Washington State University
Nuclear Radiation Center

50-27

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Pullman, WA 99164-1300
509-335-8641
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June 24, 2002

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

RE: Docket No. 5027
Facility License No. R-76

Gentlemen:

In accordance with section 50.4 and 50.21 of 10 CFR 50, application is hereby submitted for a 20 year license extension for Facility License R-76 for the Washington State University Modified TRIGA Nuclear Reactor. The documentation specified by Sections 50.4, 50.30, 50.33 and 50.34 of 10 CFR 50 are attached.

Sincerely,

G E Tripard

Gerald E. Tripard
Director

Gloria VanAusdle
(Approved and Notarized)

GLORIA VAN AUSDLE
COMMISSION EXPIRES
NOTARY
PUBLIC
12-1-04
STATE OF WASHINGTON

Dwight Hagihara
Dwight Hagihara
Chairman, Reactor Safeguards Committee

George A. Hedge
Vice Provost for Research

A020

APPLICATION FOR TWENTY YEAR LICENSE EXTENSION
FOR FACILITY LICENSE NO. R-76
FOR THE WASHINGTON STATE UNIVERSITY
MODIFIED TRIGA NUCLEAR REACTOR

1. GENERAL INFORMATION (10 CFR 50.33)

(a) Name of Applicant

Washington State University
Nuclear Radiation Center

(b) Address of Applicant

Washington State University
Pullman, WA 99164-1300

(c) Business of Applicant and Officers

(1) Business - Educational Institution (Land Grant College)

(2) Officers

a. University

i. President of WSU - V. Lane Rawlins

ii. Academic Vice President and Provost - Robert C. Bates

iii. Vice Provost for Research - George A. Hedge

b Nuclear Radiation Center

i. Director - Gerald E. Tripard

ii. Reactor Supervisor - Stephanie L. Sharp

(d) The applicant is an Educational Institution which is a Land Grant University in the State of Washington under the control of the Laws of the State of Washington.

(e) Class of License

Class 104 Production and Utilization Facility, Facility License No. R-76

(f) Financial Considerations

The cost of operating the WSU TRIGA reactor facility and attendant research projects during the current year is $180,000. The funds come from Program 10D of the university budget entitled "Other Organized Research." Since all funding for WSU is by action of the State Legislature, it is not possible to guarantee funding for any program within the university. However, the State of Washington is an Agreement-State and has in the past chosen to comply with all Federal regulations and commitments along with the costs thereof. It is thus deemed that it would be incumbent upon the State to continue to provide the necessary funding for operation of the facility.

2. WRITTEN COMMUNICATIONS TO NRC (10 CFR 50.4)

(a) Address - U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

(b) One copy to Regional Office:
U.S. Nuclear Regulatory Commission, Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

3. FILING OF APPLICATION (10 CFR 50.30)

(a) One notarized signed copy of the letter of application of the renewal of Facility License R-76 are herewith submitted in accordance with Paragraph (b) of 10 CFR 50.30. One copy will be sent to Regional Office IV.

(b) Ten copies of the application information constituting this document including the information required by 50.33 are hereby retained for distribution.

(c) Thirty copies of the new SAR of 2002 are hereby retained for distribution in accordance with the requirements of 10 CFR 30.30(a).

(d) The applicant hereby claims to be exempt from the Filing Fees specified by 50.30 (3) under the provisions of 1760.11 (1.4).

(e) Ten copies of the applicant's "Environmental Impact Appraisal" is hereby retained for distribution to fulfill the requirements of 50.30 (f).

4. TECHNICAL INFORMATION (10 CFR 50.34)

(a) A new Safety Analysis Report of 2002 in the format specified by NUREG-1537 is provided with the application.

(b) The Emergency Plan of September, 1963 as amended is still valid for the facility and a new plan need not be submitted.

(c) The proposed new Technical Specifications for the facility are given in section 14 of the SAR.

(d) Requalification Program (10 CFR 55)

The current existing "Operator Requalification Program for the Washington State University TRIGA Facility" of March 22, 1989 meets the current requirements and thus a new program need not be submitted.

(e) Physical Security Plan (10 CRF 50.34[c])

The current existing "Physical Security Plan for the Washington State University TRIGA Facility" approved September 12, 1984 meets the current standards and is till valid for the facility. Thus, a new plan need not be submitted.

(f) SNM Information (10 CFR 73.47)

The SNM requirements for the facility in the existing license as listed in Table I below are quite adequate and need not be changed.

TABLE I
SNM REQUIREMENTS FOR WSU TRIGA REACTOR FACILITY

Maximum U-235	Maximum Pu	% Enrichment	Exempt Status*
[redacted]		<20	Exempt 10 CFR 73.6(a)
[redacted]		>20	Exempt 10 CFR 73.6(b)
[redacted]		>20	Not Exempt
	[redacted]		Exempt 10 CFR 73.6(c)

*Material is exempt provided that it meets the requirements for exemption pursuant to the cited provisions of 10 CFR 73.

ENVIRONMENTAL IMPACT APPRAISAL FOR THE CONTINUED OPERATION OF THE WASHINGTON STATE UNIVERSITY MODIFIED TRIGA REACTOR

Submitted to:

U.S. Nuclear Regulatory Commission

WASHINGTON STATE UNIVERSITY
NUCLEAR RADIATION CENTER
PULLMAN, WA 99164

June, 2002

1.0 GENERAL

This Environmental Impact Appraisal for the continued operation of the Washington State University Modified TRIGA Reactor is submitted to enable the Commission to support and develop the EIA for the renewal of Facility License R-76. On January 23, 1974 the AEC staff concluded in the 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." Thus no formal EIA is required for the extension of the operating license of Facility R-76 for the WSU TRIGA 1 MWt Research Reactor.

2.0 LOCATION OF FACILITY

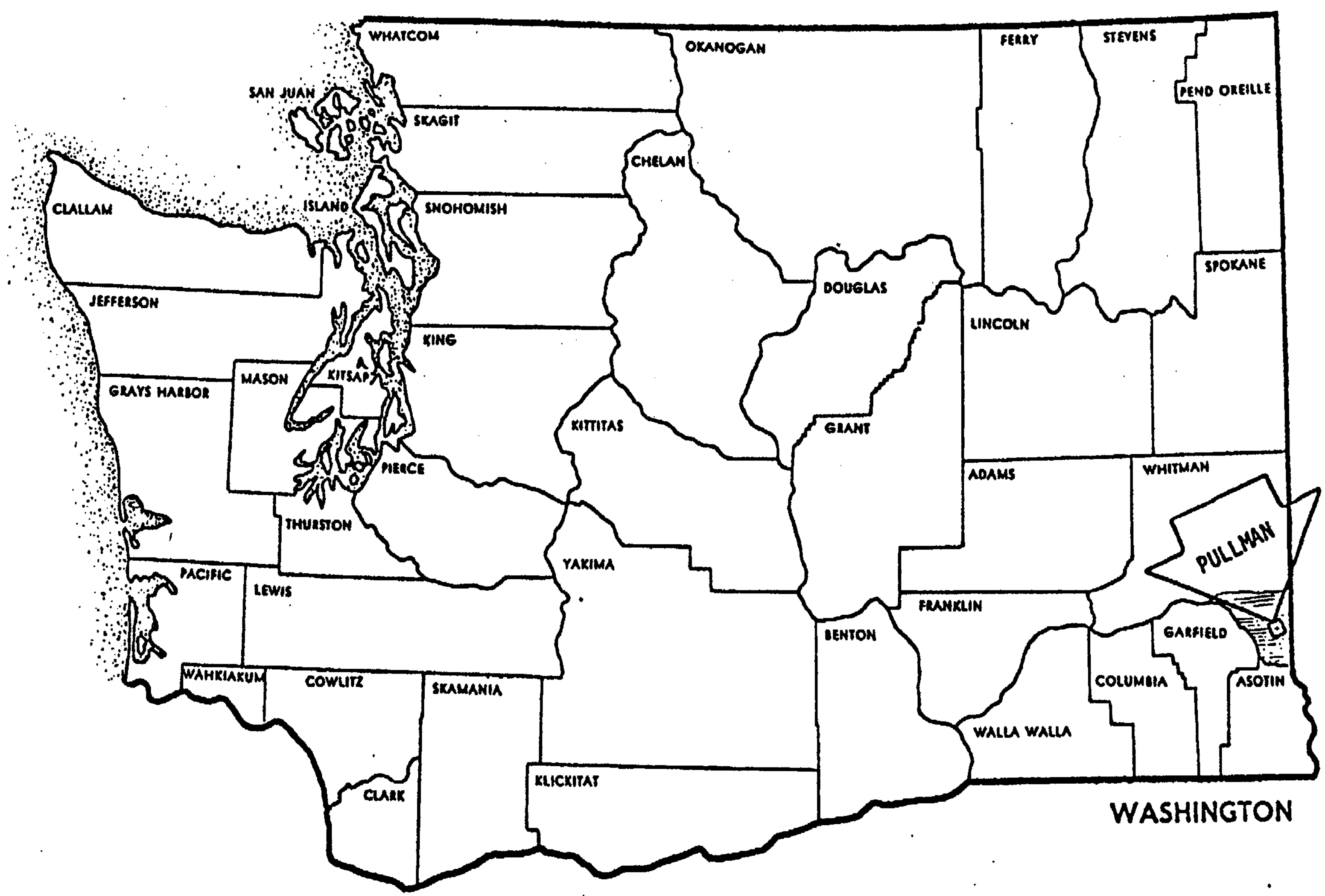
The WSU TRIGA reactor is located in the Nuclear Radiation Center on the campus of Washington State University in Pullman, Washington. Pullman is a small town in the southeast corner of the State of Washington as shown in Figure 1 and has a total population, including the university of 23,500. The Palouse region surrounding the town is a rural agricultural area devoted to dry land farming.

The actual reactor site is east of Pullman and east of the main portion of the WSU campus as shown in Figure 2. The site is surrounded by university property used for grazing livestock as shown in the site photograph of Figure 3, and the closest occupied dwelling is 411 meters west of the facility. Additional details on the site are given in the facility SAR of June, 2002.

3.0 PHYSICAL CHARACTERISTICS OF THE FACILITY

The WSU Reactor is a modified TRIGA reactor and operates with a core of mixed Standard and FLIP fuels. The reactor was originally designed to use MTR plate-type fuel but was converted to TRIGA fuel in 1967 by replacing the MTR fuel elements with 4-rod clusters of TRIGA fuel. The reactor is housed in the WSU Nuclear Radiation Center which is a 1200 square meter laboratory devoted to nuclear related research and educational activities. The core of the reactor is situated in a 242,000 liter water pool which functions as shield, moderator, and coolant.

The WSU modified TRIGA reactor, like all TRIGA type reactors, has very large prompt negative temperature coefficient, thus making the reactor inherently very safe. The kinetic behavior of TRIGA reactors permits them to be safely pulsed to very high power levels for a short duration. The pulse is automatically terminated by the effects of the large negative temperature coefficient. The WSU reactor operates at maximum continuous steady state power level of 1 MWt and may be pulsed with a $2.50 insertion. The peak power during a pulse is on the order of 2000 MW.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WASHINGTON

Figure 1

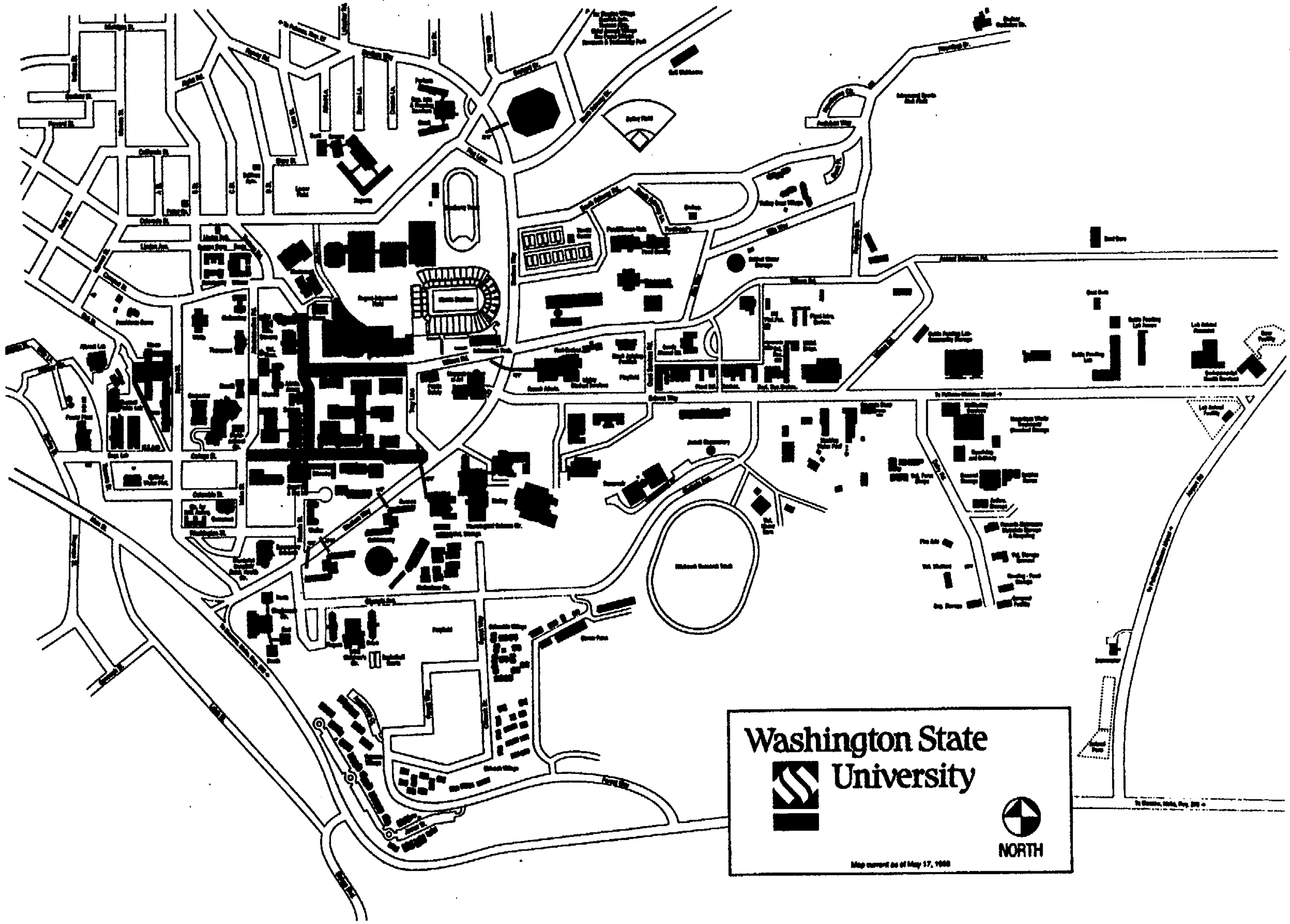

Figure 2

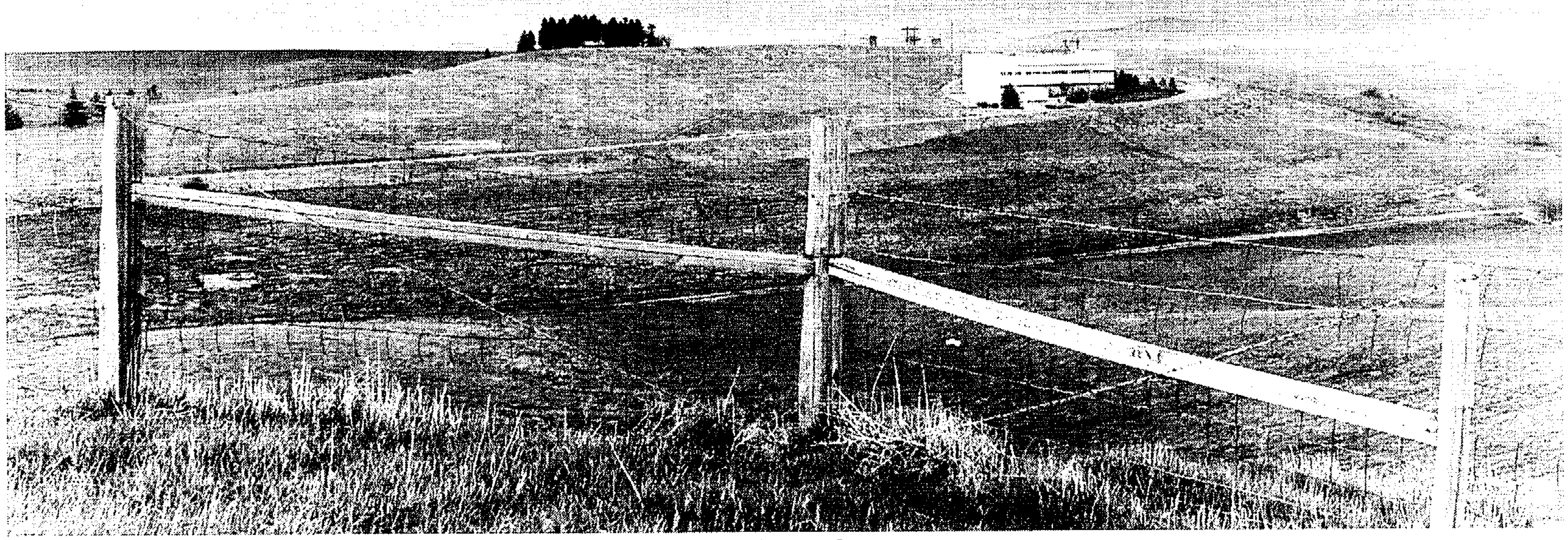

Figure 3

4.0 ENVIRONMENT IN THE AREA

The reactor site lies approximately 3.2 kilometers east of the center of the town of Pullman and 1.6 kilometers east of the center of the WSU campus. The land surrounding the site for at least 400 meters in all directions is uninhabited grass land owned by the university and used for the grazing of livestock. Geologically the site is located at an elevation of 808 meters on the south slope of a typical Palouse formation hill.

Pullman is situated near the eastern margin of the Columbia Plateau and the associated lava flows. The site is thus underlaid with basaltic rock produced by horizontal lava flows. The bedrock was capped with silt and clay deposited during the Pleistocene Age to form the present topsoil of the Palouse Loess with their characteristics rolling hill topography. The Palouse formation (topsoil) at the reactor site is approximately 30 meters thick.

Pullman is located approximately 480 kilometers inland from the Pacific Ocean. The Cascade Mountains, which average more than two kilometers in height, separate the region from the coast. The combined effect of the distance from the ocean and the extensive mountain barrier produces a climate that is continental in character. However, because the prevailing winds blow inland from the Pacific Ocean, winters are somewhat warmer than might be expected 480 kilometers inland at a latitude of 47° north. Winters in Pullman are characterized by cloudy skies and frequent snowstorms. On the average, the sun shines only about 30% of the time during the winter months.

During the summer months, the westerly winds weaken, and continental climatic conditions prevail. This causes rainfall, cloud cover, and relative humidity to be at their minimum; the daily mean temperature and daily temperature variation are at their maximum. Summers in Pullman are characterized by warm clear days and cool nights. On the average, the sun shines in Pullman about 80% of the time during the summer months.

One of the characteristics of the Palouse region is that of being a rather windy area. The average annual wind velocity is of the order of 16 km/hr. For the most part, winds peak in January averaging about 21 km/hr and the low occurs in July averaging about 11 km/hr. The wind velocity is greater than 5 km/hr 94% of the time and greater than 8 km/hr 76% of the time. The wind is from a westerly direction of the order of 60% of the time and an easterly direction 30% of the time.

The annual precipitation in the Pullman area is 50 centimeters and the annual average temperature is 8.7°C. The highest precipitation month is January with 6.8 centimeters and the lowest is July with 1 centimeter. The daily mean temperature peaks in July at 20°C. The mean daily minimum-to-maximum for the two extremes is 15.7°C and 5.7°C respectively.

There are no unique environmental or natural characteristics of the reactor site or archaeological or historical sites located within close proximity of the reactor site. The site is in a very low population density region and east of the main population concentrations of both the town of Pullman and the WSU campus. The population centers are also upwind of the site over 60% of the time.

5.0 ENVIRONMENTAL EFFECTS OF CONSTRUCTION

No modifications to the Facility or the site will be required for the continued operation of the WSU reactor. There are no exterior conduits, pipelines, electrical or mechanical structures or transmission lines attached to the reactor facility other than utilities services which are required for other structures and laboratories on campus. Thus there will be no significant effects upon the terrain, vegetation, wildlife, nearby waters, or aquatic life due to construction-type activities.

6.0 ENVIRONMENTAL EFFECTS OF FACILITY OPERATION

(a) Water Use Consumption

Make-up water for both the reactor pool and wet cooling tower are required for operation of the reactor. The WSU campus has its own water system with water derived from wells independent of the Pullman water system. Pool make-up amounts to 4,500 liters per month on the average and the cooling tower requires approximately 180,000 liters per month in summer. The total water consumption of the reactor cooling system is approximately 185,000 liters per month.

(b) Heat Dissipation

The WSU TRIGA reactor has a maximum steady state power output of 1 MWt. The 1 MWt of heat generated by the reactor is dissipated by an evaporative mechanical draft cooling tower located on the north side of the facility. The evaporative cooling system cools the reactor pool and dissipates the heat generated by the reactor to the atmosphere through the latent heat of vaporization of water. On the average, the cooling tower consumes 1.2×10^5 liters per month of water that is added to the local atmosphere by the operation of the facility. In other words, an average of 4000 liters of water per day are added to the atmosphere at the site.

Evaporative cooling towers have the potential for creating visible plumes of water vapor under certain atmospheric conditions. The plume is a region of air with a higher temperature and higher water content than the ambient air. The climatic and atmospheric conditions at the site and the small amount of water involved preclude the development of a plume by the WSU reactor cooling tower during the summer months. However, during the winter months a very small plume is sometimes produced that rises of the order of 30 meters into the air above the cooling tower. Fogging and icing conditions at the site are not affected by the operation of the cooling tower. The amount of water added to the local atmosphere annually by the cooling tower is really insignificant compared to the 50 centimeters annual precipitation in Pullman. Thus the water added to the atmosphere by the operation of the facility will have a minimal effect on the environment.

(c) Chemical Discharges (non-radioactive)

No chemical discharges are generated directly from the operation of the reactor. The chemical discharges into the sanitary waste system at the Nuclear Radiation Center are related to conventional chemical laboratory operations at the site and are not different than those of other laboratories on campus.

The blow-down of the cooling tower also discharges into the sanitary sewer system. The blow-down discharge amounts to 9300 liters per month on the average which contains an increased amount of total dissolved solids (TDS) than the input potable water. The concentration factor will be less than 10 and thus the increased TDS is not significant.

The cooling tower and associated heat exchanger, like all boilers and other water cooling systems on campus, are maintained by the WSU water treatment group. The standard campus water treatment involves the use of MOGUL WS164 water treatment liquid at the rate of 40 ppm plus 22 ppm of algicide. The incremental increase in the discharge of treated water by the operation of the reactor is, however, insignificant compared to the total campus discharge of such water into the sanitary sewage system. Thus the environmental effects related to chemical discharges created by the operation of the reactor are not significant.

(d) Radioactive Discharges

(1) Gaseous

The ventilation system of the reactor discharges 2.12 m^3/sec of air from the pool room into the atmosphere. The principal radionuclide contained in the discharge air is Argon-41 which is produced by the activation of argon contained in air. The Argon-41 content of reactor pool room exhaust is continuously monitored with a special gamma-ray spectrometer set to detect Argon-41. Over the past 5 years the total average quantity of Argon-41 discharged from the facility amounted to 20.7% of the Technical Specification limit. On a concentration basis, taking into account the dilution of the atmospheric wake effect in the lee of the building, the 5 year average release concentration of Argon-41 was 2.1×10^{-10} μCi/cm^3. The release concentration for Argon-41 given by the EPA for reactor facilities in 40 CFR 61, subpart I is 1.7×10^{-9} μCi/cm^3 which is 58 times lower than the new 10 CFR 20 Appendix B, Table II, Column 1 limit for Argon-41 of 1.0×10^{-8} μCi/cm^3. The actual release concentration over the past 5 years amounted to 2.1% of the 10 CFR 20 limit and 12% of the EPA limit. A small amount of tritium is produced in the pool water through neutron capture in the deuterium present in the pool water. Measurements of the ^3H level in the pool water of a number of TRIGA reactors including the WSU reactor are reported on Page 170 of the August, 1976 issue of Health Physics. Measurements made by the WSU Radiation Safety Office agree with the reported value for the WSU reactor of .045 μCi/l. The pool evaporation rate amounts to 560 liters per day and the pool room exhaust discharge is 1.834×10^{11} cm^3 per day. If we make the conservative assumption that the ^3H content of the pool water and evaporated water are the same, then the pool room exhaust would contain 1.37×10^{-10} μCi/cm^3 of tritium. This is significantly below the applicable limit in 10 CFR 20 of 1×10^{-7} μCi/cm^3 and the EPA limit of 1.5×10^{-9}. No other significant quantity of gaseous radioactive material or particulate radioactive material with a half-life greater than eight days has been released by the facility during the past 20 years.

In the event of a Loss of Coolant Accident or the Design Basis Accident, the 2000 review analysis of this postulated accident has shown the gaseous radioactive discharges to be minimal. The worst case whole body dose from a cloud of fission products discharged from the facility as a result of the DBA is only 1.28 mrem/hr. The worst case maximum thyroid dose outside the facility for a 3% halogen release was found to be 17.4 mrem/hr. Thus no realistic hazard to the general public would result from the DBA or a LOCA.

(2) Liquids

No radioactive liquids are generated by the operation of the reactor in and by itself. However, the nuclear research and educational activities at the Nuclear Radiation Center generate radioactive liquids from radiochemistry experiments and from activation analysis activities. All hot drains from the laboratory flow into a holdup tank system which is monitored and diluted as necessary before being discharged into the sanitary sewer. Over the past 3 years the radioactive liquid released from the holdup tanks, on the average, contained 4×10^{-8} $\mu Ci/cm^3$ or about 10% of the applicable release limit and amounts to about .5 μCi/month.

Radiation Safety at WSU has, for over 15 years, monitored the radiochemistry level in the waters in the vicinity of WSU including the South Fork of the Palouse River, local tap water, and sewage treatment plant effluent. An increase in the activity levels attributable to the operation of the WSU TRIGA reactor has never been detected.

(3) Solids

The only solid radioactive waste generated directly by the operation of the reactor is spent ion exchange resin. Approximately .3 cubic meters of spent resin is disposed of each year. It is estimated that the long-lived components of the activity in the spent resin amounts to about .1 Ci/yr.

The entire WSU campus generates of the order of 8 cubic meters of solid radioactive waste annually containing approximately .5 curies of activity. This solid waste is predominantly generated by research activities in university laboratories other than the Nuclear Radiation Center utilizing long-lived purchased radionuclides. Thus the incremental increase in solid wastes generated by the operation of the reactor is minimal. All solid wastes are transferred to the Nuclear Engineering Company of Richland, Washington for disposal.

(e) Radiation Levels

An extensive Environmental Radiation Monitoring Program was instituted at the WSU Nuclear Radiation Center in July of 1974. The program involves measuring the integrated radiation exposure for a period of three months at 40 points at the site and associated environs. Commercially available thermoluminescent dosimeters (TLD's) of the $CaSO_4$:Dy type provided and processed by the Radiation Detection Company, Sunnyvale, California are utilized.

Table I lists the average exposure rate above ambient background per megawatt hour of reactor operation for a number of locations at the site. The two highest exposure points are on the roof directly above the pool and at the freight door to the pool room.

The maximum possible on-site exposure at a readily accessible location would be to an individual standing at the pool room freight door for the 1000 hours per year that the reactor operates. The total maximum annual exposure at this on-site point would be 87 mrem/year.

The exposure rates at points from 50 meters to 24 kilometers from the Nuclear Radiation Center have also been monitored quarterly since 1974. The average exposure rate at the 24 locations involved is 188 ± 30 μR per day. No statistically significant variations in the above background exposure rates at the sample locations have been observed or any exposure attributable to the operation of the WSU reactor . In addition, the average exposure rates at these locations which are 50 meters from the site are not statistically different on a quarterly basis than the average of the background exposure rates at 17 locations in the State of Washington monitored by the State of Washington Department of Emergency Services. Thus no significant effect on the radiation levels in the environment surrounding the facility has been observed to date.

7.0 ALTERNATIVES TO CONTINUED OPERATION OF THE FACILITY

There are no suitable or more economical alternatives which can accomplish both the educational and research objectives of the facility. These objectives include but are not limited to: the training of students in the operation of nuclear reactors; the training of students in the use of radioisotopic tracer techniques; the production of radioisotopes for use in numerous areas of the physical, biological, and animal sciences; the training of students and research applications of trace element analysis by neutron activation analysis; and also a demonstration tool to familiarize the student body and general public with nuclear reactors and their operation.

In addition, the WSU Reactor Facility is in the process of establishing and licensing a Medical Therapy Facility for cancer treatment using the Boron Neutron Capture Therapy method. The BNCT method can be done only at a nuclear reactor facility and thus there is no alternative to this new, important cancer treatment methodology.

8.0 SHORT-TERM EFFECTS VERSUS LONG-TERM GAIN OF FACILITY OPERATION

One of the chief objectives of any institution of higher education is to increase the body of knowledge available to mankind and to impart that knowledge to individuals. Accordingly, it is very difficult to compare the long-term gains from the operation of a research reactor in relation to the short-term environmental effects. However, the total environmental effects of the WSU TRIGA reactor and associated Nuclear Radiation Center are not significantly different from other research laboratories at a typical university. For the most part, the cumulative long-term benefits of university research activities far outweigh the environmental effects of such activities. This would also be true for the continued operation of the WSU reactor.

9.0 COST BENEFIT ANALYSIS

The facilities at the Nuclear Radiation Center represent an investment of the order of $2.5 million dollars. If the facility were shut down, the benefits derived from this investment would drop to zero. On the other hand, continued operation would allow the continuation of 10 ongoing research programs and the completion of about 8 graduate thesis research projects per year. The benefits also include the educational objectives mentioned in Section 6.0 and the new BNCT cancer therapy project being undertaken. Considering the minimal environmental effects of the continued operation of the reactor as previously cited in this report, the environmental cost effects are very small compared to the benefits to be derived from continued operation.

TABLE I

Median Exposure Rates per Megawatt Hour of Reactor Operation
in Close Proximity to the Nuclear Radiation Center

Location	(Adjacent to Room)	Exposure (μR/MW-Hr)
Front Entrance	50V	32
Pool Room Freight Door	21	87
North Side of Building	201B	10
Roof above Control Room	201B	16
Roof above Pool	201	152
Roof above Laboratory Area	214	0
West Side Door at Beam Room	2X	14
Storage Building	217A	21
Lower Loading Dock	123A	17

[redacted]

Figure 4

12

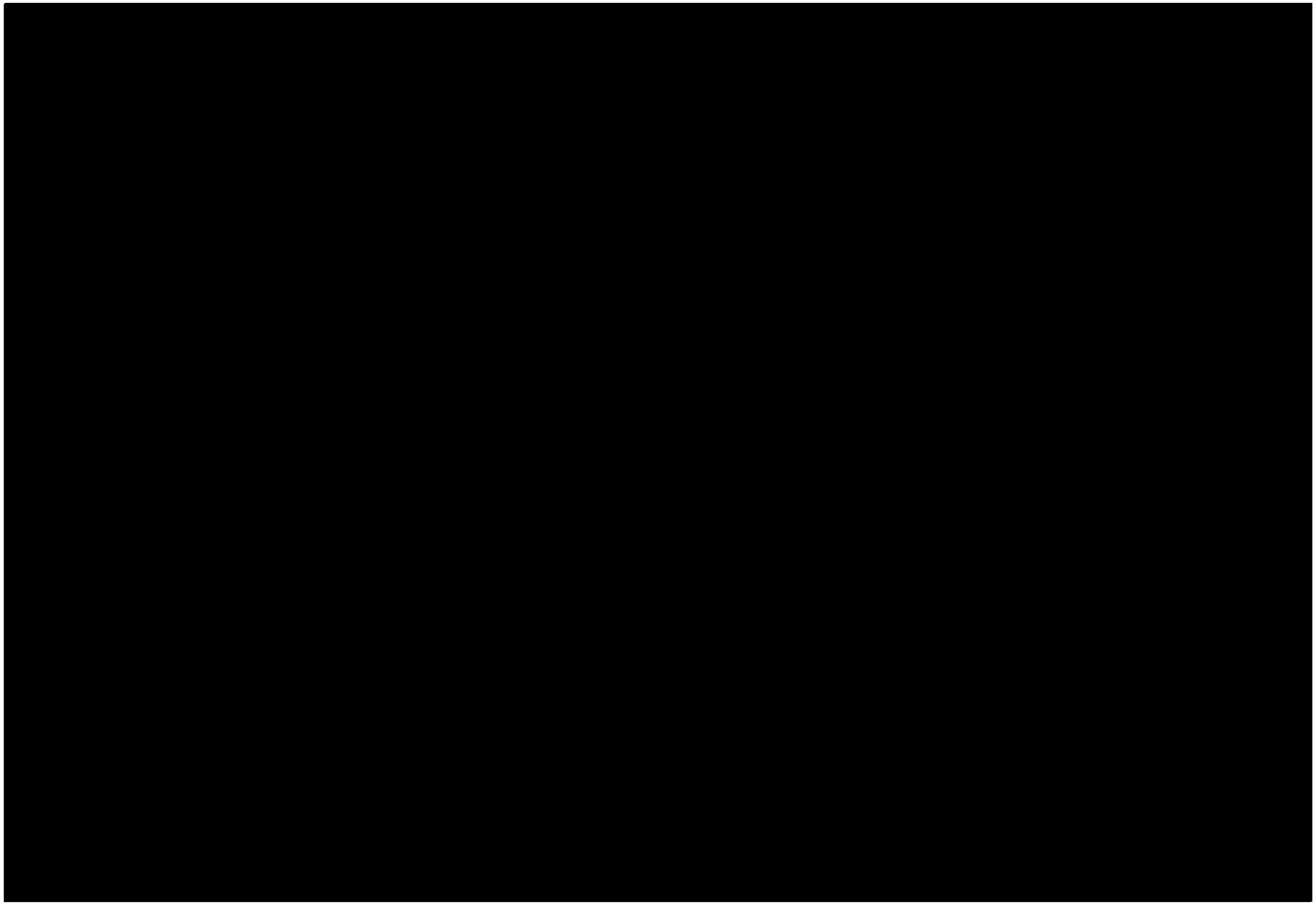

Figure 5

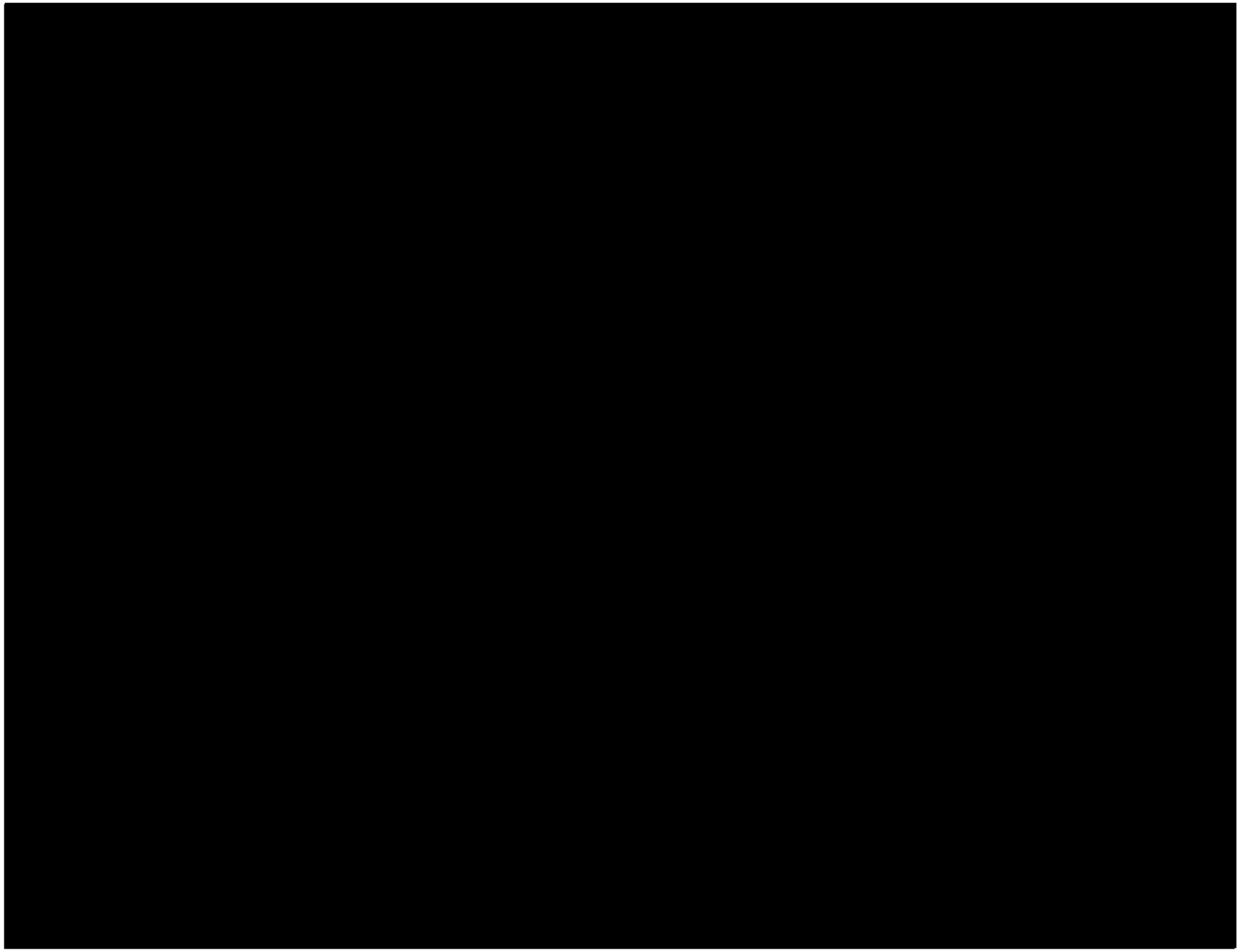

Figure 6

SAFETY ANALYSIS REPORT

for the

WASHINGTON STATE UNIVERSITY

MODIFIED TRIGA NUCLEAR REACTOR

Submitted To

U.S. NUCLEAR REGULATORY COMMISSION

FOR RENEWAL OF FACILITY LICENSE R-76

WASHINGTON STATE UNIVERSITY
NUCLEAR RADIATION CENTER
Pullman, Washington 99164-1300

June 2002

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1 THE FACILITY

1.1 Introduction

This report describes the Washington State University modified TRIGA reactor located on the WSU campus in the City of Pullman, in the State of Washington. The WSU Reactor is a TRIGA type 1 MW research reactor utilized extensively by many departments at WSU which is a land grant educational institution. This report supersedes and replaces all previous SAFETY ANALYSIS Reports and descriptions of the Washington State University Reactor.

A TRIGA type reactor has many unique features that make such a reactor ideally suited for use at educational institutions.(1) The one megawatt WSU reactor is a pool-type research reactor with a light-water moderated, heterogeneous, solid fuel reactor in which water is also used for both cooling and shielding. The inherent prompt negative temperature coefficient of a TRIGA type reactor described in more detail in section 4.2 is the most significant safety feature of such reactors. The consequences of various postulated malfunctions are analyzed in section 13 of this report. A detailed analysis of the Maximum Hypothetical Accident (MHA), which for a TRIGA type reactor is the rupture of one fuel rod in air, is given in section 13.1.1 of this report.

Over 50 TRIGA type research reactors, including 28 in the United States, have been constructed and operated safely without a single significant incident over the past 30 years. Many of these TRIGA reactors are located on university campuses and in hospitals with surrounding high population areas. This outstanding safety record and past experience at Washington State University clearly substantiate the conclusion that the continued operation of the WSU modified TRIGA reactor does not present a significant safety hazard to the general public.

Washington State University is a land grant educational institution in the State of Washington funded directly by State appropriations approved by the Legislature of the State. Financial responsibility considerations related to the continuous operation of the WSU TRIGA Reactor Facility are given in section 15 of this report. Administratively, the facility is under the Vice Provost for Research and thus is a campus wide research facility associated with University Sponsored Research and Graduate education. A block diagram of the management organization of this facility is given in Figure 1-1. Within the facility the two key management positions are the Director of the Radiation Center, Dr. Gerald Tripard, who is a senior experimental physicist and the Reactor Supervisor, Stephanie Sharp, who has a B.S. in Nuclear Engineering.

The existing WSU modified TRIGA reactor was relicensed for 20 years on August 11, 1982 and consequently has an existing updated approved set of Technical Specifications. A slightly revised set of Technical Specifications is given in section 14 of this report. The facility has recently been licensed for Boron Neutron Capture Cancer Therapy. The unique considerations for BNCT usage are covered in detail in section 16.3 of this report.

A NRC approved Physical Security Plan, Emergency Plan, and Operator Training and Requalification Plan have been in existence for a number of years. Slightly updated and revised plans are described in detail in sections 12.8, 12.7, and 12.16 of this report.

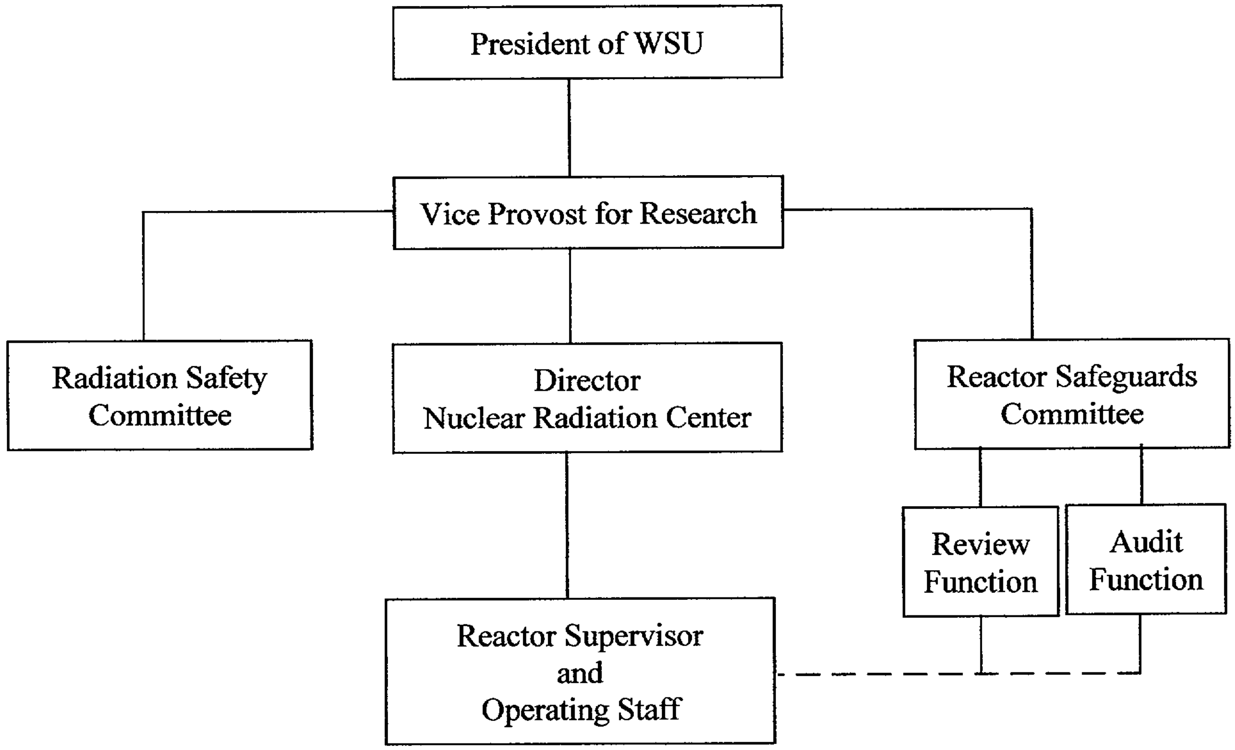

Figure 1-1. Facility organization

1.2 Summary and Conclusions on Principal Safety Considerations

The Washington State University Modified TRIGA reactor is located in the Nuclear Radiation Center which is situated in the northeast corner of the campus at Pullman, Washington. The core is located in a 242,000 liter above-ground pool which is, in turn, cooled and purified by external cooling and purification systems. Reactor experimental facilities include in-core irradiation positions, a thermal column and numerous beam tubes.

The existing reactor system operates with a mixture of Standard and FLIP* types of TRIGA fuel in the steady-state or pulsed modes. The maximum continuous steady-state power level is 1 MW and the average maximum pulsed power level is 1200 MW. Standard TRIGA fuel contains uranium-zirconium hydride enriched in ^{235}U to 20%. FLIP TRIGA fuel contains uranium-zirconium hydride enriched in ^{235}U to [redacted] The reactivity worths of both types of fuel are about equal. The increased ^{235}U content of FLIP fuel along with the burnable poison yields a fuel that has a significantly longer core life time potential than Standard TRIGA fuel. The principal design parameters of the WSU Modified TRIGA reactor are listed in Table 1-1.

The safety of the modified system, as with all TRIGA reactors, comes from the large prompt negative temperature coefficient that is inherent in a water-moderated, U-ZrH fueled reactor. The overall operating characteristics for the Washington State University Modified

* FLIP (Fuel Life Improvement Program) is a new type of long-lived fuel developed by Gulf Energy and Environmental Systems for TRIGA reactors.

TRIGA reactor fueled with various combinations of fuels is discussed in section 4.2.1 of this report. The data in this report were calculated using the EXTERMINATOR-2[2] code and multi-group cross-section data[3-5] obtained from G.A. Technologies, Inc.

TABLE 1-1
PRINCIPAL DESIGN PARAMETERS

Reactor Type	Modified TRIGA
Fuel Element Design	
Fuel-moderator material	U-$ErZrH_{1.6}$
Uranium content	[redacted]
U-235 enrichment	Std, LEU=20%, FLIP=70%
U-235 content (avg) per element	[redacted]
Burnable poison (FLIP, LEU)	natural erbium
Erbium content (FLIP, LEU)	FLIP=1.58 wt-%, LEU=.5wt-%
Shape	cylindrical
Length of fuel meat	[redacted]
Diameter of fuel meat	[redacted]
Cladding material	[redacted]
Cladding thickness	[redacted]
Core Characteristics	
No. of fuel elements	[redacted]
Vol-% water in core	30
No. of control rods	5
Total reactivity worth of control rods	$14
Neutron absorber	Boron in B_4C
Excess reactivity	$8
Neutron lifetime	24 microseconds
Prompt negative temperature coefficient	1.4¢/°C

The WSU Reactor Facility Radiation Protection Program as described in detail in section 11 of this report and meets all the requirements of 10 CFR 20, ANSI/ANS-15.11 and the associated ALARA considerations. This program insures that no one working at the facility or the general public will be exposed to radiation levels that would be hazardous to their health and safety. The primary radioactive effluent from the facility is Argon-41. The analysis given in section 13.1.1 of this report demonstrates that a 100 times normal release would not endanger the health and safety of the general public. Last but not least, the analysis associated with the MHA and the postulated related fission product release would also not endanger the health and safety of the general public.

Past experience with the Washington State University TRIGA reactor and other TRIGA reactors clearly indicates that a properly designed reactor system fueled with TRIGA-type fuel can be safely operated at steady-state power levels of 1 MW and pulsed to a power level of 1200 MW. This history of safe and conservative reactor design has permitted TRIGA type reactors to be sited in urban areas without the need for specially designed containment structures. Furthermore, the WSU reactor, fueled with a mixture of Standard and FLIP fuels, has operated for over twenty years without a single fuel-related problem.

The information presented in this safety analysis report, in section 13, indicates that the continued operation of the Washington State University Modified TRIGA reactor will pose no health or safety hazards to the public. Furthermore, the system is safe when operated in a normal manner or even if a highly abnormal condition occurs. The three major accidents considered are: (1) accidental fuel addition, (2) pulsing of the reactor (transient rod ejection) while operating at full power, and (3) accidental loss of coolant. In all of these postulated accidents, no loss of fuel cladding integrity would occur. Also, the MHA, which is loss of the integrity of the cladding on one TRIGA rod in air, is shown not to present a significant hazard to the general public.

1.3 General Description

1.3.1 Geographic Location

The Washington State University Modified TRIGA reactor is located on the WSU campus about one mile east of the main portion of campus as shown in Figure 1-2. The WSU campus is located adjacent to the town of Pullman, Washington in the south east corner of the State of Washington as shown in Figure 1-3. The land surrounding the Pullman area is devoted to dry land farming and Whitman County in which Pullman is located is a major producer of wheat. The facility is pictured in Figure 1-4.

1.3.2 Principal Characteristics of the Site

The WSU Reactor Facility pictured in Figure 1-4 is isolated from the main portion of the WSU campus and is surrounded by agricultural land used by the university for a variety of agricultural purposes. The terrain is undulating Palouse hills created during the Pleistocene epoch and capped with windblown soil during the ice ages. No known geologic hazards exist on or near the site and the nearest significant faults are the Vista and Wilma inactive faults associated with the Lewiston Downwarp over 20 miles south of the site. The closest active faults are located in the Walla Walla area some 70 miles south west of the site. Detailed information on the geology and seismology of the region is given in section 2 of this report.

1.3.3 Principal Design Criteria, Operating Characteristics, and Safety Systems

The WSU modified TRIGA reactor is located in a 1200 square meter concrete building that composes the WSU Nuclear Radiation Center. Floor plans for the first and second floors of the Nuclear Radiation Center is given in Figure 1-5 and 1-6. The principal reactor areas are: Room 201, the reactor pool room; Room 201B, the reactor control room; Room 201A, the reactor shop; and Room 201C, the heat exchanger/pump room. The other rooms on the floor plan are offices and general laboratories. An artist's sketch of the WSU pool type reactor as originally constructed showing the location of the core is shown in Figure 1-7. The control panel, however, was not located on the bridge structure but was placed in a separate room (201B of Figure 1-6). The pool is approximately [redacted] and contains 242,000 liters of very pure water which functions as shield, moderator, and coolant. The pool design originated from the Bulk Shielding Facility designed and constructed at Oak Ridge National Laboratory in the late 1950's. The original core utilized MTR type aluminum clad flat fuel plates and operated at a maximum power level of 100 kW. In 1967 the core was converted to TRIGA type rod fuel, a pool cooling system was installed, and the maximum steady-state power level increased to 1,000 kW.

