH is within local limits for discharge to the sewer.

11.3 References

1. Radiation Safety Regulations, University of Wisconsin-Madison, Revision 2, January 1997.

2. U. S. Environmental Protection Agency, COMPLY Program Rev. 2, October 1989

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12 CONDUCT OF OPERATIONS

12.1 Organization

12.1.1 Structure

Figure 12-1 is a chart indicating the operating organization. Position responsibilities and authorities are summarized in the following sections.

12.1.2 Responsibility

12.1.2.1 University Radiation Safety Committee

1. To exercise its prerogatives (as a campus-wide committee appointed by the Chancellor of the University of Wisconsin-Madison Campus to review all activities on campus which involve the use of radiation) in reviewing all activities related to the Reactor Laboratory.

2. To advise the Reactor Director of all studies and/or actions taken with regard to the Reactor Laboratory.

3. To overrule the Reactor Director where necessary in carrying out its function.

4. To supply health physics services to the University.

12.1.2.2 University Radiation Safety (Part of University Department of Environment, Health and Safety)

1. To assist the University Radiation Safety Committee by conducting inspections, making recommendations, maintaining records, and establishing procedures for emergency operations, waste disposal, etc.

2. To provide similar inspections and service functions to the Reactor Safety Committee.

12.1.2.3 Chair, Engineering Physics Department

1. Responsible for the reactor facilities licenses and charter.

2. To appoint the Reactor Director.

| 12.1.2.4 Reactor Director

1. To approve all policy decisions and all basic regulations, basic instructions, and basic procedures governing the use and operation of the reactor and related facilities.

2. To designate the Reactor Supervisor and other Senior operators.

3. To take cognizance of all recommendations and actions by the University Radiation Safety Committee (which relate to the reactor facility) and the Reactor Safety Committee.

4. To appoint qualified members to the Reactor Safety Committee as necessary.

| 12.1.2.5 Reactor Safety Committee

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

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

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

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

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

6. Review of audit reports.

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

| 12.1.2.6 Reactor Supervisor

1. To initiate and enforce policies, administrative rules, regulations, and operating procedures relating to the Reactor Laboratory, subject to the appropriate approvals of the Reactor Safety Committee, the University Radiation Safety Committee, and the Reactor Director.

2. To ensure that all activities within the Reactor Laboratory are in accordance with prior approvals from the appropriate committees or from the Reactor Director.

3. The Reactor Supervisor shall have authority to authorize experiments and/or procedures which have been approved by the Reactor Safety Committee. He will prepare specific detailed procedures based on the general procedures approved by the Committee.

4. To see that all proper records are kept.

5. To maintain a Senior Operator's License.

6. To appoint Reactor Operators.

7. The Reactor Supervisor or another Senior operator shall be in charge of the Reactor Laboratory at all times (although not necessarily physically present). The individual in charge, if physically present, shall be responsible for prompt execution of emergency procedures. The Reactor Supervisor or another Senior operator will be present at the facility during fuel manipulation, reactor start-up and approach to power, and recovery from unscheduled scrams and shut-downs, and shall be available on call at other times during reactor operation.

8. To be responsible for safety in the Reactor Laboratory, including responsibility for health physics matters.

9. To advise and prepare information for the committees concerned with the Reactor Laboratory, and to present such information to the committees

12.1.2.7 Senior Operators (alternate Supervisors)

1. To accept responsibility for safe and efficient operation of the Reactor Laboratory when designated by the Reactor Supervisor.

2. To maintain a Senior Operator's License.

12.1.2.8 Reactor operators

1. To hold a Reactor operator's License.

2. To conform to all rules and regulations for operation of the reactor.

3. A reactor operator will be present at the control console at all times when the reactor is in operation.

4. To monitor laboratory activities from a health-physics standpoint.

12.1.3 Staffing

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

1. A licensed reactor operator in the control room (if senior operator licensed, may also be the person required in 3 below).

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

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

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

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

1. Initial startup and approach to power.

2. All fuel handling or control-element manual manipulations.

3. Relocation of any in-core experiment with a reactivity worth greater than 0.7% ΔK/K.

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

12.1.4 Selection and Training of Personnel

The selection and training of operations personnel meets or exceeds the requirements of ANSI/ANS-15.4-1988 Sections 4-6 [1]. The operator training program includes sufficient radiation safety training to meet the requirements of 10 CFR Part 19 and the campus Radiation Safety Regulations. The operator training program is a two step process. First the candidate must take an elective four credit-hour course with a formal training manual, homework, and practical exercises (On the Job Training) included. It includes the equivalent of 0.5 weeks of reactor fundamentals, 1.25 weeks of systems coverage, 0.5 weeks of systems observation, and 0.8 weeks of control room operations administered over the period of one semester. The course is completely described on the UW Nuclear Reactor web page.

After completing the course, successful candidates are selected to participate in the candidacy program. Under this program, candidates perform approximately 12 additional weeks of control room operations and other non-licensed duties, including preventative maintenance and health physics surveys.

The operator proficiency maintenance program (re-qualification program) fully meets the requirements of 10 CFR Part 55 and is formalized as a facility procedure, UWNR 004 [2], which received NRC approval upon initial implementation and is reviewed annually by the facility operating organization along with other facility procedures. The program includes written, oral, and performance testing as well as emergency procedure drills and classes on changes in experiments, facility equipment, and procedures.

12.1.5 Radiation Safety

Radiation safety aspects of facility operation are routinely performed by members of the reactor operating staff, including routine radiation and contamination surveys and sampling of water and air samples. The campus radiation safety organization (see chapter 11), established to oversee all activities involving ionizing radiation on campus, is part of the University Department of Environment, Health and Safety, and thus is an independent organization which reports to the central campus administration. The radiation safety organization has the authority to interdict or terminate radiation safety related activities conducted under the reactor license.

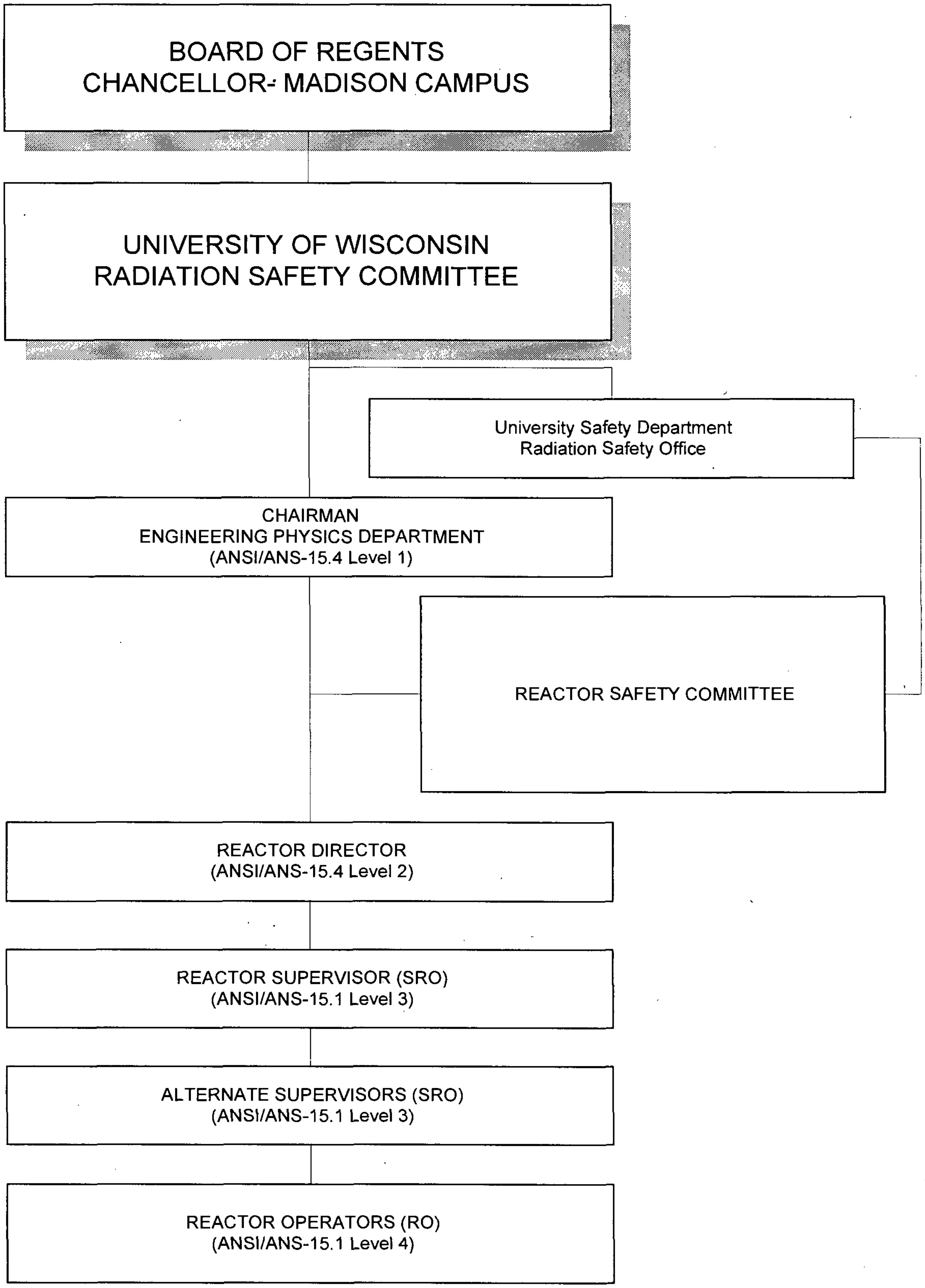

Figure 12-1 Organization Chart

12.2 Review and Audit Activities

12.2.1 Composition and Qualifications

The Reactor Safety Committee is appointed by the Reactor Director. Minimum committee size is six members, one of whom is a health physicist from the University Radiation Safety office. Other members are faculty and staff of the university selected based on expertise to assure that the following disciplines are represented:

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

Reactor operations staff is not precluded from membership on the committee as long as such members do not reach a majority of a quorum for voting. The health physics personnel who perform the monthly audits and inspections are invited to the meetings, but are not necessarily members of the committee.

12.2.2 Charter and Rules

The Reactor Safety Committee operates with a written charter which specifies the manner in which business is conducted. The charter includes rules on meeting frequency (at least annually), voting rules, agenda, quorums, use of subcommittees, minutes, and methods and content of submissions to the committee. Provisions for use of telephone polls or subcommittees for approval of items not requiring a formal meeting are also a part of the charter.

12.2.3 Review Function

The reactor director or designee reviews all written operating procedures at least annually. Results of this review, along with suggested procedure revisions, are submitted to the Reactor Safety Committee for approval, or re-affirmation if no changes are deemed necessary.

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

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

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

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

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

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

6. Review of audit reports.

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

12.2.4 Audit Function

A Health Physicist from the University Radiation Safety Office represents the University Radiation Safety Committee and conducts an inspection of the facility at least monthly to assure compliance with the regulations of 10 CFR Part 20. The services and inspection function of the Radiation Safety Office are also used by the Reactor Safety Committee, with the scope of the audit extended to cover license, technical specification, and procedure adherence.

12.3 Procedures

Written operating procedures are used to assure the safety of operation of the reactor. Procedure use does not preclude the use of independent judgement and action should the situation require such. Operating procedures are in effect for the following items:

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

2. Reactor startup, operation, and shutdown.

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

4. Fuel element and experiment loading or unloading.

5. Control rod removal or replacement.

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

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

8. Civil disturbances on or near the facility site.

Substantive changes to the above procedures may be made only with the approval of the Reactor Safety Committee. Temporary changes to the procedures that do not change their original intent may be made with the approval of two SROs. All such temporary changes are documented and subsequently reviewed by the Reactor Safety Committee.

12.4 Required Actions

In the event a safety limit is exceeded:

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

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

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

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

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

2. Operation in violation of a Limiting Condition for Operation listed in the Technical Specifications.

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

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

5. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits.

6. Abnormal and significant degradation in reactor fuel or cladding which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

In the event of an reportable occurrence as defined in the Technical Specifications, the following actions shall be taken:

1. The reactor shall be shut down.

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

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

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

5. A report shall be made to the NRC.

12.5 Reports

Reports will be made to NRC in accordance with the following:

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

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

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

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

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

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

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

    i. Liquid effluents (summarized on a monthly basis)

        (1) Liquid radioactivity discharged during the reporting period. Tabulated as follows:

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

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

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

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

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

ii. Gaseous Waste (summarized on a monthly basis)

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

(a) Gases.

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

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

iii. Solid Waste

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

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

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

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

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

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

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

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

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

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

   b. Any violation of a safety limit.

   c. Any reportable occurrences.

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

   a. Permanent changes in facility organization at Reactor Director or Department Chair level.
   b. Any significant change in the transient or accident analysis as described in the Safety Analysis Report.

12.6 Records

The following records are retained for a period of at least five years or for the life of the component involved if less than five years.

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

2. Principal maintenance activities

3. Reportable occurrences

4. Surveillance activities required by the Technical Specifications

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

6. Experiments performed with the reactor

7. Fuel inventories, receipts, and shipments

8. Approval of changes in operating procedures

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

Operator qualification and re-qualification records will be retained for at least one cycle of the re-qualification program.

The following records will be retained for the lifetime of the reactor facility. (Note: Retention of annual reports which contain the information in items 1. and 2. are considered as suitable records for those items.)

1. Gaseous and liquid radioactive effluents released to the environs

2. Offsite environmental monitoring surveys required by technical specifications

3. Radiation exposures for all personnel monitored

4. Updated, corrected, and as-built drawings of the facility

12.7 Emergency Planning

The Emergency Plan for the University of Wisconsin Nuclear Reactor was prepared to meet the requirements of ANSI/ANS 15.16-1978 [3] as amplified by Nureg-0849 [4]. This plan was submitted to NRC for review in May 21,1980, with subsequent revisions in October 25, 1982, and May 17, 1984. By letter dated July 25, 1984 NRC indicated that the plan met the requirements referenced above. The plan was again modified and submitted to NRC on May 16, 1990, with supporting information submitted on August 12, 1990. NRC notification that the revision was acceptable was received in a letter dated April 26, 1991. The plan was again modified (Revision 4) to reflect the changes in section number and nomenclature of 10 CFR Part 20 and submitted to NRC on February 17, 1994 and April 22, 1994. This version is the current version in use at the facility.

The Emergency Plan indicates response capabilities for emergency conditions arising in connection with operation of the reactor. It includes identification of various precursor conditions (loss of electrical power, fires, reactor pool leaks, riots, etc) and the consequences for various independent or simultaneous precursor. The plan includes the event classification system. The dose to which people could be exposed under various conditions is indicated, as are the actions that can be taken to minimize the consequences of the emergency. Detailed emergency implementing procedures have been developed and are referenced in the plan.

Primary responsibility for emergency planning and response is given to the Reactor Director. Delegation of responsibility and authority in the absence of the Reactor Director is specified. The Emergency Plan and implementing procedures are reviewed annually to assure that any required changes are incorporated into the plan.

12.8 Security Planning

The facility physical security plan, UWNR 003, was initially submitted to NRC on October 18, 1988. The security plan was revised and submitted again on June 17,1991 and was found to meet the applicable requirements, including the format of Regulatory Guide 5.59 [5]. The plan will require revision as a result of this Safety Analysis Report, since some figures from the previous Safety Analysis Report are included by reference. These changes will be made during the usual annual reviews of the plan once this SAR becomes the document referenced in the reactor operating license.

The security plan indicates the measures provided to protect special nuclear material, including details of the protective equipment and police agencies, and is thus withheld from public disclosure. The Reactor Director is responsible for administering the security program and assuring that it is updated as required.

12.9 Quality Assurance

Since no construction permit is sought in the application for renewal of the license for the University of Wisconsin Nuclear Reactor, no description of a quality assurance program for the design and construction of the structures, systems, and components of the facility is included. This section describes the Quality Assurance program that is in place to govern safe operation and modification of the facility. This program meets the applicable requirements of Regulatory Guide 2.5 [6] and ANSI/ANS-15.8-1995[7]

The Reactor Director has responsibility for the quality assurance activities, and thus has the authority to identify problems, to initiate corrective actions, and to insure that corrective actions are performed. He exercises QA oversight by assuring that operating and maintenance procedures include specific requirements to assure that modification, maintenance, and calibration of safety-related systems are performed in a manner that maintains the quality and reliability of equipment. Further, experiment reviews use written requirements to assure that installation and operation of the experiment does not degrade the performance of safety equipment. Modification of safety-related equipment is planned and reviewed using formal written checklist-type procedures that assure that equipment continues to meet the original specifications. Most of the reactor equipment in use in the facility does not have formal QA documentation because it was built before the QA requirements were in effect. This equipment is covered under the provisions of section 4 of ANSI/ANS-15.8. Several instruments are replacements for the vacuum-tube electronics originally provided by the reactor manufacturer, General Electric Company. This replacement equipment was designed, built, and tested to meet the original specifications stated in the equipment manuals provided with the General Electric equipment. After-maintenance checks, alignment, and calibration of the replacement equipment still assures the equipment meets the original equipment specifications.

Procedures include schedules of equipment maintenance and calibration, and provide records that such functions have been completed. Calibration procedures include requirements that critical equipment and instruments used in the calibrations are themselves currently calibrated (when appropriate).

12.10 Operator Training and Requalification

Operator training and Requalification programs are briefly described in section 12.1.4. The Requalification plan at UWNR is published as a standard procedure (UWNR 004, "University of Wisconsin Nuclear Reactor Operator Proficiency Maintenance Program") which was submitted to NRC on October 24, 1973 and revised on February 7, 1974. By letter dated March 29, 1974 we were notified by NRC that the program meets the requirements of Section 50.54(i-1) of 10 CFR Part 50 and Appendix A of 10 CFR Part 55. Since the program is a numbered procedure it is reviewed by management on an annual basis.

12.11 Startup Plan

The facility has been in routine operation for many years, so a startup plan is not included in this Safety Analysis Report for license renewal.

12.12 Environmental Reports

On January 23, 1974 the AEC staff concluded in a memorandum addressed to D. Skovholt and signed by D. R. Miller, "that there will be no significant environmental impact associated with the licensing of research reactors or critical facilities designed to operate at power levels of 2 Mwt or lower and that no environmental impact statements are required to be written for the issuance of construction permits or operating licenses for such facilities."

Since this Safety Analysis Report is written in support of extending the license expiration date for an additional 20 years, no changes in land and water use are contemplated. Emissions of radioactive materials or other effluents will not change as a result of extending the license term.

12.13 References

1. Standard ANSI/ANS-15.4-1988, Selection and Training of Personnel For Research Reactors, American Nuclear Society, June 9, 1988 ANSI Approval

2. UWNR 004, Operator Proficiency Maintenance Program

3. ANSI/ANS 15.16-1978, "Emergency; Planning for Research Reactors", ANS, LaGrange Park, Illinois, 1978

4. NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors", USNRC, October 1983

5. Regulatory Guide 5.59, Revision 1, "Standard Format and Content for A Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance, US Nuclear Regulatory Commission, February 1983

6. Regulatory Guide 2.5, Revision 0-R, :Quality Assurance Program Requirements for Research Reactors, October 1977

7. ANSI-15.8-1995, "quality assurance program requirements for research reactors", ANS, La Grange Park, Illinois, 1995

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13 ACCIDENT ANALYSIS

NUREG -1537 [1] divides accident analysis into initiating and consequences sections. In this report the two sections are combined; that is, the analyses of the consequences of accidents are grouped with the accident-initiating events and scenarios in order to reduce duplication which would otherwise occur. In addition, a better appreciation of the likelihood and consequences of accidents is afforded. The sections are numbered to correspond to the numbering system of NUREG-1537.

13.1 Accident Analysis Initiating Events and Determination of Consequences

Note: NUREG-1537 specifies separate sections for Initiating Events and Determination of Consequences. Both of these sections have been combined for clarity.

NUREG/CR-2387 [2] reports an independent study of accidents in TRIGA-type reactors and concludes "The only potential for offsite exposure appears to be from a fuel-handling accident that, based on highly conservative assumptions, would result in dose equivalents of ≤ 1 mrem to the total body from noble gases and ≤ 1.2 rem to the thyroid from radioiodines." Notwithstanding that conclusion, the following sections reiterate the analysis done for a license amendment allowing the University of Wisconsin Nuclear Reactor to operate with FLIP or mixed Standard-FLIP cores, using values specific to the UWNR reactor location and characteristics.

13.1.1 Maximum Hypothetical Accident

The maximum hypothetical accident for UWNR is postulated as damage to a fuel element resulting in failure of the fuel cladding. It is postulated that this damage occurs after a very long time of operation at 125% of full power (the power level limiting safety system setting) and that it occurs in the fuel element with the highest power density possible in permitted UWNR mixed core fuel loadings. In a compact 9-Bundle FLIP core the highest power density is 18.1 kW at 1000 kW; the corresponding number for the 15-bundle FLIP core is 17.2 kW, while the value for the currently-used all FLIP core is only 15.2 kW. Continuous operation at the power level scram setpoint is highly unlikely, but for this hypothetical computation the power level in the maximally exposed fuel element of the 9-bundle core is assumed to be 23 kW (1.25 times 18.1 kW rounded to the next highest number).

The likelihood of a major fuel element cladding failure is considered small. The elements must meet rigid quality control standards; pool water quality is carefully controlled; and much care is taken in handling fuel. Such clad failures are, however, possible and the remainder of this section is concerned with the consequences of such a failure.

The release of radioactivity by corrosion and leaching by the pool water has been measured at Gulf General Atomic. About 100 micrograms of U-ZrH per square centimeter of exposed fuel surface per day is released for shutdown conditions. This release is easily controlled by isolating the leaking element in a container provided for that purpose. The gaseous and highly volatile fission products that have collected in the space between fuel and cladding would be the activity contributing to personnel hazards.

13.1.1.1 Fission Product Inventory in Fuel Element

The quantity of these volatile and gaseous fission products was determined by the use of Perkins and King[3] data. Column B of Table 13.1 indicates the fission product activities in the fuel element exposed to the maximum power density.

13.1.1.2 Fission Product Release Fraction

The release of fission products from U-ZRH fuel elements has been extensively studied by Gulf General Atomic and others. The results of this work indicate that the release of fission product gases into the gap between fuel and cladding is given by the following relationship:

$$FR=1.5E\text{-}5 + 3.6E3\exp(-1.34E4/T)$$

where T is the maximum fuel temperature (°K) in the element during normal operation.

The maximum fuel temperature in a fuel element operated in the steady-state mode at 23 KW will be less than 440 °C. Calculations of release fraction however, are based on 600 °C in order to assure a conservative result.

The release fraction corresponding to 600 °C is 7.9 E-4. Applying this fraction to the total inventory of the fuel element as given in column B of Table 13.1 gives the released activity as shown in column C of the table.

For the purpose of further calculations, it is assumed that all gaseous fission products are released to the room air whether the pool is filled with water or not. For soluble volatiles, calculations assume all activity is absorbed in pool water for calculations of pool water activity (column D). For calculations of air activity, the assumption is made that 10% of the volatiles escape with the pool filled with water (columns E and F) and 100% escape with the pool empty.

13.1.1.3 Activity in Pool Water

If 100% of the soluble fission products are absorbed in the pool water, the resulting activity level will be 0.075 μCi/ml. Within 24 hours the level would be reduced by radioactive decay to about 0.012 μCi/ml. After 24 hours the activity decay rate would be chiefly determined by the I-131 half life (8.05 days). The demineralizer will remove most of this activity, giving a radiation dose rate of about 88 mrem/hr at one meter after the activity is deposited in the resins. The resins can be dumped to an underground storage pit or the underground liquid waste holdup tank where the activity will decay without hazard to personnel.

13.1.1.4 Fission Product Release to Air within the Reactor Laboratory

It is estimated that it would take an individual five minutes to evacuate confinement and an additional five minutes to evacuate the building. However for conservatism, it is assumed that the individual is exposed at the highest concentration, namely that found in confinement, for the full ten minutes. Calculations were performed in Appendix A to determine (1) the dose rate due to gamma emitters uniformly dispersed throughout the volume of the reactor lab, (2) the dose to the lungs from beta emitters for an individual remaining in the laboratory for ten minutes; and (3) the dose to the thyroid of an individual remaining in the room ten minutes. For the latter calculations, it is assumed that 10% of the iodine radioisotopes escape from the pool water. In addition to these calculations, a computation of the number of DAC hours indicates that a person present in the room for 10 minutes after the release would receive less than the annual limit on intake for occupational exposure.

(1) Whole body exposure due to gamma emitters

The amount of insoluble volatiles released to the room would be 5.89 Ci. If this activity is distributed uniformly in the laboratory volume, the resulting concentration would be 2.95E-3 $\mu Ci/cm^3$. The resulting maximum dose rate is calculated to be 60 mrem/hr. An individual remaining in the laboratory for 10 minutes after a release would receive a whole body dose of 10 mrem.