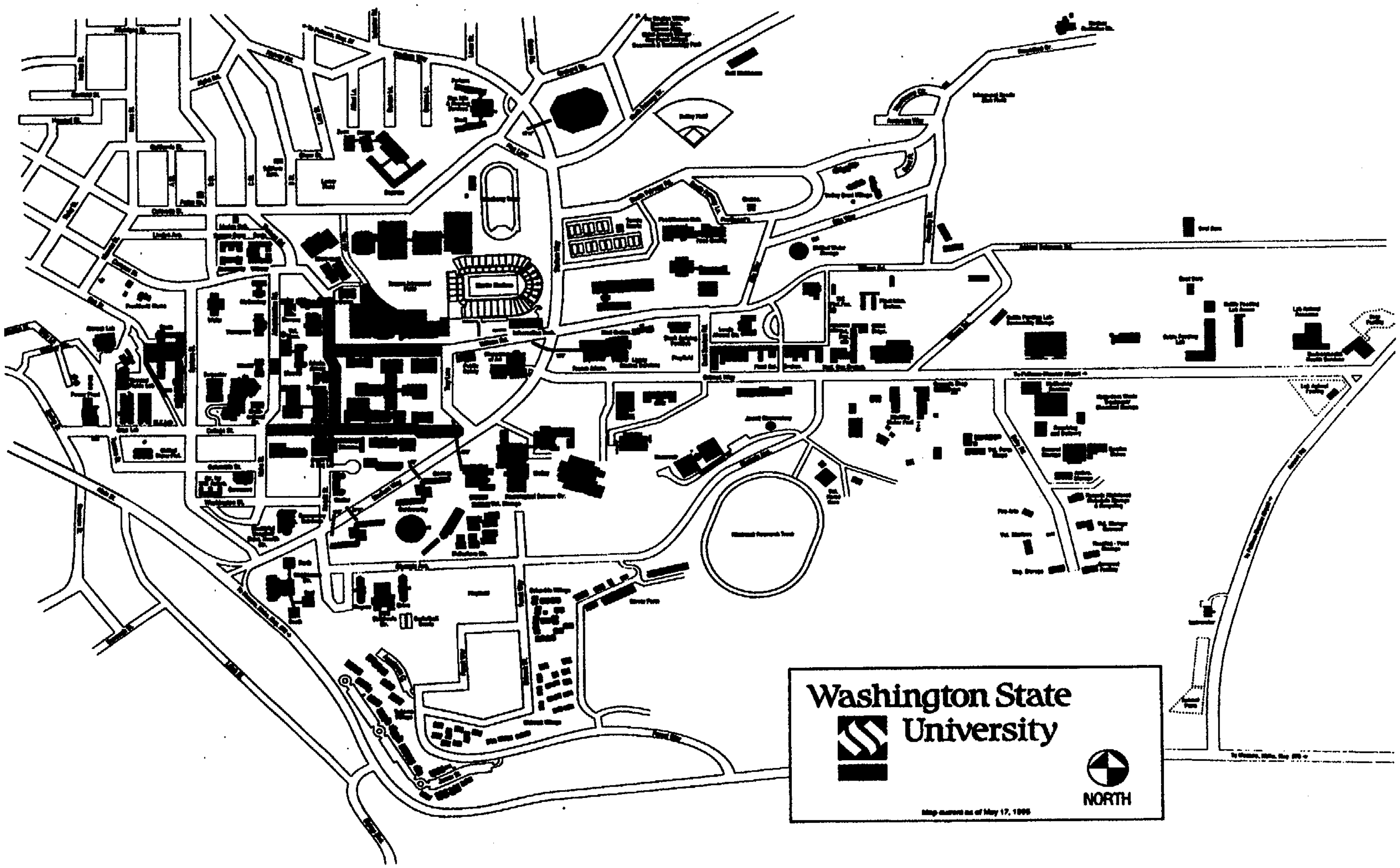

Figure 1-2

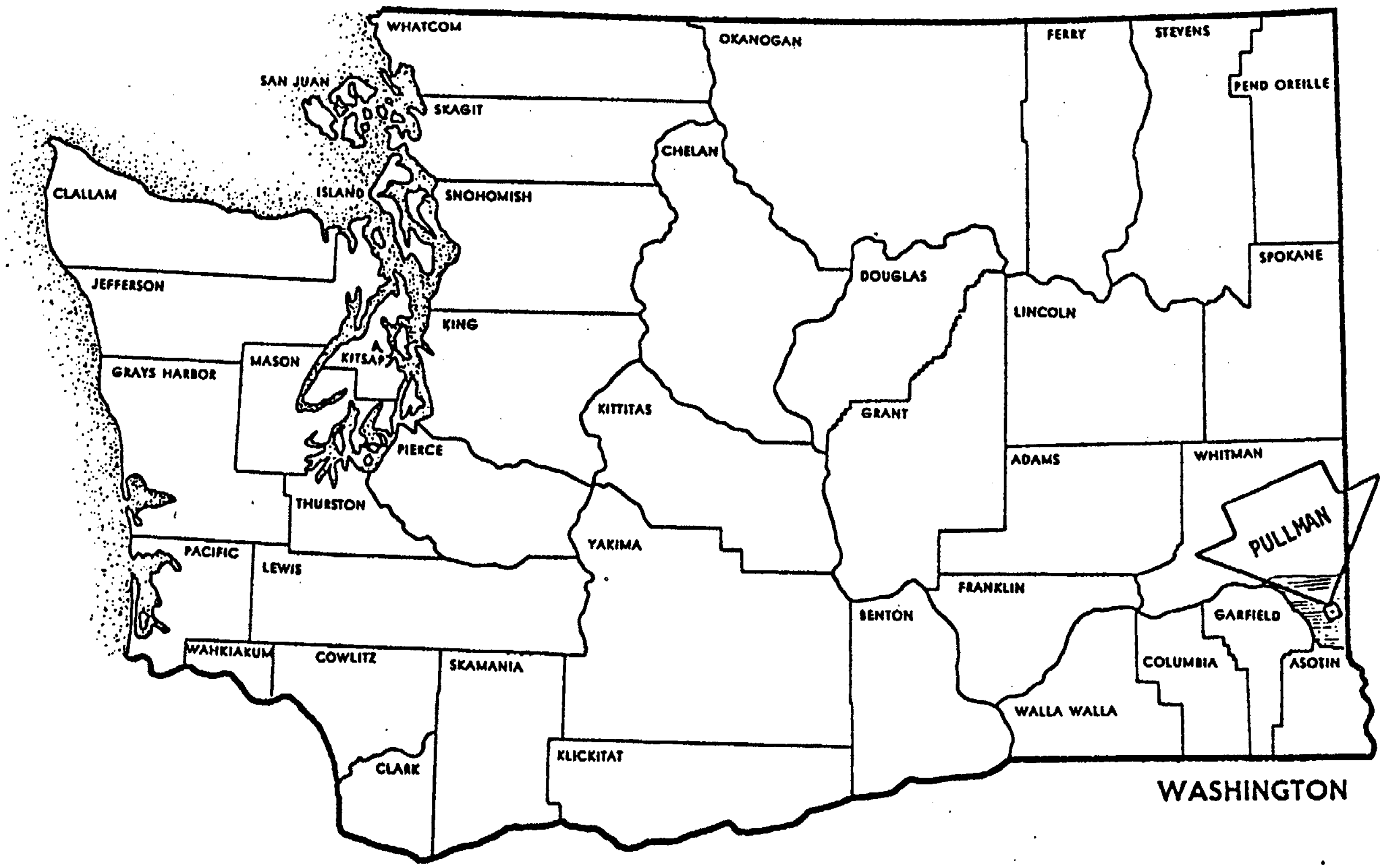

Figure 1-3
State of Washington

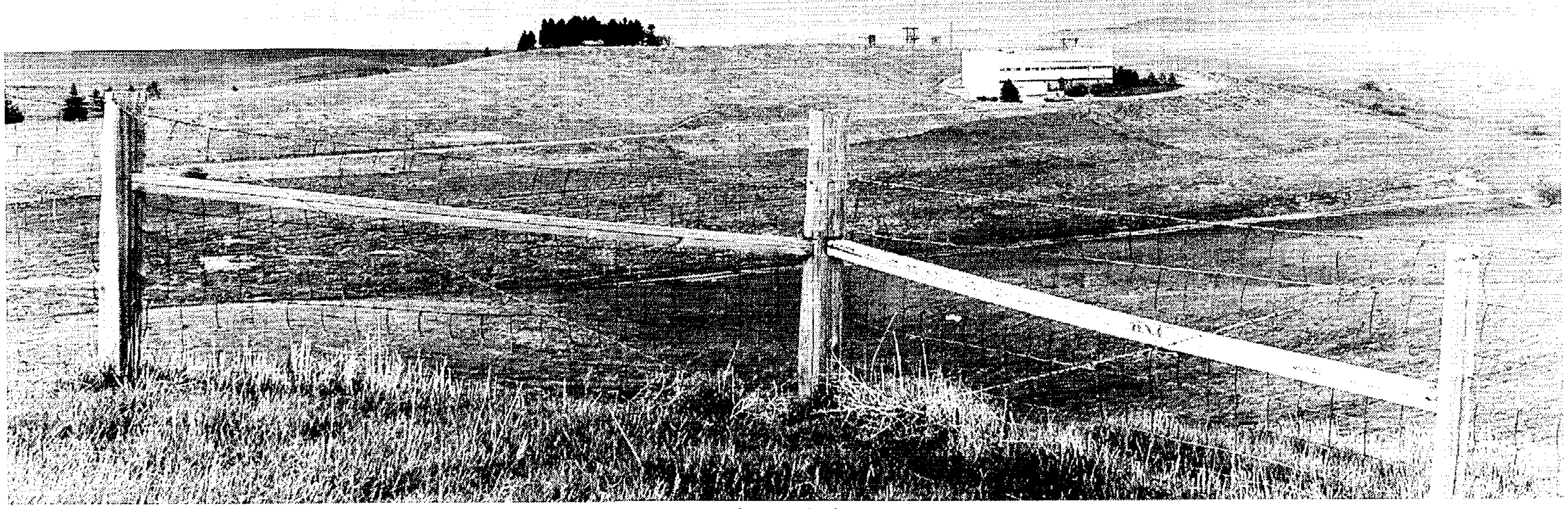

Figure 1-4
WSU Reactor Facility

[redacted]

Figure 1-5
Radiation Center First Floor Plan

[redacted]

Figure 1-6
Radiation Center Second Floor Plan

[redacted]

The WSU modified TRIGA reactor shares the safety and operational characteristics of all TRIGA type reactors. TRIGA fuel was developed around the concept of inherent safety. A core composition was sought which had a large prompt negative temperature coefficient of reactivity such that if all the available excess reactivity were suddenly inserted into the core, the resulting fuel temperature would automatically cause the power excursion to terminate before any core damage resulted. Experiments in the late 1950s at General Atomics (GA) demonstrated that zirconium hydride possesses a basic mechanism to produce the desired characteristic. Additional advantages were that ZrH has a good heat capacity resulting in relatively small core sizes and high flux values due to the high hydrogen content and ZrH could be used effectively in a rugged fuel element size.

The development and use of U-ZrH_x fuels for the TRIGA reactor have been underway at GA since 1957. Over 6000 fuel elements of 7 distinct types have been fabricated for the 50 TRIGA research reactors which have been constructed and placed in operation. The earliest of these has now passed 30 years of operation. U-ZrH fuel has exhibited unique safety features including a prompt negative temperature coefficient of reactivity, high fission product retentivity, chemical stability when quenched from high temperatures in water, and dimensional stability over large swings of temperature.

The standard TRIGA fuel contains 8.5 wt-% uranium (20% enriched) as a fine metallic dispersion in a zirconium hydride matrix. The H/Zr ratio is nominally 1.6 (in the face-centered cubic delta phase). The equilibrium hydrogen dissociation pressure is governed by the composition and temperature. For $ZrH_{1.6}$ the equilibrium hydrogen pressure is 1 atm at about 760°C. The single-phase, high-hydride composition eliminates the problems of density changes associated with phase changes and with thermal diffusion of the hydrogen. TRIGA fuel with 12 wt-% U has been proven through successful reactor operation for over two decades. A highly enriched version of TRIGA fuel called FLIP (discontinued in 1979 because of the Non Proliferation Treaty) contained up to about 3% erbium as a burnable poison to increase the core lifetime and contribute to the prompt negative temperature coefficient at higher power. The calculated core lifetime with FLIP fuel in the 2-MW TRIGA is approximately 9 MW-yr. Over 25,000 pulses have been performed with the TRIGA fuel elements at GA, with fuel temperatures reaching peaks of about 1150°C.

The WSU Modified TRIGA reactor is fueled with a mixture of Standard and FLIP fuels and is operated in the steady-state mode up to a maximum power of one megawatt. The Reactor is pulsed with a maximum insertion of the order of slightly over $2 set at a level to limit the maximum fuel temperature during pulsing to below 830°C from the lessons learned as a result of the Texas A&M FLIP fuel failure during pulsing incident. Detailed information on this matter is given in section 4.5.3 of this report. The peak power level during pulsing for a $2.20 pulse is 1200 megawatts. The principal design parameters of the reactor are listed in Table 1-1. The principal safety feature of the WSU Modified TRIGA reactor is the large prompt negative fuel temperature associated with the TRIGA type fuel discussed above.

The main safety systems of the reactor are the "Power-Level" trips that insure that the power level does not exceed the licensed limit, the "Fuel Temperature Scram" that insures that the fuel temperature does not exceed 500°C, the "Manual Scram" button that allows the operator to immediately shut down the reactor in the event of a perceived potential problem, and the diffuser system that minimizes the radiation exposure level on the reactor bridge due to the production of ^{41}Ar and ^{14}N in the reactor pool water. These safety systems are discussed in detail in sections 7.3 and 5.6 of this report.

1.3.4 Engineered Safety Features

A schematic diagram of the cooling system for the WSU modified TRIGA reactor is given in Figure 1-8. The primary consideration in the operation of any reactor is the consequence of a loss of coolant flow. Since the WSU modified TRIGA reactor has such a large pool, the effect of a loss of primary coolant flow does not instantaneously precipitate a problem. A loss of primary flow at full power will cause a slow rise in pool water temperature which will be noticed by the operator who can take appropriate corrective action.

Engineered safety features in the cooling system include: 1) operating the secondary side of the heat exchanger at a higher pressure than the primary to insure that in the event of a heat exchanger failure the flow would be into the pool, 2) a conductivity monitoring system to detect a secondary to primary leak (secondary is treated with an antifouling chemical that would greatly decrease conductivity in the event of a leak), and 3) temperature and pressure monitoring that indicate the status of the cooling system operation. The primary loop also has a Siphon Break so that the pool would not be drained in the event of a major cooling system problem.

The basic safety feature of a TRIGA type reactor is the prompt negative temperature coefficient as previously discussed. In order to add additional safety, the WSU TRIGA reactor has a Fuel Temperature Scram described in detail in section 7 of this SAR. Also, at a power level of 1 MW, a complete loss of pool water would not cause a fuel rod failure as is discussed in detail in section 13.1.4 of this SAR. Even if a single fuel rod fails after a complete loss of pool water, (MHA for a TRIGA) the consequences are minimal as analyzed in section 13.1.1 of this SAR.

Additional Engineered Safety Features of the WSU TRIGA reactor include: 1) inhibiting pulsing of the reactor from an initial power level over 1 kW, 2) a seismic scram that shuts down the reactor in the event of a major earthquake, 3) an air monitor that monitors the pool room air, 4) an Ar-41 monitor that monitors the Ar-41 content of the Reactor Exhaust, and 5) pool room air handling system that has normal, isolation, and dilute modes of operation.

1.4 Shared Facilities and Equipment

The WSU Reactor Facility is located within the WSU Nuclear Radiation Center and thus shares some building services with the Center. However, the Reactor Facility has an isolated ventilation and heating system that is described in detail in section 9.1 of this report. Thus the operation or failure of the ventilation and heating system of the laboratory and office portion of the Nuclear Radiation Center does not impact the reactor facility or vice versa. The heat source for the reactor room air is coils located in the reactor ventilation system supplied with hot water from a gas fired boiler. This boiler also serves as the heat source for the main office for the Center. The newer laboratory and office portion of the Center has gas fired heating units located within the ventilation system of that portion of the Center.

Power to the Reactor Facility is provided via a separate system with separate circuit breakers from the rest of the electrical power to the Center. The only common point is the main power transformer and associated main breaker that feeds the entire Center.

Other shared services are the demineralized water system and hot drain system. The hot drain system flows into a system of retention tanks that are sampled and monitored before being released to the campus sanitary sewer system. See section 11.2 for details on liquid radioactive waste management.

[redacted]

Figure 1-8
New Cooling System Schematic

1.5 Comparison With Similar Facilities

The WSU modified TRIGA Reactor Facility is quite similar to the facilities at Texas A&M University and the University of Wisconsin. That is, all three reactors started out as Bulk Shielding type reactors of General Electric design with aluminum clad flat plate fuel elements and a square grid box. All three reactors were converted to TRIGA type reactors by replacing the nominal 3 inch by 3 inch flat plate fuel elements with a cluster of four TRIGA type fuel rods called a "four rod cluster". Detailed information on TRIGA fuel rods is given in section 4.2.1 of this report. Both Texas A&M and the University of Wisconsin reactors now have cores that are 100% FLIP fuel whereas the WSU core is fueled with a mixture of Standard and FLIP fuel.

The old GE reactor control systems at each of the three similar facilities have been modified to correspond to the needs of a TRIGA type reactor. However, the paths taken for the required modification of each facility were significantly different. At WSU the control system is an in-house-designed system using commercially available components such as a Keithley micro-micro amp meter for one of the safety channels, a General Atomics (GA) Multi-Range Linear Channel, a G.A. Power Pulse Channel and a G.A. Wide Range Channel for startup and log power channel. More specific information is given in section 7 of this report.

In general the WSU modified TRIGA reactor shares the principal design parameters, reactor safety systems, engineered safety systems, and instrumentation and control systems of the 50 TRIGA reactors currently in existence. The outstanding safety record of TRIGA type reactors world wide is well documented and accepted by all.

1.6 Summary of Operations

Since the WSU Reactor Facility was converted to TRIGA fuel in 1967 the reactor has been operated for a total of over 20,000 hours without a single significant reactor failure incident. A few minor problems have occurred during this time period but all these problems were either associated with minor equipment malfunctions or problems with samples being irradiated in the reactor rather than major reactor system problems. Since 1967, the modified TRIGA core has accumulated a total of 18,100 megawatt hours of operation and been pulsed 950 times without a fuel element failure.

Historically, a number of TRIGA reactors have had minor fuel element leakage problems predominantly as a result of fuel element damage associated with fuel element handling. At WSU extreme caution has been exercised to minimize such problems. This matter is discussed in more detail in section 13.1.5 of this report.

The primary usage of the WSU TRIGA reactor in the past has been the irradiation of sample materials for either radioisotope production or neutron activation analysis. In the future this will continue to be the major use of the reactor, plus a small amount of usage for boron neutron capture therapy (BNCT) described in more detail in section 16.2 of this report.

1.7 Compliance With the Nuclear Waste Policy Act of 1982

In accordance with the letter from DOE (R.L. Morgan) to NRC (H. Denton) of May 3, 1983, it has been determined that all universities operating nonpower 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 Washington State University has entered into such a contract with DOE, the applicable requirements of the Nuclear Waste Policy Act of 1982 have been satisfied by the WSU Reactor Facility.

1.8 Facility Modifications and History

The Washington State University reactor has been in operation since March 1961. From 1961 to 1967, the reactor was fueled with MTR-type fuel elements and operated at a maximum power level of 100 kilowatts. In 1967, the reactor was shut down and the core and control systems were modified so that the reactor could operate with TRIGA-type fuel. The original core grid box was retained and the MTR fuel elements were replaced with a special 4-rod cluster of TRIGA fuel rods designed to replace an MTR fuel element. From July 1967 to date, the reactor has operated as a modified TRIGA reactor with a maximum steady-state power level of 1 MW. In February of 1976, the core was loaded with a mixture of Standard and FLIP fuel. A list of the most significant modifications to the WSU reactor is given in the table below.

HISTORY OF WSU REACTOR FACILITY

1)	1960 -	Facility constructed and open pool reactor with MTR plate type fuel installed with maximum power level of 100 kW.
2)	1967 -	MTR plate type fuel replaced with TRIGA rod type fuel in 4-rod clusters. Pool cooling system installed allowing operation up to a power level of 1000 kW. Pulse rod added to center of core and control system modified to allow pulsing operation.
3)	1972 -	General Atomic Wide Range Channel added to control system replacing old General Electric startup channel and Log-N and period channel. Old G.E. linear-channel replaced with new Keithley channel.
4)	1975 -	Old General Electric Compensated Ion Chambers replaced with Reuter-Stokes CIC's with integral stainless-steel water tight cable.
5)	1976 -	Replaced nine (9) TRIGA Standard 4-rod fuel clusters in the center of the core with FLIP fuel forming mixed core 30A.
6)	1976 -	Purchased new control console cabinet.
7)	1977 -	Reactor staff designed, constructed, and installed new control system in new console completely replacing old control system and console.
8)	1980 -	Replaced four (4) TRIGA Standard 4-rod fuel clusters in core with FLIP rods forming mixed core 31A.
9)	1981 -	Replaced four (4) TRIGA Standard 4-rod fuel clusters in core with FLIP rods forming mixed core 32A.
10)	1988 -	Optimized core arrangement of FLIP and Standard fuel increasing the thermal neutron flux in the rotator row by approximately 30%.
		New CIC installed on Safety Channel No. 2 to replace failed unit.
11)	1991 -	Computer interface to linear power channel installed allowing computer monitoring of reactor power level.
12)	1992 -	Power supply for GA wide range channel replaced.
13)	1994 -	Replaced Log-N, Linear Power, and Fuel Temperature channel strip chart recorders.
14)	1995 -	Added SCRAM button in the Beam Room.
15)	1995 -	New pulse rod air pressure control and low air pressure alarm.
16)	1997 -	New CAM System (Continuous Air Monitoring System).
17)	1999 -	New reactor cooling system (New cooling tower, heat exchanger, primary pump, secondary pump, system piping) and added secondary water filter.
18)	1999 -	Had leak in pool wall professionally repaired.

1.9 References

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6. "Safeguards Report for Open Pool Reactor for Washington State University," GEAP-3100, February 16, 1969.

7. "Experimental Results from Tests of 18 TRIGA-FLIP Fuel Elements in the Torrey Pines Mark F Reactor," GA-9350.

8. "Amendment I to Safety Analysis Report of October, 1966," Washington State University, Nuclear Radiation Center, Pullman, WA 99164, May 1974.

9. "Safety Analysis Report for WSU Modified TRIGA Reactor," Washington State University, Nuclear Radiation Center, Pullman, WA 99164, May 1979.

2 SITE CHARACTERISTICS
2.1 Geography and Demography
2.1.1 Site Location and Description
2.1.1.1 Specifications and Location

The facility is located on the Washington State University campus. The center of the Campus is located a[redacted] on the USGS Topography map of the Pullman Washington Quadrangle. In terms of USA local terminology, Washington State University is located in the southeastern corner of the State of Washington in the town of Pullman as shown in Figure 2-1. The town of Pullman, Washington has a population of 25,010 and is located in Whitman County, about 11 kilometers from the Washington-Idaho border as shown in Figure 2-2. In addition to the town of Pullman, the town of Moscow, Idaho is located approximately 13 kilometers east of the site, just across the Washington-Idaho border. Moscow has a population of 20,550 and is the location of the University of Idaho. The Palouse region surrounding the towns of Pullman and Moscow is a rural agricultural area devoted to dryland farming.

The WSU campus is east of the town of Pullman, Washington as shown in Figure 2-3. The actual reactor site is 3.2 kilometers east of the center of the town of Pullman and 1.6 kilometers east of the main portion of campus as shown on Figure 2-4. The site is surrounded. by University-owned property for at least .4 kilometers in all directions which is used for the grazing of livestock as shown on the aerial photo shown in Figure 2-5 and the site photograph of Figure 2-6. The Moscow-Pullman Airport is located 3 kilometers east of the site and can be seen in the upper right corner of aerial photo Figure 2-5. The closest occupied dwelling is 690 meters west of the site.

2.1.1.2 Boundary and Zone Area Maps

The exact location of the reactor facility on the WSU campus is shown in Figure 2-4. A site topography map is given in Figure 2-7. The exclusion zone associated with the facility is the perimeter of the facility building. A floor plan of the 2nd floor of the facility is shown in Figure 2-8. Additional detailed drawings of the facility are given in section 1. The actual reactor operating areas associated with the facility license include: the pool room (201), the control room (201B), and the pump room (201C). The other rooms shown on Figure 2-8 are office and laboratory spaces not directly associated with the operation of the reactor.

2.1.2 Population Distribution

The population distribution about the site in 500 meter increments out to 3 kilometers in eight directional segments is shown in Figure 2-9 and tabulated in Table 2.1-1. The population distribution was calculated for a typical day, with the University students, faculty and staff present on the campus. A circle with a radius of 500 meters about the site has no permanently occupied dwellings.

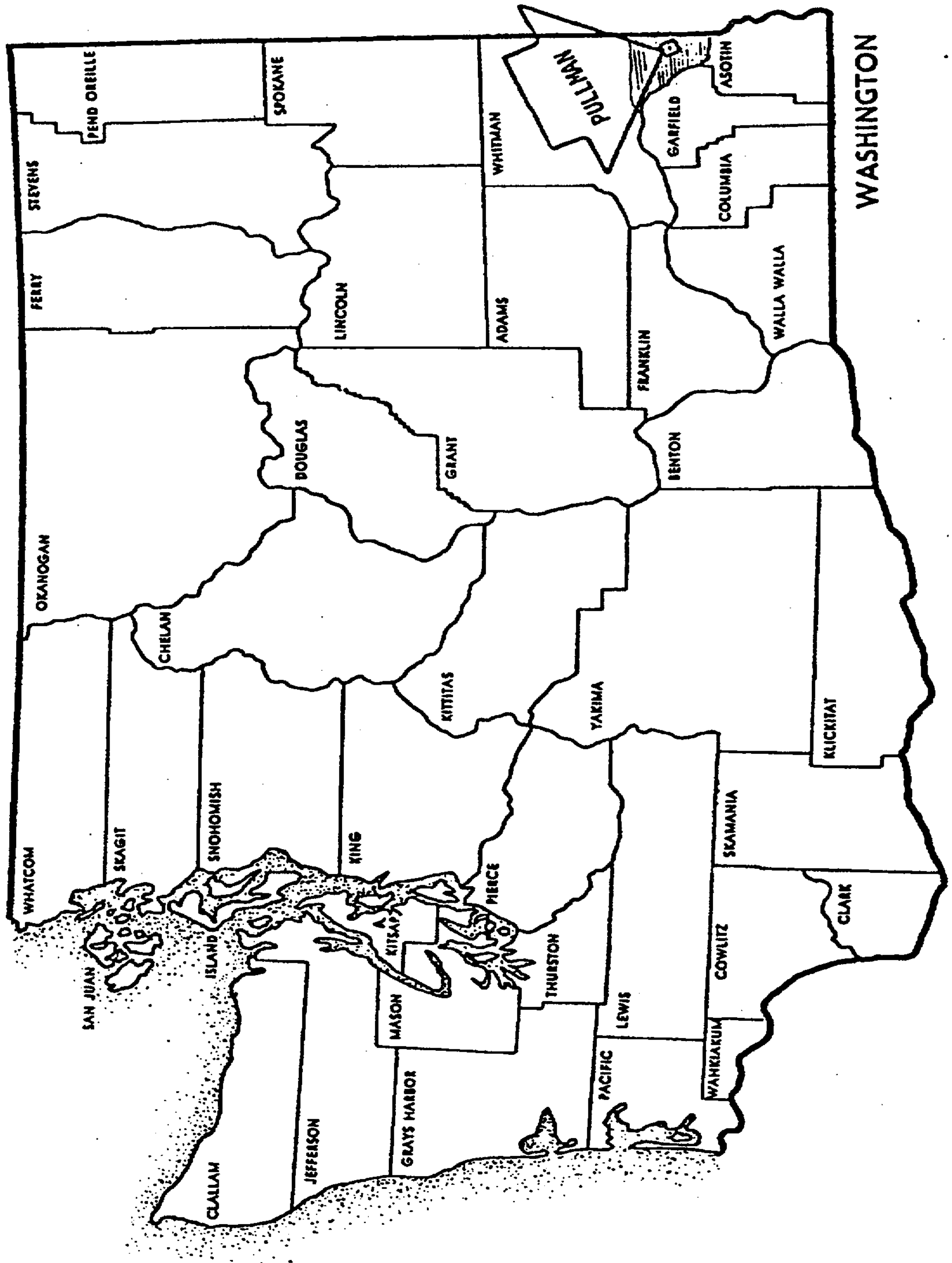


Figure 2-1
Washington State Map

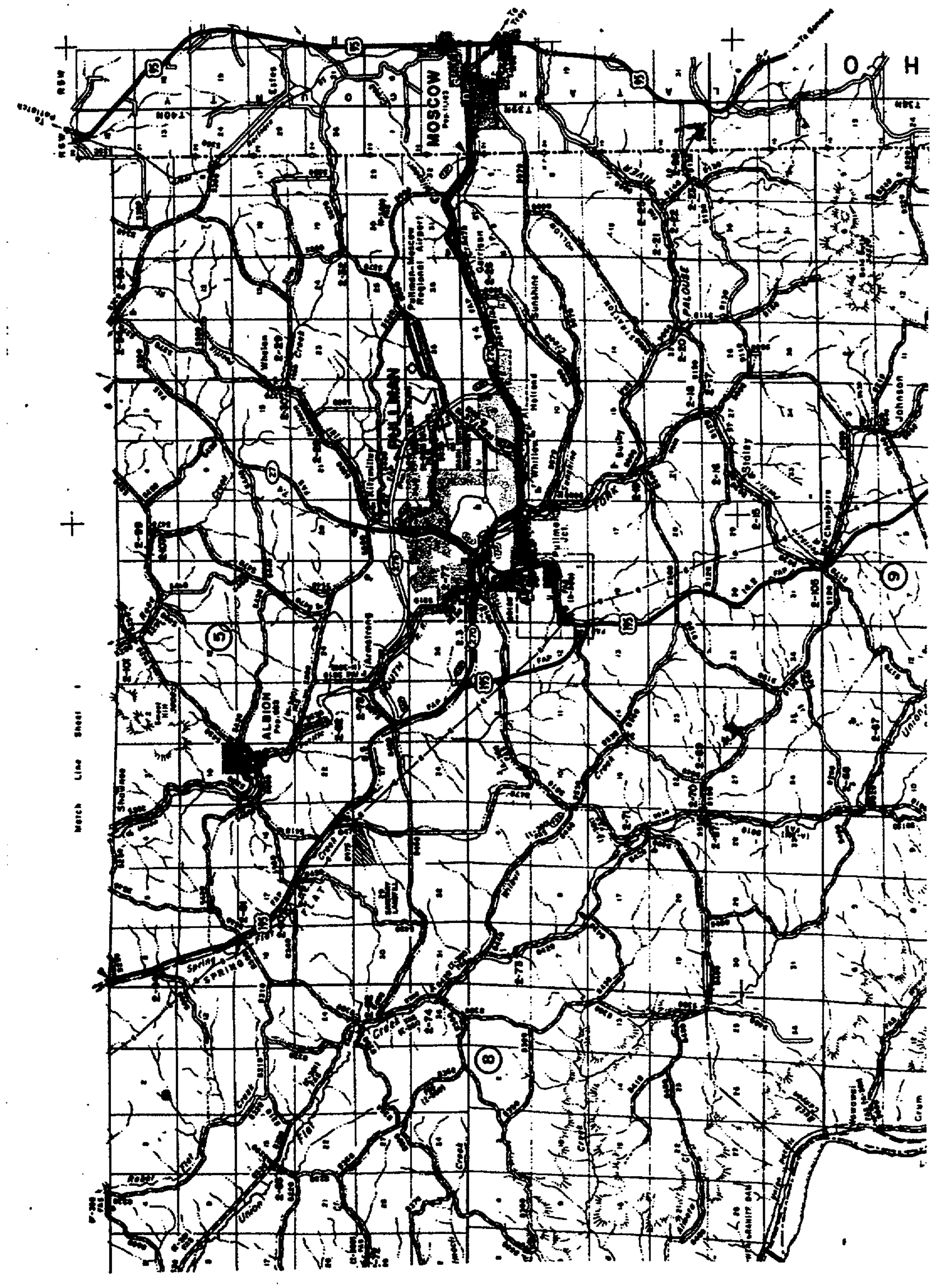


Figure 2-2
Whitman County Map in Pullman Area

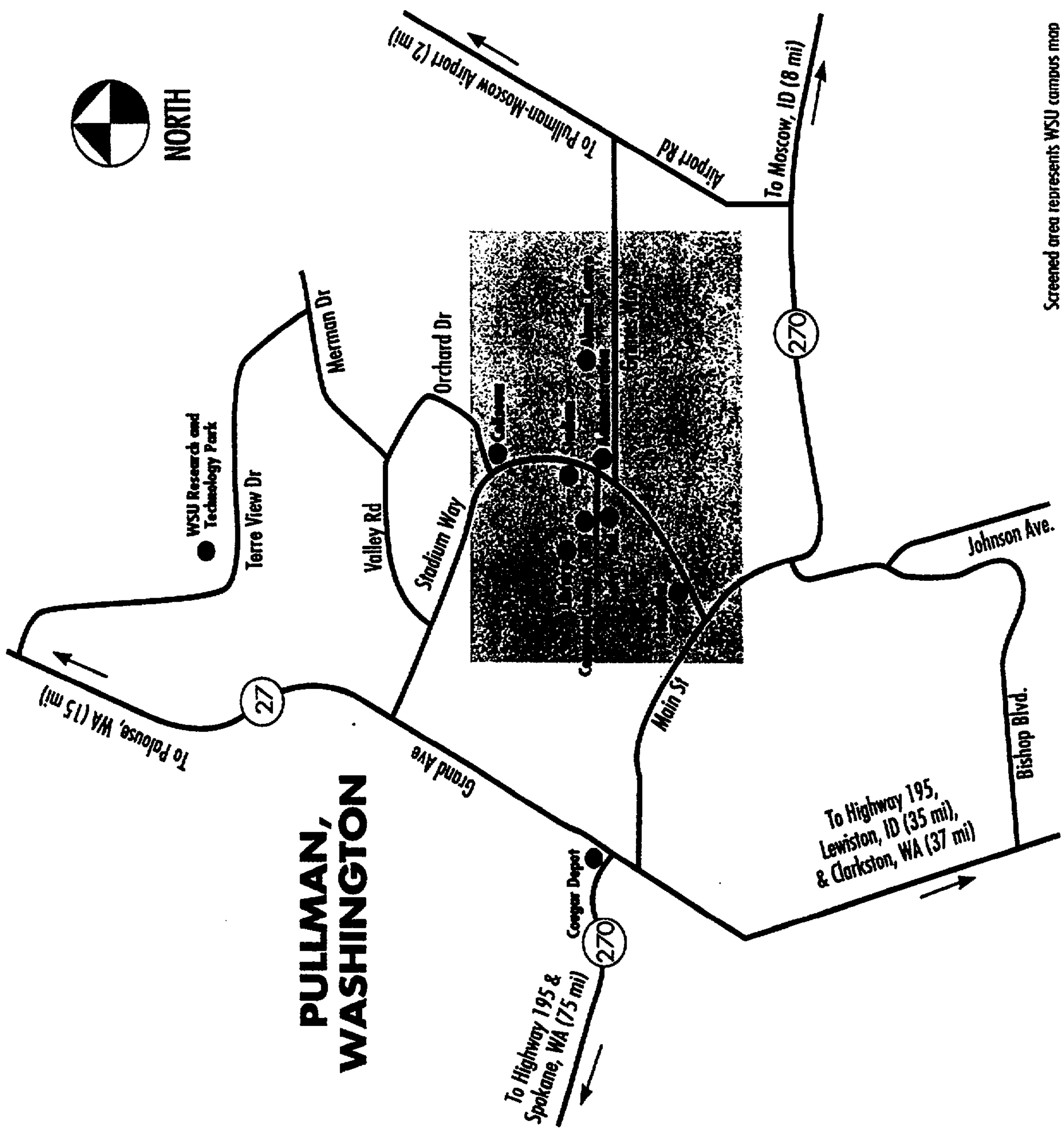


Figure 2-3
WSU Campus at Pullman, Washington

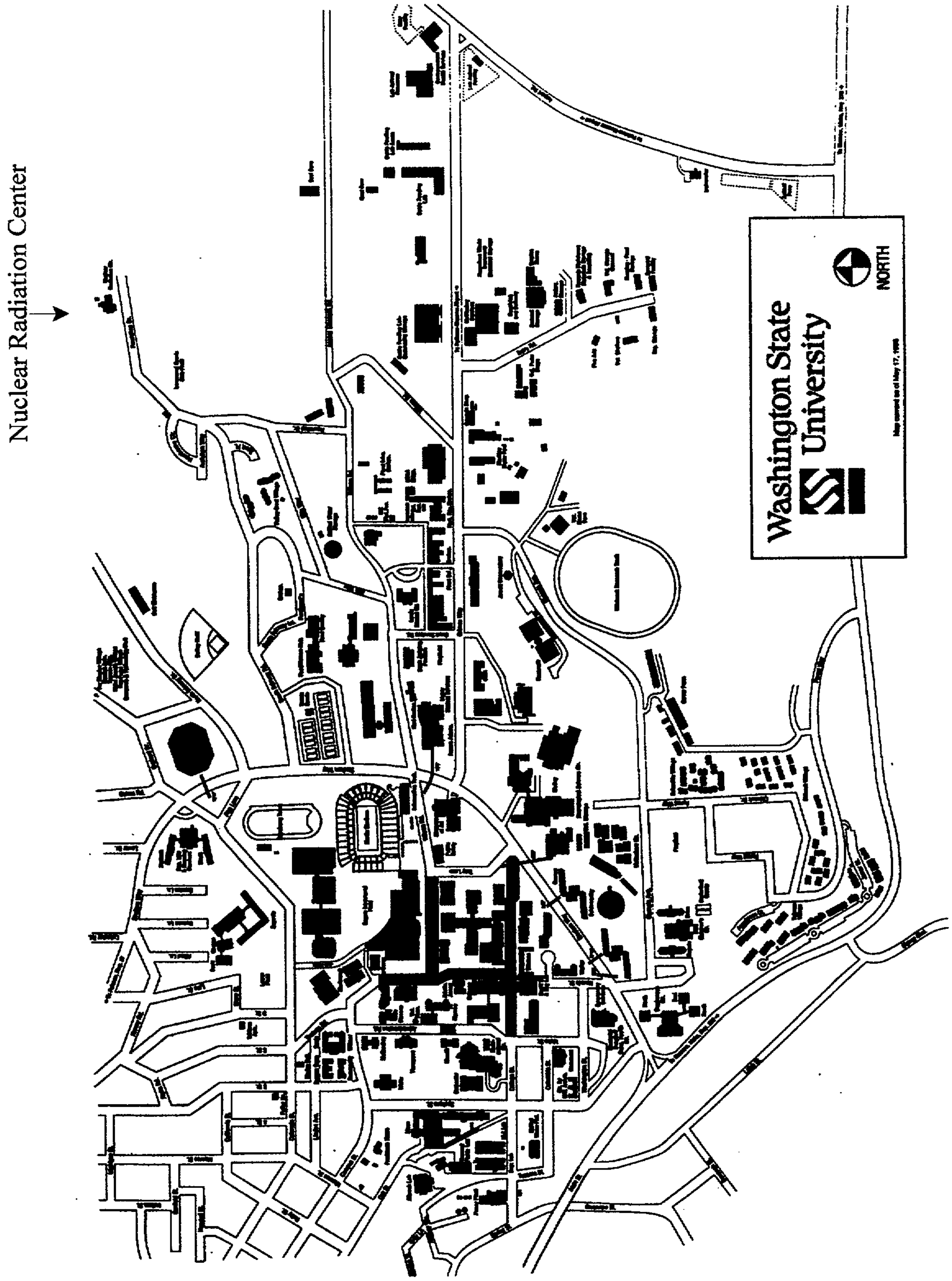

Nuclear Radiation Center
Washington State University
NORTH

Figure 2-4

Figure 2.5
WSU Aerial Photo

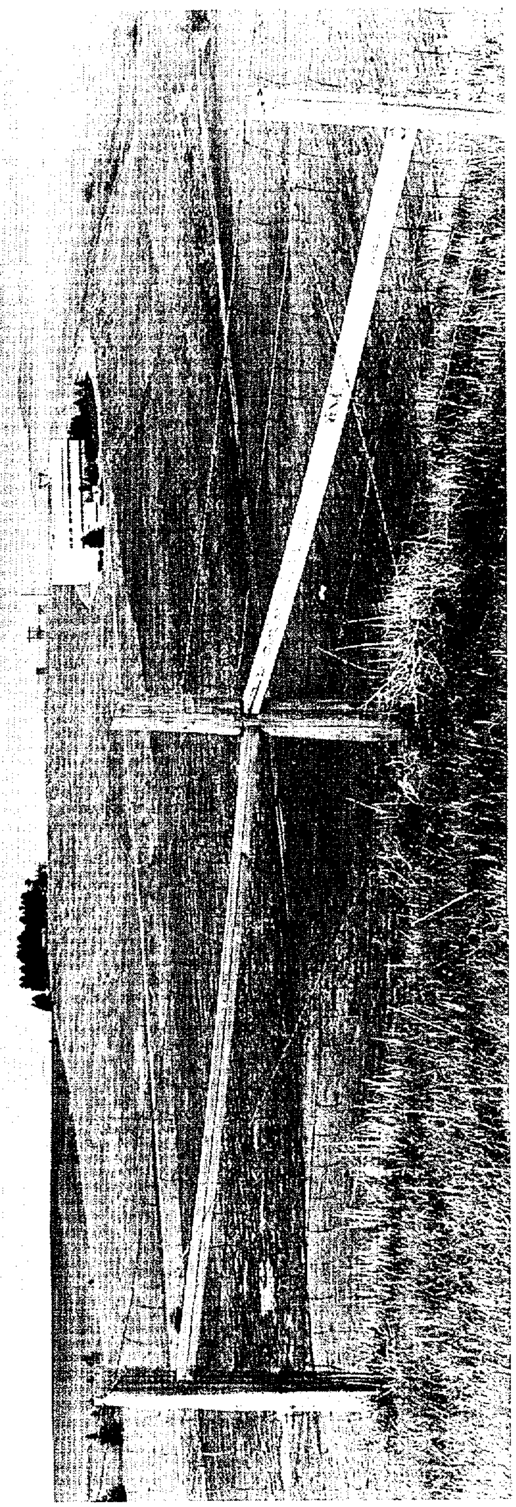


Figure 2-6
WSU Reactor Facility

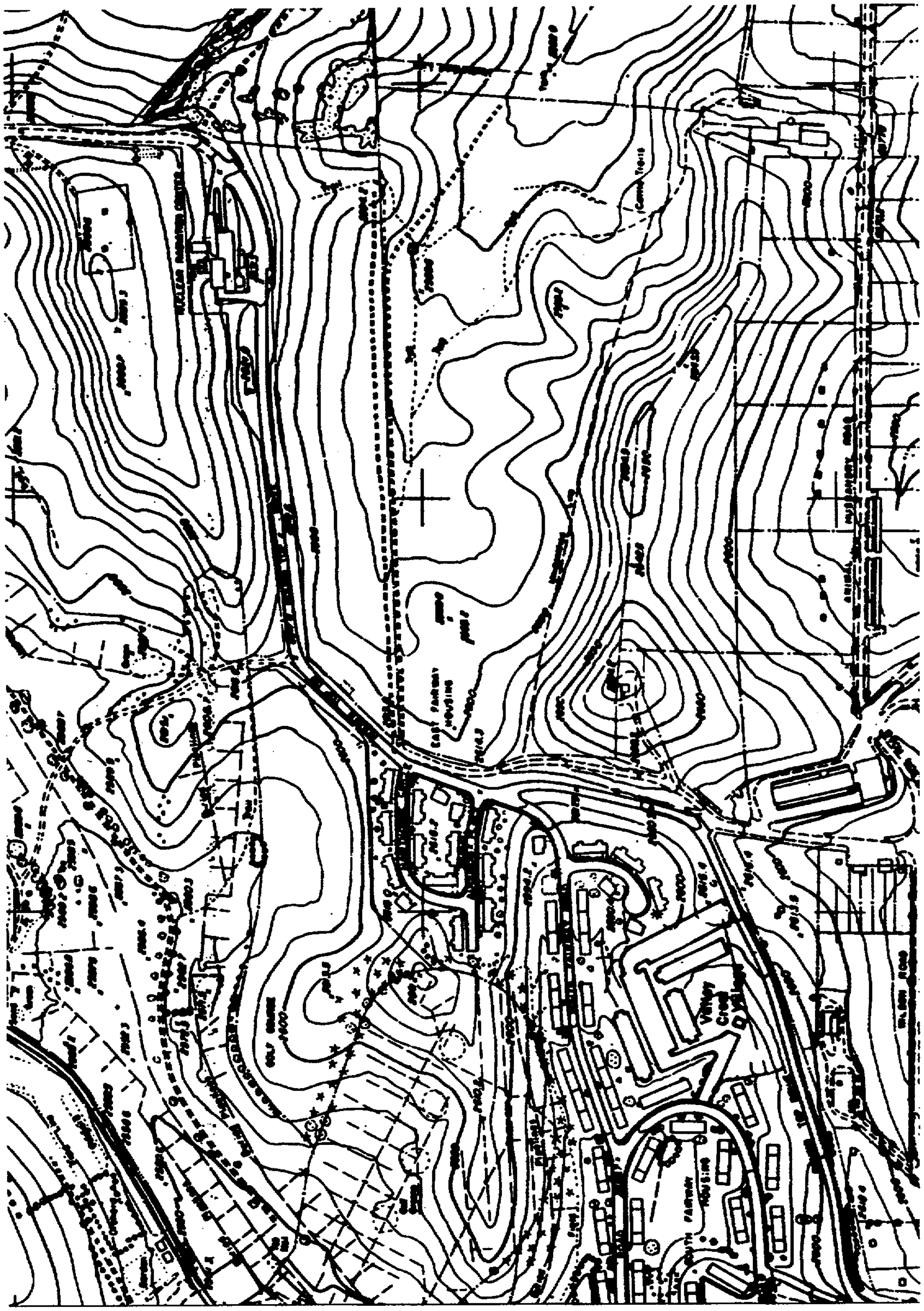


Figure 2-7
East Campus Topography Map Including Reactor Site

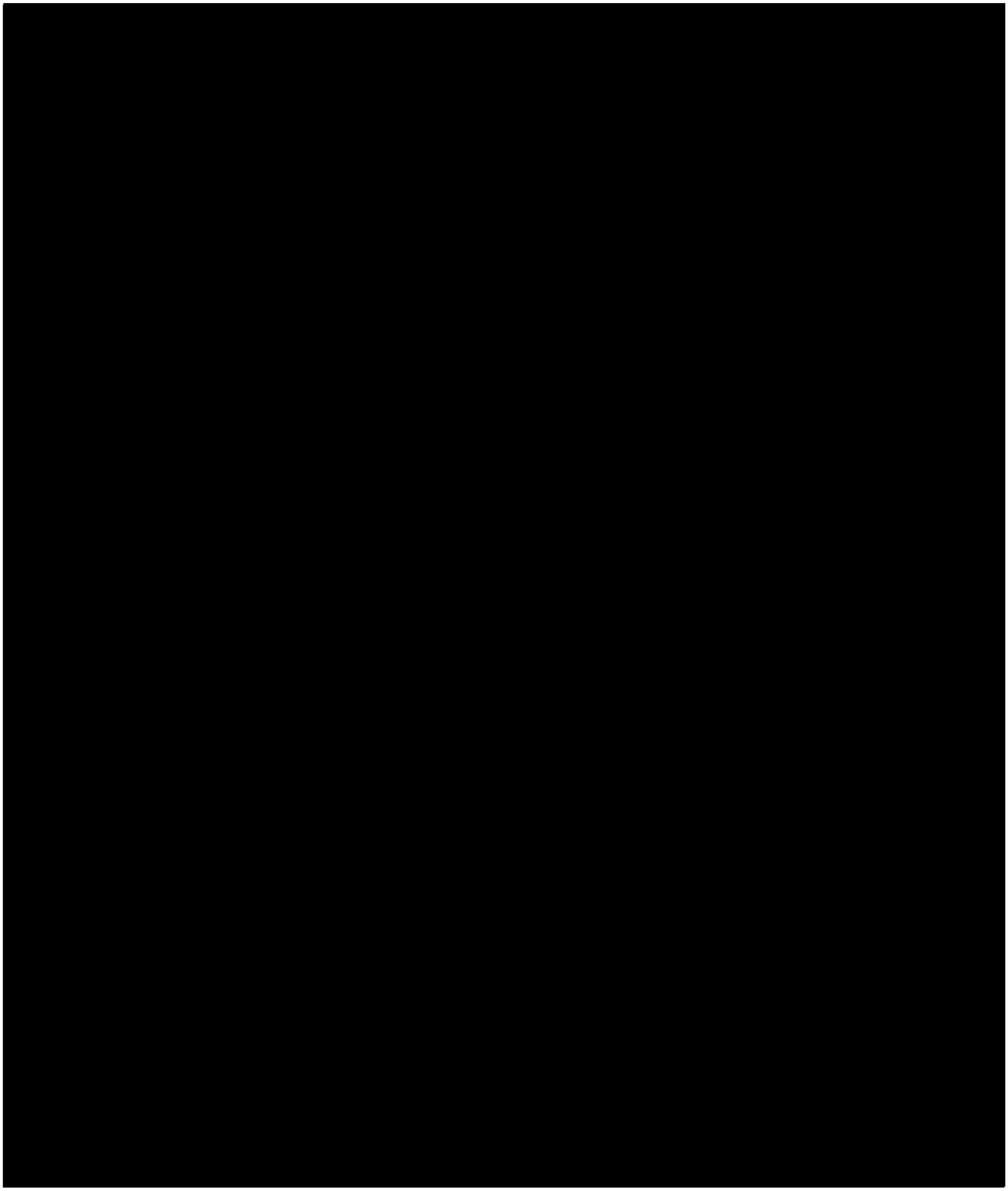


Figure 2-8

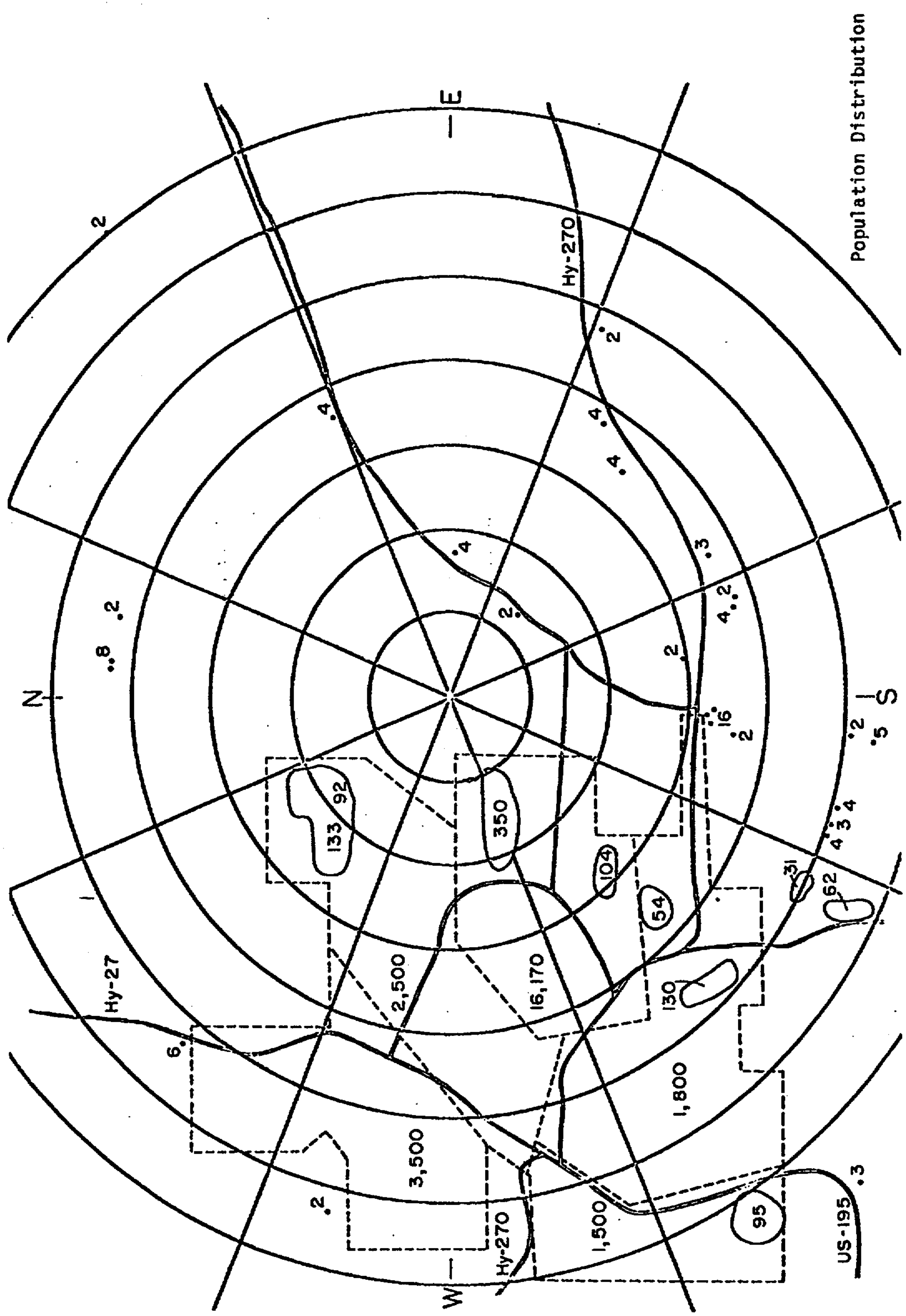


Figure 2-9
Site Area Population Distribution

TABLE 2.1-1

POPULATION DISTRIBUTION AROUND REACTOR SITE
Number of Residents per Octant

Distance in Meters	N	NE	E	SE	S	SW	W	NW
0 - 500								
500 - 1000	0	0	4	2	800	1,032	66	92
1000 - 1500	0	0	0	0	27	1,574	3,393	233
1500 - 2000	0	0	0	15	18	5,454	5,280	0
2000 - 2500	0	0	0	0	0	700	2,800	76
2500 - 3000	10	2	4	2	30	588	3,200	280
3500	4	0	4	9	4	340	233	8
TOTALS	14	2	12	28	879	9,688	14,972	689

2.2 Nearby Industrial, Transportation, and Military Facilities

No industrial, transportation, or military facilities are located in the vicinity of the facility except the Pullman-Moscow regional airport which is discussed in section 2.2.2.

2.2.1 Locations and Routes

Figure 2-3 shows the major transportation routes in the Pullman, Washington and WSU campus areas. Highway 270 to the west connects to highway 195 which is the Lewiston-Spokane highway and highway 270 to the east is the Moscow-Pullman highway. The road out to the facility is Round-Top road on the WSU campus which connects with Grimes Way on the map near the Alumni Center.

2.2.2 Air Traffic

The air traffic pattern in the vicinity of the facility is shown in Figures 2-10 and 2-11 in terms of the "Approach Surface" to the Pullman-Moscow regional airport. The west approach surface includes a large portion of the WSU campus. That is, aircraft landing from the west pass over the portion of campus that includes the facility.

The west end of Pullman-Moscow Airport- runway is at an elevation of 2537 feet and is located about 5800 feet east of the facility. The elevation of the basement of the facility is 2632 feet. In terms of the normal flight pattern, aircraft pass a number of hundred feet south of the facility and a few hundred feet above the roof level of the facility. The Pullman-Moscow Airport- does not have an electronic glide path landing system so aircraft only land by visual means in clear weather. There is, however, a VOR system to the west that is used by aircraft to approach the airport. The ceiling requirements are 603 feet vertical and one mile visibility for landing. The airport itself is sometimes fogged in so aircraft can not land but in such cases the fog generally does not extend up to the campus area.

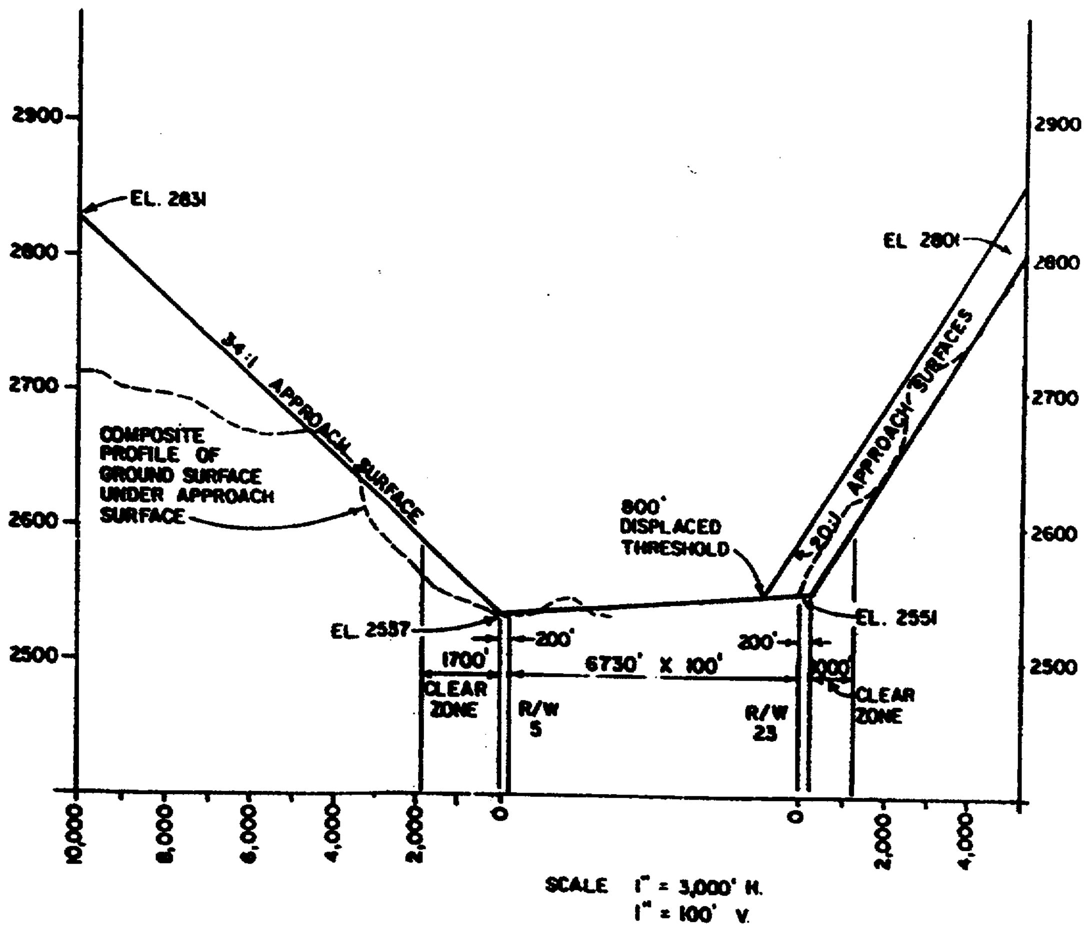

2900
2800
2700
2600
2500
EL. 2831
EL 2801
34:1 APPROACH SURFACE
COMPOSITE PROFILE OF GROUND SURFACE UNDER APPROACH SURFACE
800' DISPLACED THRESHOLD
APPROACH SURFACES
EL. 2537
200'
1700'
6730' X 100'
CLEAR ZONE
R/W 5
R/W 23
EL. 2551
CLEAR ZONE
10,000
8,000
6,000
4,000
2,000
0
0
2,000
4,000
SCALE 1" = 3,000' H.
1" = 100' V.