(2) Dose to the lungs

The lung is the critical organ when considering the effects of inhaling the insoluble volatiles from a ruptured fuel element. The beta emitting nuclides become more important than those emitting gamma rays since all the decay energy is absorbed in lung tissue. The calculation outlined in the appendix indicates the lung exposure for an individual remaining in the laboratory for 10 minutes after a clad rupture to be 1.0 rad.

(3) Thyroid dose

The thyroid dose to a person in the reactor room was calculated assuming that he remained in the laboratory for 10 minutes after the fission product release. If the pool water is not lost and 10% of the halogens released escape into the atmosphere, the concentrations of the various iodine isotopes would be as presented in Table 13.1. In a ten minute period the lungs would be exposed to the iodine isotope activities shown in Table 13.2. As before, it was assumed that the "standard man" breathes 1.25 m^3/active-hour and his lungs hold 3 liters of air. A conservative calculation results in a dose to the thyroid of 18.9 rads. Although all doses were calculated based on an individual remaining in the laboratory for ten minutes, emergency procedures require immediate evacuation after scramming the reactor, and re-entry to the area is made using a powered air purifying respirator. Actual doses in the event of the accident would be a factor of 10 less than calculated, considering reasonable evacuation times.

TABLE 13.1 FISSION PRODUCT RELEASE FROM CLAD RUPTURE

A ISOTOPE		B SATURATED INVENTORY (Ci)	C RELEASED INVENTORY (Ci)	D AMOUNT IN WATER (Ci)	E AMOUNT IN AIR (Ci)	F LABORATORY CONCENTRATION (µCi/ml)	G Part 20 TABLE 1 DAC (µCi/ml)	H CONCENTRATION DISCHARGED (µCi/ml)	I Part 20 TABLE 2 MAX.EFFLUENT CONCENTRATION (µCi/ml)	J RATIO COL. H/I
Br	82	30	0.024	0.024	0.002	1.2E-06	2E-06	1.46E-10	6E-09	0.0244
	83	105	0.083	0.083	0.008	4.2E-06	3E-05	4.95E-10	9E-08	0.0055
	84	194	0.153	0.154	0.015	7.7E-06	2E-05	9.23E-10	8E-08	0.0115
	'85	253	0.200	0.200	0.020	1.0E-05	1E-07	1.20E-09	1E-09	1.2038
	'87	600	0.473	0.475	0.047	2.4E-05	1E-07	2.85E-09	1E-09	2.8463
TOTAL Br			0.933		0.093					
I	'130m	200	0.158	0.158	0.016	7.9E-06	1E-07	9.45E-10	1E-09	0.9450
	131	563	0.446	0.446	0.045	2.2E-05	2E-08	2.68E-09	2E-10	13.3875
	132	855	0.677	0.677	0.068	3.4E-05	3E-06	4.06E-09	2E-08	0.2031
	133	1282	1.015	1.015	0.102	5.1E-05	1E-07	6.09E-09	1E-09	6.0863
	134	1554	1.230	1.230	0.123	6.2E-05	2E-05	7.38E-09	6E-08	0.1230
	135	1185	0.938	0.938	0.094	4.7E-05	7E-07	5.63E-09	6E-09	0.9375
	'136	602	0.477	0.477	0.048	2.4E-05	1E-07	2.86E-09	1E-09	2.8575
TOTAL I			4.941		0.494					
Kr	83m	105	0.084		0.084	4.2E-05	1E-02	4.95E-10	5E-05	9.9E-06
	85m	253	0.200		0.200	1.0E-04	2E-05	1.20E-08	1E-07	0.1204
	85	51	0.040		0.040	2.0E-05	1E-04	2.40E-09	7E-07	0.0034
	87	486	0.386		0.385	1.9E-04	5E-06	2.31E-08	2E-08	1.1531
	88	699	0.556		0.555	2.8E-04	2E-06	3.32E-08	9E-09	3.6875
	'89	855	0.669		0.669	3.4E-04	1E-07	4.06E-08	1E-09	40.6125
TOTAL Kr			1.935		1.935					
Xe	131m	5	0.004		0.004	2.0E-06	4E-04	2.36E-10	2E-06	0.0001
	133m	31	0.025		0.025	1.6E-05	1E-04	1.50E-09	6E-07	0.0025
	133	1282	1.015		1.015	5.1E-04	1E-04	6.09E-08	5E-07	0.1217
	135m	350	0.277		0.277	1.4E-04	9E-06	1.67E-08	4E-08	0.4163
	135	1243	0.984		0.984	4.9E-04	1E-05	5.91E-08	7E-08	0.8438
	'137	1185	0.938		0.938	4.7E-04	1E-07	5.63E-08	1E-09	56.2500
	138	894	0.707		0.707	3.5E-04	4E-06	4.24E-08	2E-08	2.1206
TOTAL Xe			3.950		3.950					

' indicates generic <2 hr half-life values used

Continuation of Table 13.1

SELECTED RELEASE TOTALS	
Halogen Gamma Emitters	5.2 Ci
Halogen Beta Emitters	5.8 Ci
Total Halogens	5.87 Ci
Insoluble Gamma Emitters	3.52 Ci
Insoluble Beta Emitters	5.50 Ci
Total Insoluble Volatiles	5.89 Ci

13.1.1.5 Release of Fission Products to Unrestricted Areas

Columns H, I, and J of Table 13.1 are concerned with the exposure of personnel outside the restricted area. Calculations were performed as indicated in Appendix A. The maximum concentrations which might be expected in unrestricted areas were calculated under the assumption that venting took place in the time required for the ventilation system to make one complete change in the laboratory (1569 seconds). Wind velocity was assumed to be the lowest average for any month (3.54 m/s).

The total dose to personnel in the unrestricted area is independent of whether the ventilation system is operating in normal mode (one exhaust fan running) or emergency venting mode (both exhaust fans running); the concentration would be considerably higher in the case of emergency venting mode, but the period of exposure would be proportionally shorter. It is also emphasized that the total exposure figure is a maximum to be expected at any point other than within the areas evacuated in the event of an accidental release.

The total of the ratios of instantaneous individual concentrations to 10 CFR Part 20 Appendix B Table 2 maximum air concentrations for discharge (sum of column J from Table 13.1) is calculated to be 134.0, where the maximum air concentration values are for unrestricted areas, 168 hours per week (24 hours per day, 7 days per week). When averaged over a year's time of ventilation (instead of just 1569 seconds), the resulting average concentration is 0.007 of the maximum indicated by 10 CFR Part 20 for non-occupational exposure in unrestricted areas. Even with the effluent discharge from normal operation (see Section 11.1.1) the total concentration to which personnel might be exposed is below the 10 CFR Part 20 limits.

A more conservative calculation which assumes zero stack height (see Appendix A) was performed. This analysis is applicable to a situation in which the laboratory ventilation system fails and the release takes place through building leaks. For purposes of comparison, it was again assumed that the release occurred in the time required for the ventilation system to make an air change in the laboratory (1569 seconds). The effect of this analysis is to multiply the values in columns H and I by a factor of 2.6, giving a resulting average concentration (yearly average) of 0.018 times 10 CFR Part 20 Appendix B Table 2 limits.

Finally, an additional calculation was performed assuming 100% release of Br and I and the more conservative calculation (zero stack height) of atmospheric dilution. The resulting summation of ratio of concentrations to release limit in this case would be 1039. Averaged over a year's time, the resulting concentration (yearly average) is 0.052 times the 10 CFR Part 20 Appendix B Table 2 limits, still within the permissible release concentration when averaged over a period of one year.

As indicated in the table, releases are in all cases less than the 10 CFR Part 20 Appendix B Table 2 limits when averaged over one year. As a backup check to assure that these calculations were conservative, cases equivalent to the maximum hypothetical acident were entered into the EPA

COMPLY program[4] at level 4. Six of the radioisotopes in Table 13.1 are not in the COMPLY listing of radioisotopes (BR-85, Br-87, I-130m, I-136, Kr-89, and Xe-137) and are thus not included in the COMPLY results (these all have half-lives less than 10 minutes). Further, COMPLY does not permit zero height releases from a building, so the stack height was input as 1 meter. For the release with the ventilation system operable and the pool filled, COMPLY indicated an annually averaged dose of 1.8 mrem, with 1.4 mrem due to Iodine. For the release with pool water lost and the ventilation system inoperable (assumed stack height of 1 meter) the COMPLY program indicated an annual dose of 14.2 mrem and 13.7 mrem from Iodine.

Table 13.2 Maximum Exposures In Unrestricted Areas from Maximum Hypothetical Accident

Assumed Failures	Total Body Dose	Thyroid Dose	Fraction of Part 20 Annual Limits
Fuel clad leak with normal operation of ventilation system; pool filled	0.006 rem	0.010 rad	0.007
Fuel clad leak with failure of ventilation system; pool filled	0.084 rem	0.102 rad	0.018
Fuel clad leak with failure of ventilation system and concurrent loss of pool water	0.153 rem	1.019 rad	0.052

13.1.2 Insertion of Excess Reactivity

The worst case result of insertion of excess reactivity would be insertion of the maximum allowed experiment reactivity worth or ejection of the transient rod (1.4% ΔK/K) while the reactor is operating at maximum steady-state power.

Calculations[5] performed by Gulf General Atomic indicate that a peak temperature of 1150 °C in FLIP fuel will not produce a stress in the fuel clad in excess of the ultimate yield strength. Further, TRIGA fuel with a H/Zr ratio of at least 1.65 has been pulsed to temperatures of about 1150 °C without any damage to the clad[6]. In a mixed FLIP-Standard TRIGA core the peak temperatures in FLIP fuel are much higher than in standard fuel due to the peaking of the power distribution near water gaps. For this reason the subsequent analysis in this section is concerned with internal temperatures in FLIP fuel elements.

A worst case core arrangement is considered, in which a FLIP element is located adjacent to a 3-inch square water gap. The power density in the FLIP element is at the maximum permissible value based on consideration of the loss of coolant accident (23 KW when the core is operating at 1 MW). The core is operating at the power level scram point of 1.25 MW, and the transient control rod is fired to initiate a pulse.

Pulses of 2.1% ΔK/K fired in standard TRIGA at this facility have had energy releases of less than 20 MW seconds. FLIP and mixed cores have been operated with maximum reactivity insertions for pulses reduced to 1.4%ΔK/K because of the shorter prompt neutron lifetime in FLIP fueled cores. Typical 1.4% reactivity pulses in an all-FLIP core have energy releases of only about 14 MW seconds, while mixed cores have slightly lower releases. Computations will be done for 20 MW second release.

The limitation of experiment reactivity to 1.4% ΔK/K will insure that reactivity insertions from experiment removal or failure will insure that such an accident will result in consequences no worse than those considered here.

Firing the transient rod while at full power is prevented by interlocks and administrative requirements. Removal of an experiment while operating at full power would not result in a reactivity insertion rate as large as that resulting from firing the transient rod, and the most likely result of experiment removal under the conditions assumed would be a reactor scram from power level, and fuel temperature trips. Further, experiments having worths approaching 1.4% Δ K/K are fastened to prevent inadvertent removal, and administrative restrictions do not allow such manipulations while the reactor is in operation. The predicted conditions establish an upper limit for a reactivity accident.

13.1.2.1 Fuel Temperatures from Operation at the Scram Point

Calculations for the SAR[7] of the Puerto Rico Reactor resulted in the information presented in the lower curve in **Figure 13-1** . This curve shows the fuel temperature distribution at the axial centerline in a FLIP fuel element operating at conditions of slightly higher power density than that assumed here. The Puerto Rico case is an element operating at a power density in the maximum element of 1.4 times the average of 22.3 KW/element. The axial peaking factor is 1.3. Calculations done for UWNR considered the case of an element operating at 23 KW times the ratio of 1.25 of scram setting/licensed power level, with the same axial peaking factor of 1.3. Using these numbers, the fuel centerline and average temperatures will be lower in the UWNR core, but the temperature at the outer surface of the fuel would be approximately the same in both cases.