Figure 2-10

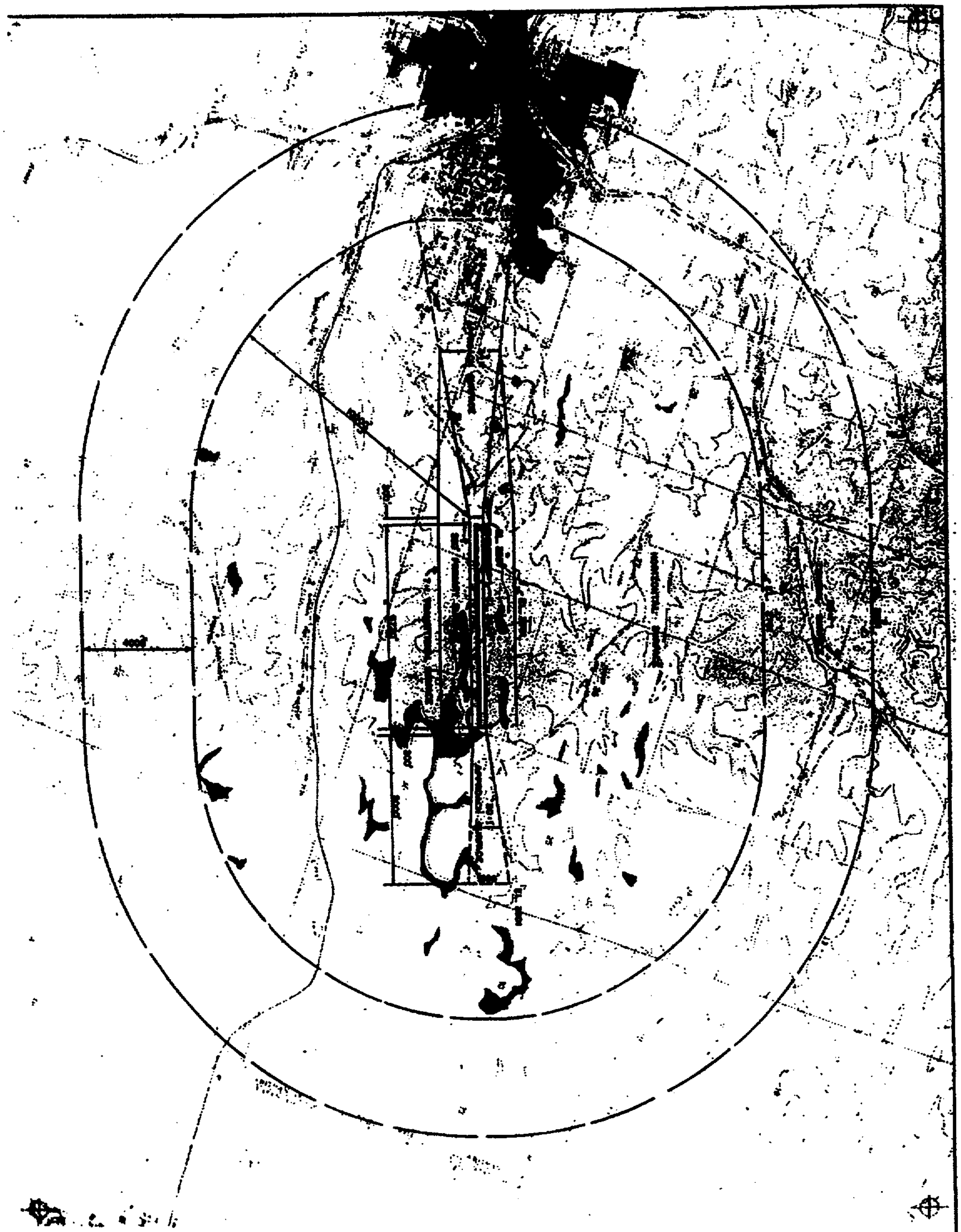


Figure 2-11

The Pullman-Moscow Airport has been in existence for over 30 years during which period the reactor facility has also been in existence. The airport does not keep any records on the number of aircraft that take off and land. However, the only commercial airline using the airport, Horizon Air, has 2,050 scheduled flights into the Pullman-Moscow Airport per year. All private aircraft are quite small and the airport runway and landing aids will not accommodate large passenger planes. Appendix 2-B to this SAR analyzes the probability and effects of a commercial airplane crashing into the WSU facility. It is estimated that over a million aircraft have landed or taken off from or to the west since the reactor facility has been in existence. The results of the analysis in Appendix 2-B demonstrates that an aircraft crash precipitated significant reactor accident represents an insignificant hazard.

2.2.3 Analysis of Potential Accidents at Facilities

There are no facilities located near the reactor facility other than the airport covered in section 2.2.2 that could create a potential accident at the facility.

2.3 Meteorology

2.3.1 General and Local Climate

Pullman is situated at latitude 470 north of the equator and consequently is about midway between the equator and the North Pole. From May to August when the sun remains above the horizon from 14 to 16 hours a day, Pullman receives more solar radiation than does the equator. In December, the sun rises only about 20° above the southern horizon at noon and is in the sky only about eight hours. Therefore, the daily accumulation of solar radiation in winter is less for two reasons: 1) the days are shorter, and 2) the sun's rays, striking the earth at an angle, are spread over a larger area. Because of this great variation in energy intake, Pullman experiences pronounced differences in temperature and other weather conditions from summer to winter.

The latitude of Pullman is only one factor influencing the climate pattern at the site. Other factors are its location with respect to land and water areas, mountain barriers and prevailing winds. Pullman is approximately 480 kilometers inland from the Pacific Ocean, and the Cascade Mountains, which average more than two kilometers in height, separate Pullman from the coast. The combined effects of the distance from the ocean and the existence of the mountain barrier create a climate with a continental character. However, because the prevailing winds blow inland from the Pacific Ocean, winters are considerably warmer than otherwise might be expected 480 kilometers inland at a latitude of 47° north. Winters in Pullman are characterized by cloudy skies and frequent snowstorms. On the average, the sun shines in Pullman only about 30% of the time during the winter months.

During the summer months, the westerly winds weaken and continental climatic conditions prevail. Rainfall, cloud cover and relative humidity are thus at their minimum; the daily mean temperature and daily temperature variation are at their maximum. Summers in Pullman are characterized by warm, clear days and cool nights. On the average, the sun shines in Pullman about 80% of the time during the summer months.

2.3.2 Site Meteorology

2.3.2.1 General

Washington State University is located in eastern Washington in a dryland agricultural area known as the Palouse region. The climate of this region is moderate, being a transitional region between the Columbia Basin and the mountains of Idaho. Precipitation and temperature data at the Washington State University campus have been accumulated since 1893 by the Department of Agronomy at the school. This extensive backlog of data was utilized to prepare the temperature and precipitation tables given in this section. The wind data were obtained from a detailed analysis of wind velocity and direction data for the year 1953.

2.3.2.2 Wind Velocity and Direction

In order to obtain wind data that are relevant to the site, charts taken from a wind recorder located on top of Wilson Hall during 1953 were analyzed in detail. Wilson Hall is approximately 1.6 kilometers WSW of the site. The monitoring station was at an elevation of 824 meters and the reactor site at 808 meters. All more recent wind data are taken at the Pullman-Moscow Airport- which is about 3.2 kilometers ENE at an elevation considerably below the site and thus not valid.

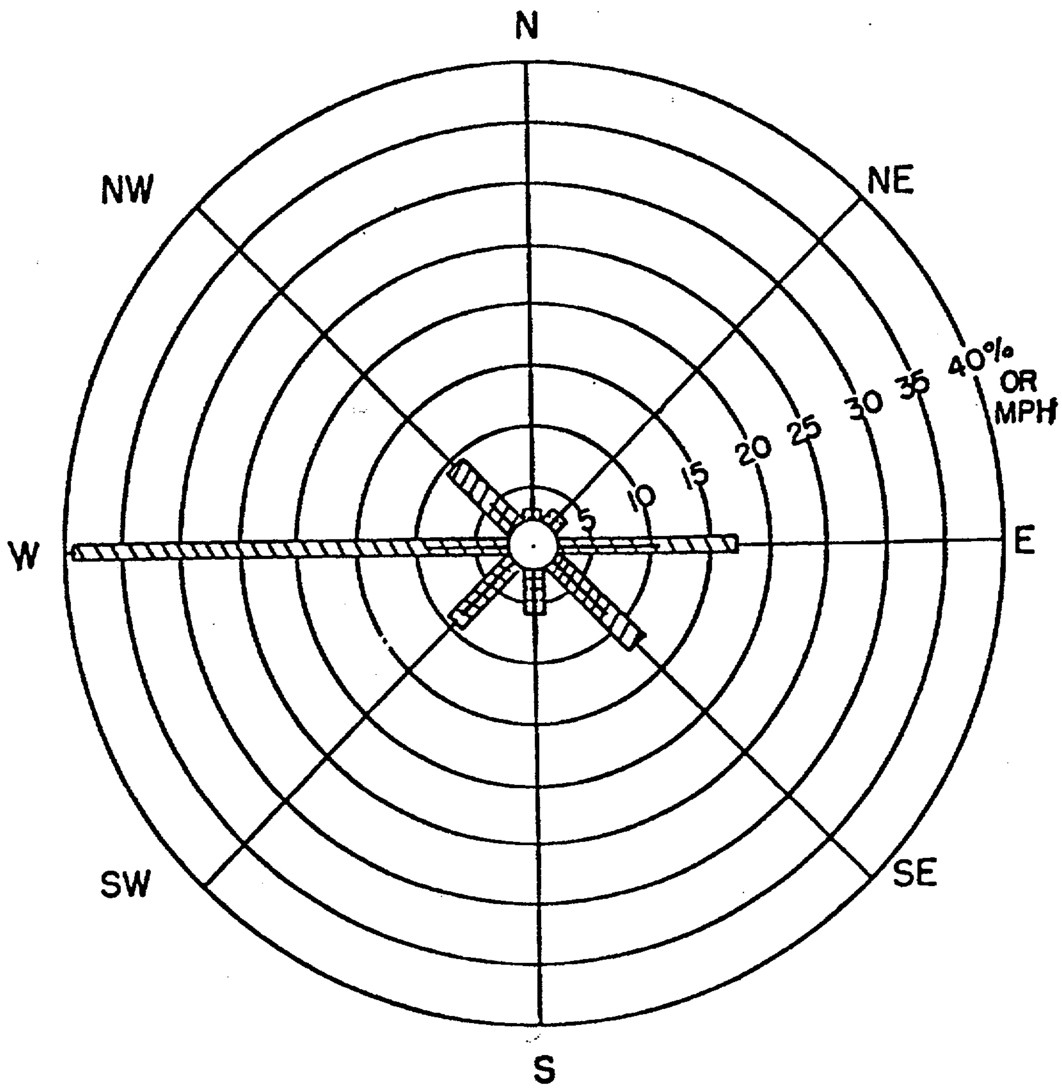
A wind rose indicating the frequency of occurrence of winds at the site is given in Figure 2-12. It is to be noted that the prevailing winds are from a westerly direction and blow over Pullman and the campus toward the site. The major population density is upwind from the site about 57% of the time and downwind only about 21% of the time. Furthermore, about 79% of the time the wind blows in a direction in which there are no inhabitants for about .8 kilometers around the site

The total number of hours of wind by direction and velocity is given in Table 2.3-1 and total time for winds of all velocities for each month is given in Table 2.3-2. These tables indicate that the average annual wind velocity is 16 kilometers/hr. Furthermore, the wind velocity was greater than five kilometers/hr 94% of the time and greater than eight kilometers/hr. 76% of the time. In general, one may conclude that there is almost always a light breeze blowing over the site.

Table 2.3-1

Total Number of Hours of Wind by Direction and Velocity, 1953

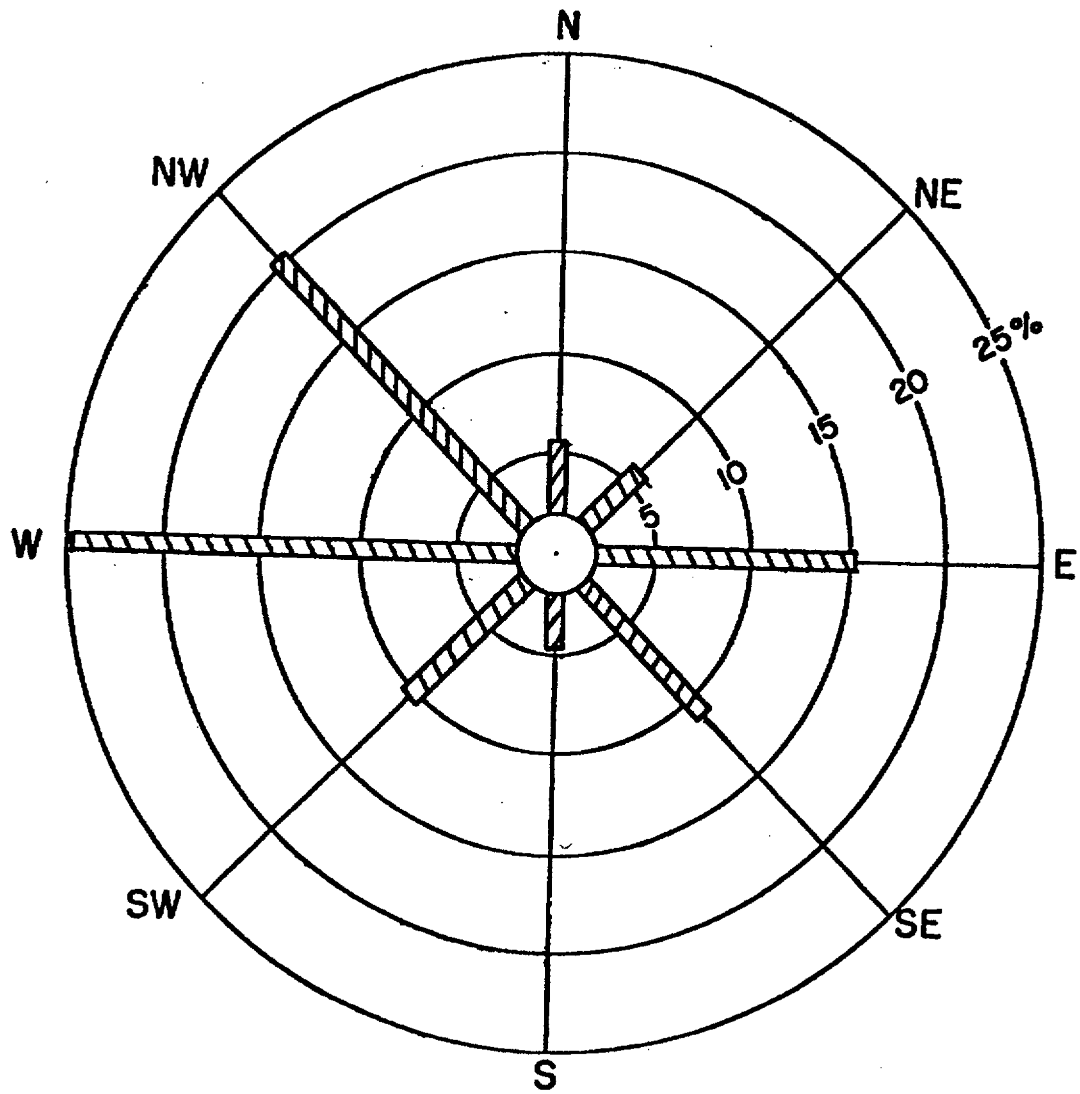
	DIRECTION							
Velocity Kilometers Per Hour	N	NE	E	SE	S	SW	W	NW
0-3	31.6	33.2	84.4	63.8	25.8	57.2	134.4	112.5
4-6	135.3	109.0	131.9	141.7	90.8	123.5	514.5	273.5
7-10	34.6	71.6	185.8	191.0	97.0	118.8	612.1	229.2
11-13	15.5	39.9	201.3	260.7	114.6	136.9	548.0	104.2
14-16	13.6	17.7	227.4	161.6	63.6	76.6	490.1	46.3
17-21	0.2	0.5	296.3	130.5	75.8	109.5	454.0	8.6
22-24			103.7	76.8	25.5	72.6	164.0	0.5
25-32		1.1	230.1	78.6	19.6	72.2	256.9	2.0
33			19.2	6.8	8.5	17.2	49.4	



Frequency Distribution and Mean Wind Speed at Site
from Direction Shown

(Hatched Area is Percent Occurrence -
Solid Line is Mean Speed in MPH)
- 1 M.P.H. = 1.61 Kilometer/Hr -

Figure 2-12



Frequency of Occurrence of Winds
of Less Than 3 Kilometers/Hr
from Direction Shown

Figure 2-13

TABLE 2.3-2

Total Time for Winds of all Velocities

Month 1953	N hours	NE hours	E hours	SE hours	S hours	SW hours	W hours	NW hours
January	2.3	3.4	43.3	123.7	93.7	135.3	243.4	16.0
February	18.0	5.7	52.4	87.0	63.2	56.9	278.7	90.0
March	14.2	10.0	146.5	123.5	31.5	81.8	308.3	19.4
April	13.9	24.0	152.2	45.5	31.6	73.2	310.4	69.0
May	23.9	29.3	157.4	51.1	32.2	56.8	296.6	88.2
June	27.0	38.1	64.5	25.5	18.1	68.8	347.0	106.5
July	58.0	57.3	81.3	63.8	17.3	31.0	257.5	124.1
August	24.4	39.3	96.3	92.2	54.1	48.3	267.1	110.6
September	28.5	34.4	141.0	81.5	46.7	75.1	265.5	38.4
October	15.0	37.0	208.5	106.0	34.6	54.8	225.0	62.2
November	5.5	3.9	257.2	134.9	33.5	36.3	158.7	37.5
December	1.5	0.1	112.4	157.6	71.1	78.4	284.9	36.3
Total	232.2	282.51	1,513.0	1,092.3	527.6	796.7	3,243.1	798.2
Percent	2.7	3.3	17.8	12.9	6.2	9.4	38.2	9.4
Av. Duration (hrs.)	.94	1.26	1.46	.85	.75	.66	1.47	.73

2.3.2.3 Precipitation and Temperature

The monthly average precipitation, monthly mean temperature and monthly mean daily variation from minimum to maximum temperature at the site are tabulated in Table 2.3-3. The seasonal variations depicted in this table are a graphic representation of the climatic conditions that prevail at the site as previously described.

2.3.2.4 Temperature Inversions

Quantitative data on temperature inversions in the vicinity of the site are non-existent. The closest points for which inversion data are available are at Spokane and Richland. However, the meteorological conditions at these two cities are significantly different from those at Pullman, making these data inapplicable. The frequency distribution of winds of less than three kilometers/hr. is depicted in Figure 2-13. These low-velocity winds blow only about 6% of the time, whereas winds in the five to seven kilometer/hr. range blow about 18% of the time.

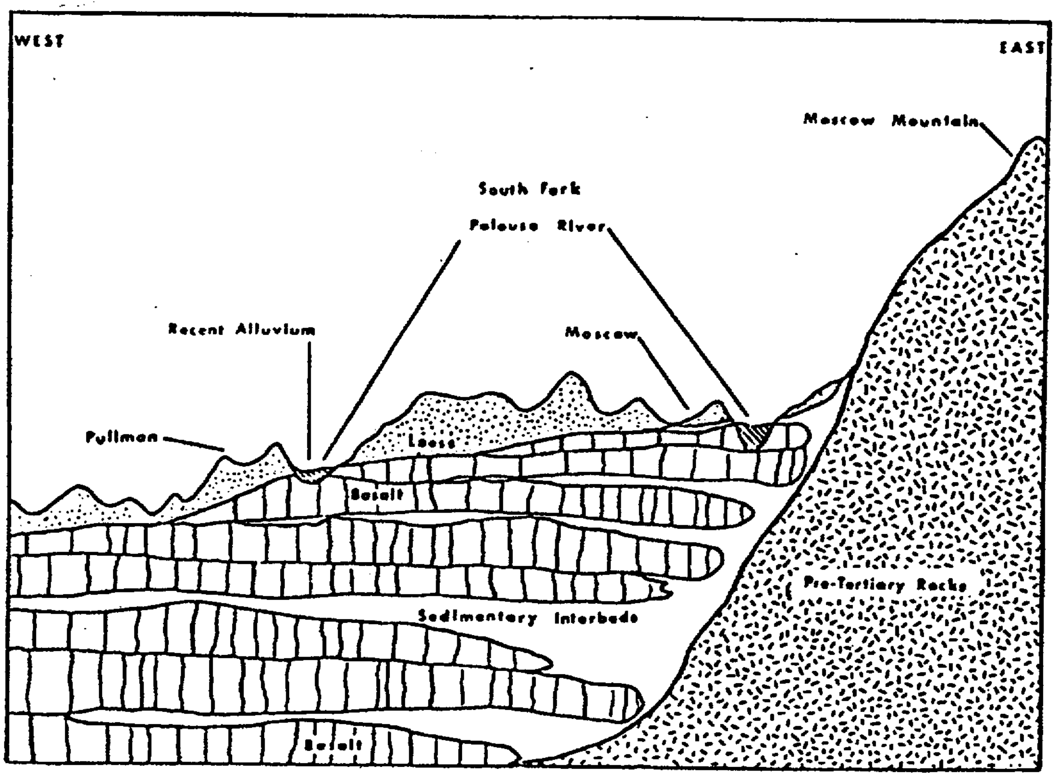
If the assumption is made that a temperature inversion can only be maintained with winds of below three kilometers/hr., then inversions could occur only about 6% of the time. The distribution of the low-velocity winds further indicates that the population center west of the site would be downwind only about 22% of the time during which inversions could possibly occur.

TABLE 2.3-3

Monthly Average Precipitation, Daily Mean Temperature,
And Mean Daily Minimum To Maximum Temperature Difference,
1893-1970

Month	Precipitation in Centimeters	Daily Mean Temperature in Degrees C	Mean Daily Minimum to Maximum
January	6.78	-2.6	5.7
February	5.33	0.2	6.7
March	5.38	3.8	8.4
April	3.78	8.5	10.8
May	3.71	12.7	11.8
June	3.91	15.4	12.7
July	0.99	19.9	15.7
August	1.32	19.1	15.3
September	2.74	14.3	13.0
October	4.85	10.0	10.4
November	6.27	3.2	6.7
December	6.96	0.1	5.7

Annual Total Precipitation - 49.50 centimeters
Annual Average Temperature - 8.7°C
Annual Average Difference between Minimum and Maximum Temperatures - 10.2°C



Schematic geologic section through the Pullman-Moscow basin (from Ichimura, 1978).

Figure 2-14

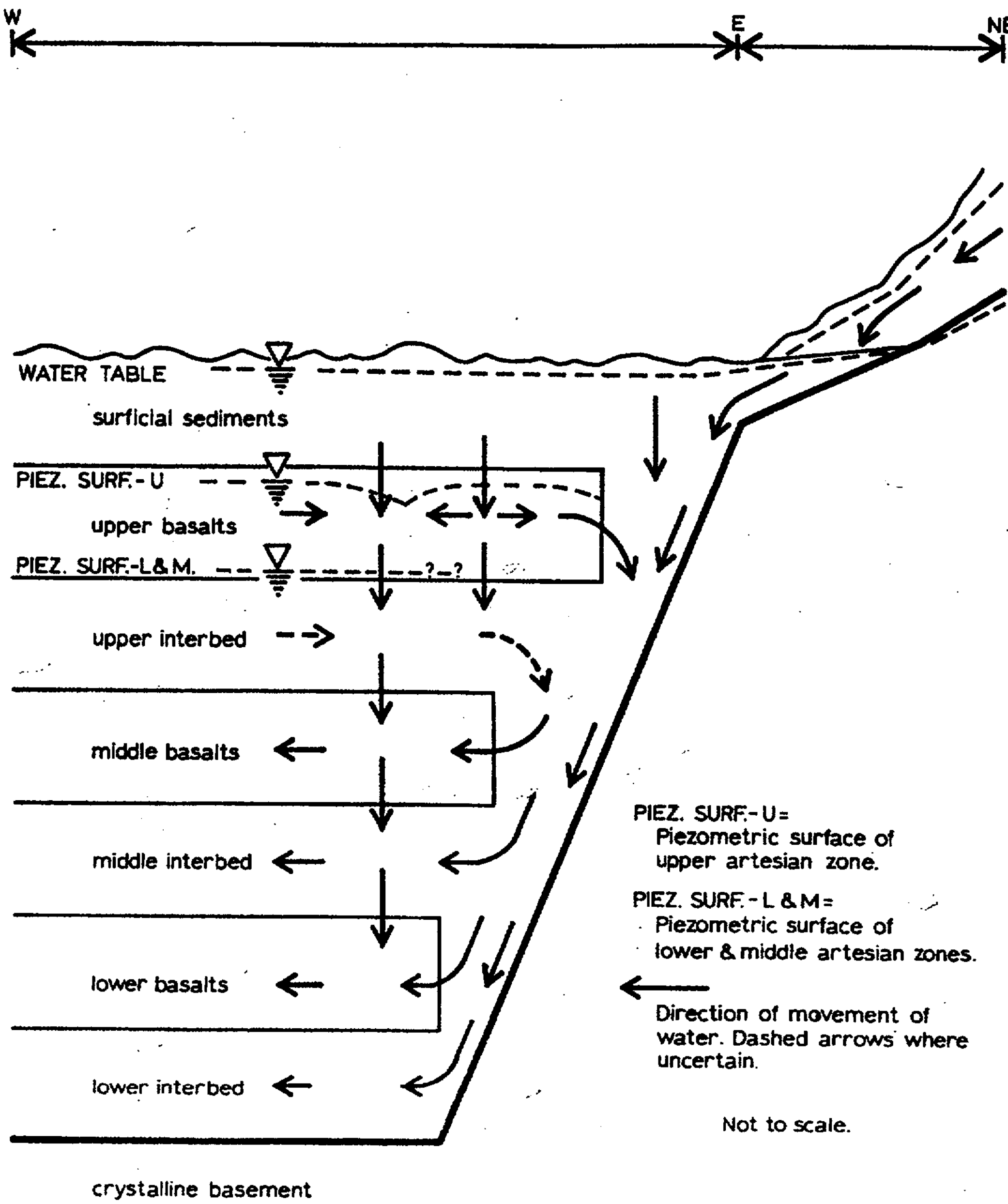


Figure 2-15
Aquifer Recharge Diagram

2.4 Hydrology

The main aquifers in the Pullman area are associated with the Latah Formation interbeds between basalt flows, as shown in Figure 2-14. Horizontal migration within an aquifer may also occur in the vesicular or porous top of the basalt layers. The cities of both Pullman and Moscow obtain their water from deep aquifers over 200 meters below the surface. Carbon-14 dating of the water from the deep aquifers indicates that no measurable recharge has occurred in recent times. Accordingly, it is believed that a layer of impervious basalt about 100 meters below the surface prevents the downward migration of surface waters.

Recharge of the shallow aquifers is believed to occur at the eastern end of the Moscow-Pullman basin where the basalts contact the pre-Tertiary Moscow Mountain Formation (as shown in Figure 2-15). Additional recharge also occurs by infiltration from streams and precipitation waters. However, surface waters percolate slowly downward due to the high water retention capacity of the Palouse Formation as well as the thickness of such soils. Accordingly, liquids discharged at the reactor site in the event of an accident will not enter the local aquifer. In addition, there are no rivers or streams within one kilometer of the site.

In February of 1955 Mr. J.J. Mundorff, the District Geologist for the USGS wrote the following report concerning the hydrology and geology of the Pullman area.

> Reference is made to your letter of January 28 regarding the hydrology and geology of the area around Pullman, with particular reference to our investigation in the vicinity of Pullman.
>
> During the course of this investigation, conducted in cooperation with the State of Washington, Department of Conservation and Development, Division of Water Resources, we have obtained considerable data on the geology and on ground-water conditions in this area. It will be several years before the investigation of the entire county is completed and a report prepared, but we do plan on completing a preliminary report on the Pullman area at a much earlier date.
>
> Two geologic units of formations underlie the Pullman area. These are the Columbia River basalt and the Palouse formation. The Palouse formation, which is composed of silt and clay, ranges from a few feet to about 150 feet in thickness and forms a mantle over most of the area. The basalt lava flows, which underlie the Palouse formation, are nearly horizontal. However, the lava evidently was extruded at many different places and individual flows are not continuous throughout the area. For this reason the original upper surface of the basalt may have been quite irregular even though the individual flows are nearly horizontal. In addition to this primary irregularity, erosion of the basalt prior to deposition of the Palouse formation increased the irregularity of the upper surface of the basalt. The elevation of the basalt surface ranges from about 2,340 to 2,550 feet in the immediate vicinity of Pullman.
>
> Ground water in the Palouse formation and the basalt occurs under water table (unconfined) conditions at elevations above 2,300 feet. The water table in the unconfined aquifers generally reflects, in a modified way, undulations of the topography. In the valleys, the water table is only a few feet below the surface; in the uplands it generally ranges from a few feet to 100 feet below the surface. At a

few places, small bodies of ground water may be perched above the regional water table.

Ground water in the basalt (and at places in sand layers interbedded with the basalt) below an elevation of about 2,300 feet is confined. Aquifers within this zone supply the city, college and many domestic users. The piezometric surface on the confined ground water apparently is nearly horizontal for a distance of several miles in every direction from Pullman. This piezometric surface has been declining at a rate of 1 to 2 feet per year for many years and at the present time is at approximately an elevation of 2,325 feet. As the lowest point on the water table in the area about Pullman is about 2,340 feet, in the valley of the South Fork of the Palouse River, it seems apparent that everywhere in the area around Pullman, except possibly northwestward down the South Fork valley, the water table has a greater head than the head on the deeper aquifers. This is significant because it means that water reaching the water table can percolate downward into the deeper aquifers unless prevented by some intervening, completely impermeable barrier. Study of a large number of well logs, and information obtained in mapping the geology indicates that there may be local impermeable barriers, such as hard unbroken basalt, or clay lenses, but there is no general impermeable blanket under the entire Pullman area which separates the water table aquifers from the deeper aquifers. Instead, it is believed that a large part of the recharge to the deeper aquifers may occur by slow downward leakage from the overlying water table aquifers.

Travel of Water through the Palouse formation is by slow percolation in the small pores between the silt grains. The rate of percolation is not known but hydrographs of water levels in observation wells show very pronounced season trends with a lag of a few weeks to a few months. The time required for water falling on the surface of the earth to reach the water table in the Palouse formation may range from days to months. However, because of the small amount of precipitation during summer months, there usually is little or no recharge during these months. Most water entering the ground during this period is used to replace lost soil moisture and is consumed or transpired by vegetation, or evaporates directly from the soil. Thus a spill of radioactive material during the summer might not be carried downward to the water table until fall or winter.

Water moving downward through a basalt lava flow moves through joints and other fractures and, depending on the size of the opening, difference in head, and other factors, may move very slowly or very rapidly. The most permeable zones in the basalt apparently are at or near the contacts of two successive lava flows, where the later low has incompletely filled the irregular surface of the underlying lava flow The fact that the piezometric surface in the vicinity of Pullman is nearly horizontal even though most of the pumpage is concentrated in the city suggests that the aquifer is quite permeable and that the ground water moves in the aquifer with comparative rapidity.

When the first wells were drilled into the deeper aquifer at Pullman, in the late 1880's and early 1890's, the piezometric surface was at an elevation of about 2,360 feet, about 20 feet higher than the floor of the valley. It is probable that there was some upward leakage from the deeper aquifer into the South Fork of the Palouse River. At the present time however the piezometric surface is below the water table in the valley at Pullman so that any natural discharge from the deeper aquifer would be at lower elevations to the west; possibly along the lower reaches of the South Fork, possibly as far west and south as the canyon of the Snake River.

Discharge of the water table aquifers, except for downward leakage as mentioned earlier, would be into the South Fork of the Palouse River.

2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Regional Geology

Pullman is situated in eastern Washington near the eastern margin of the Columbia River Plateau. In early Miocene times, the area was mountainous with a relief of over 1400 meters. These mountains, composed mostly of pre-Cambrian sedimentary and metamorphic rocks and cretaceous granite, formed the basement rock across which the Columbia River basalts would flow during the Miocene epoch. These basalt flows were numerous as well as extensive and advanced from the west and the south into the region.

The basalts of the Columbia plateau are somewhat unique, in that, a large thickness of volcanic material accumulated in a relatively short period on the geologic time scale. The lava flows extended over a 160,000 square kilometer area in the short span of about three million years about 16 to 13 million years ago. The total thickness of the basalt varies from 1000 meters in the Pullman area to a maximum of over three kilometers in the Pasco Basin. Individual flows were enormous and involved on the order of 300 cubic kilometers of lava. The source of the immense amount of heat needed to create the lava flows is postulated to be a "hot spot" in the magma below the region. Some geologists believe that the "hot spot" remained stationary as the Pacific Plate moved west. This theory accounts for the young basalts in southern Idaho and the geothermal activity in Yellowstone National Park. The "hot spot" is thus postulated to presently reside under the Yellowstone Park region.

The basalt that flowed into the pre-flow terrain of the region progressively submerged the basement features and dammed up the well-established drainage systems. Numerous lakes were created along the margin of the growing basalt plateau. Weathering of the exposed basement uplands produced detritus materials which rapidly filled in the temporary Miocene lakes established by the advancing basalt. Such lacustrine deposits were subsequently buried by flows from renewed basaltic eruptions triggering a repetition of the accumulation cycle. The solidified lava flows were nearly horizontal, however, the lava evidently erupting from many different locations at different times so that individual flows are not continuous across the plateau. The original upper surface of the basalts were probably quite rough but very low in relief.

At the end of the outpourings of the lavas of the Columbia River basalt in early Pliocene times, mild folding of the basalt began. The folding continued through middle and late Pliocene and into Pleistocene time. Deformations in this age include the Cascadian orogeny which greatly affected the climatic conditions of the region. The main tectonic events during this period include the uplift of the Cascade Range, Oregon Coast Range, Olympic Mountains and Blue Mountains; the downwarping of the Lewiston Grade, the Snake River Region and the

Walla Walla Plateau; a slightly westerly increasing subsidence of the Columbia Plateau; the isostatic depression of the Pasco Basin; and, block faulting of the Great Basin and Payette section. Volcanic eruptions accompanied the deformations, particularly in the Middle Cascades, giving rise to the volcanic peaks of the Cascade Range.

Following the cessation of the major igneous activity in early Pliocene times, the basalts and lacustrine deposits became subjected to moderate erosion as drainage patterns began to develop. This initiated the dissection of the plateau surface. During the Pleistocene epoch, the modified surface was capped with the loess of the Palouse formation and produced the rolling-hill topography of the region. The most significant geological event during the past million years is the Spokane flood at the end of the Ice Age. The advancing ice sheet dammed the Columbia, Spokane and Clark Fork Rivers. The water that was impounded behind the dams filled the tributary valleys for many miles.

The lake created by the damming of the Clark Fork contained an estimated 1000 cubic kilometers of water or about half the volume of present day Lake Michigan. When the ice dam at the mouth of the Clark Fork failed, the lake drained at an estimated flow rate of 15 cubic kilometers per hour. The incredible force of the massive flood scoured the Rathdrum Prairie and Spokane Valley creating the "Channeled Scablands" in the Sprague-Cheney area. Similar events during the Ice Age created the present features of the Columbia Plateau, including Grand Coulee.

The region has seen a very unique sequence of geological events, beginning with a vast series of lava flows. The lava flows were followed by a regional tilting of the land and by the deposition of a 30-60 meter layer of wind blown silt. The great glacial lake formed by the damming of the Clark Fork and the destructive flood created by the sudden release was the final event that brought this area to its present character.

2.5.2 Site Geology and Geologic Hazards

The Columbia River formation in the Pullman area is approximately 1000 meters thick and consists of alternating layers of basalt and the silts and clays of the Latah formation. A geologic cross-section of the Pullman area is shown in Figure 2-16. The Palouse Formation Soil at the site is 35 to 55 meters thick. Structurally, the layers of basalt in the Pullman area have not been disturbed since their disposition. The major movements in this section of the Columbia Plateau have been the Lewiston downwarp and the westerly subsidence.

No known geologic hazards, such as Karst terrain, cavernous conditions, tectonic depressions, surface or subsurface subsidence or uplifts, or active volcanoes, are present at the site or in the immediate vicinity of Pullman. Also, there are no conditions present which could produce rockfalls, avalanches, floods, tsunamis, mud flows or permafrost at the site.

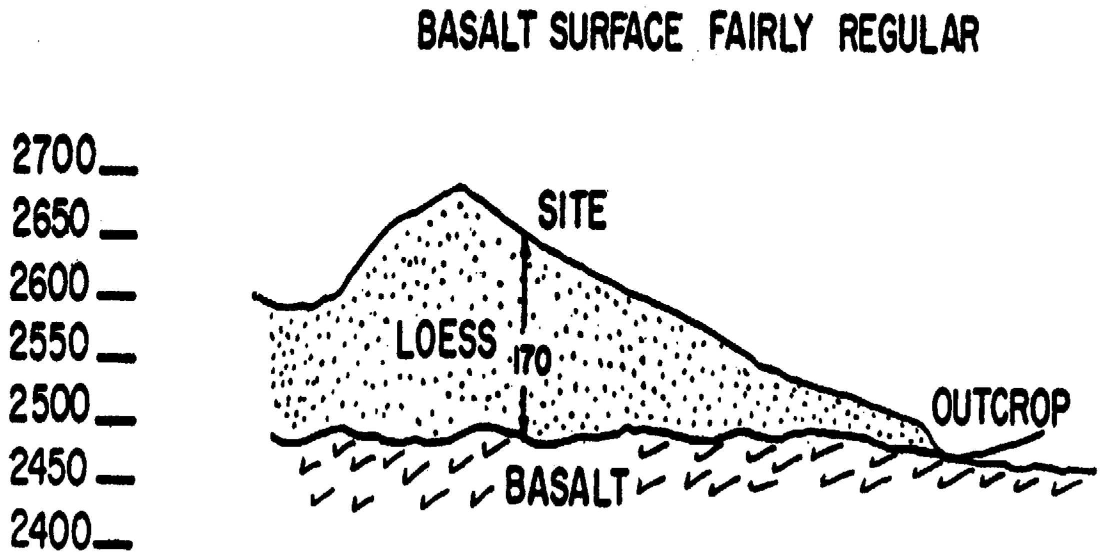


Figure 2-16
Geologic Cross Section of Site

In March of 1955 James H. McLerran, an Engineer for the WSU Division of Industrial Research made an examination of the then proposed reactor site and wrote the following report:

> The soil conditions at the site are rather typical of soils in the vicinity of Pullman. They are derived from the Palouse formation. This material is, in large, aeolian material with important depositional differences with depth. The surface soils are loose, silty, and permit rapid percolation. Below this are usually more compact layers due, in part, to the process of illuviation. This more compact underlying zone retards percolation and the water then moves laterally through the soil.
>
> The soils of the area belong in the Palouse Catena and as the site is on the South slope of the ridge the soil is probably of the Palouse Series. This soil exists from near the top of the ridge to the proximity of the drainage channel to the south that runs southeast to the South Fork of the Palouse River.
>
> Near this drainage channel the soil will change to one that has within its profile a well developed A_2 horizon which, due to its extreme silty nature, is extremely permeable. This allows for rapid lateral movement of water through the soil.
>
> The soil conditions outlined above indicated that vertical percolation will not be a problem in the area but that lateral subsurface water flow is predominant. Rapid lateral movement of water will occur in the soil along the drainage channel. However, it should take several days before any water moving from the site would reach this area and, in the meantime, in case of a harmful discharge preventive measures could be taken to intercept the water zone.

In March of 1955 Dr. W. Frank Scott of the Department of Geology at WSU made a geological examination of the proposed reactor site in order to determine the geologic conditions at and near the site. Dr. Scott's evaluation of the proposed site was as follows:

> The surface formation at and near the site is the Palouse loess, a windblown deposit of silt and clay of Pleistocene to recent age. This loess is underlain by the generally flat lying flows of the Columbia River basalt of probable late Miocene age. These basalt flows were subjected to some erosion prior to the deposition of the loess, so the loess-basalt contact is somewhat irregular.
>
> The reactor site is located at an elevation of about 2,650'. The nearest basalt outcrops located during my brief examination are tabulated below:

Outcrop No.	Dir. from Site	Dist. from Site	Elevation
I	S. 47°E.	2,300'	2,470'
II	S. 57°E.	3,000'	2,471'
III	S.	3,700'	2,486'
IV	S. 38°W.	4700'	2,440'

These elevations would indicate that basalt-loess contact at the reactor site should be at about 2,480', or 170' below the surface, if the preloess basalt surface is fairly regular. The enclosed drawing (Figure 2-16) shows a geologic cross section of the proposed site.

2.5.3 Seismicity

Realistic predictions regarding earthquakes, or earth shocks, as well as their frequency and severity, can only be based upon the seismic history of the area. Significant geological features, such as known slip-planes or faults, play an important role in the seismic history of any region. Thus, these features as well as past shocks must be taken into consideration in depicting the seismology of the site.

The overall earthquake activity of the State of Washington is shown in Figure 2-17 taken from the USGS publication "Seismicity of the United States, 1568-1989", USGSPP 1527, 1993. This drawing clearly depicts the fact that past earthquakes have generally occurred in the Puget Sound basin with an insignificant number occurring in the south eastern corner of the state where the facility is located. The overall significant earthquake events for the western USA with magnitude over 6 is shown in Figure 2-18 taken from the USGS National Earthquake Information Center web site. This figure clearly shows that the WSU reactor site region located in the south east corner of the State of Washington has not experienced any significant past seismic events.

The historic earthquake data for the region 200 km around the WSU reactor facility for a span of 43 years taken from the University of Washington Department of Geology web site contained a total of 166 events that spanned a magnitude range of 2.6 to 4.0. This data is summarized below:

Table 2.5-1

Eastern Washington Earthquake Distribution Data over 43 Years

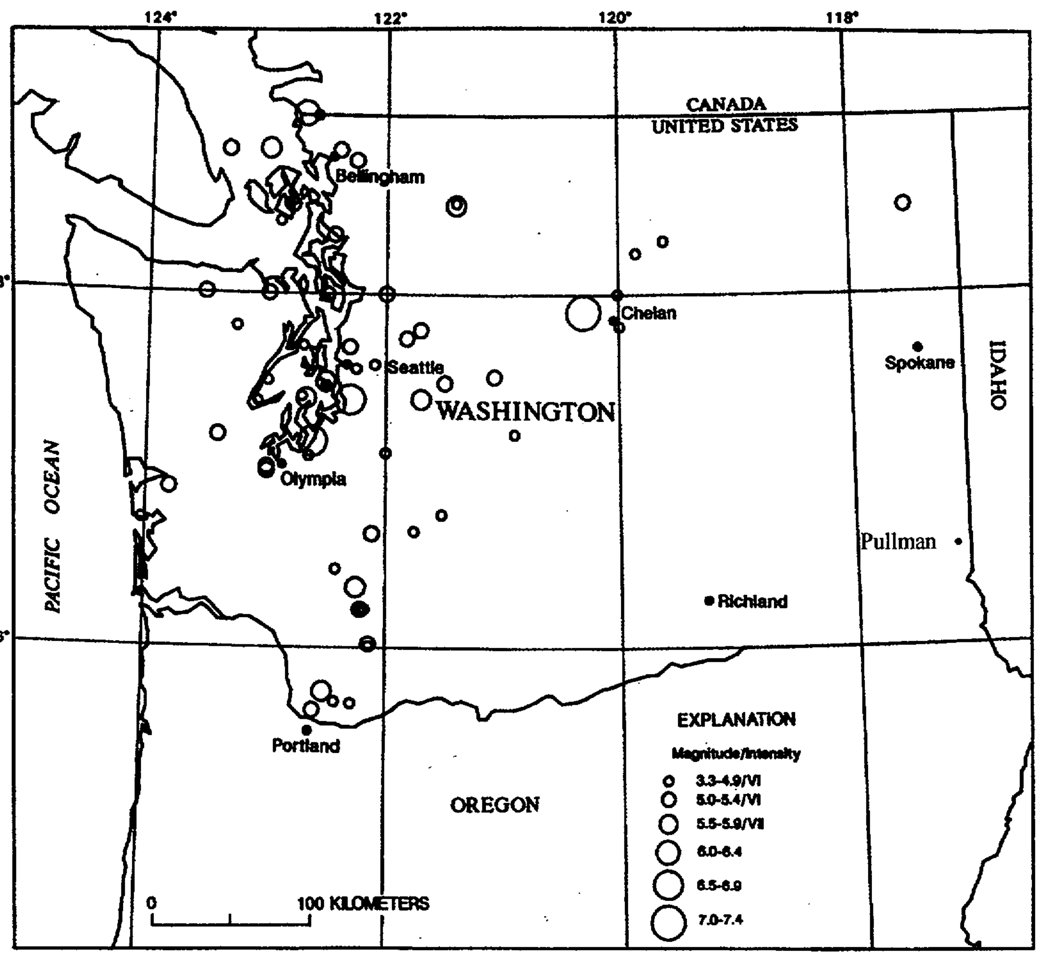
Magnitude	2.7 (2.6-2.8)	3.0 (2.9-3.1)	3.3 (3.2-3.4)	3.6 (3.5-3.7)	4.0 (3.8-4.0)
$T \cdot \Delta N$	93	25	19	17	12
$T \cdot N$	166	73	48	29	12
Frequency N	3.86	1.698	1.116	.674	.279
log(N)	.587	.230	.0478	-.171	-.554

$T \cdot \Delta N$ = Events/Magnitude range, $T \cdot N$ = Total events, max to this magnitude range, T = Period in years

It is a generally accepted fact that the distribution of seismic events over time which is called the "recurrence relationship" may be represented by an equation of the form

$$\log(N) = a - bM$$

WASHINGTON



Earthquakes in Washington with magnitudes ≥ 4.5 or intensity ≥ VI.

Figure 2-17

wmm.cc, 5<=mag<6 used to compute hazard

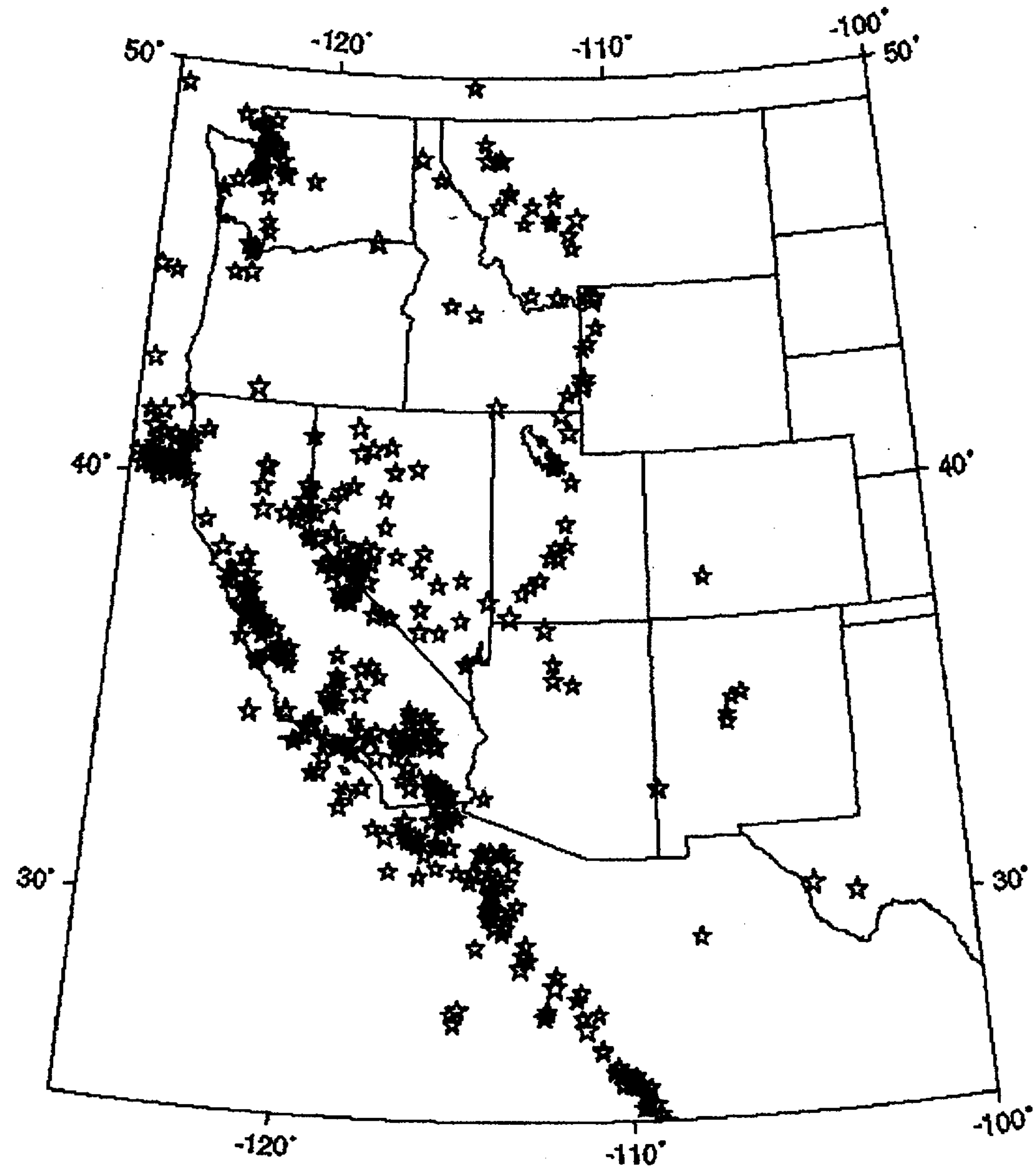


Figure 5b. The WUS catalog used to compute hazard, 5 <= magnitude < 6.

Figure 2-18

where N is the number of shocks of magnitude M or greater per year and a and b are constants (7). Using a HP 11 calculator to perform a least squares fit the results were that this data fit the recurrence relationship log (N) = 2.82 - 0.84 M. An analysis of the 64 events taken from the USGS web site over a span of 25 years yielded a similar result of Log (N) = 3.09 - 0.968 M. The result with more events is believed to be more accurate. Accordingly the maximum expected quake once in 50 years within 200 km of the WSU facility is 5.4 and one in 100 years 5.7.

An Isoseismal map for the most recent large earthquake (April 13, 1949) that has occurred in the State of Washington is given in Figure 2-19. It is to be noted that the Moscow-Pullman area is situated in the II-III intensity region on this map.

The overall geological features of the Pullman area are described in the section on geology and will not be repeated here. In this section we are concerned with the geological features of this area that could possibly produce earthquakes. The significant faults within a 100-mile radius of the site are shown in Figure 2-20. From this drawing, it is evident that there are no known significant faults in the immediate vicinity of Pullman. The closest active fault is the Walla Walla fault, some 70 miles from Pullman. The closest inactive faults are the Vista and Wilma faults associated with the Lewiston Downwarp 37 miles south of the site.

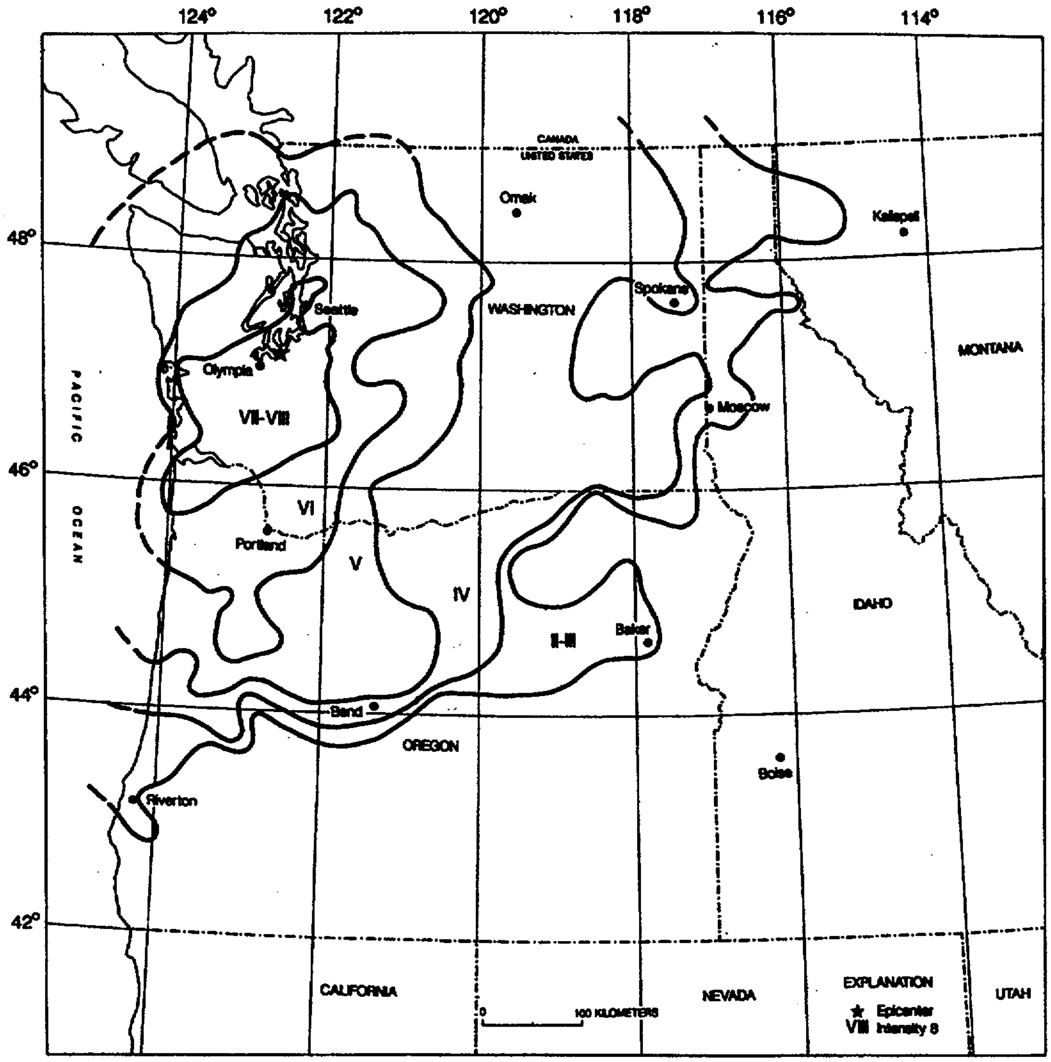
Historically, the seismic activity within 200 miles of the site is low, with infrequent earthquakes of low intensity (magnitude). The occurrences of earthquakes within 200 miles of Pullman are listed in Table 2.5-2. It is noteworthy that only two shocks have occurred at Pullman in recorded history, both of them of low intensity.

Based on the geology of the Pullman area and the past seismic activity, the probability of the occurrence of significant earthquakes in the future can be said to be very small.

2.5.4 Maximum Earthquake Potential

The recurrence relationship for the site region as previously mentioned estimates that the maximum 50 year event for the site region is magnitude 5.4 and for a 100 year period 5.7. Modern seismic hazards analysis is not based on event magnitude but rather on maximum expected peak ground acceleration in the area. It will be shown in the next section that the above estimated maximum events are consistent with the USGS National Earthquake Information Center 1997 National Seismic Hazards Mapping Project evaluation for the site region.

EARTHQUAKES IN WASHINGTON



—Isoseismal map for the Puget Sound, Washington, earthquake of April 13, 1949.