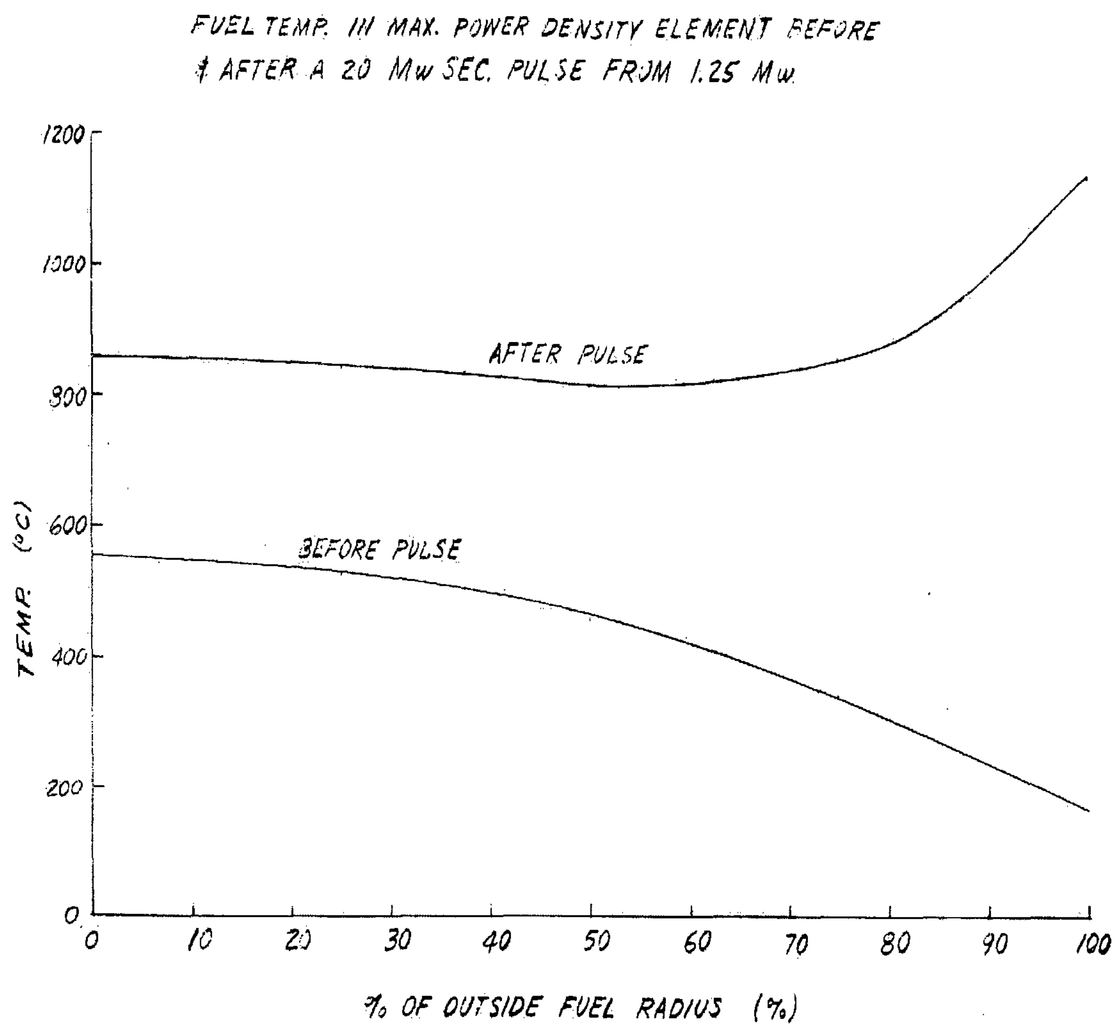

Figure 13-1 Fuel Temperature Distribution in a Fuel Element

13.1.2.2 Temperature after Pulse

Firing a pulse while at the scram point would cause the reactor to scram from power level and fuel temperature scrams. The entire pulse energy release is used, however, in the following analysis.

The temperature distribution in the fuel element immediately after a 20 MW second pulse is plotted as the top curve in **Figure 13-1**. The peaking factor within a FLIP element adjacent to a 3-inch square water-gap is 2.49, and an axial peaking factor of 1.3 is used as in the steady state conditions. The energy deposited in the element under consideration is calculated using the same peaking factor (power in maximum element/ power in average element in core) which resulted in the 23 KW steady state level.

The maximum adiabatic temperature reached in the element will occur at the outer surface of the fuel element adjacent to the water-gap. This maximum temperature would be 1133 C, slightly below the safety limit of 1150 C.

Although such an event is considered highly unlikely, it would not cause fuel damage or release of fission products from the reactor.

After these computations were completed, NRC requested that the accident be re-evaluated for the permitted cores under the technical specifications that were proposed for UWNR. The major changes were limiting the minimum FLIP content to be 9 fuel bundles (35 elements, since the transient rod is located within the 9 central fuel bundles in the UWNR core).

The re-analysis was based on a lower power in maximally exposed fuel element (18.1 kW instead of 22.3 kW) and a limitation of the reactivity insertion from 2.1% ΔK/K to 1.4% ΔK/K. First, the temperature of the fuel in the maximum element will be lower at the beginning of the pulse by about 70°C. Second, the use of a compact array of nine (9) FLIP bundles reduces the possible peaking factor within a FLIP element from the 2.49 value used in the original calculation to a value of 2.03 for a FLIP element beside the transient rod guide tube (this is the position with highest power density in the core.) Finally, reduction of allowable pulsed reactivity insertion from 2.1% ΔK/K to 1.4% ΔK/K will substantially reduce the energy generation in a pulse, while the limitation of experiment worth to 1.4% ΔK/K will provide similar safeguards for experiment failure or removal. Measurements performed on the Puerto Rico Nuclear Center TRIGA-FLIP reactor indicated that a pulse insertion of 1.4% ΔK/K resulted in a maximum fuel temperature rise of approximately 400 °C [8], and measurements at Wisconsin confirmed that prediction.

Consideration of all these differences shows a peak fuel temperature of about 450 °C lower than that indicated above. It is therefore concluded that fuel damage would occur in neither case, but with a much larger safety margin in the more restrictive case considered here.

13.1.3 Loss of Coolant

Although there is little likelihood of complete loss of water from the reactor pool, an analysis is made to demonstrate that such loss will not damage reactor fuel.

13.1.3.1 Possible Means of Water Loss

The pool is contained within the thick reinforced concrete reactor shield which will maintain its integrity under the most severe earthquake that would be expected in this area.

The only credible scenario for draining the pool would be a sheared and open beam port. For analysis in the next section, the time to drain the pool is estimated. The pool water level is routinely maintained at least 20.875ft above the top of the core (this is approximately the low pool level alarm point) which corresponds to 21.5ft above the fuel center. However, the limiting assumption is made that the water level is only 19ft above the top of the core, or 19.625ft above the fuel center. The pool has a surface area of 89.13ft^2. The inner beam port diameter is 0.5ft, and the beam ports are at core center. Using these values, the following equation[9] can be used to estimate the time required to drain the pool.

$$t_d = \sqrt{\frac{2}{g}}\frac{A_p}{A_o C_d}\sqrt{h_0}\left[1 - \sqrt{\frac{h}{h_0}}\right]$$

where: t_d = time to drain pool to height h (sec)
g = acceleration due to gravity (32.174ft/s^2)
A_p = cross-sectional area of pool surface (ft^2)
A_o = cross-sectional area of drain opening (ft^2)
C_d = discharge coefficient (0.6)
h_0 = initial height of water above drain opening (ft)
h = final height of water above drain opening (ft)

The calculated drain time is 836s. A sheared and open beam port could drain the water level to mid-core height, but water would still be in contact with the fuel and would prevent excessive temperatures.

The 8-inch stainless steel pipes built into the pool walls for possible future use in a forced convection cooling system are flange sealed on the outer ends. In addition, one of these pipes has a loop and a siphon breaker extending well above the core so that a rupture cannot lower pool level below the core. The other pipe is flange sealed inside the pool and penetrates the shield wall well above the core. Rupture of either of these lines will not uncover the core.

Rupture of the piping in the demineralizer could cause only slight water loss due to location of the outlet lines from the pool and a check valve at the demineralizer outlet.

13.1.3.2 Radiation Levels in Confinement Due to Unshielded Core

Calculations of radiation levels at various points in the Reactor Laboratory were made assuming continuous operations at 1.02MW. The fission product inventory was determined by the use of the ORIGEN2 computer code version 2.1, using the PWRUS cross-section library. A single fuel pin was simulated at the power level of the hottest rod, with an assumed exposure of 500MWd. The resulting gamma source term was multiplied by the number of fuel elements in the core and then divided by the pin power peaking factor to calculate the core gamma source term at various decay times. The decay time of 836s represents the time required to drain the pool (see section 13.1.3.1). Dose rates from direct and scattered radiation were modeled using the MCNP5 computer code. Results of the calculations are given in Table 13.3.

Table 13.3 Calculated Radiation Dose Rates in Confinement After Pool Water is Lost

Time After Shutdown	Beam Port Floor (R/hr)	Console (R/hr)	Pool Curb Over Core (R/hr)	Pool Top Behind Curb (R/hr)
836 seconds	4.89	6.72	7,430	28.1
1 day	1.23	1.69	1,720	7.07
1 week	0.682	0.938	992	3.92
1 month	0.353	0.485	492	2.02

These levels are not too high to allow emergency repairs to be made. Facility emergency procedures cover the situation of pool water loss.

13.1.3.3 Radiation Levels in Unrestricted Areas Due to Unshielded Core

The calculations from the previous section were also made for the 3rd floor non-restricted classroom to the west of the reactor. This classroom would be subject to the highest dose rate field of any non-restricted area due to the elevation above the biological shield and its line-of-sight with the reactor core.

Table 13.4 Calculated Radiation Dose Rates in Non-Restricted Area After Pool Water is Lost

Time After Shutdown	3rd Floor Classroom (R/hr)
836 seconds	4.14
1 day	0.764
1 week	0.482
1 month	0.210

The calculated dose rate to the 3rd floor non-restricted classroom is significant, but in the event of a loss of coolant accident the building evacuation alarm would alert people to evacuate these classrooms before the core was completely uncovered. In order to estimate the integrated dose received by a member of the public during the evacuation, the MCNP5 model of the unshielded core was modified to include partial water shielding at several time steps. The core gamma source term was also modified to simulate an appropriate level of decay from full power. The integrated dose to the 3rd floor classroom was calculated at various times during the pool water loss and is shown in **Figure 13-2**.

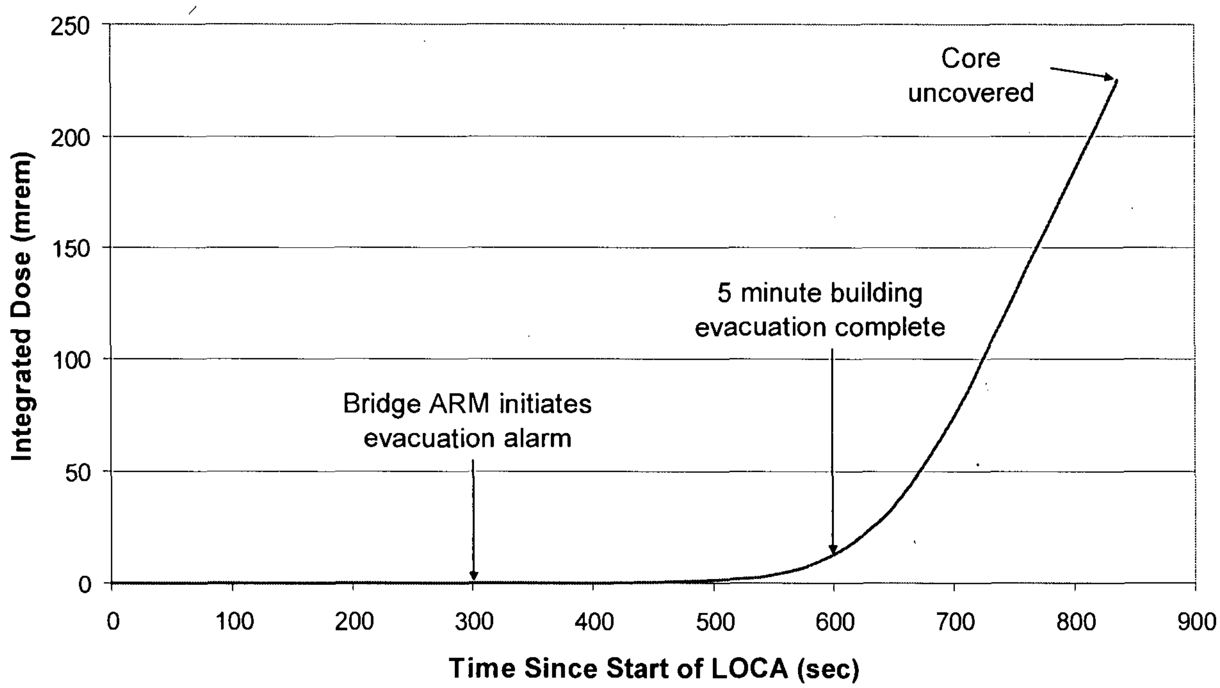

Figure 13-2 3rd Floor Integrated Dose vs. Time during LOCA

A 5 minute evacuation time from the sounding of the evacuation alarm is assumed. If the initiation of the pool water loss was observed then the building evacuation would be manually activated at the start of the LOCA, but if the accident was unobserved then the pool water would drain to approximately 7.4 ft above the core in 300 s before tripping the bridge area radiation monitor, which would in turn automatically initiate the building evacuation alarm. Therefore, the hypothetical member of the public that remains in the 3rd floor classroom for 5 minutes following the automatic initiation of the building evacuation alarm (300 s after start of the LOCA) would receive an integrated dose of about 13 mrem. This is less than the 100mrem limit (10 CFR 20.1301). Realistic doses would be far less than this, because the preceding analysis does not take into account time spent in hallways and stairwells (where the dose rate is much lower) during the evacuation. Because the time spent in the high dose rate field in the 3rd floor classroom would be far less than 5 minutes, the integrated dose would be substantially lower due to the majority of the dose being received in the final minute as shown in **Figure 13-2**.