Figure 2-19

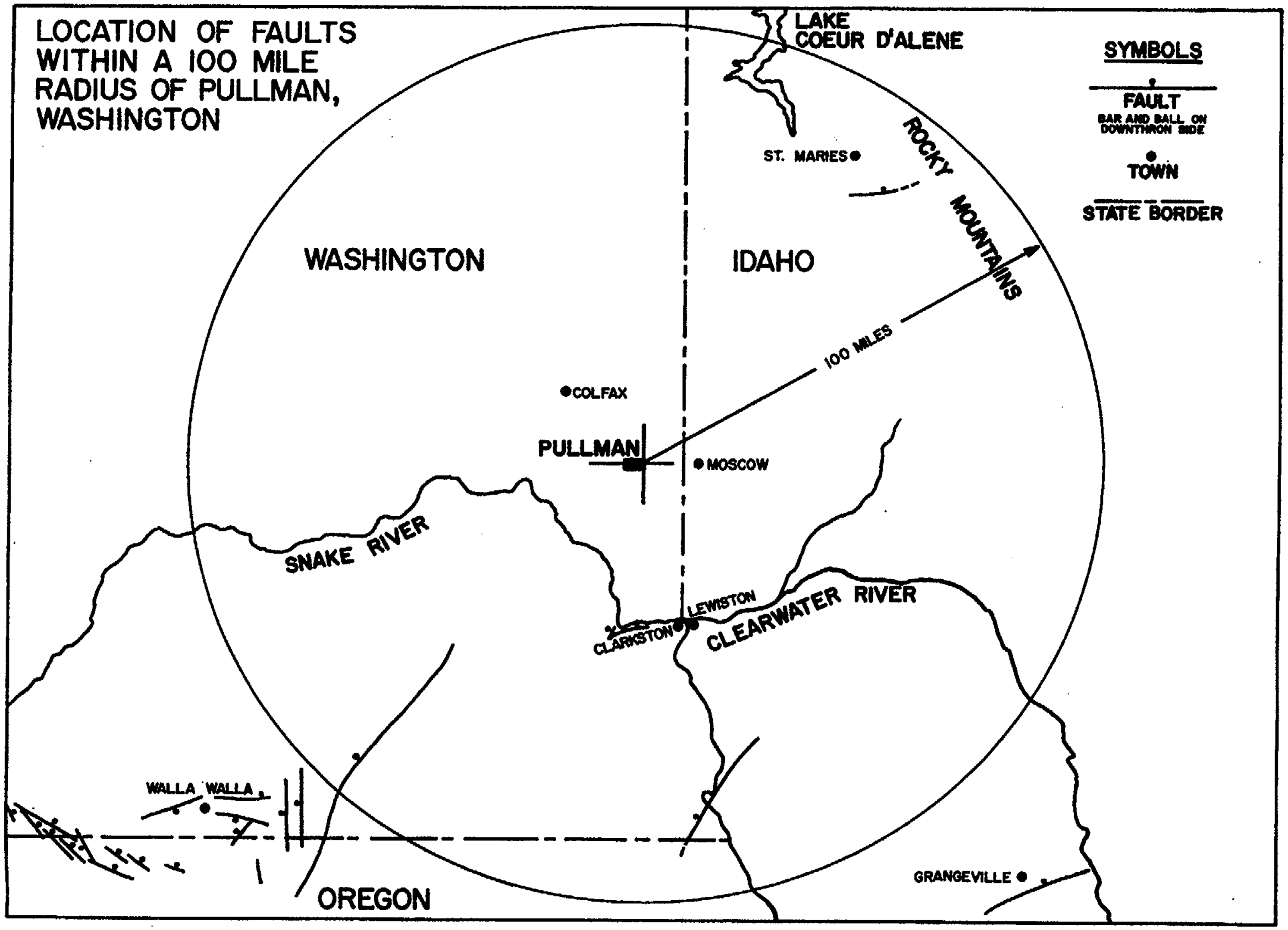
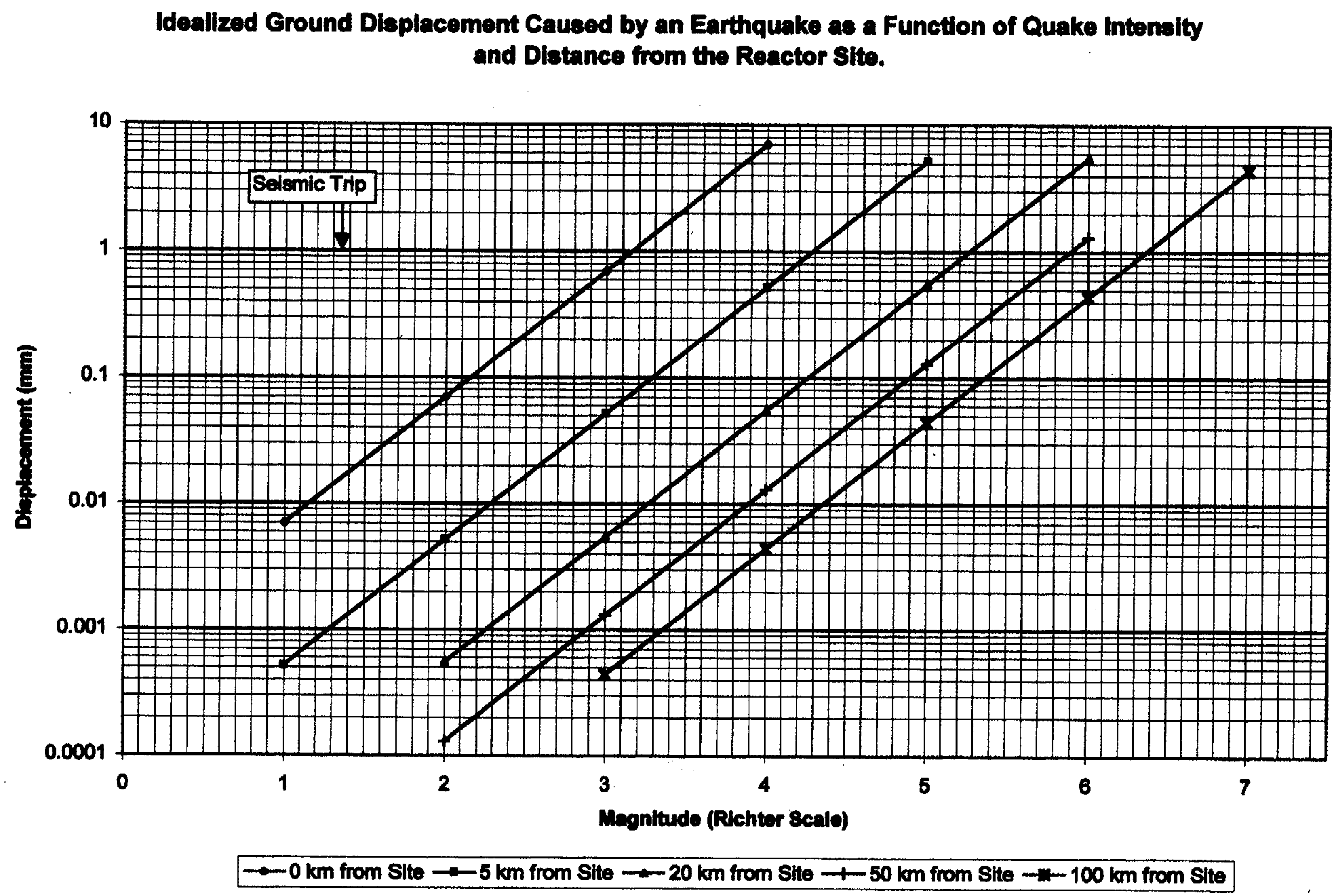

LOCATION OF FAULTS
WITHIN A 100 MILE
RADIUS OF PULLMAN,
WASHINGTON
LAKE
COEUR D'ALENE
SYMBOLS
FAULT
BAR AND BALL ON DOWNTHROWN SIDE
TOWN
STATE BORDER
ST. MARIES
ROCKY MOUNTAINS
WASHINGTON
IDAHO
100 MILES
COLFAX
PULLMAN
MOSCOW
SNAKE RIVER
LEWISTON
CLARKSTON
CLEARWATER RIVER
WALLA WALLA
GRANGEVILLE
OREGON

Figure 2-20



Idealized Ground Displacement Caused by an Earthquake as a Function of Quake Intensity and Distance from the Reactor Site.
Seismic Trip
10
1
0.1
0.01
0.001
0.0001
Displacement (mm)
0
1
2
3
4
5
6
7
Magnitude (Richter Scale)
0 km from Site
5 km from Site
20 km from Site
50 km from Site
100 km from Site

Figure 2-21

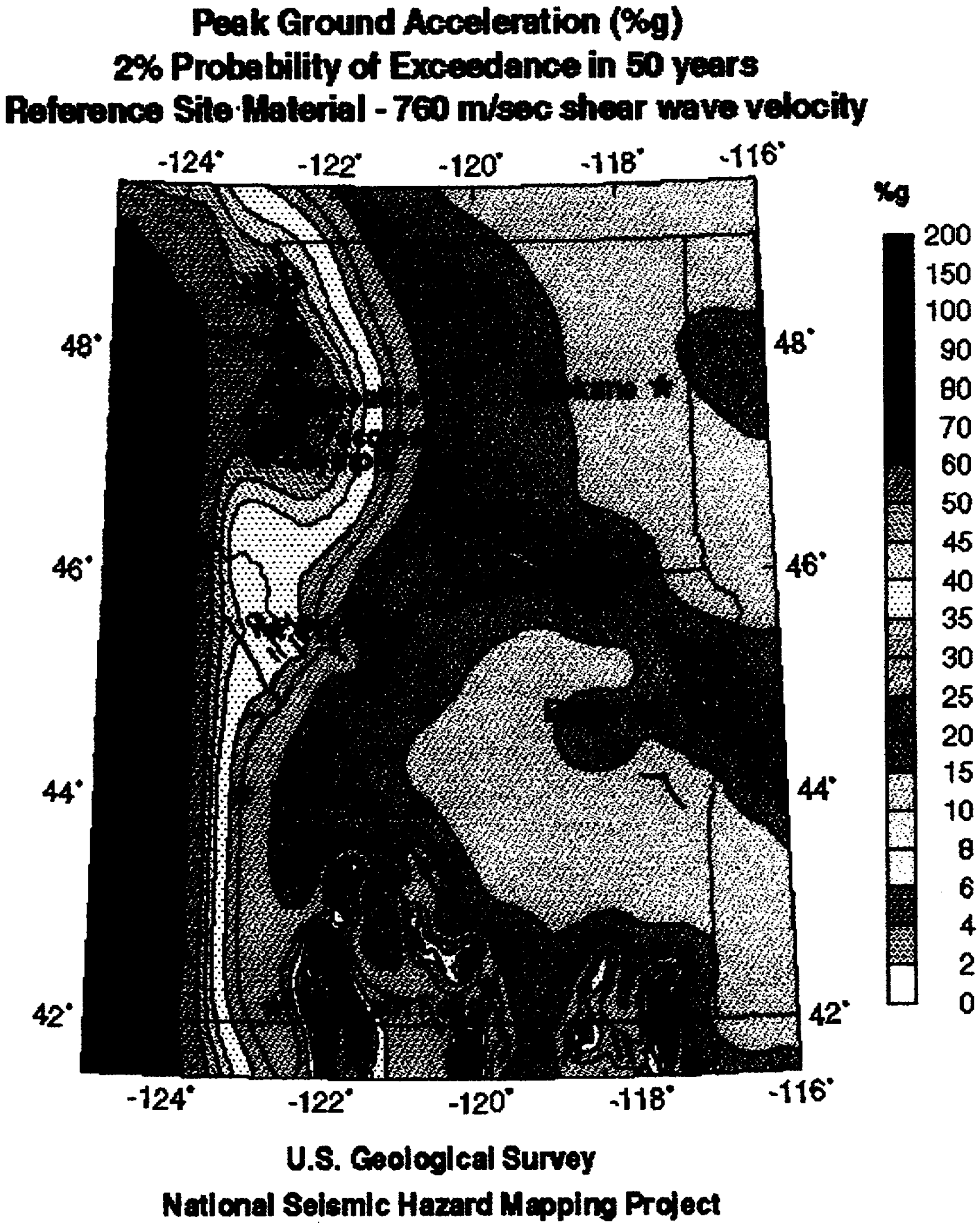

Peak Ground Acceleration (%g)
2% Probability of Exceedance in 50 years
Reference Site Material - 760 m/sec shear wave velocity
-124°
-122°
-120°
-118°
-116°
48°
46°
44°
42°
%g
200
150
100
90
80
70
60
50
45
40
35
30
25
20
15
10
8
6
4
2
0
U.S. Geological Survey
National Seismic Hazard Mapping Project

Figure 2-22

2.5.5 Vibratory Ground Motion

The underlying rock formation at the site which is as shown in Figure 2-14 basically Columbia River basalt which is quite stable, rather rigid, and not subject to faulting from minor earthquakes. Thus the ground motion would simply be that for a typical seismic event of the specified magnitude. A magnitude 3.0 or less earthquake would produce an insignificant amount of ground motion that would not even be noticed. The experts indicate that a magnitude 3.5 earthquake would only be felt by a very discriminating person and a magnitude 4.0 would be noticed by most individuals.

The ground displacement at the WSU reactor site in relation to the quake intensity and distance from the site is shown in Figure 2-21. The graph also shows the quake intensity at which the seismic switch would activate and shut down the reactor. The sensitivity of the seismic switch is such that the reactor would automatically shut down at a very low amount of ground motion significantly below that which could cause any damage to the facility. A quake of intensity 3.2 at the site would ideally activate the seismic switch with greater intensities required as distance increases out to a magnitude 6.4 at 100 km.

Figure 2-22 produced by use of use of the custom hazards mapping option at the USGS NEIC National Hazards Mapping Project web site shows the expected peak ground acceleration for the WSU reactor site region. This map shows that the site lies within a region with a peak ground acceleration of .05 to .06 g with a 10% probability of exceedance in 50 years. The 1997 map is the most accurate modern estimation of seismic hazard in the site region in terms of vibratory ground motion (7).

The recurrence relationship based on the analysis of 166 past events for the site region predicts a maximum 5.7 magnitude event in a 100 year time period. If we assume the worst case and convert this magnitude to MMI intensity, VII (2), we estimate a peak ground acceleration of about .15 g at the epicenter of the event. All significant events are located over 100 km from the site which will significantly attenuate the ground motion transmitted to the site. Applying an appropriate attenuation factor (9,10) yields an acceleration not exceeding .05 g at the site which is consistent with the USGS 1997 seismic hazards map of the site region. Thus the expected level of ground motion in the event of an earthquake would have minimal effect on the reactor facility.

2.5.6 Surface Faulting

As indicated in section 2.5.3, there are no significant faults within 8 kilometers of the site. Furthermore, because of the geology of the site and the insignificant earthquake potential at the site, the maximum likely seismicity would not create any new surface faulting at or near the site.

2.5.7 Liquefaction Potential

The soil at the site is the Palouse Loess material underlayed by the basalts of the Latah formation. Due to the nature of the local geology, soil type, and site topography, soil liquefaction at the site is an insignificant consideration.

2.5.8 Historic Earthquake Data for Pullman Area

Modern earthquake data is very complete including the recording of insignificant earth tremors that would not be noticed by even the most sensitive individual. The very small tremors are however of significance in certain instances in predicting future significant events. The data since 1979 was abstracted from the online Earthquake catalog of reference at the University of Washington.

The tabulated earthquake data includes some very small quakes within 40 miles of Pullman of unknown origin. That is, there are no known significant faults in the Pullman area that could give rise to significant earthquakes. Modern geology has shown that the earth is not static but a dynamic system with significant crustal movement especially at plate boundaries. Thus it is not surprising that small quakes occur at infrequent intervals in the Pullman area which is on the eastern margin of the Columbia Plateau as discussed in section 2.5.1.

TABLE 2.5-2

Historic Earthquakes 1872-1997 Within 200 Miles of Site

Year	Date	Location of Epicenter	Approximate Distance from Pullman (miles)	Intensity at Epicenter*
1872-73	Dec. 16-Jan. 1	Walla Walla	70	Unknown
1874	Unknown	Yakima	155	Unknown
1875	May 6	Yakima	155	Unknown
1875	May 7	Yakima	155	Severe
1887	April 29	Walla Walla	70	Felt
1898	Feb. 22	Ellensburg	160	Felt
1906	Jan. 2	NE Washington	--	Felt over 200 square miles
1906	Nov. 2	Colville	130	V
1909	May 24	47.6N, 120.0W	140	Felt
1911	July 5	Ellensburg	160	V
1915	Dec. 10	Spokane	65	Felt
1918	Nov. 1	47N, 119.5W	105	V-VI
1920	Nov. 28--29	Spokane	65	Felt
1921	Sept. 14	Walla Walla	70	V-VI
1922	Jan. 31	Republic	150	Felt
1922	June 1	Spokane	65	IV
1922	Oct. 16	Hermiston, OR	120	III
1924	Jan. 6	Walla Walla	70	IV
1924	May 27	Walla Walla	70	IV
1926	April 11	Walla Walla	70	III
1926	April 23	Walla Walla	70	IV
1930	Sept. 3	47.3N, 117.8W	70	V
1935	Oct. 24	Ellensburg	160	Felt
1936	July 16	46.0N, 118.3W	70	VII
1936	July 18-20	Walla Walla	70	Felt
1936	July 30	Freewater, OR	80	VI
1936	July 30	Walla Walla	70	III-VI
1936	Aug. 4	45.8N, 118.6W	115	V
1936	Aug. 28	Walla Walla	70	IV
1936	Nov. 17	Walla Walla	70	III
1937	Feb. 8	Walla Walla	70	III
1937	Feb. 9	Walla Walla	70	IV
1937	June 4	Walla Walla	70	IV
1937	June 17	Walla Walla	70	Felt
1937	Aug. 11	Spokane	65	Felt
1937	Sept. 20	Walla Walla	70	Felt
1938	May 9	Walla Walla	70	Felt

Year	Date	Location of Epicenter	Approximate Distance from Pullman (miles)	Intensity at Epicenter*
1938	May 24	Walla Walla	70	Felt
1938	Aug. 11	Milton, OR	80	VI
1938	Oct. 27	Milton, OR	80	VI
1939	Feb. 6	Ellensburg	160	Felt
1940	Jan. 6	Ephrata	120	Felt
1940	Nov. 14	47.7N, 121.5W	165	III
1941	Jan. 3	Pullman	0	Felt
1941	April 7	Republic	150	VI
1941	July 29	Spokane	65	Felt
1942	Nov. 1	48.0N, 116.7W	85	VI
1943	April 24	47.3N, 120.6W	110	VI
1944	Sept. 2	Walla Walla	70	IV
1945	April 29	47.4N, 121.7W	150	VII
1945	April 30	47.4N, 121.7W	150	VI
1945	May 1	47.4N, 121.7W	150	V
1945	Sept. 23	Walla Walla	70	IV
1949	Feb. 6	Wapato	155	III
1949	April 14	Pullman	0	Felt
1950	June 25	Cheney	55	IV
1952	Mar. 4	Spokane	65	V
1952	July 27	47.8N, 121.9W	155	IV
1952	July 29	47.8N, 121.9W	155	Felt
1952	Nov. 10	47.6N, 121.5W	165	Felt
1955	Feb. 6	Grand Coulee Dam	120	IV
1955	July 15	Soap Lake	120	IV
1955	Nov. 3	48.1N, 121.7W	170	V
1956	Feb. 24	Electric City	120	V
1956	Nov. 18	48.1N, 121.8W	165	Felt
1957	Feb. 11	47.5N, 121.7W	150	VI
1957	Nov. 1	47.0N, 121W	185	V
1958	Apr. 12	48N, 120W	150	VI
1958	Apr. 12	Electric City	120	IV
1959	Jan. 21	Walla Walla	70	IV
1959	Aug. 6	47.8N, 120.0W	145	VI
1959	Nov. 23	46.7N, 121.7W	140	V
1961	May 22	47.6N, 120.2W	145	IV
1961	June 28	Rocky Reach Dam	145	IV
1961	Oct. 31	48.4N, 120W	170	V
1961	Nov. 7	Spokane	65	Felt
1962	Jan. 15	47.8N, 120.2W	155	VI
1963	Jan. 25	La Grande, OR	105	III
1963	Dec. 22	48.3N, 119.3W	130	V
1964	Oct. 18	47.9N, 121.9W	155	IV

Year	Date	Location of Epicenter	Approximate Distance from Pullman (miles)	Intensity at Epicenter*
1966	Dec. 24	47.9N, 121.3W	155	III
1967	June 6	48.2N, 119.1W	125	IV
1969	Oct. 9	46.8N, 121.7W	140	VI
1969	Nov. 1	47.9N, 121.9W	165	V
1969	Nov. 10	48.5N, 121.4W	190	V
1971	Oct. 25	46.7N, 119.6W	105	IV
1974	July 14	47.6N, 120.7W	177	IV
1975	June 28	46.2N, 119.7W	125	III
1975	Sept. 18	47.8N, 118.2W	89	III
1975	Dec. 3	45.6N, 118.9W	113	III
1976	Apr. 13	45.24N, 120.2W	174	IV
1976	May 15	47.71N, 120.03W	151	III
1976	June 15	46.45N, 117.68W	31	III
1976	June 15	47.63N, 120.3W	160	III
1976	July 23	46.08N, 118.75W	87	III
1976	Aug. 30	47.62N, 120.18W	154	III
1976	Dec. 13	47.64N, 120.13W	153	III
1977	Jan. 27	46.94N, 119.59W	115	III
1977	Mar. 10	45.89N, 119.68W	132	III
1977	Apr. 21	49.12N, 117.67W	168	IV
1977	July 13	47.06N, 120.95W	179	IV
1978	June 27	46.94N, 121.14W	188	III
1979	Jan. 19	47.92N, 119.69W	144	IV
1979	April 8	46.0N, 118.42W	72	IV
1979	Jan. 1	47.90N, 119.68W	174	3.9
1979	Feb. 6	47.90N, 119.68W	174	2.8
1979	Feb. 17	46.15N, 119.91W	192	3.6
1979	March 1	46.03N, 118.90W	125	2.7
1979	March 15	46.53N, 119.96W	193	3.5
1979	April 24	47.71N, 119.43W	157	2.8
1979	July 22	46.83N, 119.40W	155	2.9
1979	Nov. 11	47.75N, 119.40W	162	2.7
1979	Nov. 12	46.91N, 119.58W	167	2.7
1979	Nov. 18	46.88N, 119.56W	167	2.8
1979	Nov. 21	46.91N, 119.56W	167	2.8
1979	Nov. 22	46.91N, 119.58W	167	2.6
1979	Nov. 24	46.91N, 119.56W	167	3.4
1979	Dec. 1	46.91N, 119.55W	165	2.6
1980	Jan. 3	47.88N, 118.15W	88	3.2
1980	Jan. 5	46.80N, 119.41W	155	2.9
1980	March 12	46.11N, 119.01W	131	2.6
1980	Nov. 7	46.93N, 119.46W	159	2.6
1980	Nov. 19	46.93N, 119.46W	159	3.3

Year	Date	Location of Epicenter	Approximate Distance from Pullman (miles)	Intensity at Epicenter*
1980	Dec. 3	46.91N, 119.35W	151	2.6
1981	Feb. 19	46.66N, 119.30W	148	2.7
1981	Feb. 27	46.93N, 119.55W	165	2.7
1981	April 23	47.61N, 119.88W	188	2.7
1981	July 10	46.28N, 118.43W	98	2.6
1981	Aug. 28	46.95N, 119.65W	172	2.6
1982	Feb. 18	47.65N, 119.73W	183	2.8
1982	March 22	47.85N, 119.93W	199	2.7
1982	Aug. 10	46.81N, 119.41W	156	2.6
1983	April 13	46.03N, 118.16W	69	2.6
1983	May 16	46.81N, 119.35W	151	2.6
1983	June 14	47.53N, 118.78W	118	2.6
1983	Sept. 10	47.88N, 118.33W	98	3.1
1983	Oct. 20	46.71N, 119.58W	167	3.4
1984	Jan. 13	46.26N, 118.13W	71	2.7
1984	Feb. 2	47.65N, 117.56W	64	2.6
1984	March 28	47.46N, 118.63W	108	2.6
1984	April 29	46.66N, 119.86W	187	2.8
1984	April 30	46.03N, 119.86W	190	2.8
1984	Aug. 14	47.11N, 118.76W	112	2.6
1984	Aug. 19	46.96N, 119.18W	140	2.7
1984	Aug. 23	46.66N, 119.45W	158	2.7
1984	Sept. 16	47.80N, 119.36W	161	2.7
1984	Sept. 17	47.80N, 119.36W	161	2.8
1984	Oct. 10	47.90N, 119.06W	143	3.0
1984	Oct. 28	47.93N, 119.58W	177	2.7
1984	Dec. 18	47.26N, 117.13W	26	2.8
1985	Jan. 14	46.80N, 118.28W	77	2.6
1985	Jan. 31	46.70N, 119.98W	195	2.7
1985	Feb. 2	46.71N, 119.98W	195	2.6
1985	Feb. 19	46.95N, 118.55W	97	2.7
1985	March 9	46.98N, 118.58W	99	3.3
1985	April 30	46.88N, 117.6W	31	2.7
1985	June 9	46.66N, 118.96W	124	3.2
1985	June 29	46.90N, 119.11W	135	2.7
1985	July 24	47.76N, 119.45W	158	2.7
1985	Aug. 24	46.21N, 117.90W	57	2.7
1985	Oct. 10	46.38N, 119.18W	141	2.8
1985	Nov. 18	46.25N, 119.61W	170	2.9
1985	Nov. 22	47.25N, 119.35W	153	3.2
1985	Dec. 3	46.15M. 110.60W	171	2.9
1985	Dec. 3	46.91N, 119.58W	167	2.6
1985	Dec. 8	46.93N, 119.55W	165	2.8

Year	Date	Location of Epicenter	Approximate Distance from Pullman (miles)	Intensity at Epicenter*
1985	Dec. 19	46.25N, 119.60W	170	2.8
1986	Jan. 16	46.25N, 119.61W	170	3.0
1986	Jan. 22	46.45N, 118.98W	126	2.6
1986	Jan. 29	46.25N, 119.60W	169	2.9
1986	Jan. 29	46.45N, 119.00W	128	2.6
1986	Feb. 1	46.45N, 118.98W	126	2.6
1986	Feb. 4	46.03N, 118.80W	118	3.2
1986	Feb. 5	46.25N, 119.60W	169	2.8
1986	March 2	46.30N, 119.78W	182	2.8
1986	April 9	47.15N, 119.95W	194	2.6
1986	Sept. 1	46.71N, 119.28W	146	3.4
1987	July 16	47.46N, 117.01W	37	3.0
1987	Dec. 20	47.76N, 119.36W	160	2.7
1988	Jan. 17	46.73N, 119.35W	159	2.6
1988	Feb. 2	46.73N, 119.38W	151	2.9
1988	Feb. 3	46.73N, 119.38W	151	2.9
1988	March 17	46.11N, 119.76W	182	2.6
1988	March 18	46.35N, 119.25W	146	2.6
1988	April 1	46.90N, 119.10W	134	2.6
1988	April 7	47.03N, 119.93W	192	2.6
1988	April 22	47.85N, 119.91W	198	2.9
1988	May 2	47.76N, 119.35W	151	2.6
1988	May 28	46.80N, 119.41W	156	3.5
1988	May 31	46.80N, 119.41W	156	2.9
1988	July 9	46.83N, 119.70W	176	3.7
1988	July 9	46.83N, 119.70W	176	2.6
1988	July 14	46.88N, 119.40W	155	3.3
1988	Aug. 26	46.06N, 118.76W	115	2.8
1989	Jan. 27	46.03N, 118.70W	112	2.8
1989	Feb. 21	46.73N, 119.41W	155	2.9
1989	March 17	46.96N, 119.71W	177	2.7
1989	May 24	47.11N, 118.55W	98	2.6
1989	June 13	46.93N, 118.53W	95	3.0
1990	March 21	47.80N, 119.41W	164	2.7
1990	April 22	46.53N, 119.71W	176	3.3
1990	April 24	47.81N, 119.80W	190	2.6
1990	June 19	46.83N, 119.31W	149	3.3
1990	Sept. 17	46.61N, 118.88W	119	2.6
1990	Oct. 10	46.86N, 117.36W	16	2.9
1990	Oct. 23	46.61N, 118.88W	119	3.0
1990	Dec. 14	46.61N, 118.90W	120	2.8
1990	Dec. 15	46.80N, 119.98W	195	3.1
1990	Dec. 22	46.78N, 119.98W	195	3.4

Year	Date	Location of Epicenter	Approximate Distance from Pullman (miles)	Intensity at Epicenter*
1991	Feb. 2	46.95N, 119.03W	130	2.7
1991	Feb. 14	47.96N, 119.98W	195	3.1
1991	Feb. 18	46.61N, 118.86W	118	2.8
1991	Feb. 19	46.61N, 118.88W	118	2.6
1991	Feb. 26	46.71N, 119.88W	188	3.0
1991	April 2	47.8N, 119.80W	190	2.7
1991	April 11	47.46N, 118.26W	84	2.6
1991	April 16	46.58N, 119.76W	180	2.9
1992	Jan. 20	47.33N, 119.01W	134	2.6
1992	Feb. 7	46.16N, 118.38W	89	2.7
1992	Feb. 18	46.95N, 119.55W	165	3.2
1992	March 6	46.58N, 117.05W	11	2.6
1992	April 27	46.61N, 118.90W	120	3.0
1992	June 4	46.31N, 117.56W	34	2.7
1992	Aug. 6	46.00N, 118.40W	93	2.8
1994	Jan. 13	46.88N, 118.68W	105	3.4
1994	March 8	46.58N, 119.73W	178	2.6
1994	May 27	46.86N, 119.31W	152	2.6
1994	Nov. 13	46.58N, 119.48W	160	3.3
1994	Dec. 20	46.81N, 117.63W	33	2.6
1995	June 12	46.40N, 119.25W	145	3.3
1995	Aug. 29	46.20N, 119.90W	191	3.1
1995	Nov. 2	46.15N, 119.55W	168	3.1
1996	June 25	47.18N, 119.50W	163	3.1
1997	May 24	46.83N, 119.35W	151	2.6
1997	May 27	46.83N, 119.36W	151	3.3
1997	Aug. 4	46.90N, 117.43W	20	2.7

*Modified Mercalli Intensity Scale of 1931.

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APPENDIX 2-A
Earthquakes and Ground Motion

The magnitude or intensity of an earthquake as measured by the Richter scale is determined by the displacement caused on a Wood-Anderson photographic instrument by an earthquake. The Richter magnitude scale is a logarithmic scale defined by the relationship:

$$ML \text{ (Magnitude Level)} = \log (A/Ao)$$

where: A is the zero-to-peak amplitude in mm on a Wood-Anderson instrument, and Ao is function of distance given by the relationship for the 0 to 200 km range.

$$\text{Log } (Ao) = 1.5 - 1.6 \log (\text{distance in Km})$$

In order to compute the actual ground motion in relation to the Richter scale one must know the sensitivity of the Wood-Anderson instrument which is somewhat frequency dependent. In the frequency range above 2 Hz, a Wood-Anderson instrument magnifies the ground motion by a factor of 2080 (7). Using the above information a series of curves was developed given in Figure 2-21 that show the ground displacement as a function of earthquake intensity on the Richter scale for events occurring at various distances from 0 to 100 Km from the site. These curves represent the idealized situation since the actual events occur at various depth, in the earth and have frequency components that vary from event to event due to the complex nature of earthquakes.

The seismic switch in the WSU Reactor control system is a specially designed and constructed oil damped pendulum with a contact spacing of .05 cm. Accordingly a ground motion of 1 mm with a period of 3 seconds or less would activate the switch.

APPENDIX 2-B

PROBABILISTIC ASSESSMENT OF THE AIRPLANE CRASH RISK FOR THE WASHINGTON STATE UNIVERSITY MODIFIED TRIGA REACTOR

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1. INTRODUCTION

The objective of this assessment is to estimate the risk of aircraft crash precipitated reactor accidents at the WSU Nuclear Radiation Center and the associated 1-MW(t) TRIGA nuclear reactor located at Washington State University. The main concern motivating this assessment is the potential release to the environment of radioactive material as a result of an aircraft striking the reactor building or its vicinity. Over the past two decades, probabilistic methods have gained increasing use for evaluating the risks of nuclear power reactors. Probabilistic risk assessment (PRA) is designed to evaluate the probability of adverse conditions that impact the reactor will lead to a prompt or delayed severe health hazard to the general public.

As a result of numerous reactor studies, it has been concluded that airplane crashes become a significant contributor to public health risks when the probability of a significant aircraft induced radiological accident exceeds approximately 1 in 10,000,000 reactor years; i.e., 10^{-7} per reactor year. If this probability is smaller than 10^{-7}, the aircraft induced accident risk becomes insignificant or, equivalently, the aircraft induced radiological accident becomes an incredible accident scenario.

The results of aircraft accident evaluation will demonstrate that the structure of the WSU Nuclear Radiation Center building and the location of the TRIGA reactor within the facility give rise to (1) the probability of an aircraft related radiological accident is less than 10^{-7} per year, or (2) the maximum credible radiological accident will have inconsequential effects, considering both air and water releases per 10 CFR 20 and ANSI 15.7 limits. The analysis will show that the WSU TRIGA reactor actually meets both requirements. Therefore, the aircraft radiological accident is an "incredible" event.

The analysis performed here has been realistic (best estimate) where possible and very conservative where there were data limitations. The derived conclusions about aircraft accident-related risk are therefore very conservative from the safety standpoint.

2. ASSESSMENT METHODOLOGY

The probability P per year that an airplane crash will lead to critical reactor damage is essentially the product of three probabilities:

1. P_c, the probability of an airplane crash at the reactor site,
2. $P_{p/c}$, the conditional probability that the reactor building will be penetrated as a result of an airplane crash,
3. $P_{d/p}$, the conditional probability of critical damage to the reactor as a result of airplane-crash-induced reactor building penetration,

$$P = P_c \bullet P_{p/c} \bullet P_{d/p}. \qquad \text{(B-1)}$$

The crash probability, P_c, is the product of three factors: the number of aircraft movements, the accident probability per aircraft movement per unit area, and the effective area of the target of interest.

$$P_c = \sum_i \sum_j \sum_k \left(N_{ijk} C_{ijk} A_{ijk}\right), \qquad \text{(B-2)}$$

where N_{ijk} = number of annual movements of type j for aircraft type i in flight pattern k,
C_{ijk} = crash probability per movement of type j for aircraft type i in flight pattern k,

A_{ijk} = effective target area associated with the structure of interest for aircraft type i in movement type j and flight pattern k.

The types of aircraft for the Pullman-Moscow Airport include small private aircraft and medium size commercial aircraft, and other miscellaneous aircraft. The aircraft movements include takeoff and landing only.

In the present analysis, aircraft were grouped into two categories: commercial aircraft and small private planes that represent an insignificant fraction of the total flights and whose weights are such that if they crash near or into the facility, the associated hazard is insignificant compared to that for the commercial aircraft. The Operations Department of Horizon Air, the only commercial airline using the Pullman-Moscow airport, indicates that there are 2,050 scheduled flights per year into the airport and that approximately 75% use runway 5, the approach to which passes over the WSU campus south of the facility. The aircraft employed are Dash-8 Turboprop planes that have a maximum gross landing weight of 34,500 pounds.

The conditional probability of penetration as a result of an airplane crash is generally evaluated for both direct and indirect hits. The latter refer to the impact of missiles generated as a result of the airplane crash in the immediate vicinity of the target of interest, as well as the possibility of lateral aircraft skid into the structure.

For the present analysis it was conservatively estimated that indirect hits are inconsequential because the reactor itself is protected by the concrete walls of the Nuclear Radiation Center building shown in Figures B2, B3, and B4. The reactor proper is surrounded on all sides by rooms and, therefore, it was estimated that no missile generated by an aircraft crash in the immediate vicinity of the Nuclear Radiation Center building could penetrate to and through the reactor pool. Thus, only direct hits were considered in the analysis.

For direct hits the conditional probability of structure penetration by an aircraft was calculated based on the method of Reference B.1. This method very conservatively evaluates the conditional probability of penetration of a reinforced concrete wall when impacted by a heavy aircraft at full flight speed. The probability is given as a function of wall thickness as shown in Figure B.6.

The conditional probability of radionuclide release from the reactor by fuel element breach due to reactor structure penetration by an impacting aircraft ($P_{d/p}$) is difficult to evaluate analytically because it strongly depends on the collision history and on the likelihood that all or most of the reactor water will be lost and that a radioactive release from the damaged fuel will enter the atmosphere or groundwater resulting in radiological environmental hazard.

The WSU TRIGA Reactor core is surrounded on all lateral sides and on the bottom by a continuous concrete structure 7 ft thick. Therefore, total loss of coolant is highly unlikely. Also, Reference B.1 conservatively estimates that 7 ft of reinforced concrete are impenetrable by an aircraft. If the reactor water is not lost, then it will be an effective radionuclide filter. Water scrubbing under accident conditions was estimated to be a very efficient radionuclide remover when estimating nuclear reactor accident source terms (Reference B.2).

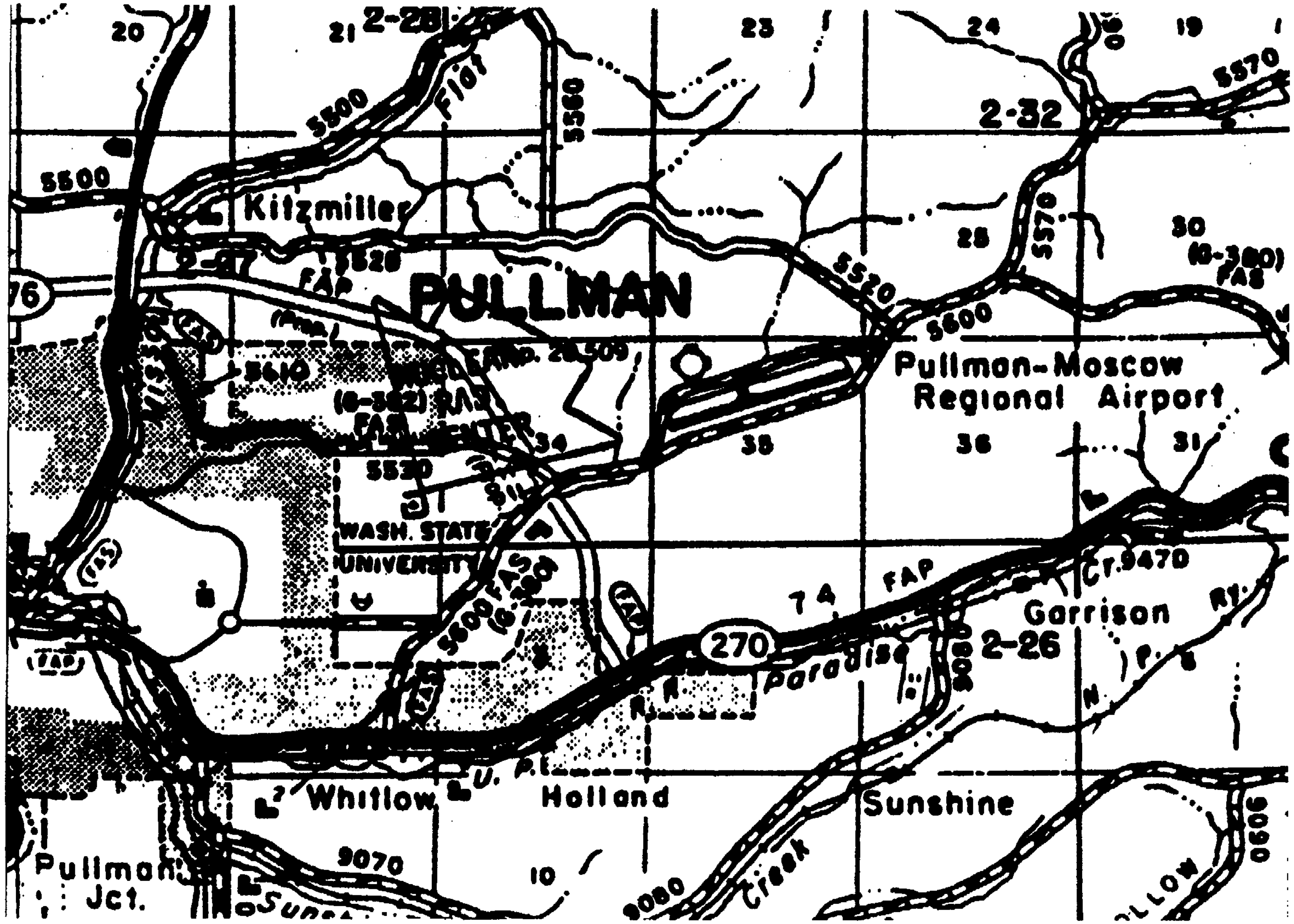

Figure B-1
Pullman-Moscow Airport in Relationship to WSU Nuclear Radiation Center

[redacted]

Figure B.2
WSU Nuclear Radiation Center Floor Plan for 2nd Floor

[redacted]

Figure B-3
WSU Nuclear Radiation Center Floor Plan for 1st Floor

[redacted]

Figure B-4
WSU Nuclear Radiation Center Floor Plan for Ground Floor

[redacted]

Figure B-5
Vertical Cross Section of WSU Reactor Facility

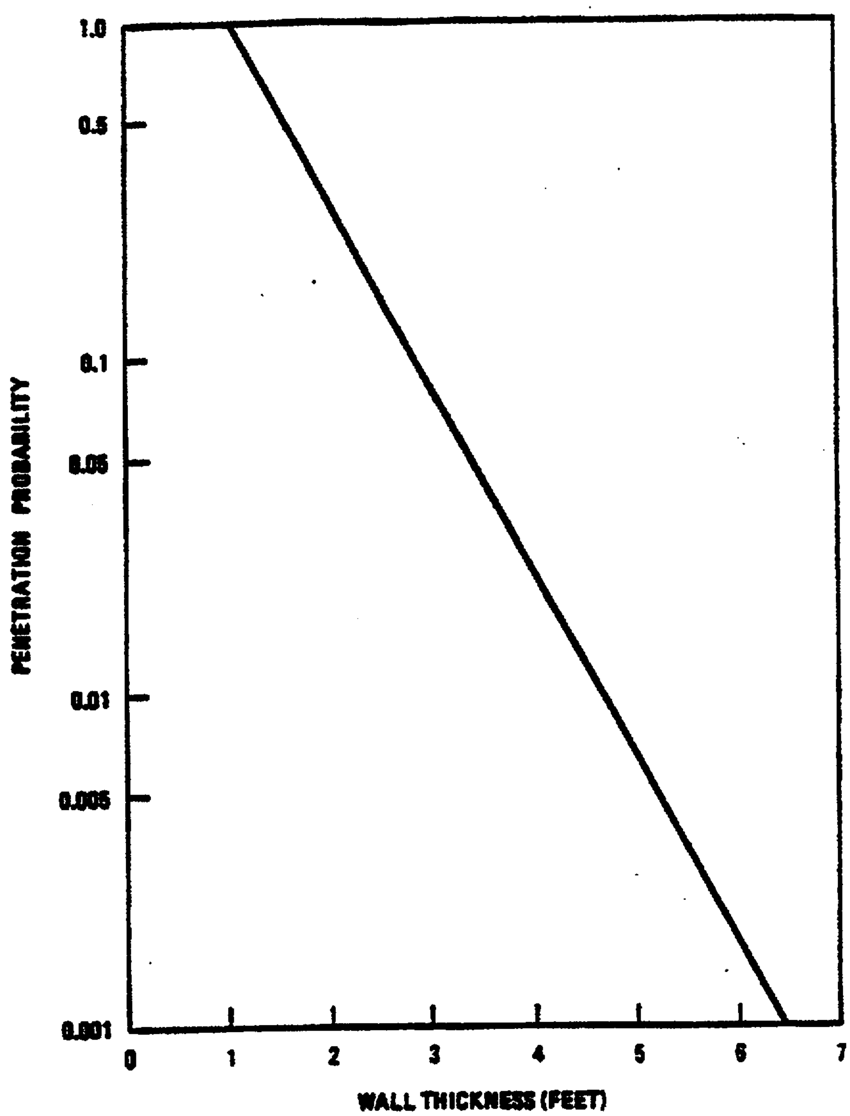

Figure B-6
Conditional Wall Penetration Probability

3. ANALYSIS

The location of the WSU Nuclear Radiation Center in relationship to the Pullman-Moscow Regional Airport is shown in Figure B-1. The west end of the main runway is at an elevation of 2537 feet and is located about 5,800 feet east of the facility. The elevation of the basement of the facility is 2632 feet. In terms of the normal flight pattern, aircraft pass a number of hundred feet south of the facility and a few hundred feet above the roof level of the facility. Figures B-2 to B-5 show the horizontal and vertical structure of the WSU Nuclear Radiation Center Building. These drawings clearly show that the only portion of the facility that is vulnerable to an aircraft accident is the west end of the building or right side of Figure B-5. That is, an off-course and low landing aircraft is the only situation in which an aircraft could crash into the facility with the potential of causing damage to the reactor core.

The basement level or "Beam Room" area is [redacted] impenetrable by a crashing aircraft according to reference B.1. Thus, an aircraft crashing into the Beam Room wall could not precipitate a significant reactor accident such as the draining of the reactor pool.

Above the Beam Room is the "Radio Chem Lab area" which is also protected by the [redacted]

Above the Radio Chem Lab is the "Reactor Control Room" and actual Pool Room area. If an aircraft were to crash into the building wall at the control room level, it would destroy the reactor control system and possibly cause severe damage to the reactor bridge. However, the reactor would automatically shut down and the core would be protected by the 20 ft of water in the pool above the core. Thus a simple crash into the control room wall area would not precipitate an accident that would endanger the general public. However, if the crash were near the control room floor, the impact could fracture the [redacted]

A very conservative approach is to use a pool wall penetration probability of .024/crash and assume a target crash area of the width of the building times the Radio Chem Lab wall height (value) plus half the Control Room wall height (value) for a total [redacted] Assuming the traditional glide angle of 20 degrees, the effective "shadow" area for analysis purposes becomes:

$A_{eff} =$ [redacted] $+$ [redacted] $= .000073 + .000084 = 1.57 \times 10^{-4}\ mi^2$

Using equation B-2 we may now calculate the probability that an aircraft will crash into the WSU Reactor building using the following conservative assumptions: 1) based on the data in reference B.5, the probability of crash during landing is 1.5×10^{-7} per square mile per landing, and 2) we will ignore the fact that only 75% of the planes land or take off on the runway that passes near the reactor facility and use the total landings per year. Accordingly,

$$P_c = 4.83 \times 10^{-8} \text{ crashes/year.}$$

We now can calculate the probability that the above crash will result in pool penetration and the possible precipitation of a reactor accident.

$$P = .024/\text{crash} \times 4.83 \times 10^{-8} \text{ crashes/year}$$

$$P = 1.16 \times 10^{-9}/\text{year aircraft-related reactor accidents.}$$

This value is considerably less than the 10^{-7} threshold for a significant risk.

4. CONCLUSION

The probability that an aircraft crash accident at the WSU Nuclear Radiation Center could precipitate a reactor related radiological accident was conservatively calculated to be 1.2 x 10^{-9} per year which is a factor of 80 below the probability for a credible aircraft related radiological accident. Thus we may conclude that an aircraft related radiological accident is an "incredible" event for the WSU facility and thus is an insignificant risk to the health and safety of the general public.

Furthermore, the Maximum Hypothetical Accident and Loss of Coolant accidents analyzed in Section 13 of this SAR are below the 10 CFR 20 and ANSI 15.7 limits. Accordingly, if an aircraft crash accident were to precipitate one of these accident scenarios, the result would also not create a significant health risk to the general public.

5. REFERENCES

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$$P_c = 2050\left(\frac{\text{landings}}{\text{yr}}\right) \times 1.5 \times 10^{-7}\left(\frac{\text{crashes}}{\text{landing} \cdot \text{mi}^2}\right) \times 1.57 \times 10^{-4}\,\text{mi}^2$$

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The principal consideration in the design, construction, and operation of a non-power reactor is that of protecting the health and safety of the general public as well as those who work at the facility. The first area of consideration is the reactor core and the associated fuel elements. In the case of the WSU modified TRIGA reactor, the basic design of a TRIGA reactor as described in sections 1.2 and 4.2.1 of this report is the most important design and safety aspect of a TRIGA type reactor. The large inherent prompt negative temperature coefficient of TRIGA type fuel provides a built in safety feature for a TRIGA type reactor. Also, in the case of a modified type TRIGA reactor in which 4-rod clusters of TRIGA fuel replace old plate type fuel, the fuel rod size is an important design parameter. The fuel rod size was selected to provide a slightly undermoderated core so that loss of water in the core has a negative reactivity effect.

The prompt negative temperature coefficient for the TRIGA fuel core is based on the spectrum hardening characteristic that occurs in all ZrH based fuel rods. The spectrum hardening is caused by heating of the fuel-moderator elements. The rise in temperature of the hydride increases the probability that a thermal neutron in the fuel element will gain energy from an excited state of an oscillating hydrogen atom in the lattice. As the neutrons gain energy from the ZrH, the thermal neutron spectrum in the fuel element shifts to a higher average energy (the spectrum is hardened), and the mean free path for neutrons in the element is increased appreciably. For a standard TRIGA element, the average chord length is comparable to a mean free path, and the probability of escape from the element before being captured is significantly increased as the fuel temperature is raised. In the water, the neutrons are rapidly rethermalized so that the capture and escape probabilities are relatively insensitive to the energy with which the neutron enters the water. The heating of the moderator mixed with the fuel in a standard TRIGA element thus causes the spectrum to harden more in the fuel than in the water. As a result, there is a temperature-dependent disadvantage factor for the unit cell in which the ratio of absorptions in the fuel to total cell absorptions decreases as fuel element temperature is increased. This brings about a shift in the core neutron balance, giving a loss of reactivity.

In the TRIGA-FLIP and TRIGA-LEU fuel, the temperature-hardened spectrum is used to decrease reactivity through its interaction with a low-energy-resonance material. Thus, erbium, with its double resonance at ~0.5 eV, is used in these fuels as both a burnable poison and a material to enhance the prompt negative temperature coefficient. The ratio of the absorption probability to the neutron leakage probability is increased for TRIGA-FLIP and TRIGA LEU fuels relative to the standard TRIGA fuel because the U-235 density in the fuel rod is greater and also because of the use of erbium. When the fuel-moderator material is heated, the neutron spectrum is hardened, and the neutrons have an increasing probability of being captured by the low-energy resonances in erbium. This increased parasitic absorption with temperature causes the reactivity to decrease as the fuel temperature increases. The neutron spectrum shift, pushing more of the thermal neutrons into the Er-167 resonance as the fuel temperature increases. As with a standard TRIGA core, the temperature coefficient is prompt because the fuel is intimately mixed with a large portion of the moderator; thus, fuel and solid moderator temperatures rise simultaneously, producing the temperature-dependent spectrum shift.

The operational parameters for the WSU modified TRIGA reactor as set forth in the Technical Specifications were selected to insure integrity of the fuel within the specified range of operating conditions. This matter is covered in more detail in section 3.5. The primary criteria are the maximum operational power level and pulsing limit. The maximum steady state power level of 1 MW is based on cumulative evidence that natural convection cooling is adequate to insure proper fuel rod cooling and TRIGA fuel integrity up to this power level. The pulsing limit at WSU involves insuring that the fuel rod temperature during pulsing does not exceed 830°C from the lessons learned as a result of the FLIP fuel rod failure at Texas A&M.

One unique operational criteria for the WSU modified TRIGA reactor is Technical Specification 3.5 concerning core configuration limits. This operational criteria is for cores of mixed FLIP and Standard type fuel to minimize possible fuel temperature peaking at the interface between FLIP and Standard fuels. A mixed core must contain at least 22 FLIP fuel rods in a contiguous block in the central region of the core and water holes in the FLIP region shall be limited to nonadjacent single rod holes. FLIP fuel has a much higher thermal neutron absorption cross section than standard fuel and thus a large water hole at the interface between FLIP and Standard would create significant flux peaking and temperature peaking in the standard fuel.

In summary, the inherent prompt negative temperature coefficient, undermoderated core, core configuration limit, 1 MW steady state power limit, and fuel temperature limit of 830°C during pulsing establish a set of safe operating limits for the WSU modified TRIGA reactor. That is, these design and operation criteria cover the complete range of expected operating conditions of the WSU reactor insuring that all operations within these boundaries from start-up to full power are safe. Also the detailed analysis of potential accidents given in chapter 13 of this SAR shows that the above operational boundaries are adequate to preclude any significant operational accident that would endanger the health and safety of the general public.

The next important area is that of reactor control and instrumentation. The control rod system was designed to provide fail-safe operation and an adequate shut-down margin under all allowable conditions of operation. The control rods fall into the core under the influence of gravity which ensures insertion in the case of some sort of failure. The shut-down margin is set to insure shut-down even with the most reactive rod stuck out as a safety criteria. The pulse rod is held out by air pressure and in the event of loss of air, the pulse rod falls into the reactor.

The reactor control system was designed to insure high reliability, ease of maintenance, and optimum layout from a human engineering aspect. Extensive information on the control system is given in section 7 of this report. The primary design criteria were: 1) provide automatic scram in the event of excessive power level or excessive fuel temperature, 2) automatic scram in the event of an important system failure (low pool level and loss of chamber high voltage, and 3) manual scram for operator intervention. The control system meets all the requirements of ANSI/ANS 15.20.

3.2 Meteorological Damage

The building that houses the WSU reactor is a concrete structure designed to meet applicable State of Washington building codes. The basic building has existed for over 30 years and has been exposed to a variety of local severe weather conditions without any structure failure. During this period extreme weather conditions that have occurred are cold weather down to minus 46°C (-50°F), very heavy snow fall, and wind gusts up to about 90 MPH with no ill long term effects. Thus there is reasonable assurance that no future meteorological condition will cause significant damage to the facility.

The only portion of the reactor system that is exposed to the weather is the cooling tower. The cooling tower is located on the central north side of the building and thus shielded from severe winds that come from the west. In order to preclude icing at moderately low temperatures, the cooling tower has a heat tape on the cooling water line. Tornados and hurricanes do not occur in the Pullman, Washington area. During extreme cold weather (below about -28°C) the reactor is not operated so as not to create potential icing problems.

3.3 Water Damage

The facility is located on the side of a small hill as can be seen from examining the topographic map of the site given in section 2 of this SAR. Furthermore, this hill is the highest ground in the area for many hundreds of meters in all directions except directly in back of the facility to the North. Also, there are no water sources or streams in the area except the water contained in the pool itself. Thus there is absolutely no potential for flooding at this site.

The facility is four stories high in the area of the reactor proper. The control area is on the second floor with the pool extending from the basement, part of the first floor, and up to the bottom of the second floor. Thus even a severe pool fracture would just gush water out of the basement level into the surrounding area and not impact the control room area. All fans are located on the third floor or roof area of the facility and can not be flooded. Thus there is no potential for flood damage at the WSU reactor site.

3.4 Seismic Damage

The control system of the reactor includes a "Seismic Switch", built by the California Academy of Sciences in 1957 which is basically an oil dampened pendulum with a contact spacing of .05 cm, which will automatically scram the reactor in the event of a seismic event of 5.2 20 km from the reactor site. In recorded history there have only been three seismic events that have even been felt in the Pullman area, all of very low intensity. Based on the geology of the Pullman area and the past seismic activity discussed in detail in chapter 2 of this SAR, the probability of the occurrence of a significant earthquake in the future is very very small.

The pool of the WSU reactor was constructed in the 1950's to conform to the then existing construction standards. In the event of a significant seismic event at the site, the worst possible effect would be a fracture of the pool structure allowing all the pool water to drain out. The loss of pool water accident is covered in detail in chapter 13 of this SAR which demonstrates that a loss of coolant accident would not precipitate a fuel cladding failure that would release fission products into the pool room. The loss of water in and of itself would shut down the reactor even if the control rods are not inserted into the core. Thus a significant seismic event at the site could not precipitate a significant reactor accident that would endanger the health and safety of the general public.

3.5 Systems and Components

Detailed information on the design basis for all the important systems and components of the WSU modified TRIGA reactor are contained in the Technical Specifications for the facility given in section 14 of this report. The design features are covered in section 5.0 of the Technical Specifications and limiting conditions of operation are given in Technical Specification section 3.0. The design basis may be grouped in five key areas which are: 1) fuel system, 2) control rod system, 3) radiation monitoring system, 4) ventilation system, and 5) pool water system.

An ANSI/ANS standard does not exist for TRIGA type fuel but General Atomics fabricates TRIGA fuel to stringent standards. Technical Specification 5.1 insures that the fuel rod's uranium content does not exceed [redacted] fuel and that the hydrogen-to-zirconium ratio does not exceed 1.8%. Technical Specification 3.5 limits core configurations that are known be safe.

Technical Specification 5.3 applies to the control rods and requires that they contain appropriate quantities of B_4C powder or boron. More importantly the WSU preventive maintenance program, SOP #5, ensures the continued functionality of the control rods at all times. SOP #5 embodies the basic requirements set forth in ANSI/ANS 15.15 and ANSI/ANS 15.8 for all portions of the WSU TRIGA reactor systems and components.

Technical Specifications 5.4 and 3.7 apply to the radiation monitoring systems. These specifications insure that the Reactor Operator is made aware of the radiation levels in key areas and that the Argon-41 stack monitor and continuous air monitor are functioning. SOPs #18, 27, and 29 provide explicit details for the testing and maintenance of these systems and embody the requirements of ANSI N323.

The Emergency Procedures Manual for the WSU modified TRIGA reactor has very explicit procedures for all possible types of emergency situations, including "Earthquakes" and "Fire and Explosion" at the facility. The Emergency Procedures in this manual all conform to the requirements of ANSI N323, ANSI/ANS 15.11, and ANSI/ANS 15.17.

3.6 References

American National Standards Institute, ANSI N323, "Radiation Protection Instrumentation Tests and Calibration," 1978.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.7, "Research Reactor Site Evaluation," 1977.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8, "Quality Assurance Program Requirements for Research Reactors," 1986.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiological Controls at Research Reactors," 1993.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.15, "Criteria for Reactor Safety Systems for Research Reactors," 1978.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.17, "Fire Protection Program Criteria for Research Reactors," 1981.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.20, "Criteria for the Reactor Control and Safety Systems of Research Reactors," draft.

4 REACTOR DESCRIPTION

4.1 Summary Description

The WSU modified TRIGA reactor is a one megawatt open top pool-type research reactor using light-water as a moderator, coolant, reflector, and shield. The light-water cools the TRIGA type solid fuel rods via natural convection. Heat generated by the fission process is first transferred to the pool water which in turn is dissipated to the atmosphere by means of a cooling-tower heat-exchanger arrangement. Two types of TRIGA fuel rods are currently used in the core, that are 8.5 weight percent uranium; standard TRIGA rods have an enrichment of 20% and FLIP rods have an enrichment of 70%. Eventually the core will be shifted to LEU type fuel with 20% weight percent uranium and has a 20% enrichment. Detailed data on TRIGA fuel rods is given in Table 1.1 in section 1 of this SAR and Table 4.1-1 below.

TABLE 4.1-1
Standard, LEU and FLIP Fuel Parameters

Fuel Element Type	FLIP	STANDARD	LEU
Fuel-moderator material	$U\text{-}ZrH_{1.6}$	$U\text{-}ZrH_{1.7}$	$U\text{-}ZrH_{1.6}$
Uranium content	8.5 wt%	8.5 wt%	20%
^{235}U enrichment	70%	20%	20%
^{235}U content (avg) per element	[redacted]	[redacted]	[redacted]
Burnable poison	natural erbium	none	natural erbium
Erbium content	1.58 wt%	--	.5 wt%
Shape	cylindrical	cylindrical	cylindrical
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
Cladding material	Type 304 SS	Type 304 SS	Type 304 SS
Cladding thickness	0.020 in.	0.020 in.	0.020 in.

The pool is spanned by a manually-operated bridge structure from which the core support structure is suspended. The core is situated in a grid box into which 4-rod clusters of TRIGA fuel are positioned. Control over the reactor is exerted by inserting and withdrawing neutron absorbing control elements suspended from a control driver mounted on the bridge. One of the control rods is a pulse rod which allows the reactor to be instantaneously pulsed to a very high power level which is terminated by the unique large prompt negative temperature coefficient of TRIGA type fuel. The principal uses of the reactor are for Neutron Activation Analysis and Radioisotope production in vertical irradiation tubes that extend down from the bridge into the core area. The bridge is movable and thus the core may be moved in the pool structure including being placed adjacent to a thermal column experimental facility. The thermal column is being used for medical purposes and Boron Neutron Capture Therapy discussed in section 16.2 of this SAR.

4.2 Reactor Core

The core of the reactor is situated in a grid box in the reactor pool suspended from a bridge structure shown in Figure 4-1. The grid box as shown in Figure 4-2 consists of a cast aluminum grid plate suspended from the bridge by four corner posts that form a suspension frame. The grid plate provides a 7 by 9 array of square holes for fuel clusters and two slots for control blades. The sides of the grid box are aluminum sheeting positioned to direct the convection currents of cooling through the core. The grid box accepts the 4-rod TRIGA fuel clusters described in detail in the next section of this SAR as well as reflector elements shown in Figure 4-3. A typical mixed Std-FLIP core arrangement is given in Figure 4-4.

Very detailed information on the various components that make up the reactor core region are given in the following subsections of this SAR. The neutronic design considerations are given in detail in section 4.5.

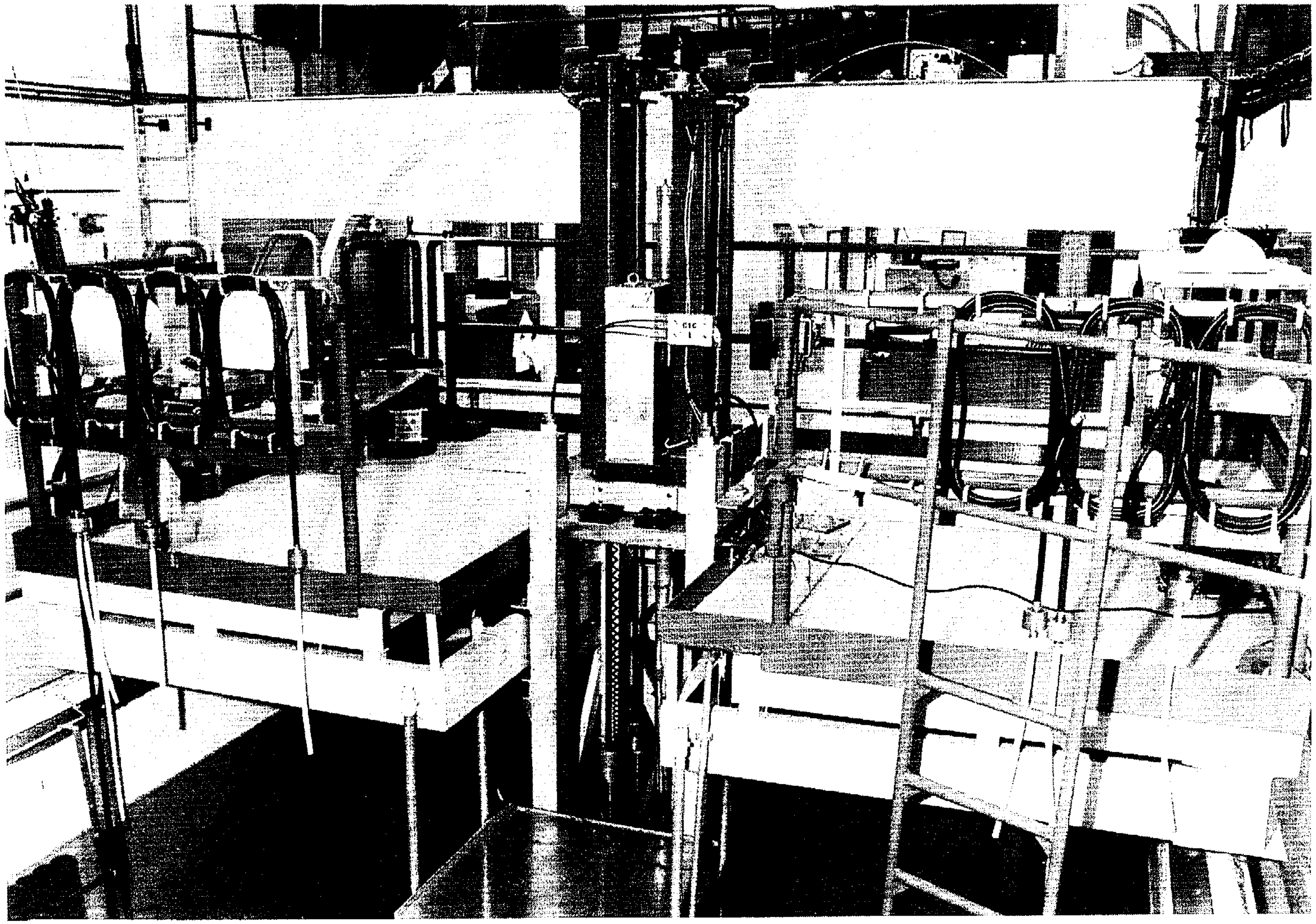


Figure 4-1
Reactor Bridge Structure

Figure 4-2
Grid Box and Grid Plate

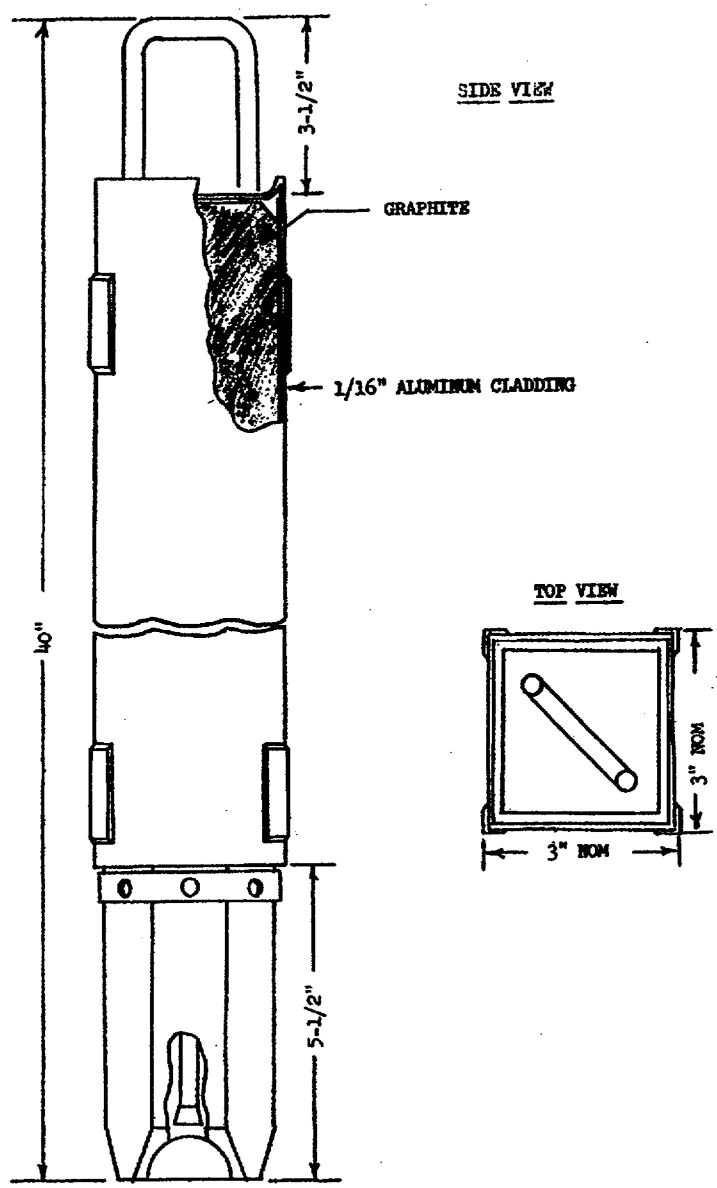


Figure 4-3
Graphite Reflector Element

[redacted]

4.2.1 Reactor Fuel

The fuel elements consist of 3-rod or 4-rod clusters of TRIGA-type fuel as shown in Figure 4-5 and 4-6. The 4-rod fuel cluster was developed as a simple replacement for MTR-type plate fuel bundles. The top handle and bottom end fitting on the 4-rod cluster serve to adapt TRIGA rod type fuel to the square grid array used with plate-type fuel.

The individual fuel rods are similar in construction to standard TRIGA fuel rods with the exception of the rod diameter and modified rod end fittings. Two types of TRIGA fuel rods, Standard and FLIP, with the parameters listed in Table 4.1-1 are currently used in the WSU reactor. Eventually the core of the reactor will be converted to LEU fuel with the paramet[redacted] listed in Table 4.1-1. [redacted], and clad in a .020-inch type 304 stainless steel cylinder, as shown in Figure 4-7.

The zirconium hydride active portion of the fuel rod is [redacted]. A 3.45-inch graphite reflector plug is positioned in each end of the fuel rod and top and bottom end fittings are welded onto the cladding.

In addition to standard fuel rods, one or more instrumented fuel rods, as shown in Figure 4-8, are used in the core. This type of fuel rod is fitted with three thermocouples used to measure the fuel temperature. A special 3-rod cluster with a transient rod guide tube, as shown in Figure 4-9, is positioned in the center of the reactor grid. The transient control rod is positioned inside the guide tube, as described in Section 4.2.2.

The specific characteristics that make TRIGA type fuels uniquely suited for use in extremely safe research type reactors are covered in detail in the following portions of this section. A summary of the characteristics is given below (18,19):

1. $ZrH_{1.6}$ is single phase up to 1200°F (delta phase region).
2. Low hydrogen equilibrium disassociation pressure at normal fuel temperatures.
3. High hydrogen retention.
4. High heat capacity.
5. Low thermal expansion coefficient.
6. Relatively low reactivity in water.
7. No significant damage or swelling due to irradiation effects.
8. High fission product retention.
9. Very large negative temperature coefficient of reactivity.
10. High burnup possible by addition of burnable poison.
11. High loading of uranium possible with insignificant change in fuel material properties.

[redacted]

Figure 4-6
TRIGA 4-Rod Fuel Cluster

[redacted]

Figure 4-7
Fuel-Moderator Rod

[redacted]

Figure 4-8
Instrumented Fuel Rod

Figure 4-9
Three-Rod Fuel Cluster with Control Rod Guide Tube

TRIGA Fuel Development

The development and use of U-ZrH_x fuels for the TRIGA reactor has been underway at General Atomics since 1957. Over 6000 fuel elements of 7 distinct types have been fabricated for the 60 TRIGA research reactors in various countries around the world. The earliest of these has now passed 30 years of operation. U-ZrH fuel has exhibited unique safety features including a prompt negative temperature coefficient of reactivity, high fission product retentivity, chemical stability when quenched from high temperatures in water, and dimensional stability over large swings of temperature. The first TRIGA reactor to be exported was for the U.S. exhibit at the Second Geneva Conference on the Peaceful Uses of Atomic Energy in 1958.

The standard TRIGA fuel contains 8.5 wt-% uranium (20% enriched) as a fine metallic dispersion in a zirconium hydride matrix. The H/Zr ratio is nominally 1.6 (in the face-centered cubic delta phase). The equilibrium hydrogen dissociation pressure is governed by the composition and temperature. For $ZrH_{1.6}$ the equilibrium hydrogen pressure is 1 atm at about 760°C. The single-phase, high-hydride composition eliminates the problems of density changes associated with phase changes and with thermal diffusion of the hydrogen. A highly enriched version of TRIGA fuel (FLIP with a 70% enrichment) contains up to about 3% erbium as a burnable poison to increase the core lifetime and contribute to the prompt negative temperature coefficient. The calculated core lifetime for FLIP fuel in a typical TRIGA reactor is approximately 9 MW-yr. Over 25,000 pulses have been performed with the TRIGA fuel elements at GA, with fuel temperatures reaching peaks of about 1150°C.

TRIGA fuel was developed around the concept of inherent safety. A core composition was sought which had a large prompt negative temperature coefficient of reactivity such that if all the available excess reactivity were suddenly inserted into the core, the resulting fuel temperature would automatically cause the power excursion to terminate before any core damage resulted. Experiments performed in the late 1950's demonstrated that zirconium hydride possessed the basic mechanism needed to produce the desired characteristic. Additional advantages were that ZrH has a good heat capacity, allows construction of a reactor with a relatively small core size and high flux values due to the high hydrogen content of the fuel rods, and could be used effectively to fabricate rugged fuel rods.

In early 1976, General Atomics undertook the development of fuels containing up to 45 wt-% uranium (3.7 gm U/cc) in order to allow the use of low enriched uranium (LEU) (under 20% enrichment) to replace the highly enriched fuels while maintaining long core life. The 45 wt-% fuel contains a relatively modest ~20 volume percent of uranium. These fuels were fabricated successfully, with the required hydrogen content and erbium loading. The structural features of the hydrided LEU fuel were similar to those of the well-proven 8.5 and 12 wt-% fuels, as shown by metallographic, electron microprobe analysis, and x-ray diffraction examination. Detailed evaluations of the new LEU fuel have shown that it performs essentially identically to the older standard TRIGA fuel in all critical cores.

Additional evaluations included analytical assessments of the prompt negative temperature coefficient of reactivity and the core lifetime (Table 4.2-1). Nuclear design and analytical studies have shown that the prompt negative temperature coefficient for the 20 wt-% uranium fuel is essentially the same as that for standard fuel over the temperature range of interest (20° to 700°C) and greater than that for the FLIP fuel which it replaces. The prompt negative temperature coefficient for the more highly loaded LEU fuel shows a small temperature dependence, whereas the coefficient is relatively constant for standard fuel. The value of the prompt negative temperature coefficient of reactivity is slightly lower for the 30 wt-% uranium fuel compared to the highly enriched fuel it replaces; however, it is still large and significantly higher than the prompt negative temperature coefficients for any other type of reactor fuel.

Table 4.2-1
Calculated Beginning of Life Prompt Negative Temperature Coefficient (α) and Core Lifetime

TRIGA Fuel Type	Wt % U	Wt % Er	Uranium Enrichment (%)	x $10^{-5} = \alpha$ Average (23°-700°C)	Core Lifetime (MWd)
Standard	8.5	0.00	20	10	~100
LEU	20	0.50	20	11	1200
LEU	30	0.92	20	8	3000
FLIP	8.5	1.58	70	10	3500

Inclusion of erbium burnable poison in the TRIGA LEU fuel has enabled core lifetimes of up to 3000 MWd to be predicted for the 30 wt-% fuel. It is emphasized that this is the core life from the time of initial refueling to end of useful life.

Dissociation Pressures

The hydrogen dissociation pressures of hydrides have been shown to be comparable in the alloys containing up to 75 wt-% U. The concentration of hydrogen is generally reported in terms of either weight percent or atoms of H/cm^3 of fuel (N_H). In the delta phase region, the dissociation pressure equilibria of the zirconium-hydrogen binary mixture may be expressed in terms of composition and temperature by the relation

$$\log P = K_1 + (K_2 \times 10^3)/T,$$

where $K_1 = -3.8415 + 38.6433\, X - 34.2639\, X^2 + 9.2821\, X^3$
$K_2 = -31.2982 + 23.5741\, X - 6.0280\, X^2$
P = pressure, atm
T = temperature, K
X = hydrogen-to-zirconium atom ratio.

The higher-hydride compositions (H/Zr > 1.5) are single phase (delta or epsilon) and are not subject to thermal phase separation on thermal cycling. For a composition of about $ZrH_{1.6}$, the equilibrium hydrogen dissociation pressure is 1 atm at about 760°C. The absence of a second phase in the higher hydrides eliminates the problem of large volume changes associated with a phase transformation at approximately 540°C in the lower hydride compositions. Similarly, the absence of significant thermal diffusion of hydrogen in the higher hydrides precludes concomitant volume change and cracking. The clad material of stainless steel or nickel alloys provides a satisfactory diffusion barrier to hydrogen at long-term (several years) sustained cladding temperatures below about 300°C.

Hydrogen Migration

Under nonisothermal conditions, hydrogen migrates to lower-temperature regions from higher-temperature regions. The equilibrium dissociation pressure obtained when the redistribution is complete is lower than the dissociation pressure before redistribution. The dimensional changes of rods resulting from hydrogen migration are of minor importance in the delta and epsilon phases.

Hydrogen Retention

The rates of hydrogen loss through 250-µm-thick stainless steel cladding are low at cladding temperatures characteristic of TRIGA fuel elements. A 1% loss of hydrogen per year occurs at about 500°C (900°F) clad temperature.

Density

The density of ZrH decreases with an increase in the hydrogen content. The density change is quite high up to the delta phase (H/Zr = 1.5) and then changes little with further increases in hydrogen. The bulk density of massively hydrided zirconium is reported to be about 2% lower than the results from x-ray defraction analysis.

For TRIGA fuel with a hydrogen-to-zirconium atom ratio of 1.6, the following relationships for the uranium density, $\rho_{U(A)}$ and weight fraction, ${}^{w}U$ in the U-$ZrH_{1.6}$ alloy apply:

$$\rho_{U(A)} = \frac{{}^{w}U}{0.177 - 0.125\ {}^{w}U}$$

$${}^{w}U = \frac{0.177\ \rho_{U(A)}}{1 + 0.125\ \rho_{U(A)}}$$

The relationship between the uranium density and the volume fraction of uranium in the alloy is given by:

$$\rho U(A) = 19.07\ V_f^{U(A)}$$

where $V_f^{U(A)}$ = volume fraction of uranium in the U-$ZrH_{1.6}$ alloy.

Thermal Conductivity

Thermal conductivity measurements have been made over a range of temperatures. A problem in carrying out these measurements by conventional methods is the disturbing effect of hydrogen migration under the thermal gradients imposed on the specimens during the experiments. This has been minimized at GA by using a short-pulse heating technique to determine the thermal diffusivity and hence to permit calculation of the thermal conductivity. From the recent measurements at GA of thermal diffusivity coupled with the data on density and specific heat, the thermal conductivity of uranium-zirconium hydride with an H/Zr ratio of 1.6 is 0.042 ± 0.002 cal/sec-cm-°C and is insensitive both to the weight fraction of uranium and to the temperature.

Heat Capacity

The heat content of zirconium hydride TRIGA fuel is a function of temperature and composition. The volumetric specific heat of 8.5 wt-% U-$ZrH_{1.6}$ is calculated to be

$$C_p = 2.04 + 4.17 \times 10^{-3}\ T\ (\text{W-sec/cm}^3)\ °\text{C (from 0 °C)}$$

and for 20 ct-% U-$ZrH_{1.6}$ is calculated to by

$$C_p = 2.17 + 4.36 \times 10^{-3}\ T\ (\text{W-sec/cc})\ °\text{C (from 0 °C)}$$

Chemical Reactivity

Zirconium hydride has a relatively low reactivity in water, steam, and air at temperatures up to about 600°C. Massive zirconium hydride has been heated in air for extended periods of time at temperatures up to 600°C with negligible loss of hydrogen. An oxide film forms which inhibits the loss of hydrogen.

The hydride fuel has excellent corrosion resistance in water. Bare fuel specimens have been subjected to a pressurized water environment at 570°F and 1230 psi during a 400 hr period in an autoclave. The average corrosion rate was 350 mg/cm^2-month weight gain, accompanied by a conversion of the surface layer of the hydride to an adherent oxide film. The maximum extent of corrosion penetration after 400 hr was less than 2 mils.

In the early phases of development of the TRIGA fuel, water-quench tests were carried out from elevated temperatures. Fuel rods (1-in. diam) were heated to 800°C and end-quenched to test for thermal shock and corrosion resistance. No deleterious effects were observed. Also, a 6-mm diam fuel rod was heated electrically to about 800°C and a rapid stream of water was sprayed on it; no significant reaction was observed. Small and large samples were heated to 900°C and quenched in water; the only effect observed was a slight surface discoloration. Finely divided U-ZrH powder was heated to 300°C and quenched to 80°C in water; no reaction was observed. Later, these tests were extended to temperatures as high as 1200°C, in which tapered fuel rods were dropped into tapered aluminum cans in water. Although the samples cracked and lost hydrogen, no safety problem arose in these tests. Recently, the low-enriched TRIGA fuels have been subjected to water-quench safety tests at GA.

Quench tests were performed on 20%-enriched TRIGA fuel samples (45 wt-% uranium, 53 wt-% zirconium, 1 wt-% erbium, 1 wt-% hydrogen) to simulate cladding rupture and water ingress into the TRIGA reactor fuel rods during operation.

These results indicate satisfactory behavior of TRIGA fuel for temperatures to at least 1200°C. Under conditions where the clad temperature can approach the fuel temperature for several minutes (which may allow formation of eutectics with the clad), the results indicate satisfactory behavior to about 1050°C. This is still about 50° to 100°C higher than the temperature at which internal hydrogen pressure is expected to rupture the clad, should the clad temperature approach that of the fuel. It should be pointed out that thermocouples have performed well in instrumented TRIGA fuel elements at temperatures up to 650 °C in long-term steady-state operations, and up to 1150°C in very short time pulse tests.

Irradiation Effects

Most of the irradiation experience to date has been with the uranium-zirconium hydride fuels used in the SNAP (containing about 10 wt% uranium) and TRIGA reactors. The presence of uranium influences the radiation effects because of the damage resulting from fission recoils and fission gases. Some significant conclusions may be drawn from the results of these experiments. The uranium is present as a fine dispersal (about 1 μm diam) in the U-ZrH fuels, and hence the recoil damage is limited to small regions within the short (~10 μm) range of the fission recoils. The U-ZrH fuel exhibits high growth rate during initial operation, the so-called "offset" growth period, which has been ascribed to the vacancy-condensation type of growth phenomenon over the temperature range where voids are stable.

The swelling of the U-ZrH fuels at high burnups is governed by three basic mechanisms:

1. The accommodation of solid fission products resulting from fission of U-235. This growth is approximately 3% $\Delta V/V$ per metal atom % burnup. This mechanism is relatively temperature insensitive.
2. The agglomeration of fission gases at elevated temperatures (above 1300°F). This takes place by diffusion of the xenon and krypton to form gas bubbles.
3. A saturable cavity nucleation phenomenon which results from the nucleation and growth of irradiation-formed vacancies into voids over a certain range of temperatures where the voids are stable. The saturation growth by this mechanism was termed offset swelling. It was deduced from the rapid decrease in fuel-to-cladding ΔT experienced during the early part of the irradiation. The saturation was reached in approximately 1500 hr.

Recent burnup tests performed by GA have shown that TRIGA fuels may successfully be used without significant fuel degradation to burnups in excess of 50% of the contained U-235.

Erbium Additions

All available evidence and extensive operating experience indicates that the addition of erbium to the U-ZrH introduces no deleterious effects to the fuel. Erbium has a high boiling point and a relatively low vapor pressure so that it can be melted into the uranium-zirconium uniformly. The erbium is incorporated into the fuel during the melting process. All the analyses that have been made on the alloy show that the erbium is dispersed uniformly, much as is the uranium. Erbium is a metal and forms a metallic solution with the uranium-zirconium; thus there is no reason to believe that there will be any segregation of the erbium. Erbium forms a stable hydride (as stable as zirconium hydride) which also indicates that the erbium will remain uniformly dispersed through the alloy. Also, since neutron capture in erbium is an n-γ reaction, there are no recoil products.

Prompt Negative Temperature Coefficient

The basic parameter which provides the greatest degree of safety in the operation of a TRIGA reactor system is the prompt negative temperature coefficient. This temperature coefficient (α) allows great freedom in steady-state operation, since the effect of accidental reactivity changes occurring from experimental devices in the core is minimized.

The prompt negative temperature coefficient for TRIGA fuels is based on the neutron spectrum hardening characteristic that occurs in a zirconium hydride fuel. The spectrum hardening is caused by heating of the fuel-moderator elements. The rise in temperature of the hydride increases the probability that a thermal neutron in the fuel element will gain energy from an excited state of an oscillating hydrogen atom in the lattice. As the neutrons gain energy from the ZrH, the thermal neutron spectrum in the fuel element shifts to a higher average energy (the spectrum is hardened), and the mean free path for neutrons in the element is increased appreciably. For a standard TRIGA element, the average chord length is comparable to a mean free path, and the probability of escape from the element before being captured is significantly increased as the fuel temperature is raised. In the water, the neutrons are rapidly rethermalized so that the capture and escape probabilities are relatively insensitive to the energy with which the neutron enters the water. The heating of the moderator mixed with the fuel in a standard TRIGA element thus causes the spectrum to harden more in the fuel than in the water. As a result, there is a temperature-dependent disadvantage factor for the unit cell in which the ratio of absorptions in the fuel to total cell absorptions decreases as fuel element temperature is increased. This brings about a shift in the core neutron balance, giving a loss of reactivity.

In the TRIGA FLIP and LEU fuel, the temperature-hardened spectrum is used to decrease the fuel's reactivity through its interactions with a low-energy-resonance material. Thus, erbium, with its double resonance at ~0.5 eV, is used in the TRIGA, FLIP and LEU fuels as both a burnable poison and a material to enhance the prompt negative temperature coefficient. The ratio of the absorption probability to the neutron leakage probability is increased for TRIGA, FLIP and LEU fuel relative to the standard TRIGA fuel because the U-235 density in the fuel rod is greater and also because of the use of erbium. When the fuel-moderator material is heated, the neutron spectrum is hardened, and the neutrons have an increasing probability of being captured by the low-energy resonance in erbium. This increased parasitic absorption with temperature causes the fuel's reactivity to decrease as the fuel temperature increases. The neutron spectrum shift pushes more of the thermal neutrons into the Er-167 resonance as the fuel temperature increases. As with a standard TRIGA core, the temperature coefficient is prompt because the fuel is intimately mixed with a large portion of the moderator; thus, fuel and solid moderator temperatures rise simultaneously, producing the temperature-dependent spectrum shift.

For reasons just discussed, more than 50% of the temperature coefficient for a standard TRIGA core comes from the temperature-dependent disadvantage factor, or cell effect, and ~20% of each come from Doppler broadening of the U-238 resonances and temperature-dependent leakage from the core. These effects produce a temperature coefficient of $\sim -10 \times 10^{-5}$ $\Delta k/k/°C$, which is essentially constant with temperature. On the other hand, for a TRIGA, FLIP and LEU core, the effect of cell structure on the temperature coefficient is smaller. Over the temperature range 23° to 700°C, about 70% of the coefficient comes from temperature-dependent changes in the parasitic absorption in the Er-167 in the core, and more than half of this effect is independent of the cell structure. Most of the remaining component of the prompt negative temperature coefficient is contributed by Doppler broadening of the U-238 resonances. Over the temperature range 23° to 700°C, the temperature coefficient for FLIP fuel with 20% enrichment in U-235 is about 1.5×10^{-4} Δk/k/°C and for TRIGA LEU fuel with 20 wt-% U is about 1.07×10^{-4} Δk/k/°C, thus being somewhat greater than the value for standard TRIGA fuel. The temperature coefficient of fuels containing Er-167 as a burnable poison are somewhat temperature dependent.

Fission Product Retention

A number of experiments have been performed to determine the extent to which fission products are retained by U-ZrH (TRIGA) fuel. Experiments on fuel with a uranium density of 0.5 g/cm^3 (8.5 wt-% U) were conducted over a period of 11 yr and under a variety of conditions (12). Results prove that only a small fraction of the fission products are released, even in completely unclad U-ZrH fuel. The release fraction varies from 1.5×10^{-5} for an irradiation temperature of 350°C to ~10^{-2} at 800°C.

The experiments show that there are two mechanisms involved in the release of fission products from U-ZrH fuel, each of which predominates over a different temperature range. The first mechanism is that of fission fragment recoil into the gap between the fuel and clad. This effect predominates in fuel at temperatures up to ~400°C; the recoil release rate is dependent on the fuel surface-to-volume ratio but is independent of fuel temperature. Above ~400°C, the controlling mechanism for fission product release from U-ZrH fuel is a diffusion-like process, and the amount released is dependent on the fuel temperature, the fuel surface-to-volume ratio, the time of irradiation, and the isotope half-life.

The results of the U-ZrH experiments, and measurements by others of fission product release from Space Nuclear Auxiliary Power Program (SNAP) fuel, have been compared and found to be in good agreement.

The fractional release, ϕ, of fission product gases into the gap between fuel and clad from a full-size standard U-ZrH fuel element is given by:

$$\phi = 1.5 \times 10^{-5} + 3.6 \times 10^{3} e^{-1.34 \times 104/(T + 273)}$$

where T = fuel temperature, °C. This relationship has also been found to apply to new LEU TRIGA fuels (19).

The first term of this function is a constant for low-temperature release; the second term is the high-temperature portion.

The function given above applies to a fuel element which has been irradiated for a time sufficiently long that all fission product activity is at equilibrium. Actual measured values of fractional releases fall well below that calculated by the function given above. Therefore, for safety considerations, this function gives conservative values for the high-temperature release from U-ZrH fuel.

The results of recent studies in the TRIGA reactor at GA on fission product release from fuel elements with high uranium loadings (up to 3.7 g U/cm^3, 45 wt-% U) agree well with data from older similar experiments with lower U loadings. As was the case with the lower U loadings, the release was determined to be predominantly recoil controlled at temperatures

≤400°C and controlled by a migration or diffusion-like process above 400°C. Low-temperature release appears to be independent of uranium loadings, but the high-temperature release seems to decrease with increasing weight fractions of uranium. The correlation used to calculate the release of fission products from TRIGA fuel remains applicable for the high uranium loaded (TRIGA LEU) fuels as well as the 8.5 wt-% U-ZrH fuel for which it was originally derived. This correlation predicts higher fission product releases than measurements would indicate up to 1100°C. At normal TRIGA operating temperatures (<750°C) there is a safety factor of approximately four between predicted and experimentally deduced values.

4.2.2 Control Rods

A layout of the WSU modified TRIGA reactor showing the positions of the control elements within the core is shown in Figure 4-10. The control elements consists of three safety blades, one control blade, and pulse rod.

The safety and regulating control elements of the WSU reactor are blade type elements as shown in Figures 4-11. The poison section of the safety blades is a boral sheet 40.5 inches long and 10.5 inches wide. The boral sheet is 3/8 inches thick and is clad with 1/8 inch aluminum. The regulating blade is a stainless steel sheet about 11 inches wide and 40 inches long.

Each blade is guided through its travel by a shroud, as shown in Figure 4-11. The shroud consists of two thin aluminum plates 38 inches high separated by aluminum spaces to provide a 3/4 inch control blade slot. Small flow holes are drilled at the bottom of the shroud to reduce the effects of viscous damping on the blade fall time.

[redacted]

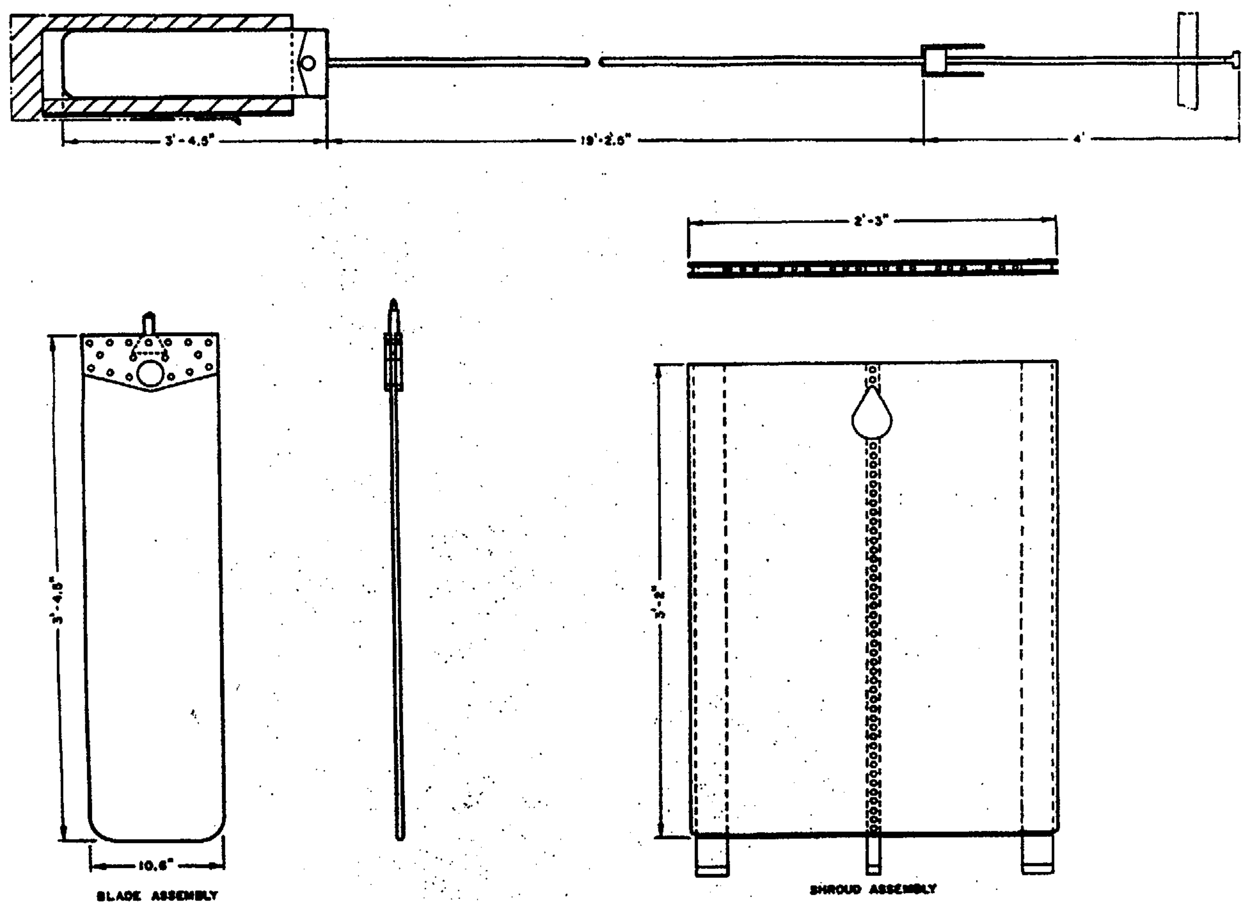


Figure 4-11
Control Element

Transient Control Rod

The transient control rod is a solid borated graphite cylinder contained in a 1 1/4 inch diameter stainless steel or aluminum tube as shown in Figure 4-12. The poison section of the transient rod is 15 inches in length. The transient rod is connected to the transient rod drive via an end fitting welded on the top end of the rod. The rod is held in position laterally by the guide tube inserted into a 3-rod cluster. A hold-down tube extends from the top of the guide tube up to the bottom of the transient rod drive, as shown in Figure 4-15.

Control Element Drive

The drive mechanism for the blade type control elements are shown in Figure 4-13 and are activated by reversible electric motors with an integral worm-gear drive mechanism. The worm-gear assembly serves to reduce the drive speed and to minimize over-travel of the drive after power is removed from the drive motor. A mechanical slip clutch on the output shaft limits the force on the blade to approximately 75 pounds. A ball-bearing screw and nut system is used to raise and lower the control element.

Each safety blade is coupled to its associated drive mechanism by means of an electromagnet and steel armature disk, as shown in Figure 4-14. De-energizing the electromagnet allows the safety blade to fall into the core by the action of gravity within 700 milliseconds. A shaft connects the armature disk to the blade and is fitted with polyethylene sleeve-bearings which control the lateral position of the blade drive shaft. A dashpot is positioned at the end of the shaft travel to decelerate the last 5 inches of fall. The blades are recovered after a scram by running the drive mechanism down and re-energizing the electromagnets.

The transient control rod drive employs a combination pneumatic-electromechanical drive assembly shown in Figures 4-15 and 16. The mechanism is designed to allow the rod to be used both as a control rod and a transient rod.

The pneumatic portion of the pneumatic-electromechanical drive, referred to herein as the "transient" rod drive, is basically a single-acting pneumatic cylinder. A piston within the cylinder is attached to the transient rod by means of a connecting rod. The piston rod passes through an air seal at the lower end of the cylinder. Compressed air is admitted at the lower end of the cylinder to drive the piston upward. As the piston rises, the air being compressed above the piston is forced out through vents at the upper end of the cylinder. At the end of its stroke, the piston strikes the anvil of a shock absorber. The piston is thus decelerated at a controlled rate during its final inch of travel. This action minimizes rod vibration when the piston reaches its upper-limit stop.

An accumulator tank mounted on the movable bridge stores the compressed air that operates the pneumatic portion of the transient rod drive. A three-way solenoid valve, located in the piping between the accumulator tank and the cylinder, controls the air supplied to the pneumatic cylinder. De-energizing the solenoid valve interrupts the air supply and relieves the pressure in the cylinder so that the piston drops to its lower limit by gravity. With this operating feature, the transient rod is inserted in the core except when air is supplied to the cylinder.

The electromechanical portion of the transient rod drive consists of an electric motor, a ball-nut drive assembly, and the externally threaded air cylinder. During electromechanical

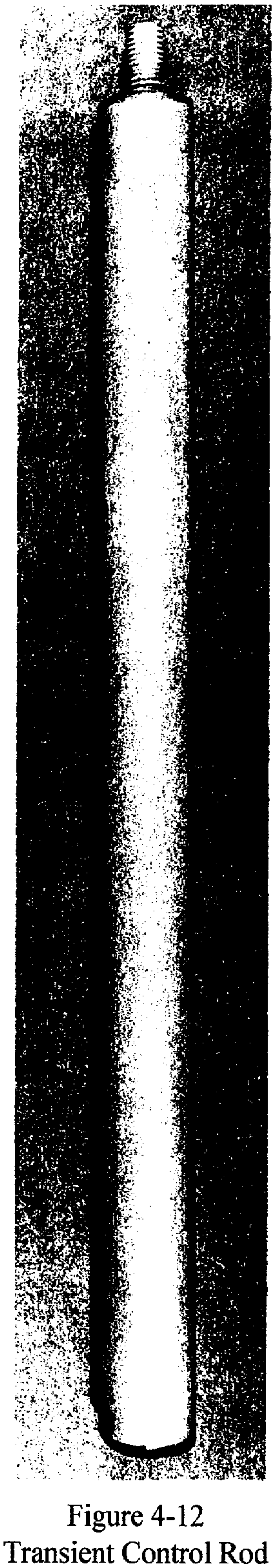


Figure 4-12
Transient Control Rod

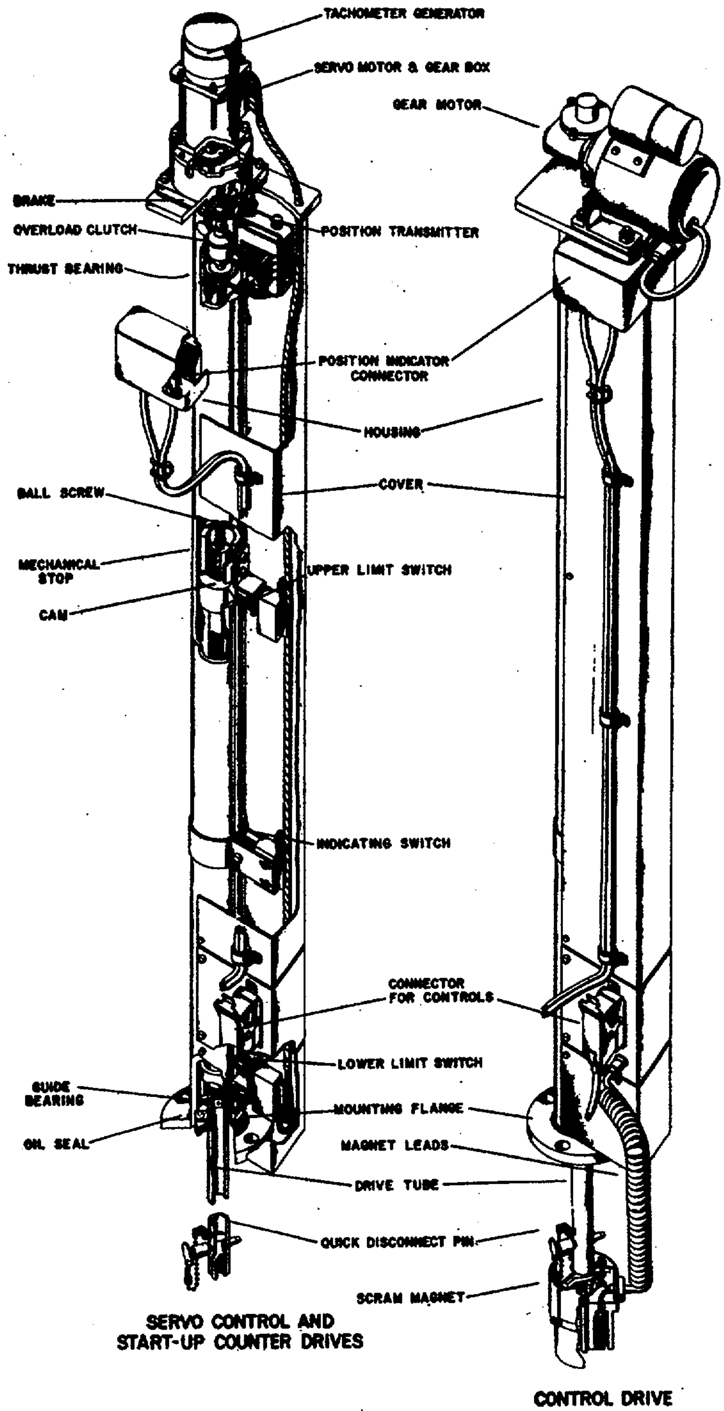


Figure 4-13
Drive Mechanisms

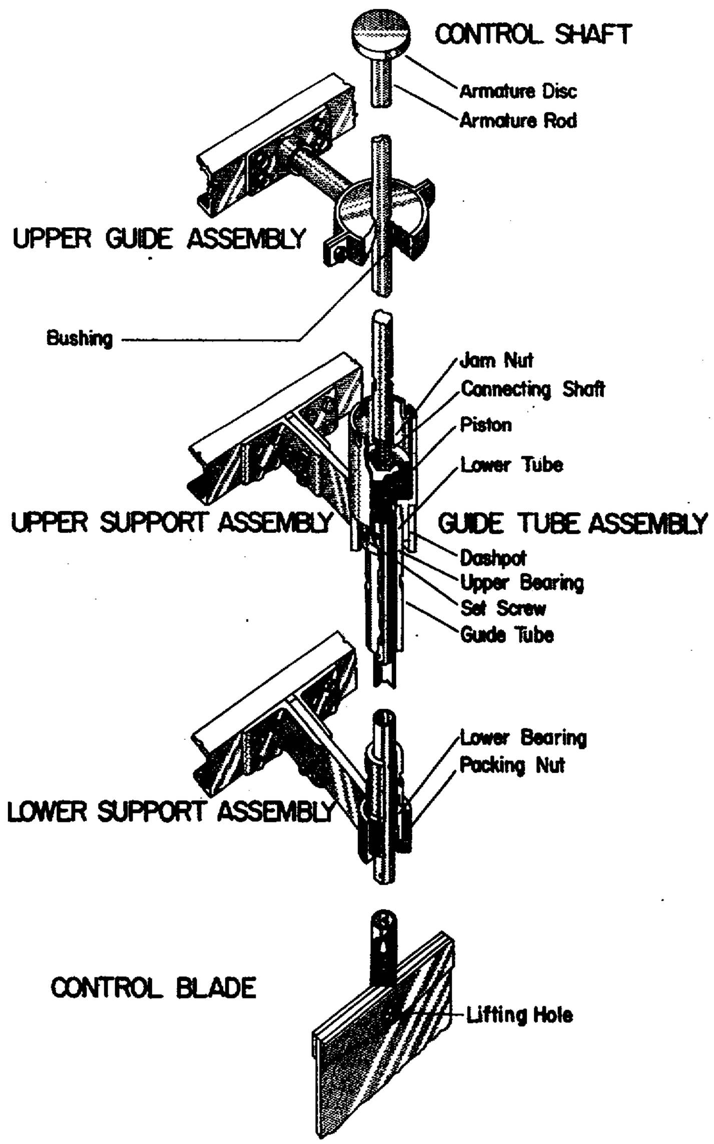


Figure 4-14
Control Shaft, Dash Pot, & Bearing Details

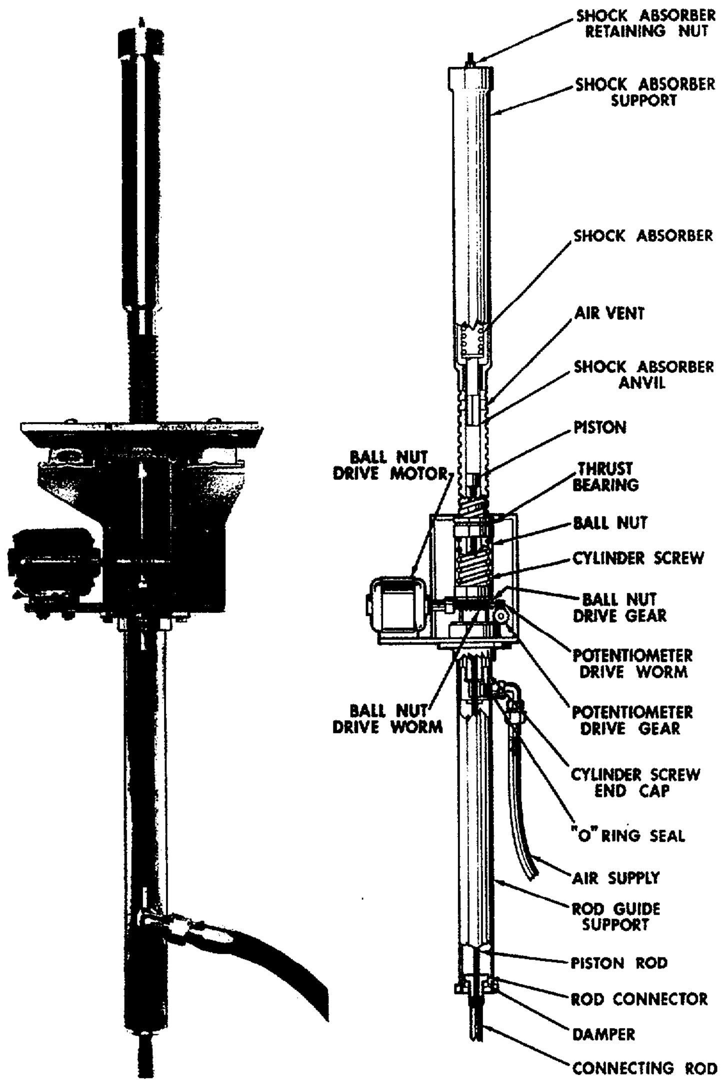


Figure 4-15
Transient Rod Drive Assembly

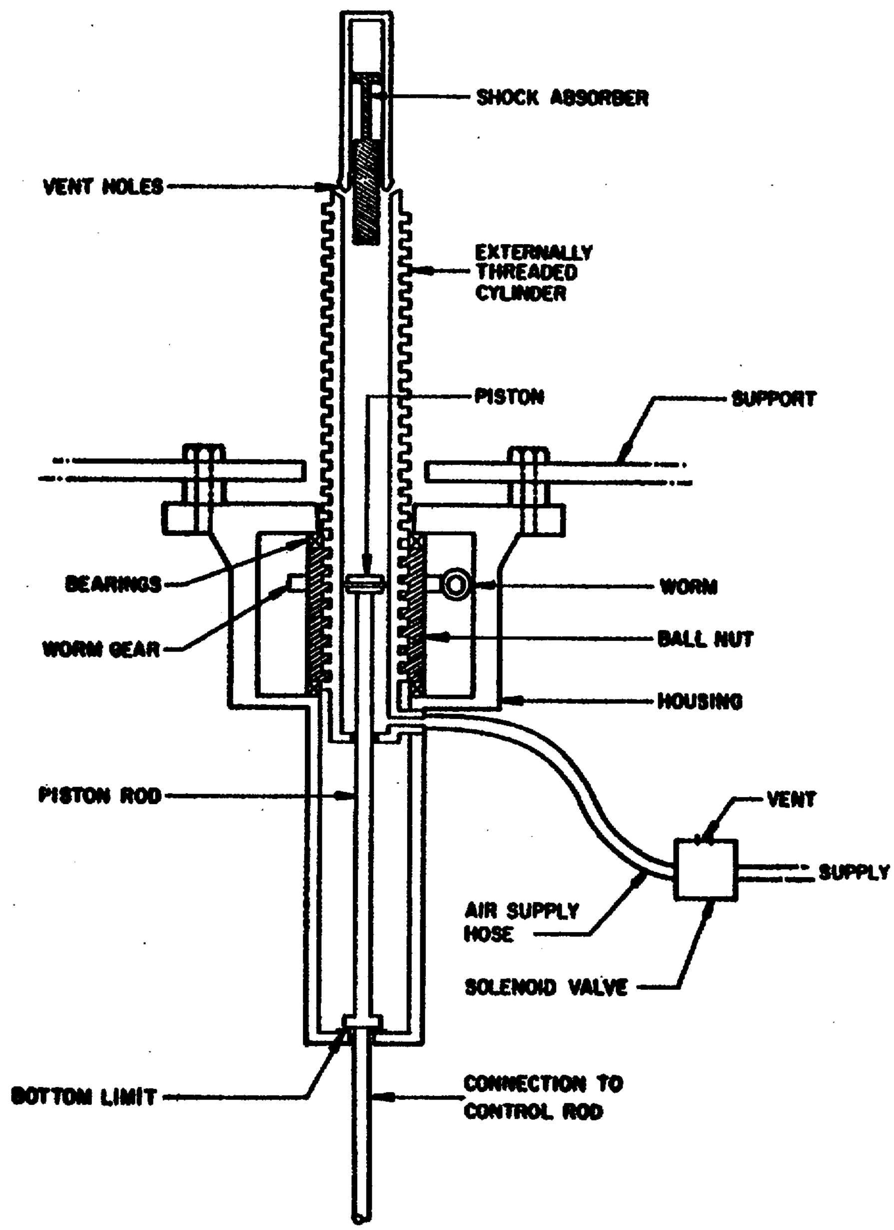


Figure 4-16
Schematic Drawing of Transient Rod Drive

operation of the transient rod, the threaded section of the air cylinder acts as a screw in the ball-nut drive assembly. These threads engage a series of balls contained in a ball-nut assembly in the drive housing. The ball-nut assembly is in turn connected through a worm-gear drive to an electric motor. The cylinder may be raised or lowered independently of the piston and control rod by means of the electric drive. Adjustment of the position of the cylinder controls the upper limit of piston travel, and hence controls the amount of reactivity inserted for a pulse.