13.1.3.4 Fuel Temperature After Loss of Pool Water

Calculations performed at Texas A & M University have treated the loss of coolant accident in detail, based on reactor shutdown 15 minutes before the core is uncovered. At Wisconsin, the pool level scram would cause automatic shutdown much sooner, as the A & M calculation is based on pool drainage by rupture of a 10-inch line. Other pertinent parameters of the two facilities are identical. The calculations employed the Gulf computer code TAC for calculation of system temperatures.

The results of these calculations (Pages 25-31 of Texas A & M University Nuclear Science Center Amendment II to the Safety Analysis Report, November 1, 1972 submitted under Docket for License R-83) indicate that for a maximum power density of less than 21 kW/element for standard fuel and 23 kW/element for FLIP fuel, loss of coolant water would not result in fuel clad failure and release of fission products.

13.1.4 Loss of Coolant Flow

Not applicable; natural convection cooling

13.1.5 Mishandling or Malfunction of Fuel

Reference 1 states that this condition produces the maximum consequence to the public. This accident is therefore included as the maximum hypothetical accident (Section 13.1/2.1), when combined with failure of the ventilation system and loss of pool water. The effect of a fuel clad failure with normal pool level and with the ventilation system operating normally has no significant effect on the public.

13.1.6 Experiment Malfunction

Experiment reactivity worth and composition are controlled and limited so that experiment failure will not insert a step change in reactivity greater than 1.4% $\Delta K/K$ (fixed experiments; movable experiments are limited to 0.7% $\Delta K/K$. Procedure for experiment review includes consideration of chemical and explosive hazards to the reactor. Any experiment containing fissionable material is limited so that production of gaseous and volatile fission products results in releases lower than that considered in section 13.1.1. Therefore, experiment malfunction will not result in consequences more severe than those listed in other parts of this chapter.

13.1.7 Loss of Normal Electrical Power

Loss of normal electrical power will cause the reactor to shut down. It will not result in any release of radioactive material or increase the dose to the population. Emergency core cooling engineered safety systems are not required. The maximum hypothetical accident analysis does include loss of the ventilation system in the analysis, thus effectively including loss of electrical power.

13.1.8 External Events

Since the safety of a TRIGA reactor is so strongly a function of the fuel composition and characteristics, none of the usual external event initiators will cause any effect on the public. As stated in chapter 2 of this report, floods and hurricanes are an insignificant threat to the safety of the reactor (Section 2.4), with the 100 year flood causing nearby Lake Mendota to expand by only 30 feet, not threatening the laboratory. Further, should the water level within the room rise above ground level it would not affect the safety of the reactor. Tornados do occur in the Midwest, but damage to the concrete shield which protects the reactor core is not credible. The seismicity of the area is extremely low, with the estimated 50 year peak ground acceleration to a seismic event is less than 0.01 g (Section 2.5.4). Since no engineered structures other than the reactor shield are required to provide protection to the reactor, such an acceleration will have no effect on the reactor. Likewise, though aircraft collisions with the building are not impossible, they are unlikely due to the location of flight paths. Further, such impacts will not breach the concrete shield at core level. Impacting the outer walls of the building will not result in radiation being released.

[redacted]

[redacted]

13.1.9 Mishandling or Malfunction of Equipment

An analysis was made of the possibility that loss of water from the reactor or from the radioactive liquid waste storage tanks could affect the city water supply, and a negative result was obtained as indicated below. Madison obtains its drinking water supply from several wells drilled into the Cambrian sandstone described above. The location of these wells is shown on **Figure 2-12**, and they supply the University as well as the city. All of these wells are cased from ground level into the sandstone so as to keep out water from the glacial deposit. The closest well to the reactor site is about 2,000 feet southeast.

The Reactor Laboratory floor drain empties into the hold tank. Should the entire contents of the pool be let out into the room, however, some water could escape into the sewer system through a drain thimble into which waste water is pumped from the hold tank. There are four methods by which water may leave the reactor room:

(1) by flow pumped from the radioactive waste storage tanks through the elevated drain thimble provided for emptying the tanks;

(2) directly into the drain thimble should the pool be completely ruptured, thus reaching a level high enough to overflow into the thimbles or escape from the laboratory and enter floor drains in surrounding areas;

(3) by loss through the floor and into the ground; and

(4) by rupture of the liquid radioactive waste storage tank directly into the soil under the laboratory floor.

Analysis for cases (1) and (2)

In so far as the first two discharge paths are concerned, the flow through the drain thimble or floor drains empties into a sanitary sewer main. From there it would travel through mains via a pumping station to the main sewage plant, located south of and outside the corporate limits of the city. From there, the sewage travels through mains an additional five miles to the south before it empties into an open ditch. On the way, any water from the reactor would become considerably diluted since the minimum flow-rate into the ditch is 7,000 gpm whereas the probable maximum rate of entry into the floor drain would not be more than 10 to 100 gpm. These facts, coupled with the fact that stringent administrative precautions will be taken to ensure that water contaminated beyond established tolerance levels is not released to the drain, tend to preclude that the city water supply could be adversely affected by this method.

Analysis for case (3)

The possibility that the city water supply could be affected via the third method (3) is also negligible. The base of the reactor is about 8 feet below ground level, and water cannot be dissipated via surface run-off. Since the walls of the building surrounding the reactor are made of concrete up to the ground level, significant water loss through the floor could result only if the concrete was breached. In fact, it would appear that the only mechanism by which contaminated water could enter the soil would be the result of an earthquake sufficiently severe to rupture both the reactor tank and shield, as well as the floor of the building, at a time when the reactor pool water was radioactive beyond tolerance levels. Such a set of coincidental occurrences is considered extremely remote. Further, even if it did occur, there is no assurance that the water supply would be adversely affected. For example, the nearest city well is about 2,000 feet from the reactor site, and it has been estimated by a ground water specialist that water would flow through the sandstone from the reactor to the well at not more than 0.1 foot per day. Thus, as long as 55 years might be required for the reactor water to reach the well.

Analysis for case (4)

Should the radioactive waste storage tanks rupture, a similar analysis to that in case (3) indicates no adverse effect on the well. Furthermore, the quantities of water likely to be lost are small and activities are expected to be low enough that no hazard exists.

13.2 Summary and Conclusions

None of the accidents considered here will result in consequences to the public health and safety. Even the maximum hypothetical accident does not result in releases of radioactivity in excess of 10 CFR Part 20 limits when averaged over a year.

13.3 References

1. NUREG-1537 Part 1, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, USNRC, February 1996.

2. Credible Accident Analyses for TRIGA and TRIGA-fueled Reactors, NUREG/CR-2387, PNL-4028, April 1982.

3. J. F. Perkins and R. W. King, "Energy Release from the Decay of Fission Products", Nuclear Science and Engineering, 3, 726 (1958).

4. COMPLY V1.5d, U.S. Environmental Protection Agency, Office of Radiation and Indoor air, Washington DC 20460, October 1989.

5. "Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor", GA-9064, Gulf General Atomic, Jan. 5, 1970.

6. "Annular Core Pulse Reactor", General Dynamics, General Atomic Division Report GACD 6977, Supplement 2, 9/30/66.

7. Safeguards Summary Report for the TRIGA-FLIP Reactor at Puerto Rico Nuclear Center, Report PRNC 123, Revision C, November 11, 1969.

8. Docket 50-120, Change No. 11 to the Technical Specifications Facility License R-83, Texas A & M University, Section 3.2 Basis.

9. Hunsaker and Rightmire, "Engineering Applications of Fluid Mechanics," McGraw-Hill Book Company, 1947 (pp. 69-70).

14 TECHNICAL SPECIFICATIONS

TS 1 INTRODUCTION

TS 1.1 Scope

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

These specifications are re-formatted from the technical specifications in force in 1999. Changes reflect only changes required by name changes or to include information not in the original technical specifications. In addition, certain additions required by NUREG-1537 are included. All substantive changes were denoted by redlining in Rev 0, but currently only changes since the last revision are redlined (indicated by vertical line in margin). These technical specifications continue to include use of TRIGA-FLIP and the original LEU TRIGA fuels, either separately or in mixed cores.

TS 1.2 Format

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

TS 1.3 Definitions

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

TS 1.3.1 Reactor Operating Conditions

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

PULSE MODE (PU)
Pulse mode operation shall mean any operation of the reactor with the mode selector switch in the pulse position.

REACTOR SECURED:
The reactor is secured when:

1. Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality upon optimum available conditions of moderation and reflection, or

2. The following conditions exist:

    a. The reactor is shut down,

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

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

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

REACTOR OPERATION:
Reactor operation is any condition wherein the reactor is not secured.

REPORTABLE OCCURRENCE:

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

1. Operation with any safety system setting less conservative than specified in the technical specifications;
2. Operation in violation of a Limiting Condition for Operation listed in Section 3;
3. Operation with a required reactor or experiment safety system component in an inoperative or failed condition which could render the system incapable of performing its intended safety function;
4. Any unanticipated or uncontrolled change in reactivity greater than 0.7% $\Delta K/K$, excluding reactor trips from a known cause;
5. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits; and
6. Abnormal and significant degradation in reactor fuel or cladding which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

SHUTDOWN MARGIN:

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

SQUARE WAVE MODE (SW)

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

STEADY STATE MODE (SS)

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

TS 1.3.2 Reactor Experiments and Irradiation

EXPERIMENT:

Experiment shall mean:

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

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

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

Classification of experiments shall be:

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

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

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

EXPERIMENTAL FACILITIES:

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

IRRADIATION:

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

NON-SECURED EXPERIMENT

Any experiment not meeting the criteria of a secured experiment.

SECURED EXPERIMENT:

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

1. Removal of mechanical fasteners
2. Use of underwater handling tools
3. Moving of shield blocks or beam port containers.

TS 1.3.3 Reactor Components

CORE LATTICE POSITION:

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

FUEL BUNDLE:

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

FUEL ELEMENT:

A fuel element is a single TRIGA fuel rod of either standard or FLIP type.

FLIP CORE:

A FLIP core is an arrangement of TRIGA-FLIP fuel in the reactor grid plate.

FLIP FUEL:

FLIP fuel is TRIGA fuel that contains a nominal 8.5 weight percent of uranium with a ^{235}U enrichment of about 70% and erbium as burnable poison.

INSTRUMENTED ELEMENT:

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

MIXED CORE:

A mixed core is an arrangement of standard TRIGA fuel elements and FLIP fuel elements with at least 35 TRIGA-FLIP fuel elements located in a central region of the core.

OPERATIONAL CORE:

An operational core may be a standard core, mixed core, or FLIP core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

REGULATING BLADE:

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

SHIM-SAFETY BLADE:

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

STANDARD TRIGA FUEL:

Standard TRIGA fuel is TRIGA fuel that contains a nominal 8.5 weight percent of uranium with a ^{235}U enrichment of less than 20% and no burnable poison.

STANDARD CORE:

A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate.