A system of limit switches is used to indicate the position of the air cylinder and the transient rod. Two of these switches, the Drive Up and Drive Down switches, are actuated by a small bar attached to the bottom of the air cylinder. A third limit switch, the Rod Down switch, is actuated when the piston reaches its lower limit of travel. During steady state operation the transient rod may be withdrawn and used as a control rod by means of the ball-nut drive.

4.2.3 Neutron Moderator and Reflector

The moderator of a TRIGA type reactor is predominantly zirconium hydride as described in section 4.2.1. The cooling water also functions as a coolant and moderator in the space between the fuel rods. A TRIGA type reactor is a special type of open pool type reactor and the pool water functions as a reflector.

4.2.4 Neutron Startup Source

The startup source is an antimony-beryllium mixture canned in a cylinder that is positioned in a special graphite reflector element. The startup source may be located in any position in the grid box but in order to be effective during startup, a position near the edge of the core and across the core from the startup channel in a typical core as shown in Figure 4-4.

4.2.5 Core Support Structure

The WSU reactor is suspended in the pool from a movable bridge which is mounted on rails. The bridge and entire reactor structure may be moved laterally. The bridge and core suspension framework are shown in Figure 4-1. The all-aluminum framework is suspended from the bridge and supports the grid box into which the fuel is inserted. The hollow corner posts of the suspension framework serve as guide tubes for the nuclear instrumentation detectors. The control element drives are connected to and supported by the bridge structure.

Deck plates mounted on the top side of the bridge structure form a floor area around the control dives. The floor area provides a work space to use and maintain the reactor and associated facilities. A railing system is connected to the bridge floor to prevent personnel from accidently falling off the bridge structure.

4.3 Reactor Tank or Pool

The reactor pool is a reinforced, above ground, unlined concrete two section pool with a volume of 247,000 liters. The pool is penetrated by a thermal column and a number of beam ports as described in Section 10. A cross section of the pool is shown in Figure 4-17. One of the significant non-hazardous problems that have occurred at a number of pool type reactors is a large water leak in the pool. A number of facilities have had to insert metal pool liners in the pool to correct for a poorly built, leaky pool. The pool of the WSU reactor is quite old but was

[redacted]

built very well on compacted soil which minimizes the probability of a future significant leak problem. The pool did have some very minor leaks that persisted for years but until recently did not worsen with time and only accounted for about 5% of the required pool makeup feed water. About 95% of the makeup feed water was accounted for by pool surface evaporation.

The WSU reactor pool has two compartments and in the event of a significant pool leak, the leak is not likely to occur in both sections at the same time. Accordingly, the reactor core can be moved to the non-leaking section, the dam inserted between the two sections, and appropriate repairs to the leaky section undertaken. Such repairs were done in 1999 by draining one section of the pool while the core remained under water in the other section.

4.4 Biological Shield

The biological shield for the WSU modified TRIGA reactor consists of a combination of pool water and the concrete of the pool structure. The original design is that of the open pool reactor designed in the 1950's at Oak Ridge National Laboratory with the old plate type fuel being replaced with 4-rod clusters of TRIGA rod type fuel.

A plan view drawing of the pool structure is shown in Figure 4-18. The biological shielding portion of the pool structure consists of [redacted] The pool has numerous penetrations consisting of beam tubes and a thermal column. The dose rate at the outside surface of the pool structure in the beam room is normally 0.2 mrem/hr when the reactor is at full power.

4.5 Nuclear Design

4.5.1 Normal Operating Conditions

The normal operating conditions of the WSU modified TRIGA reactor at full power are given in Table 4.5-1 below.

Table 4.5-1

Power Level	1.0 MW	
Pool Water Temperature	31-34°C	
Bridge Radiation Level	2.0 mrem/hr	
Fuel Temperature as measured by two instrumental fuel elements	310°C and 278°C	
Maximum Calculated Fuel Temperature	405°C	
Peak/Average Fuel Temperature	1.6	
Average Fuel Temperature	265°C	
Maximum Power Density, kW/rod	16.5	
Pool Cooling System		
	Flow Rate in GPM	ΔT in °C
Primary	450	20
Secondary	900	10
Excess Reactivity at Power	$4 to $5	
Total Rod Worth	~$14.00	

4.5.2 Reactor Core Physics Parameters

The basic core physics parameters for the WSU TRIGA reactor for a 100% FLIP core is given in Table 4.5-2. This data was calculated by General Atomics using their extensive calculational and operational experience with TRIGA reactor cores. Extensive details on the calculation on the important characteristics and performance of the WSU TRIGA reactor will be covered in the following portions of this section of this SAR. The only two parameters that change significantly from core to core and fuel type as well as burnup are: 1) the prompt neutron lifetime and 2) the prompt negative temperature coefficient. These will be described at the end of this section.

Table 4.5-2
WSU Modified TRIGA Reactor Core Physics Parameters -- FLIP Fueled

Cold, clean critical loading	[redacted]	
Operational loading	[redacted]	
Prompt Neutron Lifetime		
Beginning of Life:		
Compact core (100 elem.)	24×10^{-6} sec	
End of Life (3000 MW days):		
Compact core	$\sim 30 \times 10^{-6}$ sec	
Effective Delayed Neutron Fraction	0.0071	
Prompt Temperature Coefficient	23°C→310°	23°C→700°C
Beginning of Life:	-\$.015/°C	-\$.0145/°C
T_f Average Fuel Temperature	23°C	310°C
T_w Average Water Temperature	23°C	23°C
η Reproduction Factor	2.0657	2.0611
Thermal Utilization	0.7353	0.7247
Resonance Escape Probability	0.703	0.701
Fast Fission Factor	1.179	1.183
Infinite Multiplication Factor	1.2598	1.2381
Fermi A_{sc}	22.18 cm^2	22.18 cm^2
Buckling	.017 cm^{-2}	.017 cm^{-2}
Total Leakage	0.8097	0.8077
Finite Multiplication Factor	1.020	1.000

4.5.2.1 Core Calculations

The calculations of the characteristics of the WSU TRIGA reactor core with standard fuel, and mixtures of FLIP and standard fuel and LEU and standard fuel were performed using the EXTERMINATOR-2 code (4) and temperature dependent cross section data obtained from General Atomics. The fast cross section data were generated with the GGC-4 code (5) and the thermal cross section data using the SUMMIT and THERMIDORE codes.(6,7) Seven energy groups were used in the core calculations as well as group dependent buckling for the FLIP and LEU fuels as listed in Table 4.5-3. This group structure and energy dependent buckling is identical to that conventionally used by General Atomics in the TRIGA reactor calculations including the Puerto Rico FLIP fueled reactor calculations.

Table 4.5-3
Energy Groups and Group Dependent Buckling Use
in WSU TRIGA Core Calculations

Group		Standard Fuel Buckling	FLIP Fuel Buckling	20/20 LEU
1	15.0 -.694 MeV	0.0041	0.00546	.00601
2	639 - 9.12 keV	0.0041	0.00435	.00473
3	9.12 - 0.001125 keV	0.0041	0.00347	.00341
4	1.125 - 0.414 eV	0.0041	0.000324	.00012
5	.414 - .14 eV	0.0041	0.00469	.00245
6	.14 - .05 eV	0.0041	-0.0146	-.01398
7	.05 - .0002 eV	0.0041	-0.0614	-.08018

The results of calculations made with the EXTERMINATOR-2 on the existing core of the WSU TRIGA reactor compare favorably with measured values. Furthermore, calculations on the Puerto Rico FLIP core at WSU with the code yield results comparable to those obtained by General Atomics. Thus the calculations performed on all standard and all FLIP fueled cores are known to be accurate and reliable. Consequently, the results obtained on mixed cores can be expected to be reasonably accurate and reliable. Furthermore, Texas A&M has had good success in using this code for calculating the characteristics of mixed cores.(8)

The basic cell data used with the EXTERMINATOR code for making core calculations on the WSU modified TRIGA reactor are given in Tables 4.5-4, 4.5-5, and 4.5-6. It is to be noted that the erbium content reported in Table 4.5-6 for LEU-20/20 fuel is as presently proposed by General Atomics for this fuel. However, calculations performed both at WSU and OSU indicated that additional erbium would be needed for a practical LEU core if and when such LEU fuel is manufactured.

Table 4.5-4

Basic Cell Data for Core, Standard TRIGA Fuel Element

8.5U(20)-$ZrH_{1.65}$

Fuel Height = 15 in. (38.1 cm)
Clad Fuel OD = 1.411 in. (3.58 cm)

	Radius		Area	Volume	Volume
Cell Region	in.	cm	(cm^2)	(cm^3)	Fraction
8.5U(20)-$ZrH_{1.65}$	0.6855	1.7412	9.5246*	362.88*	0.6101
Stainless Steel Clad	0.7055	1.7920	0.5639	21.48	0.0361
Water		2.2291	5.5221	210.39	0.3537
Total Cell			15.6106	594.75	1.0000

Fuel Loading

U-235 .... [redacted]

Fuel Moderator	Cell Atomic Densities	Homogenized Atom Densities
H(ZrH)	0.056678 x 10^{24}Nuc/cc	0.034579 x 10^{24}Nuc/cc
Zr	0.035626	0.021735
U-235	0.000252	0.000154
U-238	0.000995	0.000607
Cladding		
Stainless Steel	0.0843	0.003043
Water		
$H(H_2O)$	0.0668	0.023627
Oxygen	0.0334	0.011814

*Approximately 3% of this value is removed by drilling a hole through the axial center of the fuel-moderator cylinder to facilitate hydriding. The hole is filled with a zirconium rod. This is reflected in the weights and densities quoted.

Table 4.5-5

Basic Cell Data for Core, TRIGA-FLIP Fuel Element

8.5U(70)-$ZrH_{1.60}$

Fuel Height = 15 in. (38.1 cm)
Clad Fuel OD = 1.411 in. (3.58 cm)

	Radius		Area	Volume	Volume
Cell Region	in.	cm	(cm^2)	(cm^3)	Fraction
8.5U(70)-$ZrH_{1.60}$	0.6855	1.7412	9.5246*	362.88*	0.6101
Stainless Steel Clad	0.7055	1.7920	0.5639	21.48	0.0361
Water		2.2291	5.5221	210.39	0.3537
Total Cell			15.6106	594.75	1.0000

Fuel and Burnable Poison Loading

U-235 . . . . [redacted]
Natural Er. . . . . .31.01 g/rod

Fuel Moderator	Cell Atomic Densities	Homogenized Atom Densities
H(ZrH)	0.054992 x 10^{24}Nuc/cc	0.033551 x 10^{24}Nuc/cc
Zr	0.035645	0.021747
U-235	0.000883	0.000539
U-238	0.000373	0.000228
Er-166		0.0000648
Er-167		0.0000444
Cladding		
Stainless Steel	0.0843	0.003043
Water		
$H(H_2O)$	0.0668	0.023627
Oxygen	0.0334	0.011814

*Approximately 3% of this value is removed by drilling a hole through the axial center of the fuel-moderator cylinder to facilitate hydriding. The hole is filled with a zirconium rod. This is reflected in the weights and densities quoted.

Table 4.5-6

Basic Cell Data for Core, LEU-20/20 TRIGA Fuel Element

U(20)-$ZrH_{1.60}$

Fuel Height = 15 in. (38.1 cm)
Clad Fuel OD = 1.411 in. (3.58 cm)

	Radius		Area	Volume	Volume
Cell Region	in.	cm	(cm^2)	(cm^3)	Fraction
20(20)-$ZrH_{1.60}$	0.6855	1.7412	9.5246*	362.88*	0.6101
Stainless Steel Clad	0.7055	1.7920	0.5639	21.48	0.0361
Water		2.2291	5.5221	210.39	0.3537
Total Cell			15.6106	594.75	1.0000

Fuel and Burnable Poison Loading

U-235 .... [redacted]
Natural Er. ....12.38 g/rod

Fuel Moderator	Cell Atomic Densities	Homogenized Atom Densities
H(ZrH)	0.052021 x 10^{24}Nuc/cc	0.030786 x 10^{24}Nuc/cc
Zr	0.034038	0.020921
U-235	0.000659	0.000390
U-238	0.002651	0.001569
Er-166	0.0000418	0.0000259
Er-167	0.0000287	0.0000177
Cladding		
Stainless Steel	0.0843	0.003043
Water		
H(H_2O)	0.0668	0.023627
Oxygen	0.0334	0.011814

*Approximately 3% of this value is removed by drilling a hole through the axial center of the fuel-moderator cylinder to facilitate hydriding. The hole is filled with a zirconium rod. This is reflected in the weights and densities quoted.

4.5.2.2 Fuel Element Temperature

The direct theoretical calculation of accurate fuel element temperatures for steady-state operation in a TRIGA core using natural convection cooling is very difficult. This is due to the fact that the contact coefficient between the fuel and the cladding is not known accurately, especially in fuel rods that have been pulsed. In other words, the fuel-cladding thermal contact coefficient is a function of the fuel temperature and the pulsing history and age of the fuel. In addition, the film coefficient of heat transfer between the cladding and the coolant under conditions of natural convection is uncertain. The film coefficient is a function of the coolant temperature, coolant velocity, and effective hydraulic diameter of the coolant channel.

In order to circumvent these uncertainties, we have chosen to utilize experimentally measured fuel temperature data obtained from the instrumented fuel rod coupled with calculated power distribution data obtained with the EXTERMINATOR-2 code. A graph of the maximum fuel temperature of the instrumented fuel rod as a function of reactor power in WSU TRIGA core No. 28A during steady-state operation is shown in Figure 4-19. Calculations with the EXTERMINATOR-2 code indicate that the axial average power density in the instrumented fuel rod is .283 watts per kW of reactor power per cm of core height. Combining this result with the experimental power-temperature data yields a relationship between axial average fuel rod power density and fuel rod temperature. This relationship as determined by least squares fitting of the data to a polynomial is given in Appendix 4-A.

The fuel temperatures calculated using the derived equation are shown in terms of power density in Figure 4-20. The maximum observed power density used in deriving the fuel temperature equation was 12 kW/rod and thus a linear extrapolation is used above this value. A comparison of the curve in Figure 4-20 with experimental measurements at Texas A&M and PRNC which are also displayed on the graph substantiate the validity of the relationship. That is, fuel rod temperatures calculated by the WSU equation are essentially identical or more conservative than measured values.

Due to the large number of calculations involved and the number of core configurations studied, a special program, PLDEQC (9), was written to calculate and display the neutron flux, power, and temperature distributions. This program utilized the output group flux and power generation matrixes from EXTERMINATOR-2 to make these calculations.

4.5.2.3 Prompt Negative Temperature Coefficient

Calculations of the prompt temperature coefficient of the WSU TRIGA reactor for all standard fueled cores were performed using the EXTERMINATOR-2 code and temperature dependent cross sections. The data thus obtained is plotted on the graph in Figure 4-21. The results thus obtained locally are essentially identical to those reported by General Atomics for similar type fuels. Experimental measurements of the overall temperature coefficient of WSU TRIGA core No. 28A at full power with all standard fuel yielded a result of -\$0.014/°C. The average fuel temperature for this core at 1 MW is 194°C which would give a theoretical value of -\$.0147/°C for the temperature coefficient. Since it is a known fact that a TRIGA reactor has a slightly positive bath coefficient, the calculated and measured values agree as well as can be expected. Figure 4-21 also contains temperature coefficient data for LEU 20/20 calculated by General Atomics.(20)

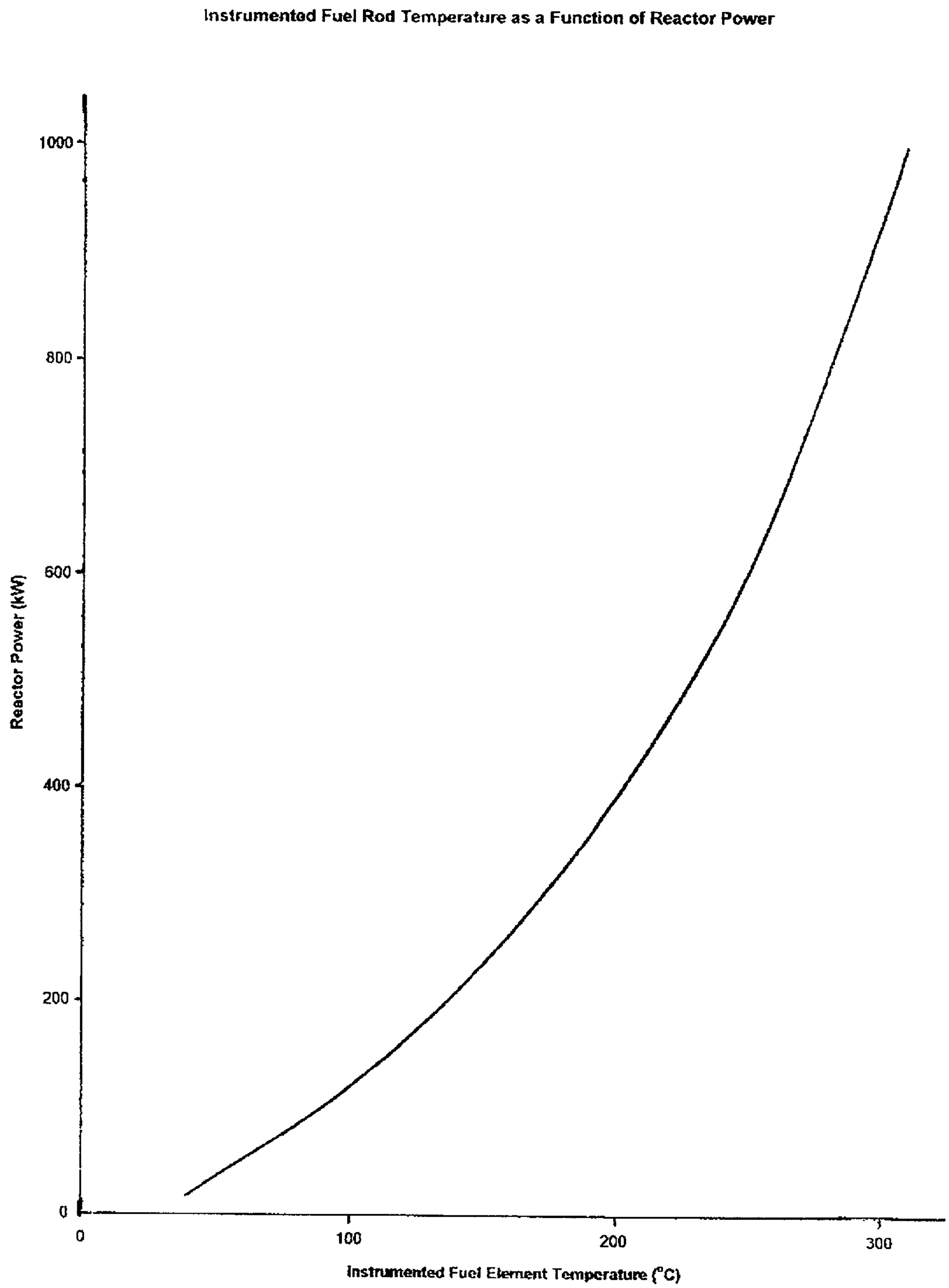
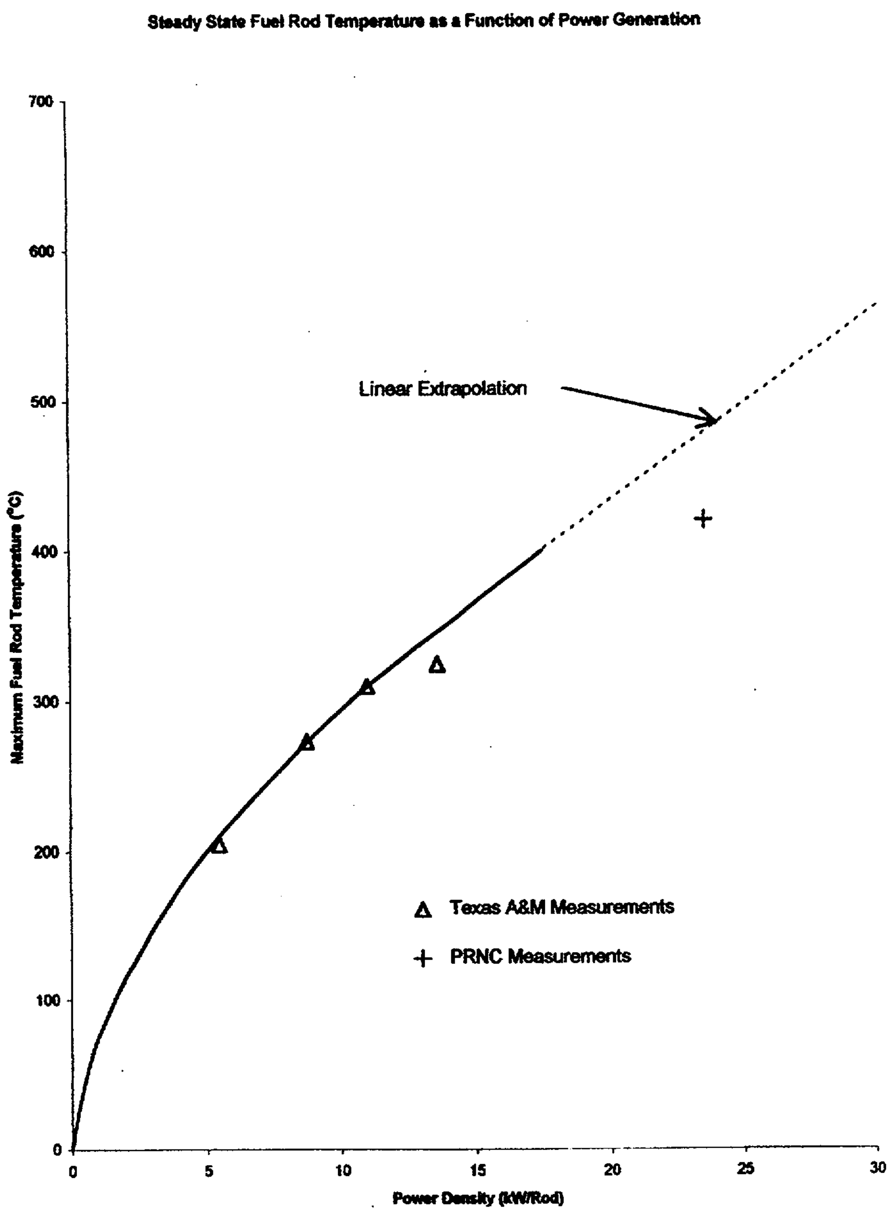

Instrumented Fuel Rod Temperature as a Function of Reactor Power
Reactor Power (kW)
1000
800
600
400
200
0
0
100
200
300
Instrumented Fuel Element Temperature (°C)

Figure 4-19



Steady State Fuel Rod Temperature as a Function of Power Generation
Maximum Fuel Rod Temperature (°C)
Power Density (kW/Rod)
Linear Extrapolation
Texas A&M Measurements
PRNC Measurements
0
100
200
300
400
500
600
700
0
5
10
15
20
25
30

Figure 4-20

CALCULATED PROMPT NEGATIVE TEMPERATURE COEFFICIENT

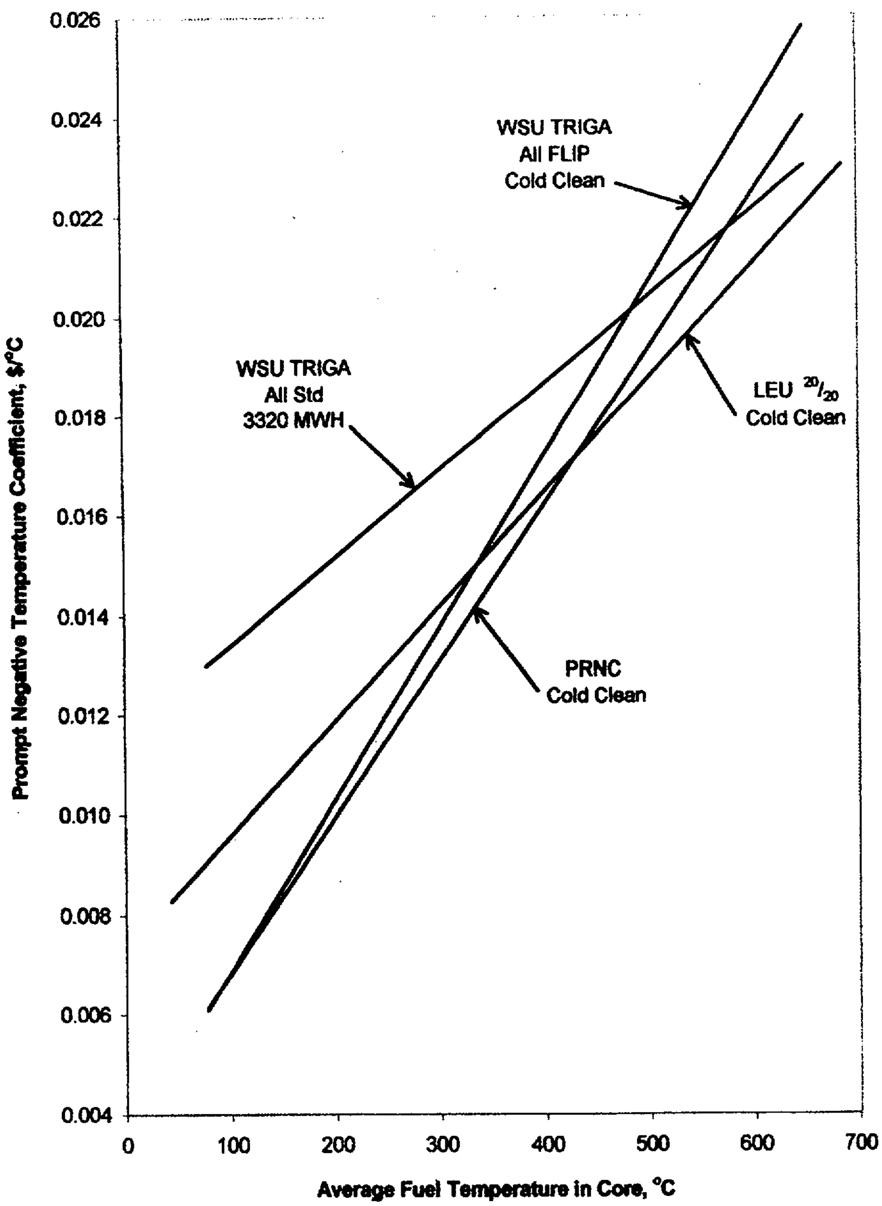


Figure 4-21

The direct calculation of a meaningful temperature coefficient for a mixed standard-FLIP or mixed standard-LEU fueled core with the EXTERMINATOR-2 code is not possible with the limited temperature dependent cross section data that was readily available. An alternate technique was adopted involving the summing of the appropriately weighted temperature coefficients of the standard and FLIP fueled regions of a mixed core (see Appendix 4-B). A similar relationship could be used for mixed standard-LEU fueled cores.

Accordingly, the mixed core prompt negative temperature coefficient is given by:

$$\text{Tcm} = \text{Tcf}\left(\overline{\text{Tf}}\right) \times \frac{\text{FF}}{\text{FT}} + \text{Tcs}\left(\overline{\text{Ts}}\right) \times \frac{\text{FS}}{\text{FT}}$$

where Tcm = Temperature coefficient of mixed core at mean core temperature.

$\text{Tcf}(\overline{\text{Tf}})$ = Temperature coefficient of all FLIP core at temperature $\overline{\text{Tf}}$

$\text{Tcs}(\overline{\text{Ts}})$ = Temperature coefficient of all FLIP core ate temperature $\overline{\text{Ts}}$

$\overline{\text{Tf}}$ = Mean temperature of FLIP fuel region

$\overline{\text{Ts}}$ = Mean temperature of standard fuel region

FF = Fissions in FLIP fuel region

FS = Fissions in standard fuel region

FT = Fissions in total core.

4.5.2.4 Pulsing Calculations

The peak power of a TRIGA reactor during a transient can be accurately described using a Fuchs-Nordheim model with variable heat capacity. Additional analysis of transient fuel rod temperatures by the group at the University of Illinois has shown that the values calculated by the model are conservative (high) (10). According to this model the peak reactor power during the transient, P(max), is given by

$$P(\max) = \text{Po} + \frac{C(\rho - 1)^2}{\alpha \ell}\left[\frac{1 + 3\sigma}{6\sigma}\right]$$

where $C = (Co + \gamma\, To)N$ = Heat capacity of core at initiation of pulse

N = Number of rods in core

$\sigma = \alpha C/[N * (\rho - 1)]$

Co = Heat capacity of TRIGA fuel at 25°C = 269 watt-sec/°C/rod

To = Average initial temperature of the core above 25°C

ℓ = Prompt neutron lifetime of core

α = Prompt negative temperature coefficient

γ = Rate of change in heat capacity of TRIGA fuel - 1.47 watt-sec/°C/rod

ρ = Reactivity inserted.

In addition, the average peak core temperature, $\overline{Tx}$, is given by

$$\overline{Tx} = \frac{(\rho - 1)}{\alpha}\left[\frac{-3}{4}(\sigma - 1) + \frac{3}{4}\sqrt{(s-1)^2 + \frac{16\sigma}{3}}\right] + (To + 25)$$

The results of calculations using the above two equations for a 100 rod core are given in Table 4.5-7 of this report.

Table 4.5-7

Pulsing Characteristics of WSU TRIGA Reactor
100 Rod Core, Temp coefficient = -0.0125
Prompt neutron lifetime = 24 microseconds
Pulse starting at To = 35°C, Po = 300 watts

Reactivity	Megawatts	Degrees C	Megawatt-sec	Watt-sec/cc
\$ = 1.50	P(max) = 234.30	Tp = 113.22	E = 8.87	E/cm^3 = 244.5
\$ = 1.55	P(max) = 284.19	Tp = 120.87	E = 9.59	E/cm^3= 264.4
\$ = 1.60	P(max) = 339.03	Tp = 128.49	E = 10.32	E/cm^3 = 284.4
\$ = 1.65	P(max) = 398.86	Tp = 136.08	E = 11.05	E/cm^3 = 304.6
\$ = 1.70	P(max) = 463.70	Tp = 143.65	E = 11.79	E/cm^3 = 324.9
\$ = 1.75	P(max) = 533.60	Tp = 151.19	E = 12.53	E/cm^3 = 345.4
\$ = 1.80	P(max) = 608.58	Tp = 158.71	E = 13.28	E/cm^3 = 366.0
\$ = 1.85	P(max) = 688.68	Tp = 166.21	E = 14.04	E/cm^3 = 386.9
\$ = 1.90	P(max) = 773.94	Tp = 173.68	E = 14.80	E/cm^3 = 407.8
\$ = 1.95	P(max) = 864.38	Tp = 181.13	E = 15.56	E/cm^3 = 429.0
\$ = 2.00	P(max) = 960.05	Tp = 188.56	E = 16.34	E/cm^3 = 450.3
\$ = 2.05	P(max) = 1060.98	Tp = 195.96	E = 17.12	E/cm^3 = 471.8
\$ = 2.10	P(max) = 1167.19	Tp = 203.35	E = 17.90	E/cm^3 = 493.4
\$ = 2.15	P(max) = 1278.74	Tp = 210.71	E = 18.69	E/cm^3 = 515.2
\$ = 2.20	P(max) = 1395.64	Tp = 218.06	E = 19.49	E/cm^3 = 537.1
\$ = 2.25	P(max) = 1517.94	Tp = 225.39	E = 20.29	E/cm^3 = 559.2
\$ = 2.30	P(max) = 1645.67	Tp = 232.69	E = 21.10	E/cm^3 = 581.5
\$ = 2.35	P(max) = 1778.86	Tp = 239.98	E = 21.91	E/cm^3 = 603.9
\$ = 2.40	P(max) = 1917.55	Tp = 247.25	E = 22.73	E/cm^3 = 626.5
\$ = 2.45	P(max) = 2061.77	Tp = 254.51	E = 23.56	E/cm^3 = 649.3
\$ = 2.50	P(max) = 2211.56	Tp = 261.75	E = 24.39	E/cm^3 = 672.2

4.5.2.5 Peaking Factors

Core power peaking factors were determined by multiplying the following terms, where $\hat{P}$ represents the maximum power density in the core:

1. $\bar{P}$ (cell with $\hat{P}$) -- The average radial power in the cell where $\hat{P}$ occurs (based on radial core average power = 1.0). This value comes from an x-y GAMBLE calculation or 1D GAZE result for a compact core. Values are listed in Table 4.5-8.
2. $\hat{P}/\bar{P}$ (cell with $\hat{P}$) -- The peak-to-average power value in the cell where $\hat{P}$ occurs. For a compact core, this can be the power distribution from an infinite medium cell calculation. For flux traps the cell is discretely described next to the water filled trap (with no experiment in the trap for maximum power peaking). Values are listed in Table 4.5-7.
3. P_T -- A peaking correction due to temperature since items 1 and 2 are computed with cold fuel. This correction is based on the difference in peaking values calculated for cold and hot cells in transport theory when disadvantage factors were computed for the core cross-sections. The value for P_T is 1.05 for FLIP fuel. There was negligible change in the value of $\bar{P}$ (cell with $\hat{P}$) for a hot core versus a cold core.
4. $\hat{P}/\bar{P}_{axial}$ -- The axial peak to average power ratio which was calculated to be 1.25 from an RX GAMBLE calculation. This is a standard value for the TRIGA size fuel elements.

Table 4.5-8
Power Peaking Factors TRIGA-FLIP

Configuration	$\bar{P}$ (cell with $\hat{P}$)	$\hat{P}/\bar{P}$ (cell with $\hat{P}$)
Compact	1.60	1.43
Single central water cell	1.72	2.24
Most central 3-cell flux trap (water flooded)	2.09	2.65
Most central 3-cell flux trap with typical thermionic experiment	1.80	----

This method of determining the core peaking factors has evolved during analysis of several fuel and reactor development programs and has served well for analysis of recent experiments. The peaking values given here are all for beginning of life conditions. Power distributions from extensive analysis of the burnup history of the core, involving several two-dimensional burnup calculations, have shown the total peaking value next to a water-filled flux trap after 3000 MW days of operation to be 0.75 times the value at beginning of life. The 3000 MW day peaking value next to a flux trap containing a typical thermionic experiment is 0.83 times the beginning of life value.

Power peaking data from rod worth calculations show that the power peaking next to flux traps containing typical thermionic experiments is very slightly reduced when the control rods are inserted.

Additional calculations have shown that the power density in an element is not strongly influenced by fractional variations in the erbium-167 content. A 5% decrease in erbium-167 in a given element relative to its neighbors, is calculated to increase the power density by <1%. It is estimated that a 2.5% to 3% increase in uranium in a given element would increase its power density by ~1%.

4.5.2.6 Delayed Neutron Fraction

The delayed neutron fraction in a specific reactor is predominantly dependent upon the basic delayed neutron parameters of the nuclear species that make up the fuel for a reactor. Of secondary importance is the type of moderator used in the reactor. Thus, all U-235 fueled water cooled reactors including TRIGA reactors have a delayed neutron fraction of the order of .007.

The specific value for a TRIGA type reactor has been calculated by General Atomics using the relationship

$$\beta\text{eff} = \frac{k_t(1-\beta_0)}{k_p}$$

and yielding βeff = .0071

where β_0 = actual delayed neutron fraction (.0065).

This calculation involves first computing the neutron multiplication with a prompt negative spectrum (k_p) and then recalculation with a neutron spectrum accounting for both the prompt and delayed neutron (k_t).

4.5.2.7 Neutron Lifetime Calculation by the l/v Absorber Addition Method

Consider a homogeneous unreflected thermal neutron reactor. If a small amount of a l/v absorber, such as boron, is distributed uniformly throughout the core the macroscopic absorption cross-section Σa, of the core would be increased by $\Delta\Sigma a$. The reactivity effect of the added boron is obviously related to the change in the core's neutron multiplication factor.

$$\text{Reactivity} = P = \frac{\Delta k}{k} = \frac{k^1 - k}{k^1}$$

where k and k^1 are the multiplication factors before and after the boron addition respectively.

The multiplication factor of the reactor is

$$k = \frac{\eta p f \varepsilon}{1 + L^2B^2} = \frac{\eta p f \varepsilon \Sigma a}{\Sigma a + DB^2} = \frac{\eta p f \varepsilon \Sigma a \Sigma \mu}{(\Sigma a + DB^2)\Sigma a} = \frac{\eta p \varepsilon \Sigma \mu}{\Sigma a + DB^2}$$

where $1/(1 + L^2B^2)$ is the thermal neutron non-leaking probability. Similarly

$$k^1 = \frac{\eta p \varepsilon \Sigma \mu}{\Sigma a + \Delta \Sigma a + DB^2}$$

Combining above three equations yields

$$P = \frac{\dfrac{\eta p \varepsilon \Sigma \mu}{\Sigma a + \Delta \Sigma a + DB^2} - \dfrac{\eta p \varepsilon \Sigma \mu}{\Sigma a + DB^2}}{\dfrac{\eta p \varepsilon \Sigma \mu}{\Sigma a + \Delta \Sigma a + DB^2}} = 1 - \frac{\Sigma a + \Delta \Sigma a + DB^2}{\Sigma a + DB^2}$$

$$P = \frac{\Delta k}{k} = -\frac{\Delta \Sigma a}{\Sigma a + DB^2}$$

If the concentration of boron atoms added to the core per unit volume is N_b, then

$$\Delta \Sigma a = \sigma_a(V) N_b.$$

Combining this fact with the previous result yields

$$P = -\frac{\sigma_a(V) N_b}{\Sigma a + DB^2}$$

By definition the prompt neutron lifetime in an infinite reactor is given by $\ell_o = 1/\Sigma a V$ and that for a finite reactor is given by

$$\ell = \frac{\ell_o}{1 + L^2 B^2} = \frac{\ell_o \Sigma a}{\Sigma a + DB^2} = \frac{1}{V\left(\Sigma a + DB^2\right)}$$

Solving this equation for $(\Sigma a + DB^2)$ and substituting the results of the previous equation gives

$$P = -\sigma_a(V) V N_b \ell$$

The absorption cross-section of boron varies inversely with neutron velocity and consequently

$\sigma_a(V)V = \sigma_o V_o$ and substituting this fact into the above equation yields

$P = -\sigma_o V_o N_b \ell$

or $\quad \ell = -P/(\sigma_o V_o N_b)$

For natural boron, $\sigma_o = 755$ barns and $V_o = 2.2 \times 10^5$ cm/sec so that

$\ell = -P/(755 \times 10^{-24} \times 2.2 \times 10^5 \times N_b)$

or $\quad \ell = -P/\omega$

where $N_b = \dfrac{\omega}{\sigma_o V_o} = 6.024 \times 10^{-9}\left(\times 10^{24}\right) \times \omega$

The results of a calculation of the neutron lifetime for WSU TRIGA core 31A using the TRG3 code by the method described above is given in Table 4.5-9.

TABLE 4.5-9
Neutron Lifetime Calculation For Core 31A

N_b	ω	k_{eff}	Δk_{eff}	$\ell(\mu sec)$
0	0	1.0504589	-	-
6.024×10^{-8}	10	1.0502224	2.37×10^{-4}	24
6.024×10^{-7}	100	1.0480728	2.39×10^{-3}	24

4.5.3 Operating Limits

4.5.3.1 General Considerations

The safety related aspects of the operation of the WSU Modified TRIGA reactor which establish the reactor operating limits will be considered in this section. First we will examine the fundamental characteristics and operational parameters of a TRIGA-type reactor as they relate to safety. Then we will examine the consequences of a number of postulated accidents.

4.5.3.2 Design Bases

The operating limits for the Washington State University TRIGA reactor are determined by the maximum safe operational capabilities of the solid-fuel moderator elements used to fuel the WSU modified TRIGA reactor and the core configuration described in this report. The combination of fuel and core configuration must be selected to provide a high degree of operational safety independent of mechanical, electrical, or human errors. To attain this, the following characteristics and properties must be inherent in the reactor system.

a. Large prompt negative temperature coefficient of sufficient magnitude to control the effects of a sudden large insertion of positive reactivity.
b. Metallurgical properties of the fuel-moderator alloy that would insure integrity of the fuel-moderator alloy during either a sudden increase in temperature or prolonged periods of operation at high temperatures.
c. A suitable cladding that would contain the fuel-moderator material and associated fission products under all operating conditions. Cladding integrity must be maintained under the expected thermal and mechanical stresses and strains resulting from sudden increases in temperature and prolonged periods of operation at high temperature.
d. A core configuration that is slightly undermoderated to provide a negative void coefficient and to insure safety in case of a loss-of-water accident.

4.5.3.3 Design Limits

A considerable amount of theoretical analysis has been performed and a large amount of operational experience has been accumulated on TRIGA-type reactors over the past decade. This accumulation of knowledge and experience has led to the establishment of certain design limits for TRIGA-type reactors. These limits may be categorized as (1) shutdown margin limit, (2) reactivity addition rate limit, (3) fuel operating temperature limit, (4) operating power limit, and (5) reactivity addition limit during pulsing.

4.5.3.3.1 Shutdown Margin

The aggregate worth of the control elements of a reactor must be set so that a safe shutdown margin is obtained with the highest worth control element fully withdrawn from the core. This requirement insures that the reactor remains sub-critical during core changes with one control element withdrawn. The total control element worth necessary is obviously determined by the excess reactivity available and the worth of the individual control elements. The safe shutdown margin for TRIGA reactors has been set at 0.2% Δk/k or about $.25. The highest worth control element in most TRIGA reactors is the pulse rod which generally has a maximum worth limit of $4.00 as determined by pulsing considerations. Accordingly, in normal operation a TRIGA reactor is shut down by $4.25 with all the control elements inserted.

4.5.3.3.2 Reactivity Addition Rate

The reactivity addition rate to a reactor during normal operation is a function of the worth of the control elements, the speed of element withdrawal, and the number of elements being withdrawn at one time. In a TRIGA reactor the control system is generally designed to allow the withdrawal of only one element at a time. Thus the maximum reactivity addition rate is equal to the product of the maximum differential worth of the most reactive element in dollars per inch times the element speed in inches per second.

In practice the maximum reactivity insertion rate is set at a level where an operator can retain control of power changes during steady-state operation. No limit is set on the transient rod during pulsing as the transient rod must be removed in a very short time to prevent clipping of the power transient resulting from the pulse. The normal reactivity insertion rate limit that has been set for TRIGA reactors is about 0.2% Δk/k per second or $.25 per second.

4.5.3.3.3 Fuel Operating Temperature

The thermal limit for the fuel used in the WSU TRIGA reactor is based on the combined characteristics of the fuel-moderator alloy and the associated cladding material. The limit depends upon the metallurgical properties of the fuel-moderator alloy, the pressure of the gases in the cladding gap, and the yield stress of the cladding (see section 4.7).

For TRIGA reactors using high hydride fuel it is a well established and documented fact that the limiting factor is the pressure buildup from out-gassing of hydrogen from the uranium-zirconium hydride fuel-moderator alloy. TRIGA fuel with a hydrogen-to-zirconium ratio of 1.6 as used in the WSU TRIGA reactor is single phase for temperatures in excess of 1150°C. The fuel-moderator alloy actually melts at about 1800°C. Furthermore, the higher hydride fuels do not undergo any significant thermal diffusion of hydrogen. These two facts and the intensive testing of the fuel-moderator alloy by the manufacturer plus extensive in-core experience clearly demonstrate that the fuel-moderator alloy characteristics would allow safe operating temperatures up to at least 1150°C.

The currently accepted limiting fuel temperatures for high hydride type TRIGA fuels are 1150°C for FLIP fuel and 1000°C for Standard fuel and LEU fuel. It is customary to employ a

200°C safety margin to yield a limiting condition of operation of 950°C in FLIP fuel and 800°C in Standard fuel and LEU fuel.

The cladding material for the WSU TRIGA reactor fuel is type 304 stainless steel with a thickness of 20 mils. It is a well known and documented fact (see section 4.7) that the tensile and yield strength of this cladding material is not significantly reduced up to a temperature of 850°C. The analysis contained in section 4.7 of this SAR establishes the fact that the ultimate strength of the 304 stainless steel cladding for TRIGA (H-Zr 1.6) fuel is 940°C. This cladding temperature is the limiting condition for a Loss of Coolant Accident.

4.5.3.3.4 Operating Power

The limitation on the maximum steady-state power level of a TRIGA-type reactor is determined by the ability of the cooling system to remove heat at a rate to assure that the fuel cladding temperature is held well below the safety limit during normal steady-state operation. A limitation on the maximum power level is also imposed by the decay heat of fission products if an accident occurs in which all or part of the cooling water is lost. Sufficient cooling must be provided under this circumstance to insure integrity of the cladding. These considerations are analyzed in section 4.7.

It is a well established fact that a TRIGA reactor can safely operate at a steady-state power level of one megawatt with natural convection cooling if the pool cooling system is designed to remove the heat produced and to limit the core cooling water outlet temperature to below 100°C. At power levels significantly above one megawatt, forced cooling systems are needed during steady-state operation and an emergency spray cooling system for cooling in case of a loss of core cooling water. The WSU TRIGA reactor utilizes natural convection cooling and is thus limited to a steady-state power level of one megawatt. Heat removal considerations are covered in section 4.6 of this SAR.

4.5.3.3.5 Pulsing Limit for WSU Reactor

The limiting factor for pulsing of a TRIGA reactor is the temperature in the hottest fuel rod during a pulse. The limiting temperature in any fuel rod during pulsing is set at 830°C as a result of the lessons learned from the FLIP fuel rod failure problem that occurred at the Texas A&M reactor in 1976. An analysis of the Texas A&M fuel rod problem is given in section 4.8 of this SAR. This limit assumes that the metallurgical properties of all TRIGA fuels are the same. In practice, the limiting temperature increase for the WSU modified TRIGA reactor during pulsing is:

$$\Delta T(\max) = 830 - 35 = 795°C$$

with a nominal initial average core temperature 35°C.

Fuel Temperature and Power Density Relationship

The WSU TRIGA reactor will have a mixed core of FLIP or LEU and Standard fuels and thus the power density per unit volume during a pulse to fuel temperature relationship for a specific core must be determined by combining the relationship for the fraction of each type of fuel in the core using the volumetric heat content relationships given below (where DT is the core temperature increase in degrees centigrade):

FLIP and LEU Fuel

$$\frac{\text{Watt - sec}}{\text{cm}^3} = 2.08 \times 10^{-3}\Delta T^2 + 2.08 \times \Delta T - 53.3$$

Standard Fuel

$$\frac{\text{Watt - sec}}{\text{cm}^3} = 2.08 \times 10^{-3}\Delta T^2 + 2.12 \times \Delta T - 54.3$$

Using the above relationships, a typical core which is composed of 75% FLIP or LEU fuel has a fuel temperature to power density relationship of:

$$\frac{\text{Watt - sec}}{\text{cm}^3} = 2.08 \times 10^{-3}(\Delta T)^2 + 2.09 \times \Delta T - 53.6$$

Limiting Maximum Power Density

In order to calculate the limiting maximum allowable power density in the core that corresponds to the maximum allowable fuel temperature, we insert the limiting temperature rise of 795°C into the power density relationship.

$$\frac{\text{Watt - sec}}{\text{cm}^3}(\text{max}) = 2.08 \times 10^{-3}(795)^2 + 2.09 \times 795 - 53.6 = 2923$$

Power Density Peaking Factors

The Fuchs-Nordheim variable heat capacity model for calculating the pulsing performance of a TRIGA reactor is really a point reactor model and thus the values calculated using this model are average values for the whole core. Also, the TRG3 code used to model the steady-state WSU TRIGA reactor is a two-dimensional diffusion code program that does not take into account variations in the vertical or axial direction. Accordingly, in order to relate values calculated by the model and the code to the real reactor case, corrections must be made for the peaking effects in the real case over the idealized cases calculated by the model and the code. The overall power density peaking factor for a specific core is the product of three correction factors as follows: 1) the radial or peak to average power density ratio as calculated by the TRG3 code, 2) the axial correction factor to account for the fact that TRG3 is only a 2D code, and 3) cell peaking factor within the worst case fuel rod to correct for the fact that the core is heterogeneous and the TRG-3 code uses a homogeneous model..

The worst case peaking factors for FLIP and LEU cores as calculated by GA and WSU are listed in Table 4.5-10.

TABLE 4.5-10
PEAKING FACTORS

Type of Peaking	FLIP	LEU	Mixed*
Radial	1.93**	1.57	1.75
Axial	1.25	1.36	1.36
Cell	2.35***	1.52	2.20

*Conservative values for mixed LEU-STD cores.
**WSU mixed FLIP-STD core 32A.
***Adjacent to water hole.

Using the worst case peaking factors for a typical mixed fuel core we obtain:

$$\text{Total Peaking Factor} = 1.75 \times 1.36 \times 2.20 = 5.24$$

Limiting Average Power Density

The limiting average power density in the core during pulsing is equal to the limiting maximum power density divided by the total peaking factor. In the case of a typical mixed core, this value is:

$$\text{Limiting Average Power Density} = \frac{2923}{5.24} = 559 \frac{\text{Watt - sec}}{\text{cm}^3}$$

Prompt Negative Temperature Coefficient

The prompt negative temperature coefficient for TRIGA type fuels is discussed in detail in a variety of General Atomics reports and in IAEA TECDOC-233. This report and other documents clearly shows the fact that the prompt negative temperature coefficient is a function of both core temperature and fuel burnup. A very conservative value for a typical FLIP or LEU core with 500 MWD burnup would be about -8.75×10^{-5}/°C or -.0125$/°C.

Neutron Lifetime

The neutron lifetime in a LEU core should be quite similar to that for a FLIP core. Calculations at WSU for FLIP cores as reported in the SAR for conversion to FLIP fuel yielded a value of 24 microseconds. GA calculated a lifetime of 24 microseconds for LEU cores and thus this value will be assumed to be correct.

Pulsing Characteristics of TRIGA FLIP and LEU Cores

The pulsing characteristics of a typical FLIP or LEU fueled modified TRIGA core are given in Table 4.5-5. The table was calculated by program PULSE using the Fuchs-Nordheim model (4) with variable heat capacity using the core parameters listed above the table.

Maximum Allowable Reactivity Insertion

A comparison of the calculated values shown in Table 4.5-4 and the limiting average power density value previously calculated leads to the conclusion that the conservative maximum allowable reactivity insertion for a modified TRIGA FLIP or LEU fueled reactor core is $2.25. A similar value was calculated from experimental pulsing data for WSU FLIP core 32A and further substantiates the selection of $2.25 as the limiting safe reactivity insertion value for ZrH1.6 TRIGA fuel.

This limiting value is predicated on the lessons learned from the FLIP fuel failure problem that occurred at Texas A&M in 1976 and is very conservative in order to insure that the type of problem that occurred at Texas A&M does not occur at the WSU TRIGA reactor. It is to be noted that Texas A&M arrived at a similar result by a completely different calculational method.

4.6 Thermal-Hydraulic Design

Very extensive thermal-hydraulic design studies and extensive actual performance tests have been done by General Atomics over the years on reactor cores utilizing TRIGA type fuel. This well known volume of analysis and testing (1,2,3) will not be repeated here in this SAR and only the results as they apply to the WSU modified TRIGA reactor will be presented.

The calculated and measured heat transfer characteristics of three TRIGA reactors including the WSU reactor are given in Table 4.6-1. The PRNC reactor is essentially equivalent to the WSU modified TRIGA reactor with the exception of a 2.0 MW licensed maximum power limit compared to the 1.0 MW for the WSU reactor. All calculations and tests were done under conditions of natural convection circulation of water through the core. Figure 4-22 shows the heat removal parameter for the three reactors as a function of the void detachment fraction in the coolant water.

An examination of Table 4.6-1 and Figure 4-22 clearly indicates that the operation of the WSU modified TRIGA reactor is significantly less severe thermodynamically and hydrodynamically than either they Torrey Pines Mark III or PRNC reactors. These two reactors have been safely operated at higher power levels and maximum heat fluxes than the WSU modified TRIGA reactor. Thus from a thermo-hydraulic standpoint, the WSU modified TRIGA reactor operation is extremely conservative. The power density and fuel rod temperatures of a typical mixed Std-FLIP core at full power are given in Figures 4-23 and 4-24. The fuel region inside the box is the FLIP fuel region.

Table 4.6-1

Comparison of Heat Removal from a Standard TRIGA Mark III
and 4-Rod Cluster TRIGAs

	Standard TRIGA	PRNC TRIGA	WSU TRIGA
Reactor Power (MW)	1.5	2.0	1.0
No. of Elements	74	[redacted]	[redacted]
Fuel Element Diameter (in.)	1.475	1.41	1.41
Hydraulic Diameter (ft)	0.0601	0.0578	.0578
Max. Heat Flux (Btu/hr/ft^2)	284,500	301,9800	150,000
Fuel Surface Area (ft^2/rod)	.4826	.4621	.4621
Heat Transfer Surface (ft^2)	35.71	43.90	46.21
Saturation Temp. (°F)	239	241	240
Inlet Temp. (°F)	60	100	92
Exit Subcooling (°F)	0	5*	2*
Mass Flow Rate (#/hr in^2)	1160*	1000	500*
Min. DNB Ratio	1.15*	1.37	1.10*

*Extrapolated

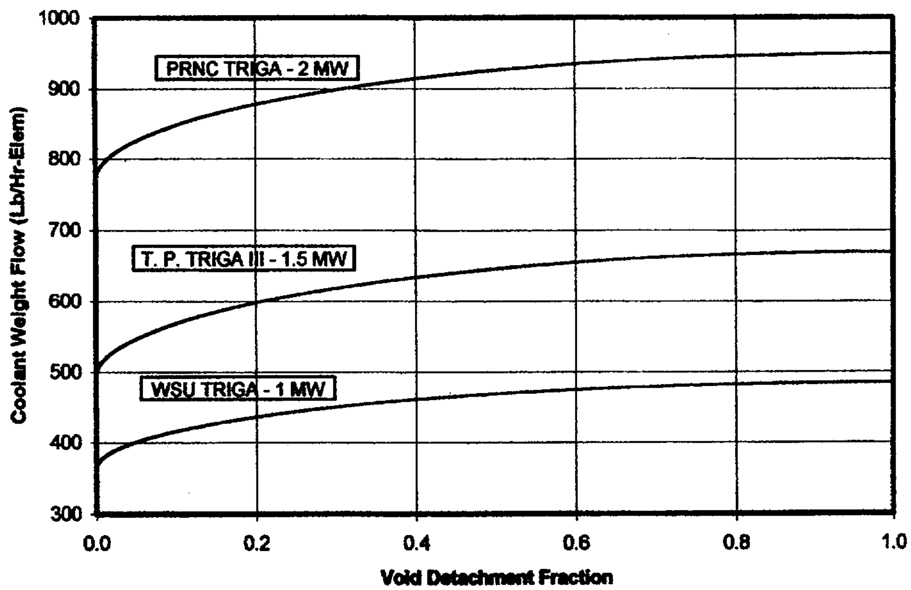


Figure 4-22
Heat Removal as a Function of the Void Detachment Fraction in Coolant

CORE 30E
POWER DENSITY IN kW/ROD:

	2		3		4		5		6		7	
A	0.0	0.0	0.0	0.0	3.6	4.3	0.0	0.0	3.9	3.0	0.0	0.0
	0.0	0.0	0.0	0.0	3.9	5.0	0.0	0.0	4.5	3.3	0.0	0.0
B	3.1	3.4	4.1	4.7	5.3	6.1	7.0	6.8	5.6	4.5	3.6	3.4
	3.9	4.8	6.1	7.2	8.1	8.6	8.9	8.7	7.9	6.8	5.6	4.9
C	4.9	6.3	7.8	7.5	17.4	15.8	16.0	15.6	15.4	17.4	6.2	6.3
	4.6	5.8	7.3	6.6	14.7	13.9	15.1	13.8	13.8	0.0	6.2	6.1
D	4.7	6.1	7.5	6.7	15.1	15.3	0.0	15.2	13.2	15.5	5.9	6.3
	4.7	6.2	0.0	6.6	14.8	14.2	15.4	14.0	12.0	12.7	5.2	6.2
E	4.5	5.9	7.1	6.4	14.1	12.4	12.7	12.3	11.3	11.9	4.9	5.9
	4.8	6.0	7.5	7.1	16.6	14.9	15.2	14.7	13.6	13.9	5.3	6.0
F	3.7	4.7	5.8	6.8	7.6	8.0	8.2	.9	7.3	6.3	5.2	4.6
	2.9	3.2	3.9	4.5	5.1	5.4	5.5	5.3	4.9	4.3	3.5	3.2

Figure 4-23

CORE 30E
FUEL ROD MAXIMUM CENTERLINE TEMPERATURE, DEGREES C:

	2		3		4		5		6		7	
A	0	0	0	0	184	204	0	0	193	163	0	0
	0	0	0	0	193	219	0	0	208	173	0	0
B	167	177	196	212	226	241	257	253	232	207	185	178
	191	216	242	260	271	278	281	278	269	254	232	218
C	218	245	268	263	442	388	396	383	378	439	244	245
	210	237	260	251	362	346	371	344	344	0	244	242
D	212	242	263	252	371	376	0	374	335	381	237	245
	212	243	0	251	366	351	379	349	317	326	224	244
E	209	237	259	247	351	323	327	321	309	316	217	238
	215	241	263	257	412	367	373	363	341	346	227	239
F	188	212	237	254	264	271	273	269	261	246	224	211
	163	172	192	209	221	228	230	226	217	202	182	173

PEAK/AVE = 1.67 MEAN = 264.9

Figure 4-24

4.7 Fuel Temperature Limitation for TRIGA Fuels

The determining factor that sets the temperature safety limit for TRIGA U-$ZrH_{1.6-1.65}$ fuels is the disassociation pressure of the hydrogen in the fuel. As the fuel temperature increases the hydrogen pressure increases and imposes a stress on the fuel cladding material. If the fuel temperature were to increase without limit, some point would be reached at which the internal pressure could cause the cladding to yield and eventually rupture. The purpose of the fuel temperature safety limit is to limit the hydrogen pressure to preclude a cladding failure.

The hoop stress exerted on the cladding by the hydrogen pressure is given by the equation

$$\delta = P_h \frac{r_c}{t_c} = P_h \cdot 35.25 \quad \left(\text{where } t_c = .02, r_c = .705\right)$$

where δ is the hoop stress in psi, P_h is the hydrogen pressure in psi, r_c is the radius of the cladding, and t_c is the thickness of the cladding. Under normal steady state operating conditions at full power in a TRIGA reactor the temperature of the type 304 stainless steel fuel cladding will not exceed 140°C. At 140°C the yield strength of type 304 stainless steel is 38,000 psi[1] and the ultimate strength is 68,000 psi (16). Thus at a 140°C cladding temperature a 1078 psi hydrogen pressure would produce a .2% deformation in the cladding and the maximum allowable hydrogen pressure would be 1929 psi.

The hydrogen-to-zirconium ratio in TRIGA fuels has a nominal value of 1.6 and a maximum value of 1.65. A detailed analysis performed by G.A. (17) indicates that the equilibrium hydrogen pressure in a U-$ZrH_{1.65}$ fuel rod which is constant temperature over the entire volume of the fuel is given by

$$P_e = 2.59 \times 10^9 \, e^{[-1.997 \times 104/(T+273)]}$$

where P_e is the equilibrium hydrogen pressure in psi and T is the fuel temperature in degrees Celsius. Solving this equation for the maximum allowable hydrogen pressure of 1929 psi produces a maximum allowable uniform fuel and temperature of 1142°C.

The equilibrium condition pressure defined above never occurs, however, in the real case because a fuel rod is not at constant temperature over the whole volume of the rod. Consequently, the hydrogen pressure will be much lower than the equilibrium value calculated from the maximum fuel rod temperature. The axial power distribution along a typical TRIGA fuel rod varies from P max in the center to about .63 P max at the end of the fuel region (1). If we make the conservative assumptions that the fuel rod temperature does not vary in the radial direction and that the axial fuel rod temperature distribution follows the fuel rod power distribution, then the average fuel temperature equals the maximum fuel temperature divided by 1.2. Under these conservative conditions the maximum allowable fuel temperature under steady state conditions that does not exceed the yield strength of the cladding is 1370°C.

In addition to the steady state case, the effects of the transient fuel temperature increase during pulsing must be considered. A detailed analysis performed by G.A. (17) has shown that

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1 Stress to produce a .2% deformation.

the hydrogen pressure during pulsing increases to a maximum value of 22% of the equilibrium value in about .3 seconds and then falls off. Under pulsing conditions some film boiling occurs and the peak cladding temperature is greater than under steady state full power operation. A conservative value of 500°C is selected as the maximum cladding temperature under conditions of film boiling.

The ultimate strength of type 304 stainless steel at 500°C is 57,000 psi. At this temperature the maximum allowable hydrogen pressure is 1617 psi. A peak transient fuel temperature of 1290°C would be required to produce this pressure during pulsing. As a safety limit the peak adiabatic fuel temperature occurring during the pulse mode of operation for U-$Zr_{1.65}$ is selected to be 1150°C. The peak hydrogen pressure that would result at this temperature is approximately 460 psi. This would produce a stress of 16,000 psi which would not exceed the ultimate strength of the cladding at a temperature of 700°C.

Actual measurements of the peak hydrogen pressure during pulsing have been made at G.A. Five special instrumented fuel rods were tested in the ATPR during these experiments that involved a total of 426 pulses. The maximum peak fuel temperature during these tests was 1175°C and the maximum observed peak transient hydrogen pressure was 41 psia. The peak pressure transient occurred during the first pulse and decreased to about 20 psia by the 220th pulse. These experiments clearly indicate that the actual peak hydrogen pressure during pulsing is at least a factor of ten below the calculated values.

The 1150°C fuel temperature safety limit for TRIGA reactor FLIP U-$ZrH_{1.65}$ fuel is seen to be a conservative limit. This limit will preclude a fuel cladding failure due to the internal hydrogen pressure during the steady state mode of operation. Other considerations concerning the TRIGA fuel temperature limit during pulsing are considered in the next section of this SAR.

4.8 Pulsing Limits for TRIGA Cores

On September 27, 1976, Texas A&M University discovered the fact that some FLIP fuel elements had been damaged in their mixed core fueled with Standard and FLIP fuel rods. The four FLIP fuel rods adjacent to the transient rod were found to be severely damaged in the form of swelled cladding and bent and bowed rods near the center of the four fuel rods surrounding the transient rod. The ultimate cause of the fuel damage was not resolved until 1982 after extensive examinations of sections from the damaged portions of the fuel rods were done in a hot cell. The cause of the damage and new temperature limit established by Texas A&M for pulsing are given in the next paragraph (3).

TRIGA fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 for FLIP and LEU fuel and 1.65 for Standard. This yields delta phase zirconium hydride which has a high creep strength and undergoes no phase changes at temperatures over 1000°C. However, after extensive steady state operation at 1 Mw, the hydrogen will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions. When the fuel is pulsed, the instantaneous temperature distribution is such that the highest values occur at the surface of the element and the lowest values occur at the center. The higher temperatures in the outer regions occur in fuel with a hydrogen to zirconium ratio that has now substantially increased above the nominal value. This produces hydrogen gas pressures considerably in excess of that expected for $ZrH_{1.6}$. If the pulse insertion is such that the temperature of the fuel exceeds

874°C, then the pressure will be sufficient to cause expansion of microscopic holes in the fuel that grow larger with each pulse. Thus a combination of extensive steady state operation at full power followed by pulsing with a fuel temperature of 874°C can cause fuel rod damage. A pulsing limit of 830°C is obtained by examining the equilibrium hydrogen pressure of zirconium hydride as a function of temperature as shown in Figure 4-25. The decrease in temperature from 874°C to 830°C reduces hydrogen pressure by a factor of two, which is an acceptable safety factor. This phenomenon does not alter the safety limit since the total hydrogen in the fuel element does not change. Thus, the pressure exerted on the clad will not be significantly affected by the distribution of hydrogen within the element.

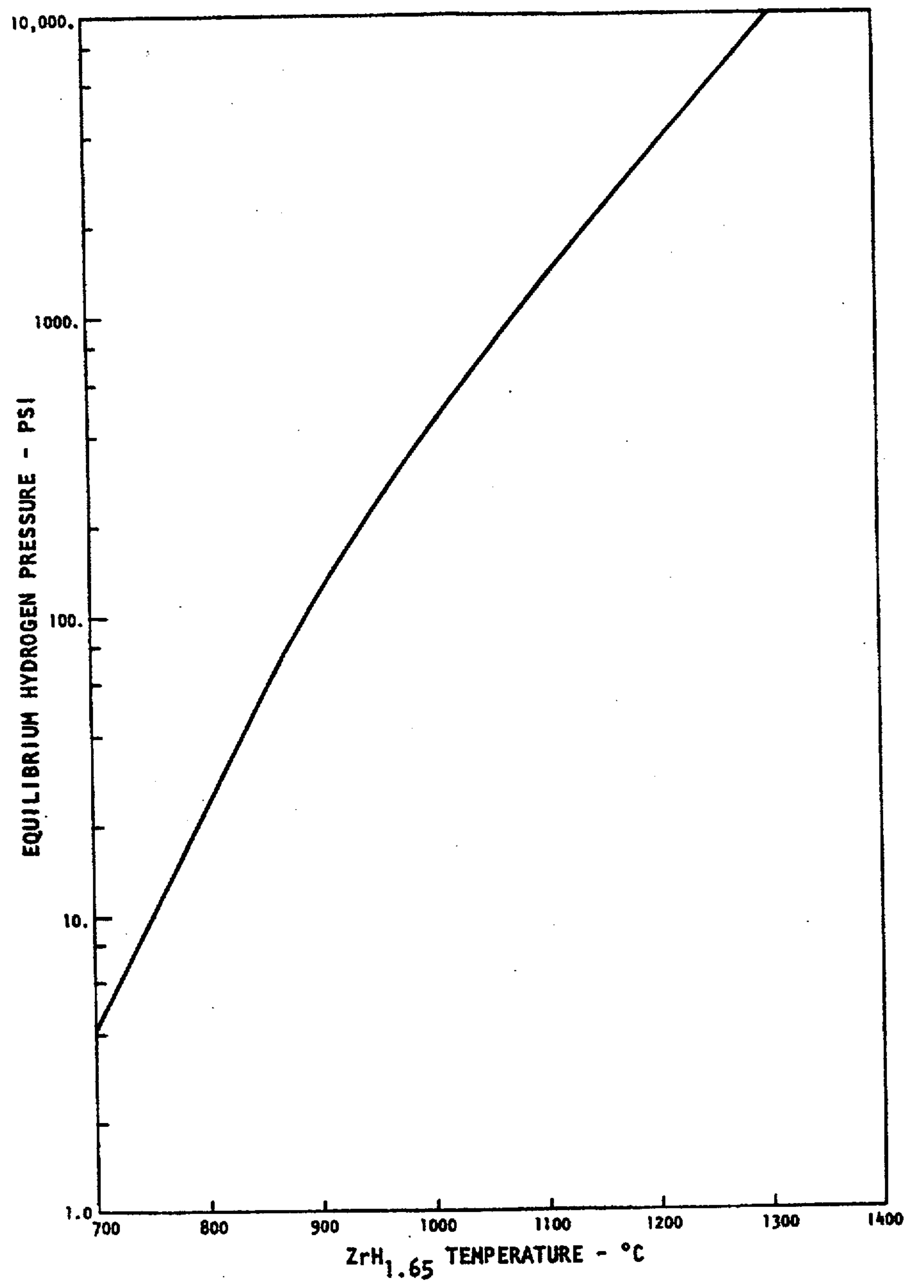

Figure 4-25
Equilibrium hydrogen pressure over $ZrH_{1.65}$ versus temperature

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13. "Energy Release from the Decay of Fission Products," Nuclear Science and Engineering, 3, 726 (1968), J.F. Pecking and R.W. King.

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19. "Fission Product Releases from TRIGA-LEU Reactor Fuels," GA-A16287, November 1980, Baldwin, Foushee, and Greenwood.

20. "Research Reactor Core Conversion Guidebook," IAEA-TECDOC-643, Vol #4: Fuels, Appendix 1-7, International Atomic Energy Agency, April 1992.

APPENDIX 4-A

Fuel Rod Temperature Calculations

The experimental instrumented fuel rod temperature data shown in Figure 1 was fit to a polynomial expression by the method of least squares and the following equation was obtained:

$$T = 31.77 + 0.6148\ (Pw) - (5.396 \times 10^{-4})\ (Pw)^2 + (1.958 \times 10^{-7})\ (Pw)^3$$

where T = Instrumented fuel rod temperature in 0C
Pw = Reactor power in kW.

Calculations with the EXTERMINATOR-2 code for this core indicated that the axial averaged power density in the instrumented fuel element is 283 watts per centimeter of fuel rod height at 1 MW. If we assume that the power density in the instrumented fuel rod as well as all fuel rods is a linear function of reactor power and that all rods behave identically to the instrumented fuel rod, then we can calculate the temperature of any rod from the data on the instrumented fuel rod. That is, axial averaged fuel rod power densities calculated with the EXTERMINATOR-2 code may be used with the experimental data on the instrumented fuel rod to calculate the centerline fuel rod temperature of any rod.

The power density, Pd, in the instrumented fuel rod at any power, Pw, is Pd = .283 x Pw, where Pd = axial averaged power density in watts per cm of core height and Pw is the reactor power in kW. Thus, Pw = Pd/.283. This may be combined with the instrumented fuel rod temperature equation to yield a relationship between fuel rod power density and fuel rod temperature. This equation is:

$$T = 31.77 + 2.172\ (Pd) - (6.738 \times 10^{-3})\ (Pd)^2 + (8.639 \times 10^{-6})\ (Pd)^3.$$

<u>Temperature Coefficient Weighting Factor for a Mixed Core</u>

The multiplication factor, k, of a reactor according to modified one group reactor theory is given by:

$$k = \frac{np f \varepsilon e^{-B^2\tau}}{1 + B^2L^2} = \frac{pe^{-B^2\tau}}{\Sigma a + DB^2} \cdot \varepsilon \nu \Sigma f(\text{fuel})$$

$$= \frac{pe^{-B^2\tau}}{\Sigma a + DB^2} \cdot \text{Fast Neutron Production}$$

$$\text{Thus, } Tc = \frac{dk}{dT} = \frac{d}{dT}\left[\frac{pe^{-B^2\tau}}{\Sigma a + DB^2} \cdot \text{Fast Neutron Production}\right]$$

for a TRIGA reactor

$$Tc \approx \frac{d}{dT}\left[\frac{Pe^{-B^2\tau}}{\Sigma a + DB^2}\right] \cdot \text{Fast Neutron Production}$$

APPENDIX 4-C

Calculation of Temperature Coefficient of Core 30E at Full Power

$\overline{Tm}$	=	2650C
$\overline{Ts}$	=	2290C
$\overline{Tf}$	=	3600C
FS/FT	=	.5105
FF/FT	=	.4895
Tcs(229°C)	=	.0155
Tcf(360°C)	=	.0152
Tcm	=	.0155 x .5105 + .0152 + .4895 = $.0154/°C

5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The heat generated within the fuel during operation of the reactor is transferred to the pool water by natural convection. A new modern type heat exchanger and fiberglass cooling tower were installed in the summer of 1999 to replace the old cooling system. The heated pool water is pumped through a plate type heat exchanger constructed with stainless-steel nozzles and plates. The primary pump has a stainless steel pump housing and rotor, and the primary piping is aluminum in the pool and schedule 80 CPVC in the pump room. A secondary side pump takes water from the sump of an induced draft type cooling tower and passes it through a filter and then the shell side of the heat exchanger. A filter was added to the secondary side to minimize fouling of the heat exchanger that has occurred in the past. The heat generated is dissipated to the atmosphere through the latent heat of evaporation of water in the cooling tower.

5.2 Primary Coolant System

The WSU Modified TRIGA reactor like all TRIGA type reactors is cooled by natural convection as discussed in section 4.6 of this SAR. The heat generation rate in the fuel elements is distributed axially in a cosine distribution chopped at the ends with a peak to average value of 1.25. The important thermo-hydraulic parameters for the WSU reactor are given in Table 5.2-1. As was indicated in section 4.6, the primary cooling system of the WSU reactor operates well within the normal range for natural convection cooling. That is, the cooling system of other TRIGA type reactors operate safely with a heat flux of twice that for the WSU reactor.

The loss of coolant accident for the WSU TRIGA reactor is analyzed in section 13.1.3 of this SAR. This analysis demonstrates that a loss of coolant accident at the WSU TRIGA reactor will not precipitate a fuel cladding failure. That is, the power density in kW/rod for all operational WSU reactor core arrangements are significantly below the limiting values of 22 kW/rod for standard fuel and 23 kW/rod for FLIP and LEU fuels. All portions of the reactor cooling system are checked for proper operation during the pre-startup check out procedure before the reactor is operated. Thus there are no significant problems associated with the primary cooling system that are likely to occur.

Table 5.2-1
Thermo-Hydraulic Parameters for WSU Reactor

Reactor Power (MW)	1.0
Number of Fuel Rods	[redacted]
Hydraulic Diameter (ft)	.0578
Max. Heat Flux (Btu/hr/ft^3)	150,000
Heat Transfer Surface (ft^2)	46.21
Saturation Temperature (°F)	240
Inlet Temperature (°F)	92
Exit Subcooling (°F)	2*
Mass Flow Rate lbs/hr/in^2	500*
Minimum DNB** Ratio	1.10*

*Extrapolated
**DNB = departure from nucleate boiling = ratio of surface temperature to coolant temperature)

5.3 Secondary Coolant System

The cooling system layout is shown in Figures 5-2 and 5-3, and schematically in Figure 5-1. The major components are a primary pump, heat exchanger, secondary pump, water filter, cooling tower and control instrumentation. The primary and secondary systems are shown in functional diagrams in Figure 5-4 and 5-5. Design data for the cooling system are given in Table 5.3-1 and 5.3-2. The absolute pressure on the secondary side of the heat exchanger is higher than the primary side so any leakage would be from secondary to primary. A differential pressure gauge is installed between the primary and secondary to insure the fail safe condition. This prevents inadvertent transfer of radioactive material from the reactor pool water to the secondary cooling tower water. A secondary to primary leak would be detected by an increase in conductivity of the primary loop discharge into the pool.

In operation, water leaves the pool through the suction pump at a point about 4' 8" below the surface of the pool, passes through the primary pump and the heat exchanger and returns to the pool through a distribution pipe located along the bottom of the pool on the west side of the

[redacted]

Reactor Cooling System

Normal flow directions shown. Local readout (analog gauge) and contron room display (digital gauge) are present for all temperature and pressure measurements. Primary flow rate 450 gpm. Secondary flow rate 900 gpm.

Figure 5-1
New Cooling System Schematic

[redacted]

[redacted]

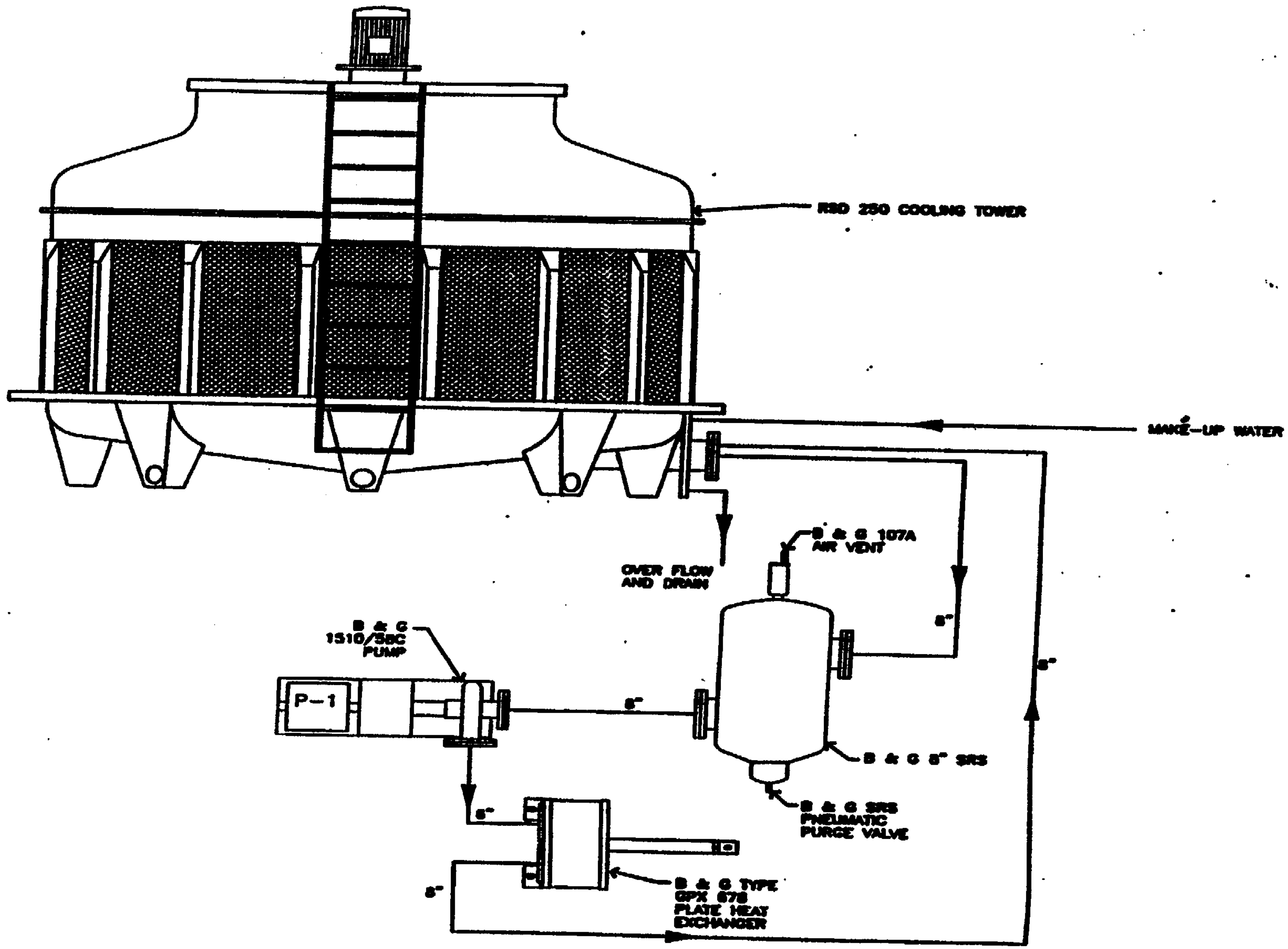


Figure 5-4
Secondary Loop

[redacted]

Table 5.3-1
Design Data for Cooling System

	Old System	New System
Primary Flow Rate, gpm (tube side)	350	450
Primary Inlet Temperature, °F	110	110
Primary Outlet Temperature, °F	90	90
Secondary Flow Rate, gpm (shell side)	700	900
Secondary Inlet Temperature, °F	80	80
Secondary Outlet Temperature, °F	90	90
System Heat Load, BTU/Hr (nominal)	3,500,000	4,440,000
Design Wet Bulb Temperature, °F	65	65
Primary Pressure Drop, psi (tube side)	10	3
Secondary Pressure Drop, psi (shell side)	6	11
Cooling Tower Water Consumption, evaporation, gph	430	425
Cooling Tower Water Consumption, blow down, gph	100	---
Maximum Pool Temperature, °F	120	120
Primary Pump, Horsepower	10	5
Secondary Pump, Horsepower	15	20
Cooling Tower Fan, Horsepower	20	7.5
Primary Pipe Size, Inches	6	8
Secondary Pipe Size, Inches	8	8

Table 5.3-2

Plate Heat Exchanger Specifications

	Hot Side	Cold Side
Fluids:	Water	Water
Flow (gpm)	450	900
T-in (deg. F)	110.00	80.00
T-out (deg. F)	90.00	89.96
Specific Gravity	0.99	0.99
Specific Hat (BTU/lb/F)	1.00	1.00
Thermal Conductivity	0.36	0.36
Viscosity (cp)	0.69	0.82
Operating Pressure (psig)		
Pressure Drops (psi)	2.88	10.67
Passes	1.00	1.00
Nozzle Diameter (inches)	6.00	6.00
Nozzle Material	316SS	316SS

Heat Exchanged (BTU/hr)	4,450,000
Model (Graham)	GPE-51
Plate Thickness (inches)	0.0197
Plate Material	304SS
Gasket Material	Nitrile
Frame Material	Carbon Steel
Design/Test Pressure (psig)	150/225
Design Temperature (deg. F)	250

Table 5.3-3

RSD-250 Cooling Tower Data

Height	114 in
Diameter	154 in
Pipe Inlet	8 in
Pipe Outlet	8 in
Fan Motor	7.5 HP
Fan Diameter	82.5 in
Air Volume	65,300 CFM
Nominal Water Flow	859 GPM

pool gate partition. A siphon break is located in the inlet line 5.05 meters above the center line of the core. Any leakage from the primary pumps and heat exchanger header is returned to the building hot drains which lead to the hold-up tank. Valves in the primary exit and return lines may be closed so that the pool water level may be maintained when it is necessary to disconnect the primary pump or heat exchanger for servicing. The primary pump is stainless steel and the primary piping and valves are CPVC, aluminum or stainless steel.

The new heat exchanger is a modern plate type with stainless steel plates, stainless steel nozzles and a carbon steel shell. Water from the pool in the primary loop will only contact stainless steel, aluminum, or CPVC. The secondary pump takes water from the cooling tower sump, pumps it through a filter, then the shell side of the heat exchanger and back to the top of the cooling tower. The pump and piping in the secondary loop are made of schedule 80 CPVC.

The heat exchanger is a plate type unit manufactured by Graham model GPE-51 and tested in accordance with the ASME section VIII Division I code specifications. The unit was tested at 225 psig, designed to operate at 150 psig maximum but is only operated at a 35 psig maximum.

The new cooling tower is a modern super efficiency induced draft type made of fiberglass Chandler model RSD-250. Air enters all around the lower perimeter of the circular cooling tower and is expelled by the fan at the top. Make-up water for the cooling tower is taken from the raw water supply, mixed with chemicals in a treating system to control corrosion, sliming and scaling and sent to the sump of the cooling tower. Sump water is periodically drained off to control salt and sludge concentration. The drained water will by-pass the hold-up tank and enter the city sewer. To prevent freezing during shut down in cold weather, a thermostatically controlled heater is provided for the cooling tower.

Instrumentation provides indication at the reactor control console of primary inlet and outlet water temperature, primary outlet water conductivity, primary and secondary inlet and outlet pressures and on-off condition of circulating pump. The starting switches for the pumps and cooling tower are located at the console. The pumps and cooling tower shut down automatically upon excessive radiation levels over the pool or upon building evacuation. The cooling system instrumentation equipment has been upgraded to modern types of units in the past few years and has operated flawlessly for a number of years. Thus the cooling system instrumentation is quite adequate to insure the safe operation of this system.

The pump house is attached to the north wall of Room 201 and is a relatively air tight structure with no windows. A door interconnects the pool room with Room 201. A door to the outside is provided for bringing in equipment and water treatment supplies. This outside door is kept locked and connected to the reactor security alarm system.

5.4 Primary Coolant Cleanup System

The purity of the water in the pool is maintained by passing a small amount of the pool water through a mixed bed ion exchanger. A detailed drawing of the cleanup loop is shown in Figure 5-6. Pool water is taken from the surface of the pool via the surface skimmer at the west pool gutter, passed through a recirculating pump, flow meter, water filter, and into the ion exchange bed. The resin bed is located in a shielded room to minimize personnel exposure from radionuclides removed from the pool water. A conductivity cell is located at the input and output

[redacted]

of the ion exchange bed so that the performance of the ion exchange bed can be monitored. From the ion exchanger bed the water flows back into one of the pool sumps, normally the west sump.

The nominal flow rate of the clean up loop is about 5 gpm. The output of the ion exchange bed normally is of the order of 0.1 micromho and a conductivity alarm occurs if the conductivity rises to 1 micromho. The ion exchange resin is normally replaced when the conductivity rises to 0.5 micromhos/cm. The resin is replaced as needed but this is usually done at a frequency of every four weeks. The pool water conductivity is maintained at 0.5 micromhos/cm or less and a pH between 5.5 and 7.5.

In order to insure that radioactive species do not build up in the pool water, the pool water is routinely monitored for the contained radionuclides. The monitoring involves the counting of a sample of pool water on a sensitive gamma ray spectrometry system and identifying the radionuclides present.

5.5 Primary Coolant Makeup Water System

The level of the water in the pool is maintained by the pool water make up system. A schematic diagram of the makeup water system is shown on Figure 5-6. The source of make up water is the Nuclear Radiation Center deionized water system. This system feeds deionized water to all the laboratories in the facility and consists of a three cartridge mixed bed deionizer in room 201-C as shown on the diagram. This system supplies water with a conductivity of 1 micromho and the deionizer beds are changed if the conductivity increased to 1.25 micromhos.

There is a float switch monitoring the pool water level attached to the side rails that controls make up water for the pool. If the pool level falls, this float switch closes opening a solenoid valve that feeds make up water into the suction side of the clean up loop recirculating pump. The make up feed line contains a water meter that is read each time the reactor is started up so as to maintain a continuous record of the amount of pool make up water utilized.

5.6 Nitrogen-16 Control System

Calculations made for the Puerto Rico Reactor (2) which is very similar to the WSU reactor predicted a bridge N-16 exposure rate of 73 mr/hr at a power level of 1 megawatt. This value checks with experiments made at the WSU reactor with the nitrogen-16 control system (diffuser) off that were terminated before complete equilibrium conditions were reached at a bridge dose rate of 60 mr/hr.

In order to reduce the radiation exposure level on the bridge, the WSU reactor has a diffuser system that is attached to the bridge structure. This closed loop system takes water from the pool, passes it through a large pump, and discharges the water through a nozzle directed down toward the top of the core. The net effect is an increase in the time that N-16 activity in the core cooling water takes to reach the surface of the pool. Since N-16 has a very short half-life, 7.4 seconds, this significantly reduces the N-16 exposure rate on the bridge. The exposure rate at the bridge is continuously monitored during operation by one of the channels of the area radiation monitoring system. The bridge channel reading during operation is normally 1.5 mr/hr. Also calculations of the N-16 activity in the air of the pool room of the PRNC reactor (2) with the diffuser off, yielded a pool room air dose rate of .4 mr/hr. Thus with the diffuser on, the N-16 pool room air dose rate is insignificant. Both the bridge and pool room air N-16 exposure rates are within the limits specified for controlled areas and the facility ALARA program.

The diffuser system is completely independent of the pool cooling system and has had no effect on the pool cooling in the more than 20 years that it has been in operation. The system is manually operated and is turned on as part of the checkout procedure for starting up the reactor. In the event of a failure of the diffuser system to operate properly the radiation level at the bridge would increase and cause an alarm of the bridge area monitoring channel. The operator would take appropriate corrective action as needed.

5.7 Auxiliary Systems Using Primary Coolant

There are no auxiliary systems at the WSU reactor that utilize primary water for cooling purposes.

5.8 References

1. Safety Analysis for the Washington State University Reactor Core Conversion and Power Increase, October 1966, submitted to the U.S. Atomic Energy Commission.
2. Safeguards Summary Report for the TRIGA-FLIP Reactor at The Puerto Rico Nuclear Center, Mayaguez, Puerto Rico, PRNC-123, November 11, 1969.

6 ENGINEERED SAFETY FEATURES

6.1 Summary Description

Section 13 of this SAR on Accident Analysis clearly demonstrates that there are no accidents whose consequences could be unacceptable without mitigation. Thus the WSU reactor facility design does not include any required engineered safety features. That is, there is no conceivable mode of operation which could create a significant threat to the health and safety of the reactor staff and general public. The maximum possible radiation exposure to an individual outside the facility under the postulated conditions for the maximum hypothetical accident is minimal. The exposures are significantly below the generally acceptable accident results for nonpower reactors of not more than 5 rem whole body and 30 rem thyroid for occupational exposure and not more than .5 rem whole body and 3 rem thyroid for members of the general public. In addition, the calculated accident exposures are well below the maximum values established in 10 CFR 20.1201 for occupational exposure and 10 CFR 20.1301 for public exposure. Thus, no realistic hazard to the staff at the reactor as well as the general public would result from any postulated accident event.

6.2 Detailed Descriptions

No engineered safety features to mitigate the consequences of a postulated accident are designed into the WSU reactor facility except those associated with the heating, ventilation, and air conditioning (HVAC) system described in sections 9.1 and 6.2.2 below. However, the reactor coolant system as described in section 5.2 and 5.3 has certain features that are safety related.

6.2.1 Confinement

The HVAC system for reactor operating portions of the WSU Nuclear Radiation Center building is separate from the HVAC for the rest of the center. This separate HVAC system serves the pool room, control room, beam room, and radiochemistry laboratory and is described in detail in section 9 of this SAR.

The primary radionuclide routinely present in the exhaust air from the facility HVAC system into the atmosphere is Argon-41. The Argon-41 content of the exhaust air is continuously monitored and constitutes an insignificant safety hazard as discussed in detail in section 11 of this SAR. In the event of the maximum hypothetical accident (MHA) and the release of fission products into the pool room, the analysis of the MHA in chapter 13 of this SAR demonstrates the maximum discharge by the HVAC of the released fission products into the environment would lead to a maximum of .002 rem whole body and .02 rem thyroid dose. These dose rates would be significantly reduced if the HVAC system is operated in the dilution mode.

6.2.2 Containment

The HVAC system for the reactor facility as described in detail in chapter 9 of this SAR has three modes of operation. These modes are normal, dilution, and isolation. In the normal mode 2.12 x 10^6 cm^3/sec of air is removed from the facility and discharged into the environment. In the dilution mode the total discharge rate is the same but the air is diluted with outside air by a factor of 6.7. The pool room air is also passed through a HEPA filter before being diluted. The HEPA filter would remove at least 90% of the iodine from the pool room air before it is diluted.

The building wake effect plus dilution would decrease the concentration of radionuclides in the pool room air being discharged into the atmosphere by a factor of 1970 or to a very safe level as is shown in chapter 13.

6.2.3 Emergency Core Cooling System

The WSU reactor system does not need or have an emergency core cooling system. Section 13.1.3 of this SAR demonstrates that a loss of coolant accident will not precipitate a fuel cladding failure.

6.3 References

1. Washington State University Reactor Safety Analysis Report of May, 1974 submitted to allow use of FLIP fuel in the WSU reactor.

2. Safety Analysis Report for the WSU Modified TRIGA Nuclear Reactor of May, 1979.

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The instrumentation and control system of the WSU modified TRIGA reactor is very similar in functionality to that for most TRIGA reactors. The inherent large prompt negative temperature coefficient of the TRIGA type reactor makes control of TRIGA type reactors much easier to accomplish than non TRIGA type reactors. The major portion of the Reactor Control System (RCS), Reactor Protective System (RPS), and Radiation Monitoring System components are all located in the reactor control console.

The RCS consists of the indicators, control switches, and monitoring devices and instruments required to exercise control over the reactor. The RCS thus primarily comprises the reactor control elements, control and element position indication devices and the reactor power level indicating devices including the start-up channel system. The RPS consists of those circuits and detectors that monitor various reactor parameters and initiate a reactor SCRAM in the event any Limiting Safety System Setting (LSSS) is exceeded. The primary LSSS setting is reactor fuel temperature.

7.2 Design of Instrumentation and Control Systems

7.2.1 Design Criteria

The control system for the WSU TRIGA reactor consists of a control console and associated instrumentation. The control console is pictured in Figure 7-1 and was designed and constructed by the Nuclear Radiation Center Staff. Considerable thought and experience went into the human engineering aspects of the console layout. All indicating devices are placed for optimum readability and accessibility. All the alarm and interlock functions are in a single block of dual color back-lighted switches as shown in Figure 7-2. Each interlock function has a separate independent reset switch and associated relay contacts that must be individually activated. There is no "Master Reset" system, the failure of which could compromise the safety of the control system. The controls for the control elements are grouped together and located for ease of use as shown in Figure 7-3. The console is positioned to allow the operator to not only watch the console but also to view the activities on the reactor bridge. A closed circuit TV system allows the operator to view the activity in the radiochemistry laboratory and beam rooms.

The electronic systems use solid-state circuitry and high reliability components wherever possible. All essential relays and terminal boards are mounted on slide-out trays for easy maintenance. A permanent record of the important parameters is provided by strip chart recorders mounted in a rack on the left side of the console. Ready access to communications facilities are provided for operator convenience.

All instrument components are high quality industrial grade and/or meet military specifications. All active components are solid state devices for high reliability and reduced size. Integrated circuits are used extensively where appropriate, and printed circuit boards are of high temperature and fire-resistant material. Components exceed specifications for improved reliability, and circuit reliability analyses are made for all modules involved in the reactor safety system. Recommended testing intervals are based on the predicted mean time between failure. Overall system reliability is enhanced by the use of plug in circuit boards or modules, thus

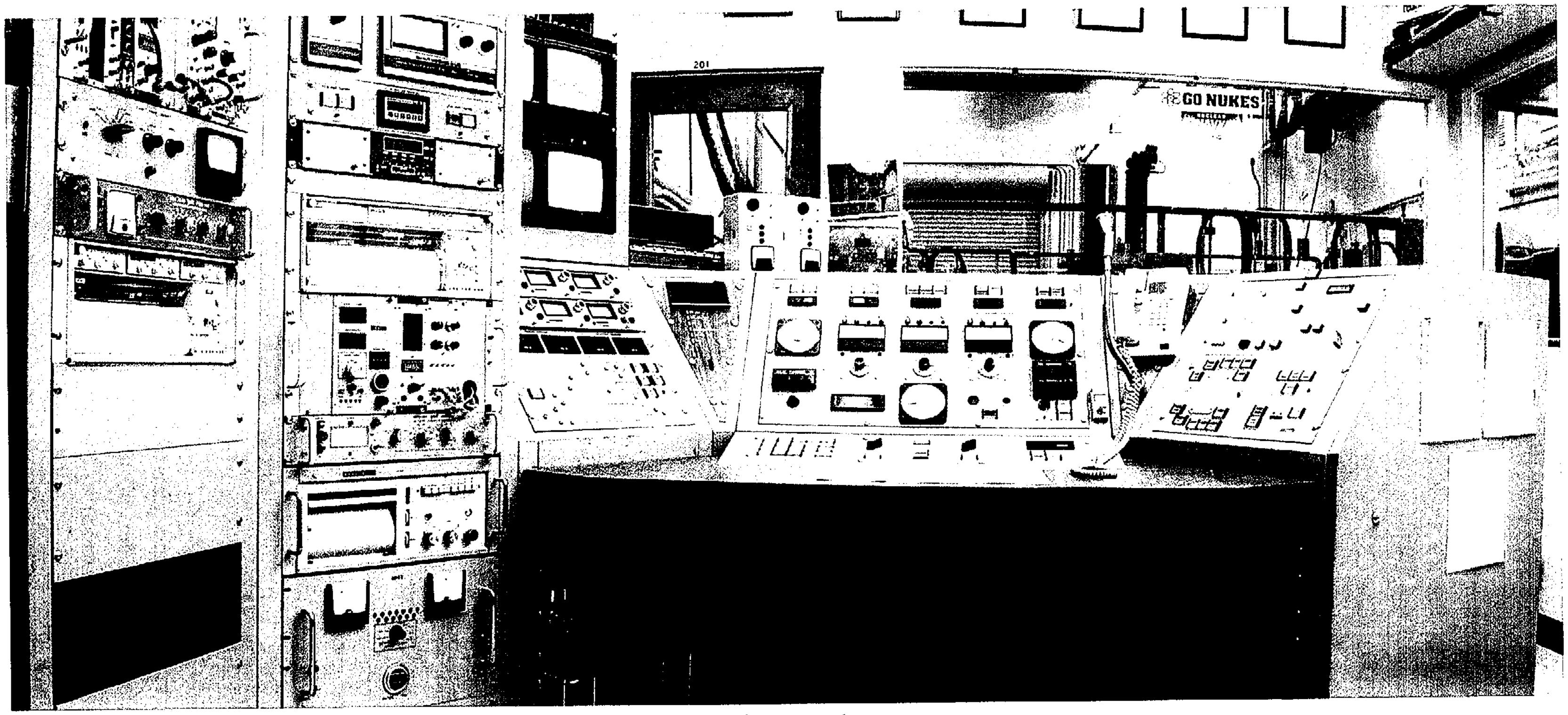


Figure 7-1

Console Front Left

(Interlock Control Block)

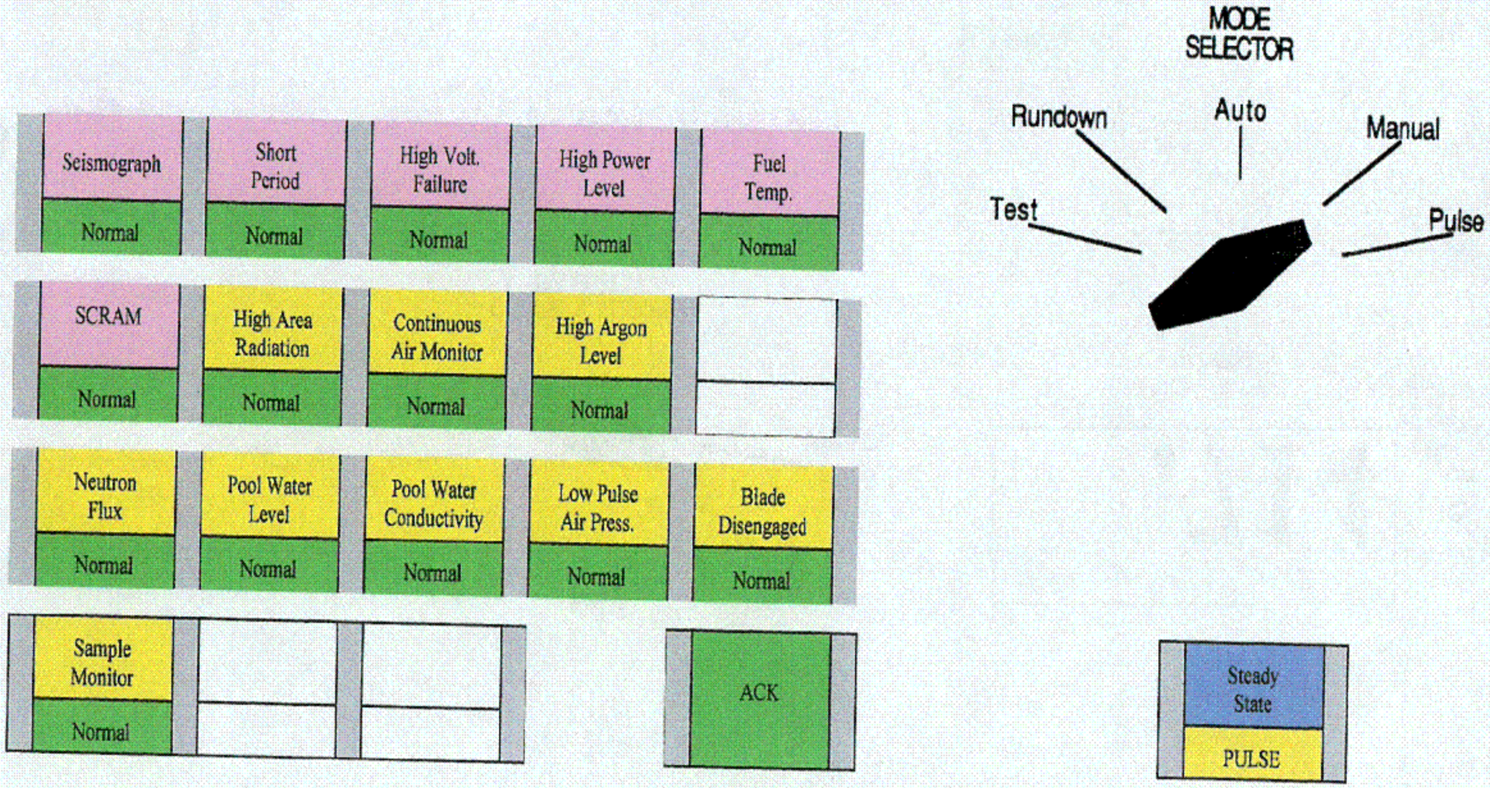


Figure 7-2

C01

Console Front Right

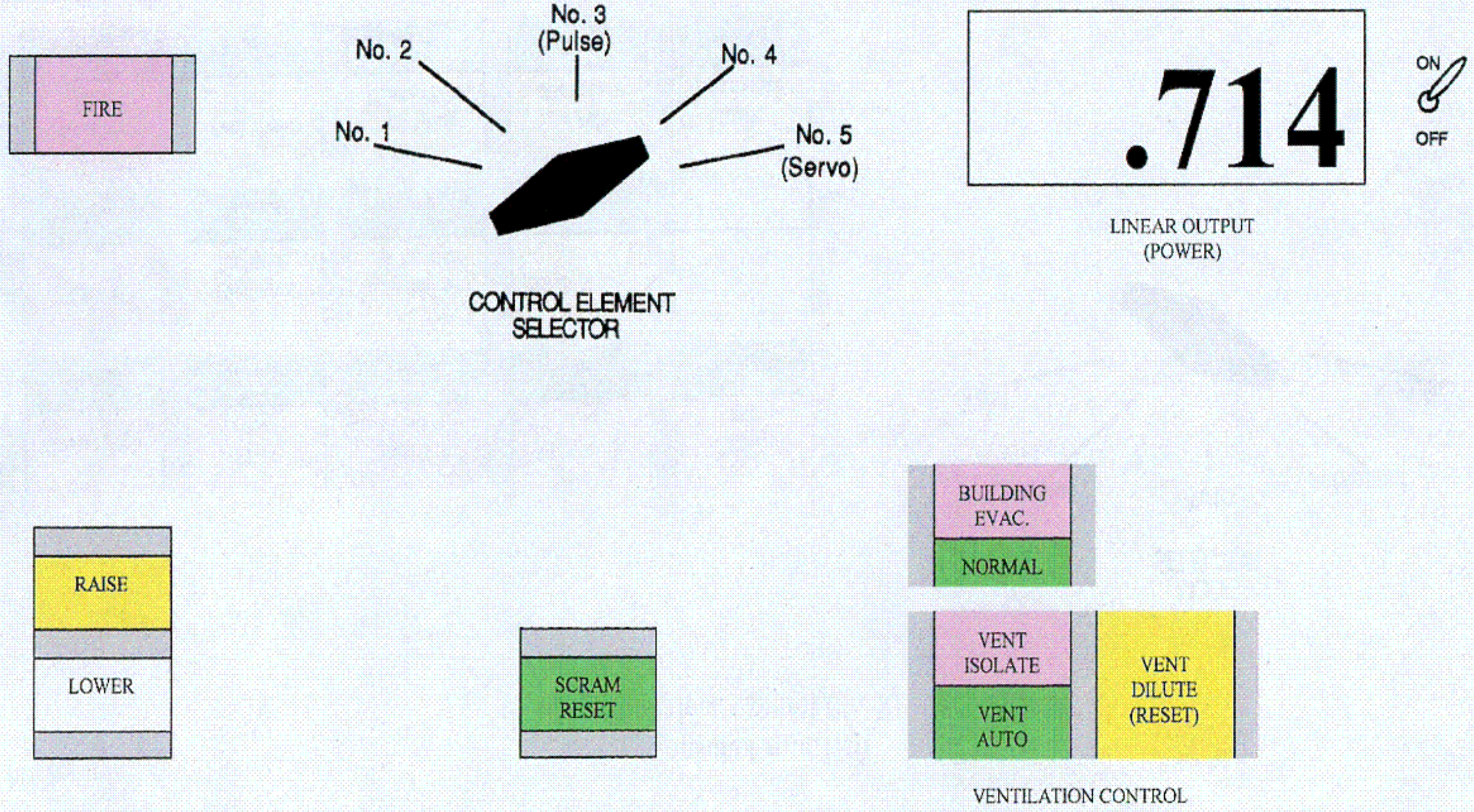


Figure 7-3

C02

reducing the mean time to repair. The use of common printed circuit boards and integrated circuits in several different instruments reduces the necessary inventory of repair components and simplifies maintenance. The RCS and RPS systems meet or exceed all the criteria set forth in ANSI/ANS-15.15.

7.2.2 Design-Basis Requirements

The basic requirement for the reactor instrumentation and control system of the WSU modified TRIGA reactor is to provide accurate safe reliable control and operation of the reactor. The Technical Specifications for the facility enumerate various parameters that are to be monitored and controlled as well as certain limiting safety system settings (LSSS) which if exceeded are to automatically shut down the reactor. The control system for the WSU reactor thus must contain instrumentation to perform the functions listed in the Technical Specifications which are listed in Tables 7.2-1 and 7.2-2.

Table 7.2-1
Minimum Measuring Channels

	Min. no.	Effective mode	
Measuring channel	operable	SS	pulse
Fuel element temperature	1	X	X
Linear power level	1	X	
Log power level	1	X	
Integrated pulse power	1		X

Note: SS = steady-state

Table 7.2-2
Minimum Reactor Safety Channels
(Technical Specifications Requirements)

		Number operable in specified mode	
Safety Channel	Function	SS	Pulse
Fuel temperature	Scram if fuel temperature exceeds 500°C	1	1
Power level	Scram if power level exceeds 125% of full licensed power	1	
Manual scram	Manually initiated scram	1	1
Wide range	Prevent initiation of a pulse above 1 kW		1
	Prevent control element withdrawal when neutron count is less than 2 cps	1	
High-voltage monitor	Scram on loss of high voltage to power channels	1	1
Pulse-mode switch	Prevent withdrawal of standard control and regulation elements in pulse mode		1
Preset timer	Transient rod scram 15 seconds or less after pulse		1
Pool level	Alarm if pool level falls below 16 ft over the core	1	1
Transient rod control	Prevent application of air unless fully inserted	1	

Note: SS = steady-state

7.2.3 System Description

The main portions of the WSU reactor instrumentation and control system are located within the reactor control console pictured in Figure 7-1. Each portion of the system is described in detail in the following sections of section seven of this SAR.

7.2.4 System Performance Analysis

The entire control system as it currently exists has been functioning for over ten years without a significant serious system failure. The facility preventative maintenance program is very comprehensive and insures each portion of the system accurately and reliably performs its intended function. All upgrades were done after an extensive safety analysis under the requirements of 10 CFR 50.59.

7.2.5 Conclusion

On the basis of past performance and a lack of significant system problems, during this period it can be concluded that the WSU reactor control and instrumentation system as it currently exists will continue to allow for the safe operation and control of the WSU modified TRIGA reactor within the limits established in the Technical Specifications.

7.3 Reactor Control System

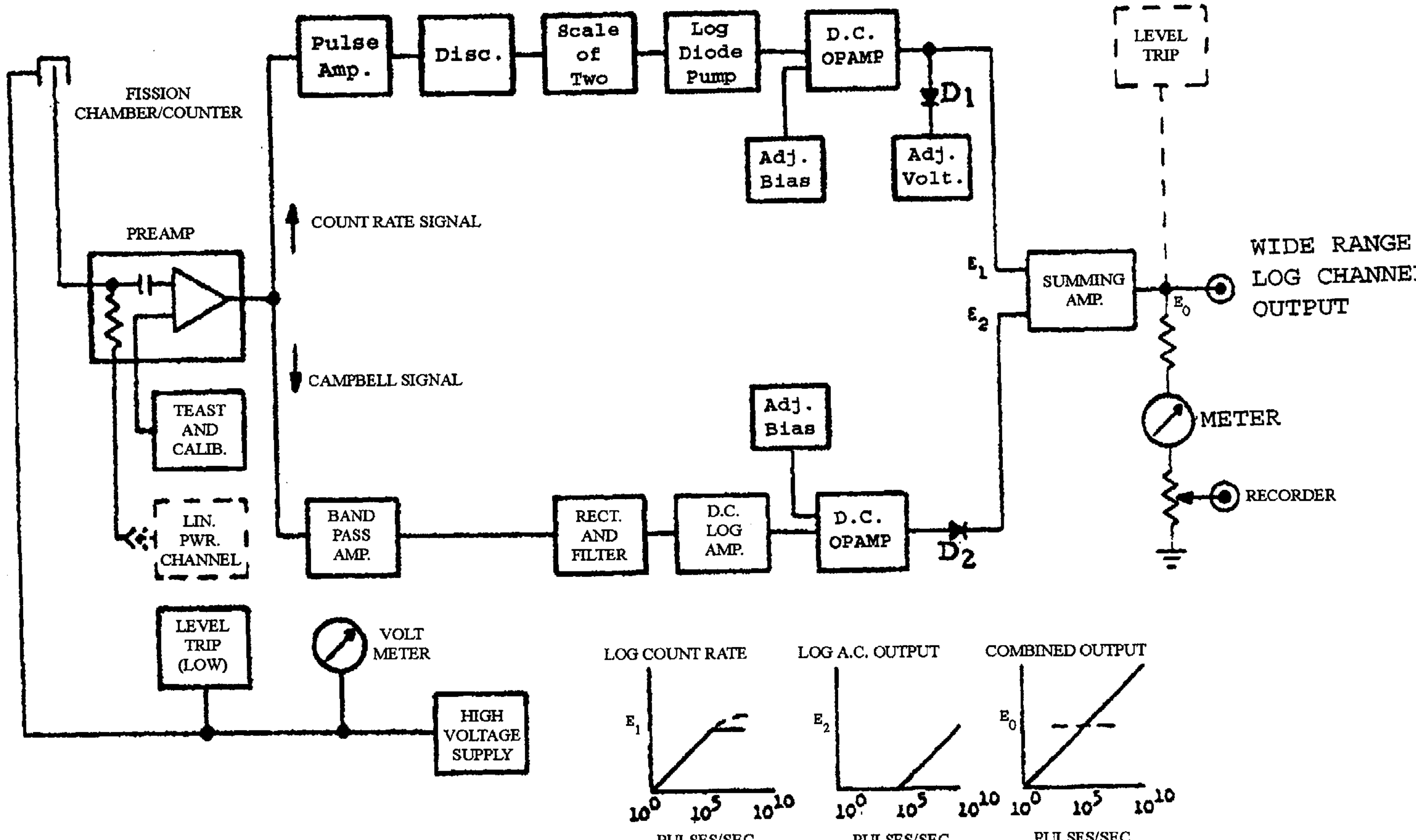
7.3.1 Startup-Channel

The startup-channel for the WSU reactor is part of the General Atomics Wide Range Channel shown in Figure 7-4. The detector is a fission chamber that feeds the log power, log count rate, and period portions of this channel. The startup-channel function is the log count rate portion of the Wide Range Channel which is a conventional log count rate circuit with a input range from 0.1 to 3 x 10^6 CPS. The low count rate inhibit circuit functions as a startup safety feature during reactor startup. The count rate must exceed 2 CPS or rods can not be withdrawn from the core. In other words, the wide range channel must be functioning, the startup source must be in the core, and subcritical multiplication must be sufficient to produce 2 CPS or rods can not be withdrawn. This precludes the occurrence of what is generally called a "startup accident" in which the reactor becomes super critical before the instrumentation starts to detect the neutron level in the core. The current version of the wide range channel uses digital electronics and was installed under the provision of 10 CFR 50.59 and NRC-Generic Letter 95-02.