TRANSIENT ROD:

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

TS 1.3.4 Reactor Instrumentation:

CHANNEL CALIBRATION:

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

CHANNEL CHECK:

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

CHANNEL TEST:

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

EXPERIMENT SAFETY SYSTEMS:

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

LIMITING SAFETY SYSTEM SETTINGS:

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

MEASURED VALUE:

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

MEASURING CHANNEL:

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

OPERABLE:

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

REACTOR SAFETY SYSTEMS:

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

SAFETY CHANNEL:

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

SAFETY LIMITS:

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

TS 2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

TS 2.1 Safety Limits

Applicability

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

Objective

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

Specifications

1. The temperature in a TRIGA-FLIP fuel element shall not exceed 1150°C under any conditions of operation.

2. The temperature of a standard TRIGA fuel element shall not exceed 1000°C under any conditions of operation.

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

Basis

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

The safety limit for the TRIGA-FLIP fuel element is based on data which indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided the temperature does not exceed 1150°C and the fuel cladding is water cooled [1].

The safety limit for the standard TRIGA fuel is based on data including the large amount of experimental evidence obtained during high performance reactor tests of this fuel. These data indicate that the stress in the cladding (due to hydrogen pressure from the dissociation of zirconium hydride) will remain below the ultimate stress provided that the temperature of the fuel does not exceed 1000°C and the fuel cladding is water cooled [2].

It has been shown by experience that operation of TRIGA reactors at a power level of 1500 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1500kW. It has been shown by analysis and by measurements on other TRIGA reactors that a power level of 1500 kW corresponds to a peak fuel temperature of approximately 600°C. Thus a Safety Limit on power level of 1500 kW provides an ample margin of safety for operation.

TS 2.2 Limiting Safety System Settings

Applicability

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

Objective

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

Specifications

1. The limiting safety system setting for fuel temperature shall be 400°C (750°F) as measured in an instrumented fuel element. For a mixed core, the instrumented element shall be located in the region of the core containing FLIP type elements.

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

Basis

The first limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. A setting of 400°C provides a safety margin of 750°C for FLIP type fuel elements and a margin of 600°C for standard TRIGA fuel elements. A part of the safety margin is used to account for the difference between the true and measured temperatures resulting from the actual location of the thermocouple. If the thermocouple element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. If the thermocouple element is located in a region of lower temperature, such as on the periphery of the core, the measured temperature will differ by a greater amount from that actually occurring at the core hot spot. Calculations and measurements made at the facility with all permitted mixed core arrangements indicate that, for this case, the true temperature at the hottest location in the core will differ from the measured temperature by no more than a factor of two. Thus, when the temperature in the thermocouple elements reaches the trip setting of 400°C, the true temperature at the hottest location would be no greater than 800°C providing a margin to the safety limit of at least 200°C for standard fuel elements and 350°C for FLIP type elements. These margins are ample to account for the remaining uncertainty in the accuracy of the fuel temperature measurement channel and any overshoot in reactor power resulting from a reactor transient during steady state mode operation. For a mixed core (i.e., one containing both standard and FLIP type elements), the requirement that the instrumented element be located in the FLIP region of the core provides an even greater margin of safety since the peak to average power ratio within that region will be smaller than over an entire core composed of elements of the same type.

Calculations and measurements for this and similar TRIGA reactors indicate at 1.25 MW, the peak fuel temperature in the most limiting core loading permitted under section 3 of these specifications (9 FLIP bundles) will be less than 600°C so that the second limiting power level setting provides an ample safety margin to accommodate errors in power level measurement and anticipated operational transients.

In the pulse mode of operation, the first limiting safety system setting will apply. However, the power level channels do not provide protection in pulse mode, and the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). The limit on transient rod worth in another specification limits the generated power so that fuel temperatures reached in a transient are smaller than those from full power operation. This transient rod worth limit is less than the reactivity required for steady-state full power. If the transient rod fails to automatically drop after the pulse, fuel temperature reached due to energy generation in the tail of a pulse will be less than that at full power. Only in the case of operation outside the permitted parameters of core composition and reactivity limitations would fuel temperature safety system actuation be needed to provide protection in any operating mode.

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TS 3 LIMITING CONDITIONS FOR OPERATION

TS 3.1 Reactor Core Parameters

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods. 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 fuel temperature safety limit will not be exceeded.

TS 3.1.1 Excess Reactivity

Specifications

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

Basis

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

TS 3.1.2 Shutdown Margin

Specifications

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

1. the highest worth non-secured experiment in its most reactive state,

2. the highest worth control element and the regulating blade (if not scrammable) fully withdrawn, and

3. the reactor in the cold condition without xenon.

Basis

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

TS 3.1.3 Pulse Limits

Specifications

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

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

Basis

Measurements performed on the Puerto Rico Nuclear Center TRIGA-FLIP reactor indicated that a pulse insertion of reactivity of 1.4% Δ k/k resulted in a maximum temperature rise of approximately 400°C. With an ambient water temperature of approximately 100°C, the maximum fuel temperature would be approximately 500°C resulting in a safety margin of 500°C for standard fuel and 650°C for FLIP type fuel. Tests done on the mixed and all-FLIP cores [3,4,5,6] indicate that the fuel temperatures measured and calculated for the core arrangements allowed by these specifications do not exceed 683°C in the worst case allowed.

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

These margins allow amply for uncertainties due to the accuracy of measurement or location of the instrumented fuel element or due to the extrapolation of data from the PRNC reactor.

TS 3.1.4 Core Configurations

Applicability

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

Objective

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

Specifications

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

2. The TRIGA core assembly may be standard, FLIP, or a combination, thereof (mixed core) provided that any FLIP fuel be comprised of at least thirty-five (35) fuel elements, located in a contiguous, central region.

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

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

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

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

Basis

1. Standard TRIGA cores have been in use for years and their characteristics are well documented. The Puerto Rico Nuclear Center and the Gulf Mark III all-FLIP cores have operated and their characteristics are available. Gulf has also performed a series of experiments using standard and FLIP fuel in mixed cores and a mixed core has been used successfully in the Texas A&M University TRIGA reactor. In addition, studies performed at Wisconsin for a variety of mixed core arrangements indicate that such cores with mixed loadings would safely satisfy all operational requirements (SAR Chapters 4 and 6).

2. In mixed cores, it is necessary to arrange FLIP elements in a contiguous, central region of the core to control flux peaking and power generation peak values in individual elements.

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

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

5-6. Manual manipulation of core components will be allowed only when a single manipulation can not result in inadvertent criticality.

TS 3.1.5 Reactivity Coefficients

Does not apply to TRIGA and TRIGA-FLIP reactors.

TS 3.1.6 Fuel Parameters

Applicability

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

Objective

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

Specifications

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

1. In measuring the transverse bend, its sagitta[7] exceeds 0.125 inch over the length of the cladding;
2. In measuring the elongation, its length of the cladding exceeds its original length by 0.125 inch; and
3. A clad defect exists as indicated by detection of release of fission products.
4. The fuel has not been visually inspected within the previous 15 months.
5. The burnup of uranium-235 in the UzrH fuel matrix shall not exceed 50 percent of the initial concentration.[8,9]

Basis

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

TS 3.2 Reactor Control and Safety Systems

TS 3.2.1 Operable Control Rods

Applicability

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

Objective

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

Specifications

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

Basis

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

TS 3.2.2 Reactivity Insertion Rates (Scram time)

Applicability

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

Objective

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

Specifications

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

Basis

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

TS 3.2.3 Other Pulsed Operation Limitations

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

TS 3.2.4 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

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

Specifications

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

Table 3.2.4 Reactor Safety System Channels

Safety Channel	Setpoint and Function	Number operable in specified mode		
		SS	SW	PU
Fuel Temperature	Scram if fuel temperature exceeds >400°C in the fuel temperature safety channel. In the event of loss of all available fuel thermocouples and inability to obtain a replacement instrumented fuel element, operation may continue in any operational core if the linear power level scram points are reduced to 110% full power.	1	1	1
Linear Power Level	Scram if power > 125% full power	2	2	-
Manual Scram	Manually initiated scram	1	1	1
Preset Timer	Transient rod scram 15 seconds or less after pulse	-	-	1
Reactor water level	Scram if < 19 feet above top of core	1	1	1
High Voltage Monitor	Scram on loss of high voltage to neutron and gamma ray power level instrument detectors	1	1	1

Basis

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

The exception is required because FLIP fuel is no longer manufactured. If a core has been tested to meet the definition of an operational core the power level scrams provide adequate protection to assure the LCO of fuel temperature is not exceeded.

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

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

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

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

TS 3.2.5 Interlocks

Applicability

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

Objective

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

Specifications

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

Table 3.2.5 Interlocks

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

Basis

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

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

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

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

TS 3.2.6 Backup Shutdown Mechanisms

Backup shutdown mechanisms are not required for this reactor.

TS 3.2.7 Bypassing Channels

Applicability

This specification applies to the interlocks in Table 3.2.5.

Objective

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

Specifications

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

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

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

Basis

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

TS 3.2.8 Control Systems and Instrumentation Required for Operation

Applicability

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

Objective

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

Specifications

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

Table 3.2.8 Instrumentation and Controls Required for Operation

Channel	Function	Number operable in specified mode		
		SS	SW	PU
Fuel Temperature	Input for fuel temperature scram. In the event of loss of all available fuel thermocouples and inability to obtain a replacement operation may continue in any operational core.	1	1	1
Linear Power Level	Input for safety system power level scram	2	2	0
Log Power Level	Wide range power indication, permissive for initiation of Pulse Mode	1	1	0
Startup Log Count Rate	Wide range power indication, permissive for control element withdrawal	1*	1*	0
Pulsing Power Level	Pulse power level indication	0	0	1

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

Basis

Fuel temperature indicated at the control console gives continuous information on the process variable which has a specified safety limit. The exception is required because FLIP fuel is no longer manufactured. If a core has been tested to meet the definition of an operational core the power level scrams provide adequate protection to assure the LCO of fuel temperature is not exceeded.

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

TS 3.3 Reactor Pool Water Systems

Applicability

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

Objective

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

Specifications

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

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

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

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

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

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

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

Basis

1. This specification is based on thermal and hydraulic calculations which show that the TRIGA-FLIP core can operate in a safe manner at power levels up to 2,700 kW with natural convection flow of the coolant water. A comparison of operation of the TRIGA-FLIP and standard TRIGA Mark III has shown operation to be safe for the above power level. Thermal and hydraulic characteristics of mixed cores are essentially the same as that for TRIGA-FLIP and standard cores.

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

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

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

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

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

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

TS 3.4 Confinement

Applicability

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

Objective

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

Specifications

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

2. All air or other gas exhausted from the reactor room and associated experimental facilities shall be released to the environment a minimum of 26.5 meters above ground level.

Basis

Calculations in Chapter 13 of the Safety Analysis Report show that exposure of occupants of the Laboratory can be kept below 10 CFR Part 20 limits for occupational exposure under accident conditions if the room volume is 2,000 m^3. Calculations in Chapter 13 of the SAR based on release of radioactive effluent at ground level show that concentrations of radioactive materials are within limits of 10 CFR Part 20 for non-restricted areas during the accidents considered. Further calculations based on release at the stack height show a further reduction by a factor of 2.6 due to operation of the ventilation system and release of effluent at a height of 26.5m.

TS 3.5 Ventilation Systems

Applicability

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

Objective

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

Specifications

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

Basis

It is shown in the SAR Chapter 11 that Argon-41 release at zero stack height results in concentrations less than the concentrations permitted for non-restricted areas. Further, the calculations indicate that operation of the ventilation system reduces the concentration to which the public would be exposed by a factor of 10 below this limit. Exposures in the event of a fuel element cladding leak are also calculated based on non-operation of the ventilation system. Therefore, operation of the reactor with the ventilation system shut down in order to make repairs assures the degree of control on which the calculations are based.

TS 3.6 Emergency Power

Emergency power systems are not required for this facility.

TS 3.7 Radiation Monitoring Systems and Effluents

TS 3.7.1 Monitoring Systems

Applicability

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

Objective

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

Specifications

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

Table 3.7.1 Radiation Monitoring Systems

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

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

Basis

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

TS 3.7.2 Effluent (Argon-41) Discharge Limit

Applicability

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

Objective

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

Specifications

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

Basis

10 CFR Part 20 Appendix B, Table II specifies a limit of 1×10^{-8} μCi/ml for Ar-41. Chapter 11 and Appendix A of the SAR substantiates a release level of 3.3E-9 μCi/ml for a 3.54 meters/second (lowest monthly average) wind speed if all Ar-41 produced were continuously discharged. The dilution factor by which emitted material is diluted is 2.5E-4 μCi/ml per Ci/second discharged.

TS 3.8 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 or excessive release of radioactive materials in the event of an experiment failure.

TS 3.8.1 Reactivity Limits

Specifications

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

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

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

Basis

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

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

TS 3.8.2 Materials

Specifications

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

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

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

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

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

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

   d. An atmospheric dilution factor of 2.5×10^{-4} μCi/ml per Ci/s for gaseous discharges from the facility.

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

Basis

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

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

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

TS 3.8.3 Experiment Failure and Malfunctions

Specifications

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

Basis

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

TS 3.9 Facility Specific LCOs

There are no facility specific LCOs at this facility.

TS 4 SURVEILLANCE REQUIREMENTS

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

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

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

General Applicability

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

Objective

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

Specifications

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

Basis

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

TS 4.1 Reactor Core Parameters

Applicability

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

Objective

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

Specifications

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

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

3. Pulse limits
   The reactor shall be pulsed semiannually to compare fuel temperature measurements (if an operating fuel thermocouple is available) and peak power levels with those of previous pulses of the same reactivity value.

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

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

6. Fuel Parameters

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

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

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

<u>Basis</u>

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

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

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

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

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

TS 4.2 Reactor Control and Safety Systems

Applicability

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

Objective

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

Specifications

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

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

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

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

5. Scram and Power Measuring Channels

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

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

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

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

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

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

Basis

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

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

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

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

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

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

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

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

TS 4.3 Coolant Systems

Applicability

This specification applies to the reactor pool water.

Objective

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

Specifications

The pool water conductivity and radioactivity shall be measured quarterly.

Basis

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

TS 4.4 Confinement

No surveillances are required.

TS 4.5 Ventilation Systems

Applicability

This specification applies to the building confinement ventilation system.

Objective

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

Specifications

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

Basis

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

TS 4.6 Emergency Electrical Power Systems

Not Applicable.

TS 4.7 Radiation Monitoring Systems and Effluents

TS 4.7.1 Radiation Monitoring Systems Applicability

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

Objective

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

Specifications

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

Basis

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

TS 4.7.2 Effluents

Applicability

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

Objective

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

Specifications

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

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

Basis

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

TS 4.8 Experiments

No surveillances are required.

TS 4.9 Facility-Specific Surveillance

Not applicable. There is no facility-specific surveillance.

TS 5 DESIGN FEATURES

TS 5.1 Site and Facility Description

Specifications

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

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

TS 5.2 Reactor Coolant System

Specifications

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

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

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

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

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

TS 5.3 Reactor Core and Fuel

Specifications

1. TRIGA-FLIP Fuel
   The individual unirradiated FLIP fuel elements shall have the following characteristics:

   a. Uranium content: maximum of 9 Wt-% enriched to nominal 70% Uranium 235.

   b. Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms.

   c. Natural erbium content (homogeneously distributed): nominal 1.5 Wt-%.

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

   e. Identification: Top pieces of FLIP fuel bundles will have characteristic markings to allow visual identification of FLIP fuel employed in mixed cores.

2. Standard TRIGA fuel
   The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

   a. Uranium content: maximum of 9.0 Wt-% enriched to a nominal 20% Uranium 235.

   b. Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.7 H atoms to 1.0 Zr atoms.

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

TS 5.4 Reactor Core

Specifications

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

2. The TRIGA core assembly may be standard, FLIP, or a combination thereof (mixed core), provided that any FLIP fuel be comprised of at least thirty-five (35) fuel elements, located in a contiguous, central region.

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

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

TS 5.5 Control Elements

Specifications

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

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

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

TS 5.6 Fissionable Material Storage

Specifications

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

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

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TS 6. ADMINISTRATIVE CONTROLS

TS 6.1 Organization

TS 6.1.1 Structure

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

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

TS 6.1.2 Responsibility

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

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

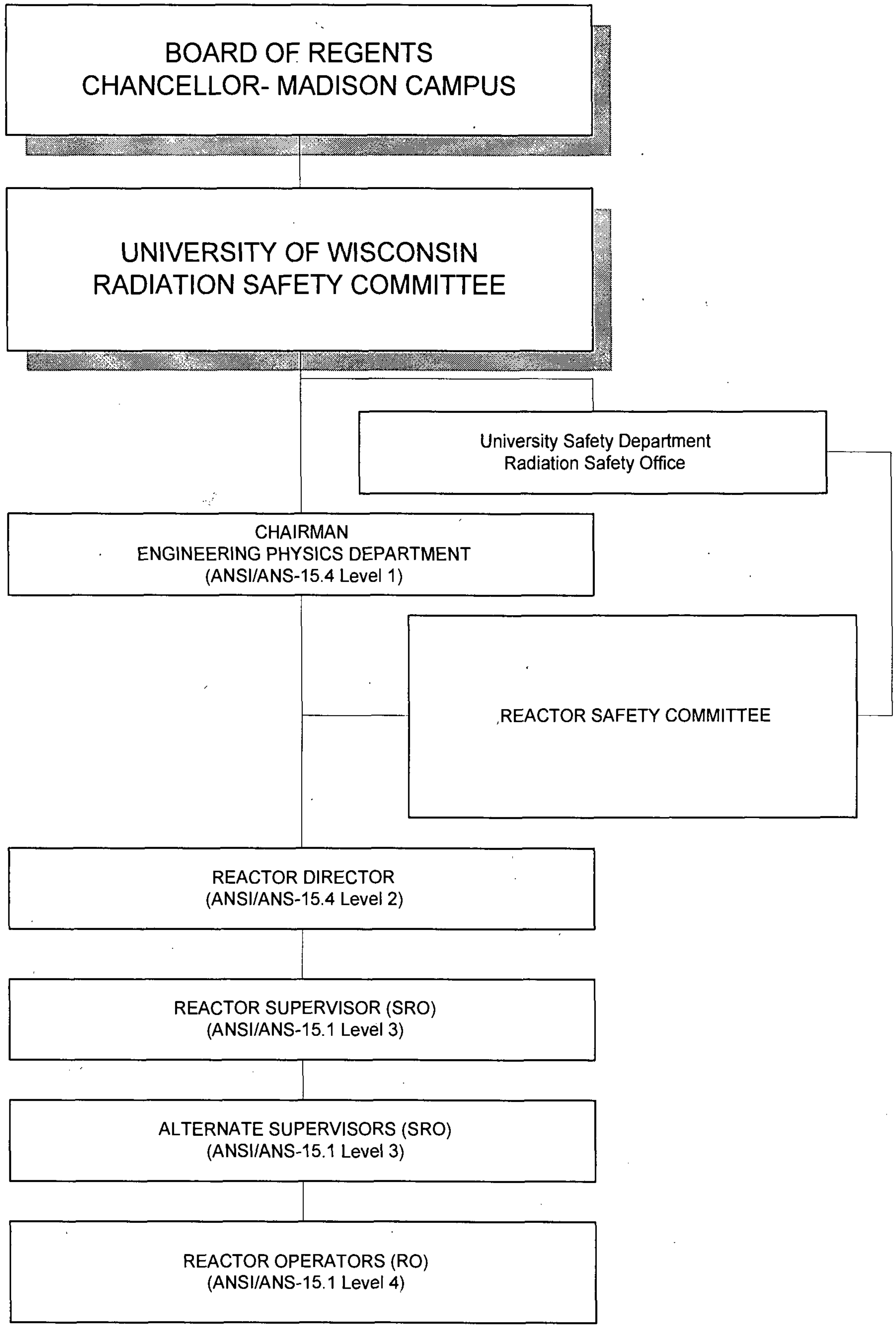

Figure 14-1 Organization Chart

TS 6.1.3 Staffing

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

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

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

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

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

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

   a. Initial startup and approach to power.

   b. All fuel handling or control-element relocations.

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

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

TS 6.1.4 Selection and Training of Personnel

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

TS 6.2 Review and Audit

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

TS 6.2.1 Composition and Qualifications

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

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

TS 6.2.2 Charter and Rules

The Committee shall meet at least annually.

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

TS 6.2.3 Review Function

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

1. Review and approval of experiments utilizing the reactor facilities;
2. Review and approval of all proposed changes to the facility, procedures, license, and technical specifications;
3. Determination of whether a proposed change, test or experiment would constitute an unreviewed safety question or a change in Technical Specifications;
4. Review of abnormal performance of plant equipment and operating anomalies having safety significance; and

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

6. Review of audit reports.

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

TS 6.2.4 Audit Function

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

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

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

TS 6.3 Radiation Safety

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

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

TS 6.4 Procedures

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

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

2. Reactor startup, operation, and shutdown;

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

4. Fuel element and experiment loading or unloading;

5. Control rod removal or replacement;

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

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

8. Civil disturbances on or near the facility site.

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

TS 6.5 Experiment Review and Approval

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

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

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

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

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

TS 6.6 Required Actions

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

In the event a safety limit is exceeded:

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

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

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

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

In the event of an reportable occurrence (1.3.1) the following actions shall be taken:

1. The reactor shall be shut down.

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

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

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

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

TS 6.7 Reports

TS 6.7.1 Operating Reports

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

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

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

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

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

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

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

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

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

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

Liquid radioactivity discharged during the reporting period tabulated as follows:

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

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

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

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

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

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

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

(a) Gases.

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

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

(3) Solid Waste

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

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

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

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

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

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

TS 6.7.2 Special Reports

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

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

   b. Any violation of a safety limit; and

   c. Any reportable occurrences as defined in Section 1.3.1 of these specifications.

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

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

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

TS 6.8 Records

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

1. Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which shall be maintained for a period of at least one year),
2. Principal maintenance activities,
3. Reportable occurrences,
4. Surveillance activities required by the Technical Specifications,
5. Reactor facility radiation and contamination surveys where required by applicable regulations,
6. Experiments performed with the reactor,
7. Fuel inventories, receipts, and shipments,
8. Approved changes in operating procedures,
9. Records of meeting and audit reports of the review and audit group.

TS 6.8.2 Records to be Retained for at Least One Cycle

Operator qualification and re-qualification records.

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

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

1. Gaseous and liquid radioactive effluents released to the environs,
2. Offsite environmental monitoring surveys required by technical specifications,
3. Radiation exposures for all personnel monitored,
4. Updated, corrected, and as-built drawings of the facility.

TS 7 REFERENCES

1. GA-9064, pages 3-1 to 3-23
2. GA-9064, pages 3-1 to 3-23
3. NE Memo No. 4, Report on Refueling the University of Wisconsin Nuclear Reactor, R. J. Cashwell, March 1968, University of Wisconsin Department of Nuclear Engineering
4. Core Test Program, UWNR Mixed TRIGA-FLIP Core (9 FLIP), R. J. Cashwell, July 1974, University of Wisconsin Department of Nuclear Engineering
5. Core Test Propgram, UWNR Mixed TRIGA-FLIP Core (15 FLIP), R. J. Cashwell, February 1978, University of Wisconsin Department of Nuclear Engineering
6. Core Test Program, All FLIP Core, R. J. Cashwell, January 1980, University of Wisconsin Department of Nuclear Engineering
7. "Sagitta" refers to the bow of the element and means the maximum excursion of the clad surface from a chord connecting the two ends of the clad surface.
8. Simnad and West, 1986
9. NUREG-1282

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15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Construct a Non-Power Reactor

Not applicable for renewal application.

15.2 Financial Ability to Operate a Non-Power Reactor

The Reactor Laboratory is a part of the Engineering Physics Department of the College of Engineering at the University of Wisconsin-Madison. The teaching mission of the laboratory takes precedence over the research and service missions; for this reason the primary fiscal support for operation of the facility is from the state-funded university budget. Reactor personnel have instructional duties, such as teaching courses, setting up laboratory experiments, and assuring that equipment used in the teaching laboratories is operable. It thus becomes somewhat difficult to allocate funding precisely to just the operation of the reactor. In the following information no attempt is made to separate the instructional component of the budget from the reactor operations part of the budget. For instance, the Reactor Director is teaching two courses during the Spring 2000 semester, but his entire salary is included in the budget below.

Recently the operating budget of the Reactor Laboratory has been considerably higher than what would be predicted from the past history of the facility. This is due in large part to providing replacement of long-term employees who are approaching retirement age with younger workers in time to allow adequate training of the new employees. The total salary expenditures will be reduced significantly after the retirements actually take place, but the estimate below uses the current funding level.

The total operating budget of the laboratory is $388,000. Of this total, state instructional funding covers $299,000, with the remaining expenses split between grants ($48,000) and income generated by reactor services ($41,000).

By expense category, the funds are spent for salary($253,000), fringe benefits ($77,300), supplies and expense ($24,000), and capital equipment ($33,000).

These numbers do not include the infrastructure provided by the University such as electrical power, heating, janitor service, and health physics coverage. Further, the fringe benefits included above are not specifically billed to the department in which the employee works for instructional funding, but come from a campus-wide fund, while the fringe benefits for salaries supported by non-instructional funding are charged to the fund paying the salary.

The instructional funding is appropriated by the state. The administration of the university has been very supportive of the reactor facility and continuation of the Nuclear Engineering curriculum. The fact that the application for renewal of the license is signed by the administration indicates this support.

Much of the capital equipment funding in recent years has come from the DOE program to update the instrumentation and experimental equipment on non-power reactors. In addition, a local utility company has provided matching grants to support instruction in traditional "fission nuclear engineering", and this has contributed to both the supply and expense and capital equipment budget. The combination of funding opportunities has resulted in the reactor and associated laboratories being in excellent condition. The reactor can continue to operate without the outside grant income, but it would not allow for upgrading the equipment as has been done for the last 10 years. However, the present instrumentation and control systems are capable of continuing to operate for another 20 years with no loss of function.

Funds from services provided to persons outside the university have been steady for the last 10 years, with some increases in the last three years. It is expected that this funding will continue for at least the next five years.

15.3 Financial Ability to Decommission the Facility

By letter dated July 19, 1990[1], the University responded to 10 CFR Part 50.75 showing the expected cost of decommissioning the reactor. At that time estimated decommissioning cost was $1,200,000 in the year 2000.

Early in 1999 the computations on which the funding plan was made were updated to extend the time of decommissioning to 2020 and to incorporate more recent figures on the cost of disposal of radioactive debris from the decommissioning. The estimate is again based on placing the facility in condition for unrestricted release three years after cessation of operations, now estimated as June 30, 2020. The result of this revised estimate is a decommissioning cost of between $3,000,000 and $8,000,000, with the wide range of cost based primarily upon the uncertainty in disposal cost. Whereas the disposal cost was a minor part of the total decommissioning effort costs for the original computation, it is now by far the largest component of the total cost. Nevertheless, as a state agency, the funding plan remains to obtain the funding when necessary.

15.4 Reference

1. Letter to USNRC Document Control Desk from R. J. Cashwell under Docket 50.156 dated July 19,1990, with attachment signed on behalf of the Board of Regents.

16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

There are no components in use at the University of Wisconsin Nuclear Reactor Laboratory that have had prior use at any other facility or organization. It is conceivable that prior use components could be integrated into Reactor Laboratory systems at some future time. Appropriate analysis and reviews of component replacement will be conducted in accordance to applicable standards, regulations and facility procedures and licensed technical specifications.

16.2 Medical Use of Non-Power Reactors

The University of Wisconsin Nuclear Reactor Laboratory is not engaged nor licensed to conduct any activities for medical use of the facility. Future medical use of the Reactor Laboratory would be conducted pursuant to appropriate license applications and approvals as authorized by the Atomic Energy Act of 1954 as amended.

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Appendix A Calculation Methods for Atmospheric Release of Radioactivity

A. Models Used for Calculations in Sections 11.1.1.1.2 and 13.1.1.5

For Sutton's diffusion model, the maximum concentration (X_{max}) at any point downwind is given as:

$$(1)\ X_{max} = \frac{2Q}{e\pi\bar{\mu}h^2} \qquad \text{(Reference 1)}$$

where $\bar{\mu}$ is mean wind speed in meters/second
Q is release rate in Ci/second
h is stack height in meters

For the generalized Gaussian Plume Model, the maximum concentration is given by the same equation (Reference 2, equation 8).

For calculations in this report, the following values are used:
$\bar{\mu}$ = lowest monthly average = 3.54 meters/second
h = stack height above ground = 26.5 meters.
Using these numbers, equation (1) reduces to

$$(2)\ X_{max} = 9.42\text{E-}5\ Q,$$

where X_{max} is in μCi/ml

Reference 2 presents a method applicable to release from buildings with zero stack height to approximate release from leaks in a containment structure. The relation given, as equation 4, is:

$$(3)\ X = \frac{Q}{(\pi\sigma_y\sigma_z + CA)\mu},$$

where X, Q, and μ are defined as above,
C is an empirical constant with a value between 0.5 to 2, and
A is the minimum building cross section.
The σ terms are concerned with atmospheric dispersion, which will be neglected in this analysis, which will result in the equation;

$$(4)\ X = \frac{Q}{CA\mu}$$

Inserting values for the UWNR facility used in the safety analysis for FLIP fuel conversion, and using a value of 1 for C yields:

$$(5)\ X = \frac{Q}{(1)(12{,}200 ft^2)(9.29E{-}2 m^2/ft^2)(3.54 m/\mathrm{sec})}, \text{ or}$$

(6) X=2.49E-4 Q with X in units of μCi/ml

B. Sample Calculations Supporting Section 11.1.1.1.2

The maximum release rate for Ar-41 activity is 13.3 μCi/second. Using the ventilation system rated flow-rate of 9600 scfm, this activity is diluted to 2.94E-6 μCi/ml at the stack outlet. The resulting maximum concentration downwind, assuming the stack height, is calculated to be, from equation (2)

(7) X_{max} = (13.3E-6)(9.42E-5) = 1.25E-9 μCi/ml.

If calculated using equation (6) (which assumes zero stack height with building wake dilution), the resulting value is

(8) X= (13.3E-6)(2.49E-4) = 3.31E-9 μCi/ml

It is obvious that the two methods used to calculate the above values cannot both be applicable. Since the reactor is not operated when the ventilation system is not in operation, the value in equation (7) is more realistic, but the more conservative value in equation (8) is used in the text.

C. Calculations Supporting Section 13.1.1.4 (1), Whole Body Exposure

The activity concentration of the insoluble volatiles in the reactor room air was determined by dividing released activity by room volume.

$$(9)\ \frac{A}{V} = \frac{5.89E6\ \mu Ci}{2.00E9\ cm^3} = 2.95E{-}3\ \mu Ci/cm^3$$

Since 3.7E4 dps = 1 μCi, A/V= 109 γ/sec-cm^3

The maximum dose rate is calculated by assuming the room is equivalent to a hemisphere with a radius of 782 cm. In addition, the average gamma energy is 0.7 MeV, the attenuation coefficient for air is 3.5E-5 cm^{-1}, and the flux-to-dose conversion factor is 4.2E4 γ/cm^2/cm^2-mr/min

Using the relationship

$$(10)\ DR = \frac{30S(1 - \exp(-R\Sigma))}{C\Sigma} \quad \text{where}$$

DR = dose rate in mr/hr
S = Volumetric source strength in γ/sec-cm^2
R = outer radius of hemisphere
Σ = attenuation coefficient for air,

yields a dose rate of 60 mr/hour.

D. Calculations Supporting Section 13.1.1.4 (2), Dose to the Lungs

The dose to the lungs was calculated by first assuming uniform dispersal of the released volatiles in the laboratory volume, giving a concentration of

$$(11)\ \frac{A}{V} = \frac{5.54E6\ \mu Ci}{2.00E9\ cm^3} = 2.75E-3\ \mu Ci/cm^3.$$

Since the "standard man" breathes 1.25 cubic meters of air per active hour, he would breathe approximately 0.21 cubic meters in 10 minutes, the assumed evacuation time. If this number is increased to 0.30 cubic meters to allow for excitement and stress, then his lungs would be exposed to an activity of

$$(12)\ \frac{A}{V}\ V = (.75E-3\ \mu C)\ (3E5\ cm^3) = 825\ \mu Ci.$$

The dose to the lungs is then calculated from the following expression to be 1 Rad.

$$(13)\ Dose\ (Rad) = \frac{ACR}{m} \sum_{I=1}^{8} \frac{F_i\ E_i}{\lambda_i}(1 - e^{\lambda_i t}),\ \text{where}$$

A = Activity exposure (825 μCi)
C = Conversion factors

$$\frac{(3.7E4\ \beta/\text{sec}-\mu Ci)(1.6E-6\ erg/MeV)}{100 erg/gm-rad}$$

R = lung retention factor (0.125 is customary)
m = mass of lungs (1000 grams)
F_i = fraction of total activity
E_i = energy of beta for nuclide i (MeV)
λ_i = radioactive decay constant + biological release constant (6.7E-8 sec^{-1})
t = time of exposure (assumed infinite)

E. Calculations Supporting Section 13.1.1.5 and Table 13.1

Release rate, Q, for an isotope is the total quantity released to air (column E of Table 13.1, Chapter 13) divided by the assumed release time. The release time used in further calculations is the time for the ventilation system (room air and beam port and thermal column exhaust systems) to make a complete change of air in the Reactor Laboratory.

$$(14)\ T_{release} = \frac{2000\ m^3}{2700\ scfm}\ \frac{35.31\ ft^3}{1\ m^3}\ \frac{60\ \sec}{1\ \min} = 1569\ seconds$$

Using the generalized Gaussian Plume Model (equation (2)), and demonstrating with data for Br-83, concentrations released to unrestricted areas (Chapter 13, Table 1 Column H) are calculated as shown below:

$$(15)\ X_{Br-83} = \frac{(0.0083Ci)(9.42E-5)}{1569} = 4.98E-10\ \mu Ci/ml$$

The remaining isotope values are calculated in the same manner.

The activity release was also evaluated through use of equation (6). This calculation would be applicable to release of the activity through the building walls with the ventilation system not operating.

Again using Br-83 as an example, and assuming the same release time as in the previous calculation

$$(16)\ X_{Br-83} = \frac{(0.0083Ci)(2.49E-4)}{1569} = 1.32E-9\ \mu Ci/ml$$

This value is a factor of 2.6 greater than that evaluated by the Gaussian Plume Model. All similar values in Chapter 13, Table 1, Columns H and I may be multiplied by this factor for a more conservative case. This calculation was done for the previous Safety Analysis Report using as the building dimensions only the minimum dimensions of the reactor laboratory, a room within the Mechanical Engineering Building. This considerably underestimates the "wake effect" of the actual building. The current analysis uses a value appropriate for the renovated Mechanical Engineering Building (cross-sectional area of 12,200 ft^2).

References

1 Meteorology and Atomic Energy, U. S. Dept of Commerce Weather Bureau, Govt. Printing Office, Washington, DC July 1955

3. F. A. Gifford, Jr, Atmospheric Dispersion Calculations Using the Generalized Gaussian Plume Model, Nuclear Safety, December 1960

4. Calculation of Distance Factors for Power and Test Reactors, (TID-14844), USAEC, March 23, 1962

Appendix B Supporting Documents

Task Order No. 2 Under Master Task Agreement No. C96-175937

Page 1

LMITCO FORM
PROC-1811b
07/99

TASK ORDER NO. 2 UNDER MASTER TASK AGREEMENT NO. C96-175937
LOCKHEED MARTIN IDAHO TECHNOLOGIES COMPANY (LMITCO)
2525 Fremont Avenue
P. O. Box 1625, Idaho Falls, ID 83415-3521
OPERATING UNDER U. S. GOVERNMENT CONTRACT NO. DE-AC07-94ID13223

To: University of Wisconsin-Madison
Research Administration
750 University Avenue
Madison, WI 53706-1490

Effective Date: August 26, 1999
Completion Date: November 1, 2001

To: Tom Handland
PI: R. J. Cashwell

This Task Order No. 2 is awarded to:

1. Transfer Reactor Fuel Assistance Subcontract No. C87-101251-002 to the new Master Task Agreement No. C96-175937 as Task Order No. 2.
2. Extend the period of performance to November 1, 2001. This extension is retroactive to November 1, 1998.
3. Confirm the Statement of Work and modification thereto remain unchanged.
4. Assignment: On September 30, 1999, LMITCO's prime contract with DOE will expire. Thus, pursuant to the article in the General Provisions entitled "assignment"" this Task Order is assigned to Bechtel BWXT Idaho, LLC, under its DOE Prime Contract No. DE-AC07-ID13727, effective October 1, 1999.

Procurement Agent: Lynda Keller	Telephone: (208) 526-5597	Cost: $0.00
Ship via: N/A	F.O.B./Trans.: N/A	Cash Terms: Net 0 Days

Billing Address:
Lynda Keller
LMITCO
P. O. Box 1625
Idaho Falls, ID 83415-3521

Signed: Lynda T. Keller 9/14/99
Lockheed Martin Idaho Technologies Company Date

Title: Procurement Agent

Signed: [signature] 10/10/99
Subcontractor's Official Date

Title: William J. Vance, Assistant Dean
Research & Sponsored Programs

Return one signed copy of this Task Order to Lynda Keller.

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