7.3.2 Log Power Channel

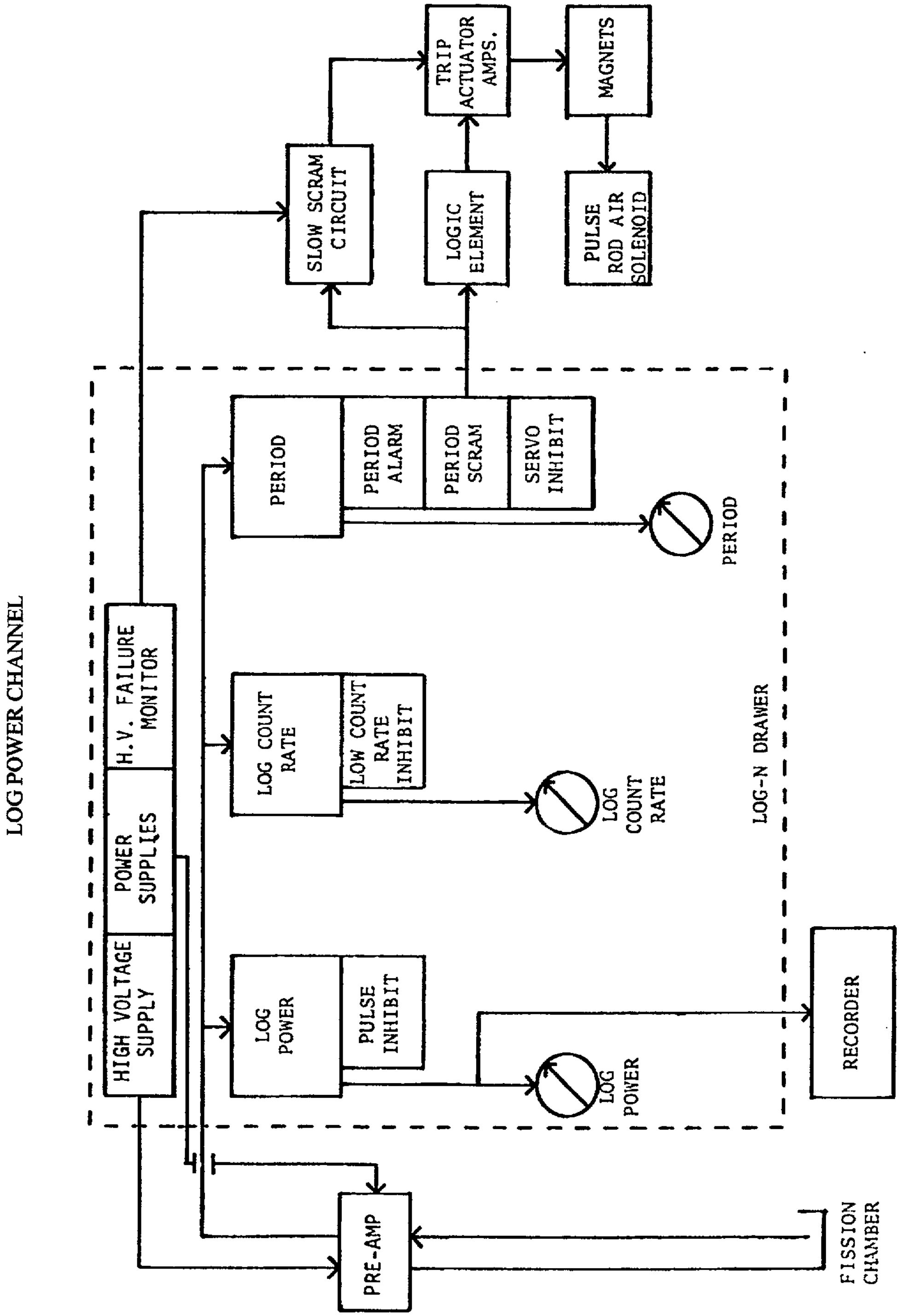
The Log Power Channel is part of the General Atomics Wide Range Channel shown in Figures 7-4 and 7-5. The Log Power Channel covers a 10 decade range from startup to full power by means of a combination of log count rate and Campbelling circuits as shown in Figure 7-4. The log count rate portion is effective at low power or startup range, and the Campbelling portion at high power levels.

The log power channel feeds a log power meter on the wide range channel chassis and a remote indicator on the main console for the operator to view. The log power level also feeds a recorder that maintains a permanent record of the reactor power level during operation. The log



WIDE RANGE LOG CHANNEL

Figure 7-4



LOG POWER CHANNEL
HIGH VOLTAGE SUPPLY
POWER SUPPLIES
H.V. FAILURE MONITOR
LOG POWER
PULSE INHIBIT
LOG COUNT RATE
LOW COUNT RATE INHIBIT
PERIOD
PERIOD ALARM
PERIOD SCRAM
SERVO INHIBIT
LOG POWER
LOG COUNT RATE
PERIOD
LOG-N DRAWER
RECORDER
SLOW SCRAM CIRCUIT
LOGIC ELEMENT
TRIP ACTUATOR AMPS.
MAGNETS
PULSE ROD AIR SOLENOID
PRE-AMP
FISSION CHAMBER

Figure 7-5

power channel also includes a period meter that indicates reactor period but is not necessarily connected to the RPS. It is used for training purposes only since the very large prompt negative temperature coefficient of a TRIGA type reactor eliminates the need for period protection. That is, a TRIGA reactor is self limiting and an operator must continue to pull rods to get the power level to increase.

7.3.3 Linear Power and Safety Channels

A block diagram of the linear power and safety channels for the WSU reactor is shown in Figure 7-6. The linear channel has a compensated ion chamber detector that feeds a NMP 1000 channel readout that indicates power level on a linear scale and auto ranges as the reactor power level increases. The output of the linear channel is recorded during startup and during all power level reactor operation. This is only a power level indication channel and does not feed the reactor scram system.

The reactor also has two safety channels that will scram the reactor in the event the reactor power level exceeds a preset level adjusted to correspond somewhat below the 125% power level stated in the Technical Specifications. Safety channel 1 is fed from the wide range channel linear output and safety channel 2 is fed from a UIC with a trip added and no recorder. In addition, the safety channels each have a high voltage failure monitor that will scram the reactor in the event of the failure of the high voltage feeding either chamber.

7.3.4 Fuel Temperature Monitoring Channel

The fuel temperature monitoring channel for the WSU reactor is shown in Figure 7-7. There are two instrumental fuel elements (IFR) located in the core of the reactor each of which have three separate thermocouple elements. The selected thermocouple in each IFR feeds a TC current transmitter, current sensor, and digital temperature readout on the reactor console. Thus the operator has continuous indication of the fuel temperature as monitored by two thermocouples in differed IFRs located at different locations in the reactor core.

One IFR feeds the scram system and will cause a reactor scram if the fuel temperature in that IFR exceeds 500°C as per the technical specifications. Details on the scram chain are covered in section 7.4.

7.3.5 Pulsing Instrumentation

One of the unique features of a TRIGA reactor is that such reactors may be safely pulsed from a low power level to a very high self terminating power due to the large prompt negative temperature coefficient of a TRIGA reactor. The instrumentation associated with pulsing operation of the WSU reactor is shown in Figure 7-8. Section A of this block diagram shows the fuel temperature channel that remains in effect during a pulse. This channel will cause a somewhat belated scram of the reactor if the fuel temperature were to exceed 500°C.

The main power monitoring channel during a pulse is a special pulse power channel that actually monitors the gamma ray level in the core during a pulse. This special channel has a very fast response time. The channel provides a complete record of the reactor pulse including the pulse power and pulse energy.

5/02 EDC

LINEAR SAFETY AND POWER CHANNELS

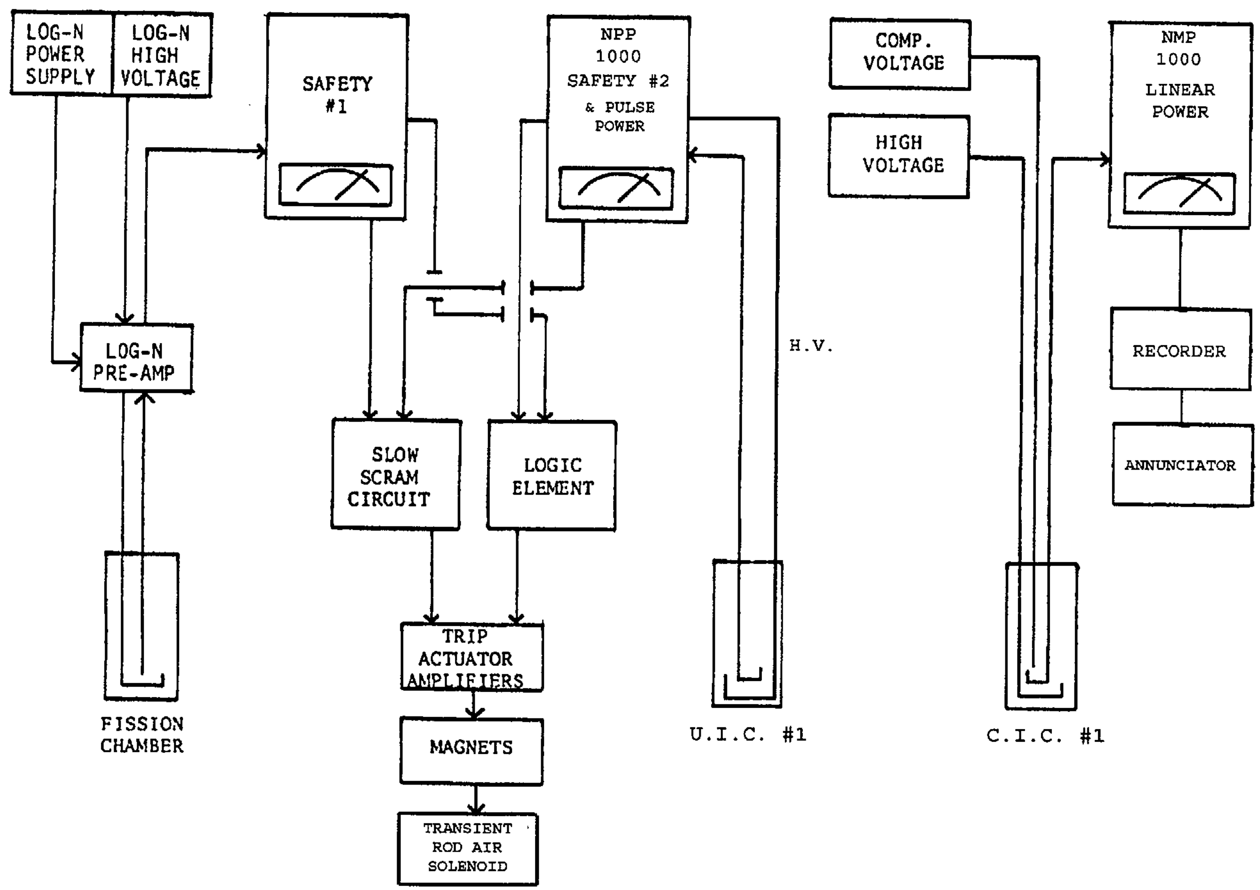
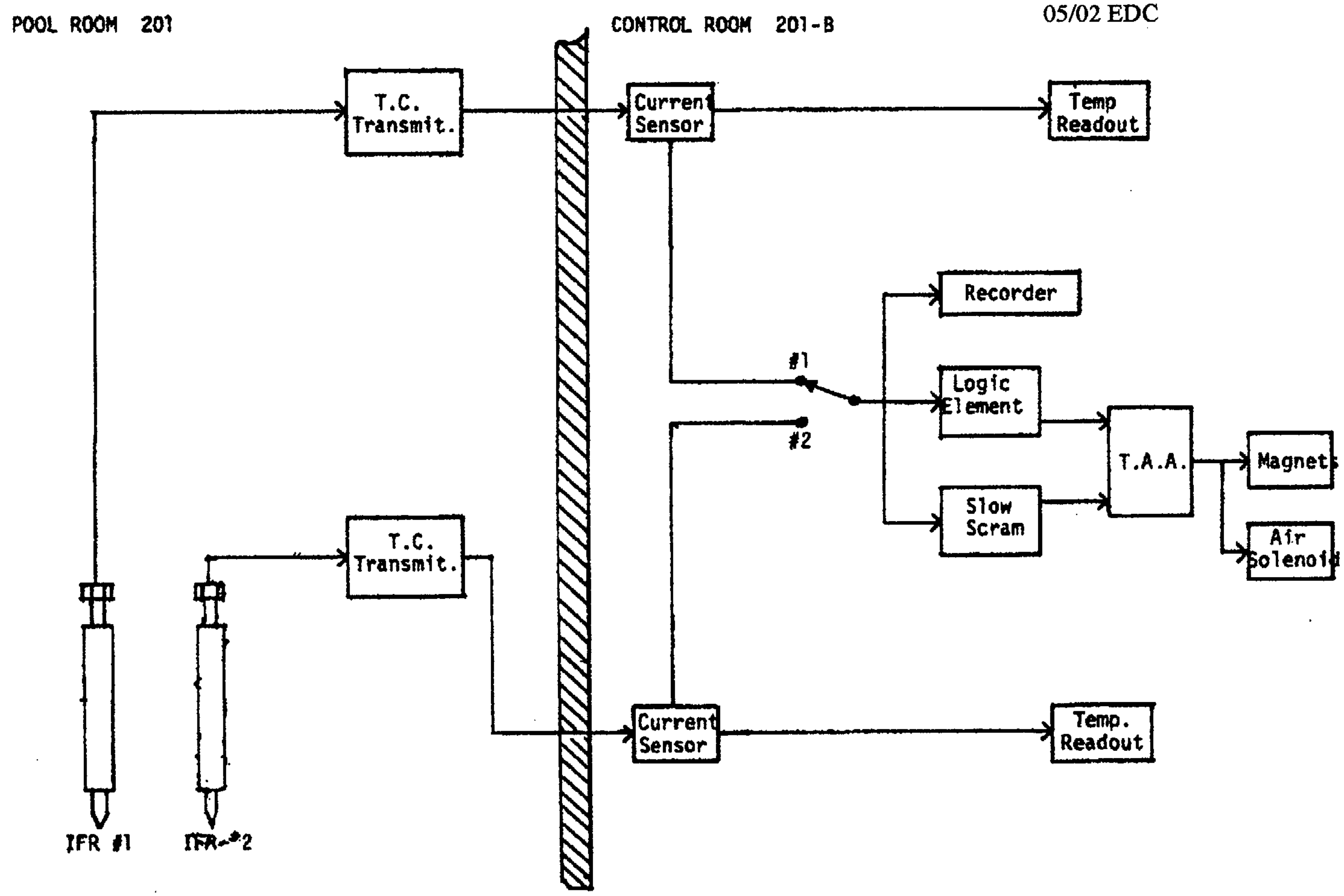


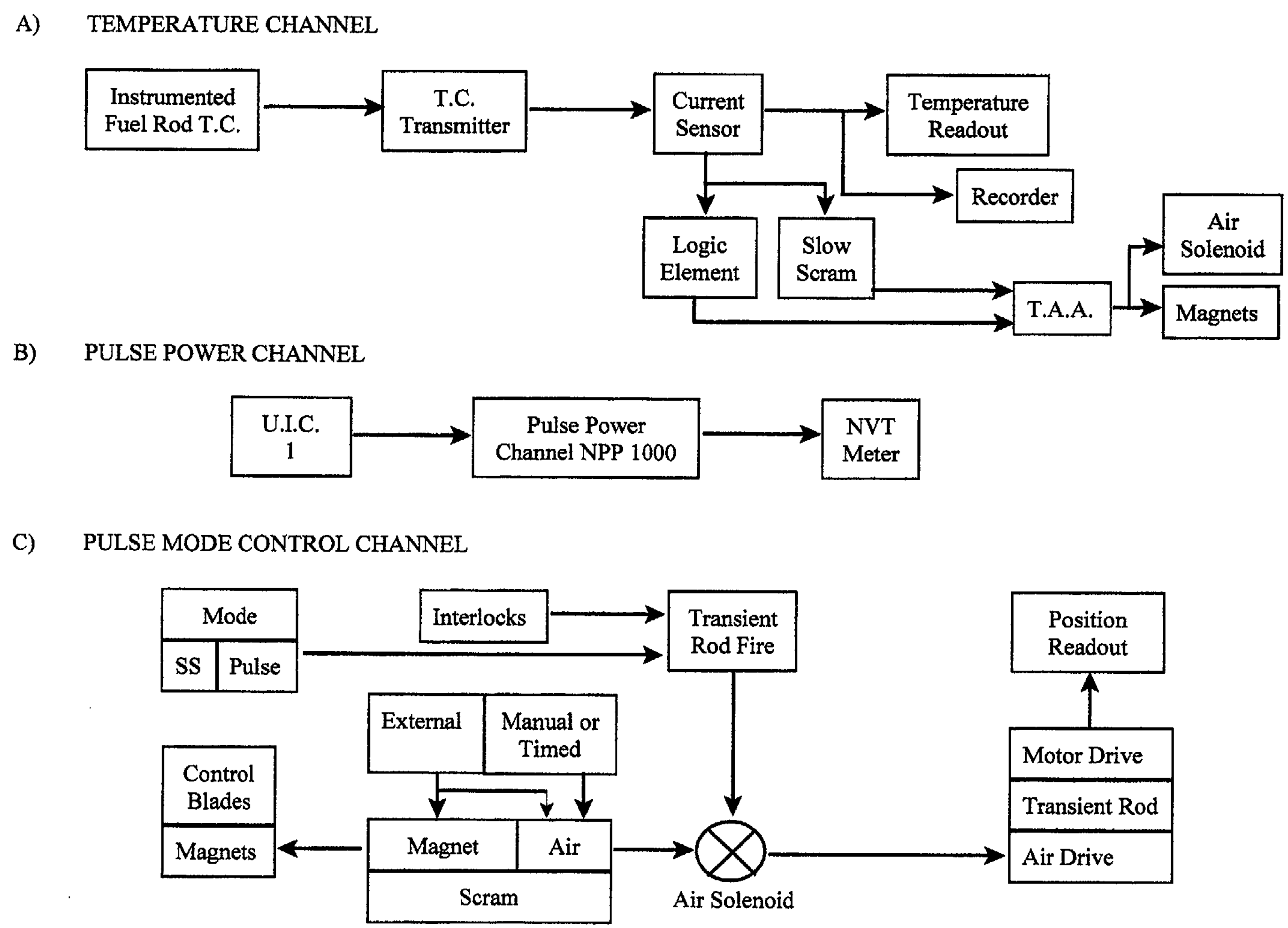
Figure 7-6

05/02 EDC



FUEL TEMPERATURE MONITORING CHANNEL

Figure 7-7



PULSING INSTRUMENTATION

Figure 7-8

The controls for the pulse mode operation are shown in the lower portion of Figure 7-8. The power level interlock is on the log power channel and prevents pulsing at a power level over 2 kW as per the technical specifications. The operator brings the reactor critical with the pulse rod fully inserted and the pulse rod stop set at the level for the desired amount of reactivity for the pulse. The power level must be below 2 kW. The mode switch is set to pulse, and the pulse button pressed opening the air solenoid valve firing the pulse rod out of the reactor up to the stop location. After 2 seconds or less, the pulse rod automatically falls back into the reactor when air pressure is removed by the pulse rod delay timer thereby clipping the tail of the pulse. The actual pulse is self terminating by the effect of the large prompt negative temperature coefficient. The operator then runs the control rods into the reactor so it will not go critical again after the core cools down. Even if the rods are not inserted, the reactor would in time only come back to a power level of below 2 kW.

7.3.6 Control System Recorder

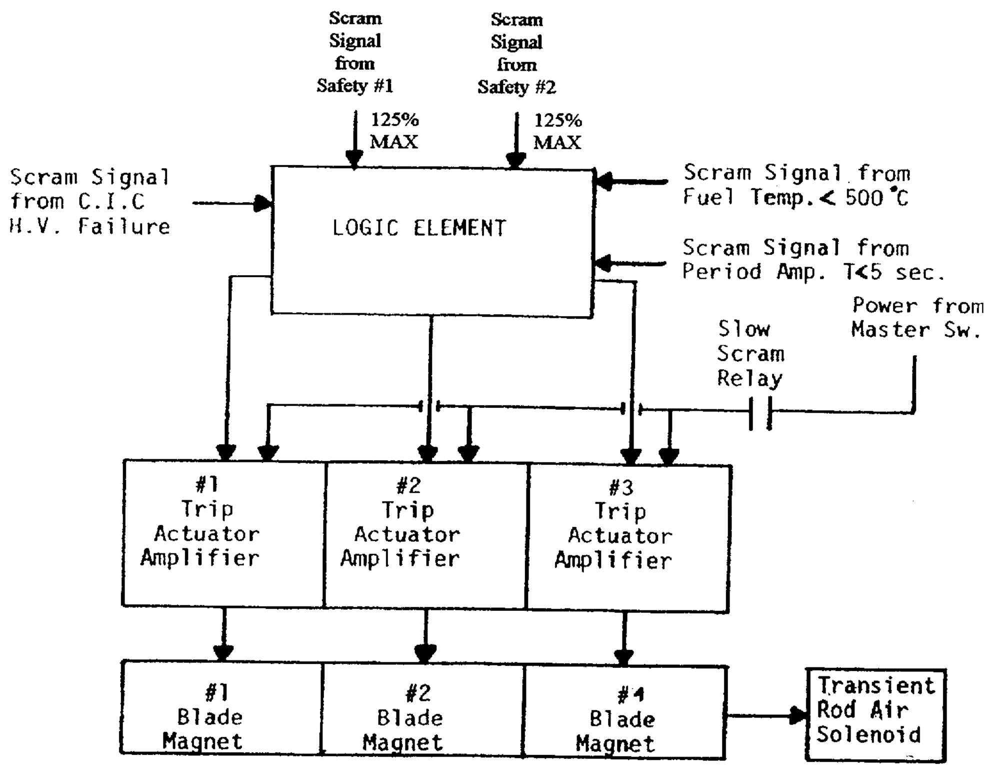
The RCS includes a modern three channel recorder with an individual plug-in adjustable sensitivity module for each channel manufactured by Omega. The recorder operates when the reactor power is on and records the output of the linear power channel, log power channel, and fuel temperature channel. Thus a permanent record of these three important parameters is made whenever the reactor is operated.

7.4 Reactor Protection System

The purpose of the RPS portion of the WSU modified TRIGA reactor is to automatically insert the control rods into the reactor thereby making the reactor subcritical in the event that certain monitored parameters exceed predetermined limits. The primary limits are exceeding a fuel temperature of 500°C or a power level of 125% of the licensed limits. The main part of the RCS circuitry that actually shuts down the reactor is the "Scram Circuitry" shown in Figure 7-9. The top portion of the circuitry is referred to as the "Fast Scram" portion and the string of relay contacts at the bottom is referred to as the "Slow Scram". The distinction is made because the upper circuitry is all electronics and operates faster than the lower portion which is based on relays.

The control rods of the reactor are held up by blade magnets, the current through which is controlled by a "Trip Actuator Amplifier" for each rod. A special pulse rod trip amplifier controls the current to the transient rod air solenoid amplifier that applies air to the transient rod mechanism holding that rod out of the reactor against the stop. Each trip actuator amplifier has a slow scram and fast scram input. A loss of signal on either or both of these inputs will cause a "Trip" cutting off current to the blade and air solenoid thereby causing the rods to fall by gravity into the reactor. The "Logic Element" is an electronic "OR" circuit that controls the inputs to the trip amps. If any of the five inputs to the logic element goes "TRUE", all outputs of the Logic Element go "FALSE" and cause all the rods to be scrammed. The five inputs to the logic element are: 1) CIC high voltage failure, 2) high power level from safety channel #1, 3) high power level from safety channel #2, and 4) fuel temperature above 500°C, and 5) period below 5 sec. The period scram is not required by the technical specifications but is available for training purposes.

SCRAM CIRCUITRY



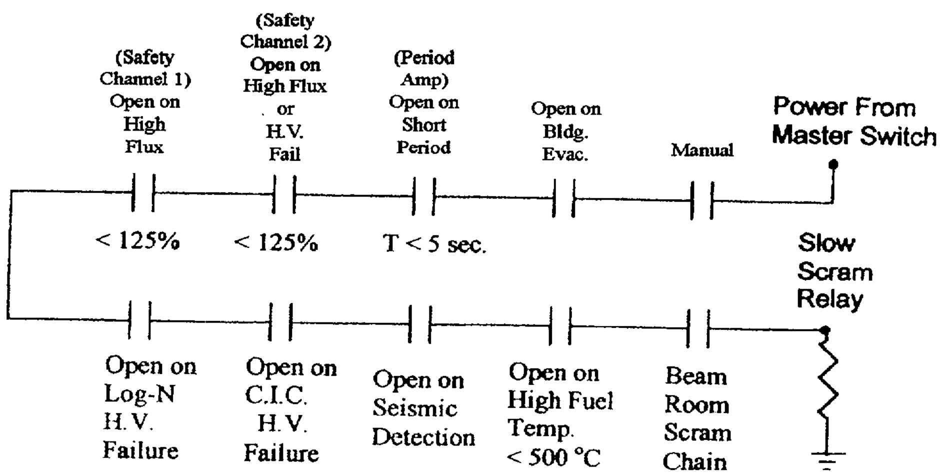


Figure 7-9

The slow scram chain is really an "AND" circuit composed of a series of relay contacts the closure of each being controlled by various system parameters.

If any one of the system parameters goes "TRUE", that relay contact opens breaking the chain causing the master slow scram relay to open and cut off power to all the trip amps. This drops all the rods including the transient rod into the reactor. Individual relays in the chain are controlled by the following RCS systems: 1) beam room scram chain, 2) fuel temperature, 3) seismometer, 4) CIC high voltage, 5) log-power high voltage, 6) safety channel #1 power, 7) safety channel #2 power, 8) short period, 9) building evacuation, and 10) manual scram button. A trip condition in any one of these nine parameters will precipitate a slow scram. Each portion of the RPS is described below.

7.4.1 Fuel Temperature

The fuel temperature monitoring and display system was described in section 7.3-4 and a block diagram is given in Figures 7-7 and 7-11. The current sensor module takes the output from the TC transmitter and sends a signal to the digital temperature indicator. The current sensor also has a built in trip function that produces an output if the temperature exceeds an adjustable limit. A special calibration box is used to set this level to correspond to 500°C. Thus, if the fuel temperature exceeds 500°C, a signal is sent to the logic element which causes a fast scram and the fuel temperature relay in the slow scram circuit opens effecting a slow scram.

7.4.2. Seismometer

The seismometer is a "Seismic Switch" built by the California Academy of Sciences that is really a very sensitive critically damped pendulum with a period of 1 second and a contact spacing of .05 centimeter. A sensitive relay is connected across the contacts of the seismic switch. In the event of any ground motion, with a displacement of .05 cm or more, the contacts of the seismic switch will make, the sensitive relay will actuate and cause the seismic relay in the scram chain to open dropping the rods into the reactor. The ground motion and earthquake intensity required to activate the seismic switch is given in Figure 2-21.

The ground acceleration required to close the seismic switch is reported to be 6 cm/second^2 or .0061g (1g = 978 cm sec^{-2}). The experts at the WSU Geology Department have estimated that an earthquake of magnitude 6.1 within 100 kilometers of the site is required to activate the seismic switch. No earthquake of this magnitude has ever occurred within 100 kilometers of the facility and is thus very unlikely to occur in the future. Section 12.3 of IAEA-TECDOC-348 recommends reactors have a seismic system with a trigger level of .025g or 24.5 cm/sec^2 for Class A facilities. The WSU facility seismic switch is significantly more sensitive than the recommended trigger level by a factor of over four.

The WSU reactor facility is deemed to be a Class A facility for seismic evaluation because:

1 During the lifetime of the facility, seismic events exceeding the design strength of the facility are not likely to occur.

2. Should such an event occur, the radiological consequences would be essentially equivalent to the MHA analyzed in section 13 and thus would be minimal.

FUEL TEMPERATURE SCRAM SYSTEM

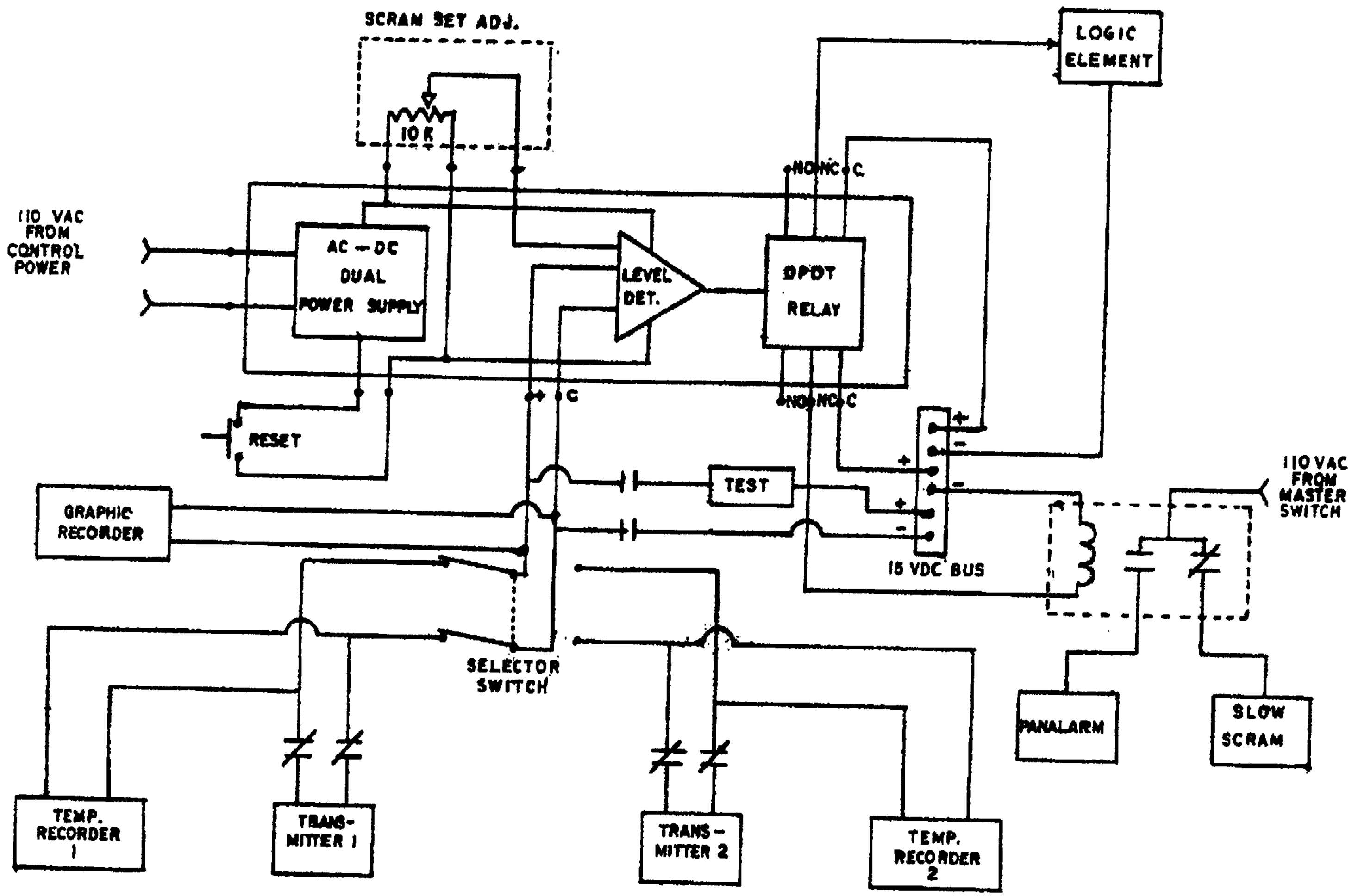


Figure 7-11

3. The seismic switch in the RPC will automatically shut down the reactor in the event of any significant seismic event.

7.4.3. High Voltage Failure Monitor

The power supply that feeds the UIC that constitutes safety channel#2 contains a high voltage monitoring circuit. If the high voltage decreases significantly, a UIC HV failure trip occurs sending a scram trip signal to the logic element causing a fast scram. Also, the CIC HV failure relay in the scram chain would open precipitating a slow scram.

The General Atomics Wide Range Channel also includes a high voltage failure monitor for the power supply that feeds the fission chamber. This signal controls one relay in the scram chain that would open in the event of a fissure chamber HV supply failure and precipitate a slow scram.

7.4.4 High Flux Scram

Safety channels #1, and #2, have a built in high-limit trip circuit for positive input currents. The adjustable trip circuit is completely solid state and is latching once the trip level is exceeded and must be reset by pressing the reset button. During the startup check out, each of the trip levels is tested and adjusted to correspond to a power level above 1 megawatt but below 1.25 megawatts. Thus during high power reactor operation, if either safety channel indicator exceeds the preset power level, a fast and slow scram is precipitated. Each safety channel separately feeds into the logic element and has its own relay in the scram chain for redundancy purposes to maximize safety.

7.4.5 Manual Scram Circuits

The RPS contains two manual scram circuits associated with the building evacuation alarm system and the manual scram button. If the reactor operator presses either the building evacuation alarm actuating button or the manual scram button, a slow scram is precipitated. Thus, in an emergency, just pressing the building evacuation button will cause the alarm to sound for people to evacuate the facility and also scram the reactor. If the operator just wants to quickly shutdown the reactor, he just pushes the manual scram button.

7.4.6 Emergency Power

The WSU reactor control system contains a large battery bank to provide emergency power to critical portions of the system during a power failure. A block diagram of the Auxiliary Reactor Emergency Supply (ARIES) is shown in Figure 7-12. The ARIES system provides emergency power to the area monitoring system, building evacuation alarm system, pool level alarm, seismograph alarm, and the security system in the controlled access areas of the facility.

7.5 Engineered Safety Features Actuation Systems

The WSU modified TRIGA reactor system does not contain any required engineered safety features as discussed in section 6 of this SAR. However, the HVAC system as described in section 6.2.2 and chapter 9 of this SAR does have a safety function. Furthermore, the Technical Specifications in section 3.9 state: "The reactor shall not be operated unless the

AUXILARY REACTOR INSTRUMENTATION EMERGENCY SUPPLY
A.R.I.E.S.

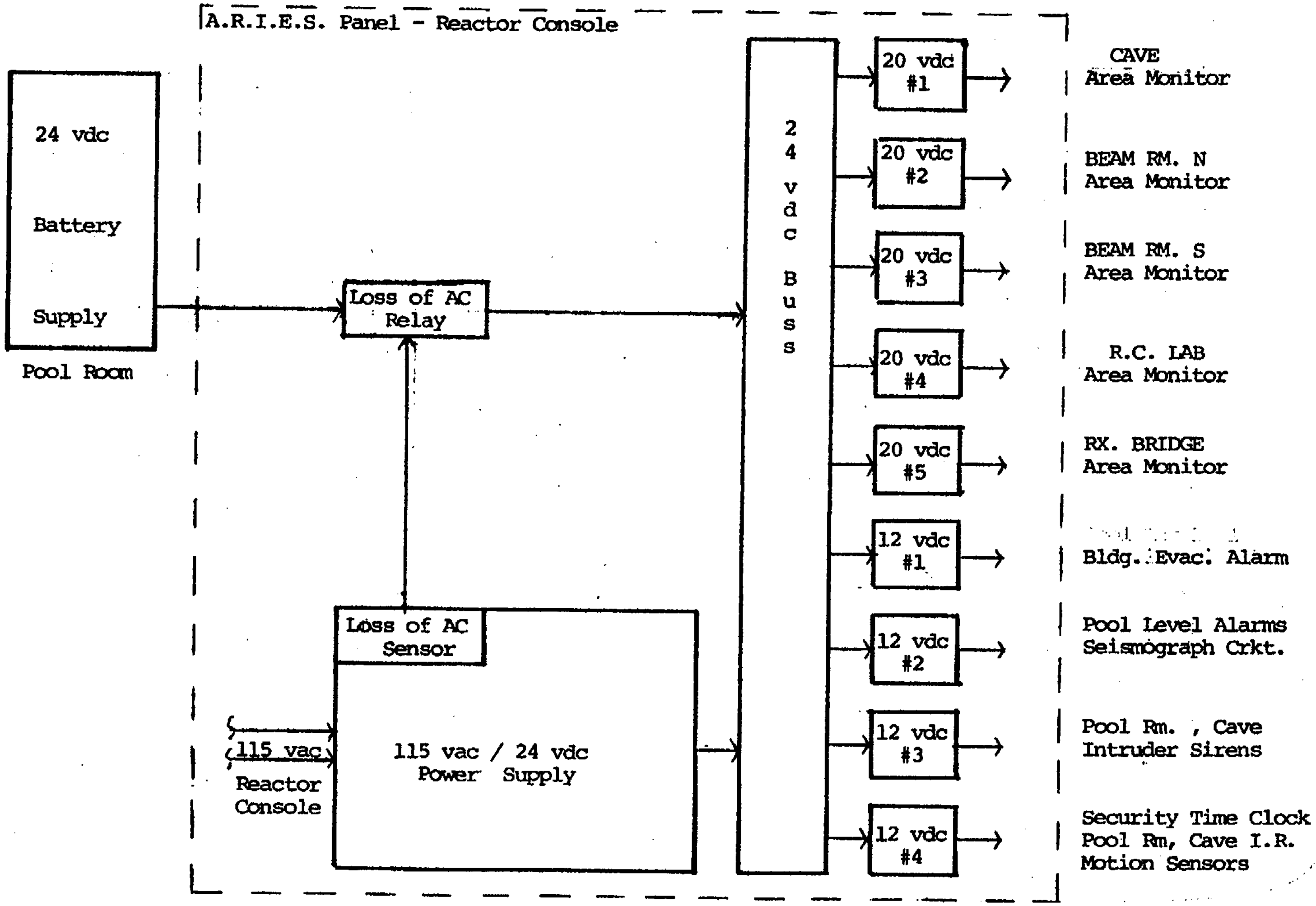


Figure 7-12

facility ventilation system is operable, except for periods of time not to exceed 48 hours to permit repair or testing of the ventilation system." The right side of the reactor console in the HVAC system control block contains three lighted latching indicator switches labeled 1) Ventilation Auto, 2) Ventilation Dilute, and 3) Ventilation Isolate. Table 7.5-1 describes the operation of fans 1, 3, and 4 and dampers 1, 2, 3, and 4 in the various modes of operation. Fan #1 is the pool room exhaust fan, fan #3 is the dilution fan, and fan #4 is the pool room air supply fan. See section 9 for more complete information of the HVAC system.

Table 7.5-1
Reactor Ventilation System Operating Modes

CONTROL MODE	FAN STATUS			AUTO-DAMPER STATUS			
	F1	F3	F4	D1	D2	D3	D4
AUTO	ON	OFF	ON	OPEN	CLOSED	CLOSED	OPEN
DILUTE	OFF	ON	OFF	CLOSED	OPEN	OPEN	CLOSED
ISOLATE	OFF	OFF	OFF	CLOSED	CLOSED	CLOSED	CLOSED

7.6 Control Console and Display Instruments

The control console is pictured in Figure 7-1 and the two primary display-control function blocks are shown in Figures 7-2 and 7-3. The design criteria for the control console and associated display instruments was discussed in section 7.2.1.

7.7 Radiation Monitoring Systems

The three systems that comprise the radiation monitoring systems as required by section 3.7 of the Technical Specifications are: 1) the area monitoring system, 2) the continuous air monitor, and 3) the Argon-41 stack monitor. The area monitoring system utilizes gamma-sensitive detectors to monitor the radiation level of various positions in the facility. The continuous air monitor is a beta-gamma detector with particulate collecting capabilities that monitors the particulate activity in the pool room air. The Argon-41 stack monitor is a gamma-sensitive detector that measures the ^{41}Ar content of the reactor exhaust. Table 7.7-1 lists the minimum monitoring channels that must be in operation during all reactor operations.

Table 7.7-1
Minimum Monitoring Channels

Channel*	Function	No.
Area radiation monitor	Monitor radiation level on the bridge	1
Area radiation monitor	Monitor radiation level in the beam room	1
Continuous air monitor	Monitor the activity of the pool room air	1
Exhaust gas monitor	Monitor the Argon-41 activity in the exhaust	1

*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 that shall be kept under visual observation.

7.7.1 Area Monitoring System

The area monitoring system at the WSU reactor facility is a Victoreen Model 855 G-M area monitoring system with detectors located at various points in the facility. A six module system is pictured in Figure 7-13. The WSU system is a 6 module system with the readout modules located on the left side of the console as shown in the console photo given in Figure 7-1. Detectors are located at the following positions in the facility: 1) reactor bridge, 2) cave room, 3) beam room north wall, 4) beam room south wall, 5) radiochemistry laboratory room 101, and 6) special sample monitor on reactor bridge.

Each readout module and its associated detector is a completely separate independent radiation monitoring channel. Each unit has five decade response range from .1 mr/hr to 1,000 mr/hr in a logarithmic scale. All circuitry is of highly reliable solid state construction with built in fail safe alarm outputs. The trip level on each module is independently adjustable and each unit has a built in check source. A special designed and calibrated fixture with an NBS certified source is utilized to calibrate the units.

Each model has a green "fail safe light" to indicate that it is functioning correctly and a red "alarm" light to indicate the set high level trip has been exceeded. The RM alarm outputs are wired into the console to activate an annunciator and light the "high area radiation" warning light. In addition, the bridge monitor is set up to scram the reactor and activate the building evacuation system if the bridge radiation level exceeds a preset high radiation level.

7.7.2 Continuous Air Monitoring System

The CAM unit at the WSU facility is a Victoreen CAM detector with a model 942A digital ratemeter for a readout. A block diagram of the CAM system is shown in Figure 7-14. The detector is a Beta-Gamma sensitive scintillator connected to a PM tube all inside a shielded detector housing. A vacuum pump takes a suction of the air over the top of the pool and passes it through the particulate collection filter in the CAM.

The readout unit has a presetable rate alarm feature that is connected to the CAM alarm light on the console as well as connected to cause the ventilation system to shift to the "Dilution Mode" in the event of a high CAM alarm. The sensitivity of the CAM system for Cs-137 is of the order of .3 counts per minute per disintegration per minute or about 30% efficiency factor. For lower energy beta emitters, the efficiency is less.

7.7.3 Argon-41 Monitoring System

The exhaust gas monitoring system of the WSU facility is a locally designed and constructed Argon-41 monitoring system shown in Figure 7-15. The unit takes a sample of the exhaust air being discharged from the facility, filters out particulates, and discharges the air into a 30 liter chamber in a lead shield. In the center of the chamber a 2" x 2" NaI(Te) detector is located that is connected to a single channel pulse height analyzer system set up to count the gamma rays emitted by Argon-41. The entire system as shown in Figure 7-15 includes appropriate electronics to count and record the Argon-41 content of the reactor exhaust.

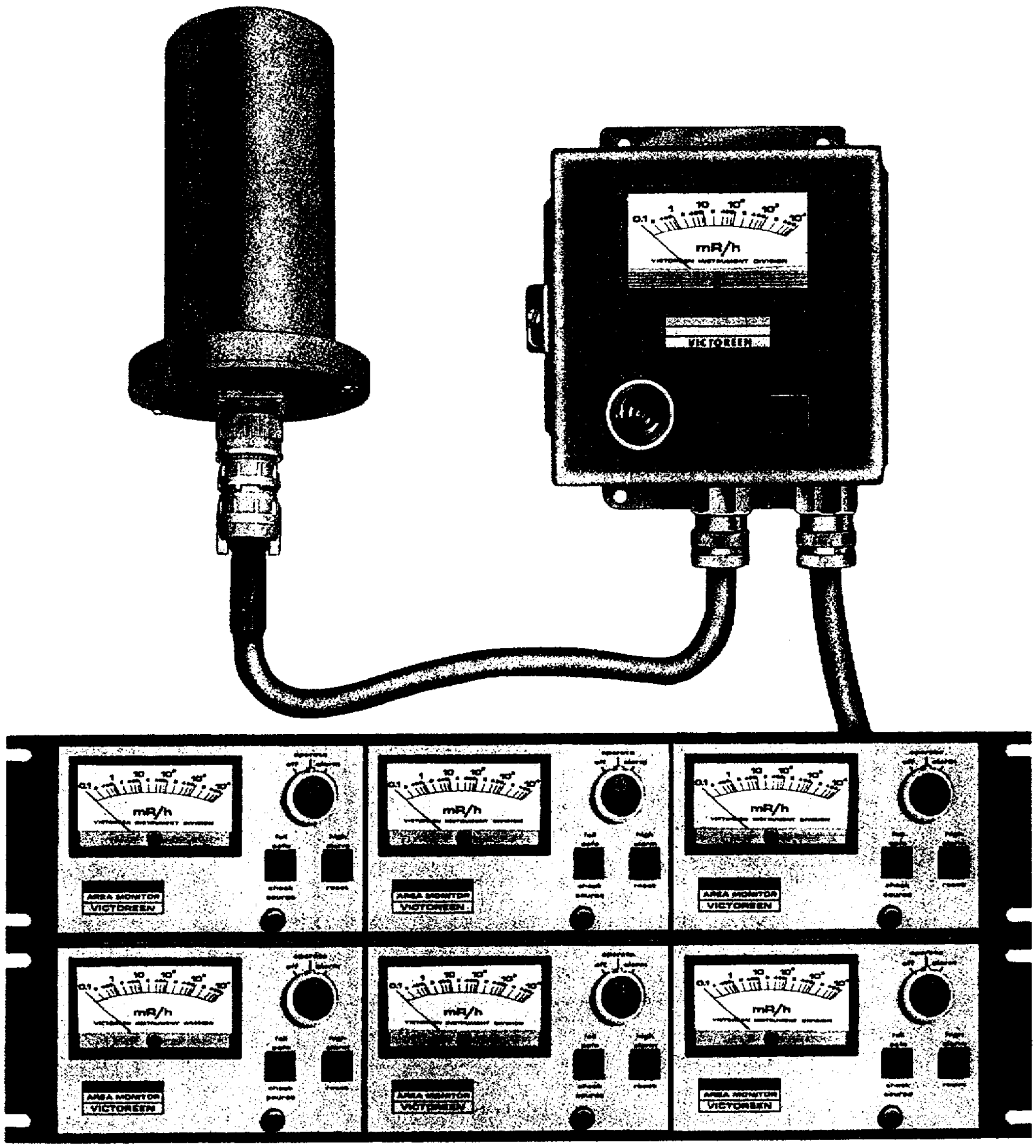


Figure 7-13
Major Components of a Victoreen 855 Series G-M Area Monitoring System

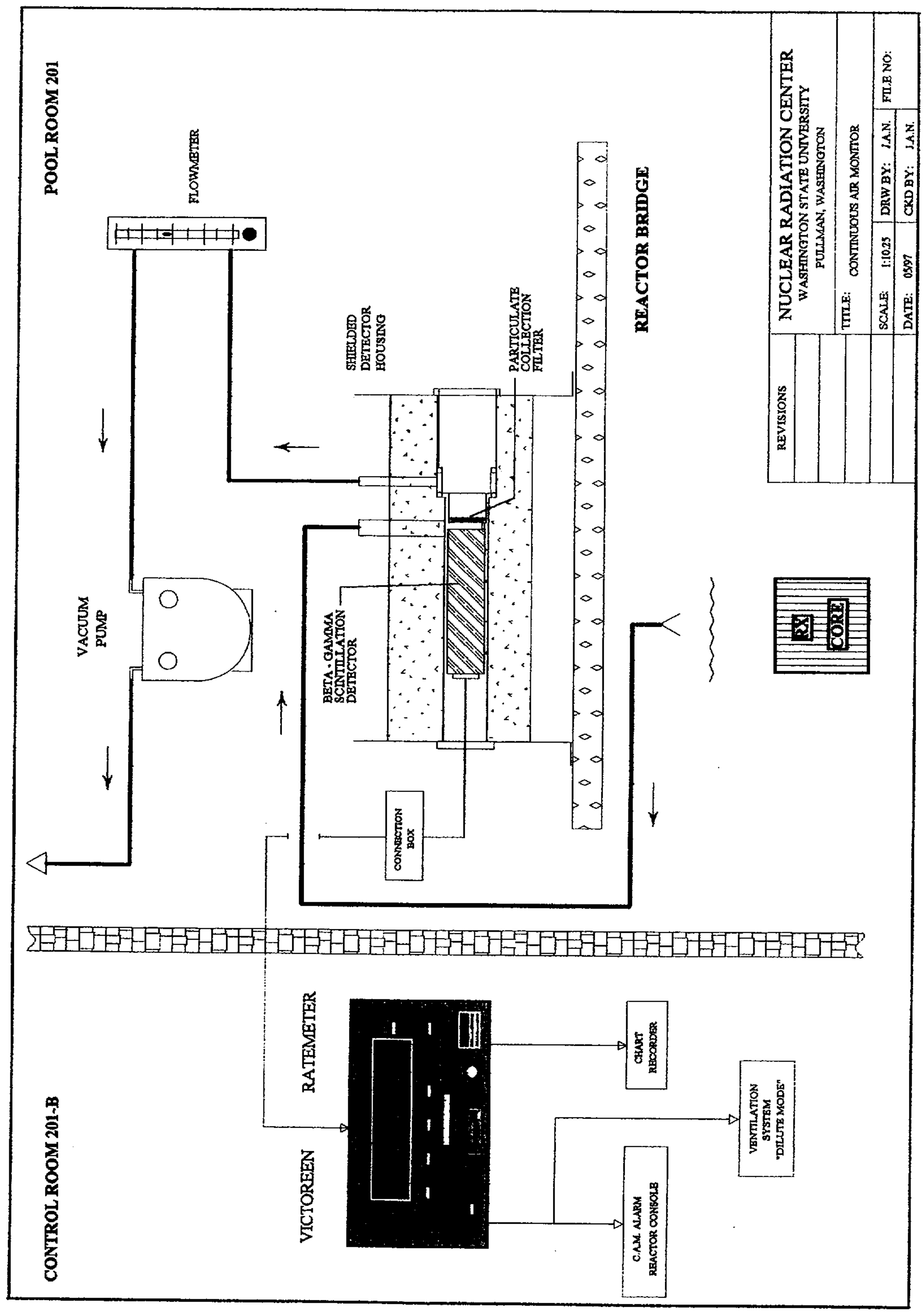
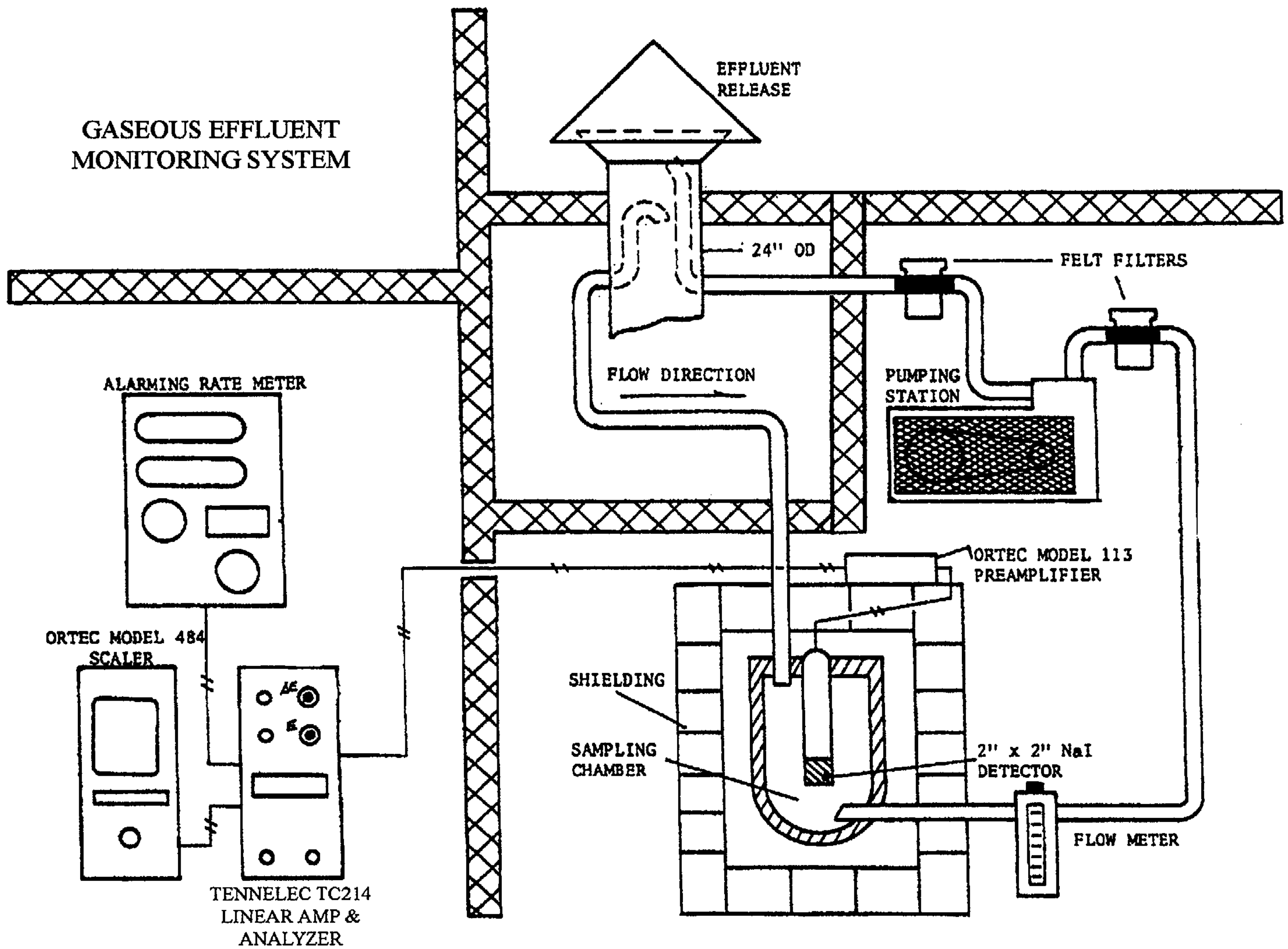

POOL ROOM 201
FLOWMETER
VACUUM PUMP
SHIELDED DETECTOR HOUSING
PARTICULATE COLLECTION FILTER
BETA-GAMMA SCINTILLATION DETECTOR
REACTOR BRIDGE
CONNECTION BOX
RX CORE
CONTROL ROOM 201-B
VICTOREEN
RATEMETER
CHART RECORDER
VENTILATION SYSTEM "DILUTE MODE"
C.A.M. ALARM REACTOR CONSOLE
REVISIONS
NUCLEAR RADIATION CENTER
WASHINGTON STATE UNIVERSITY
PULLMAN, WASHINGTON
TITLE: CONTINUOUS AIR MONITOR
SCALE: 1:10.25
DRW BY: J.A.N.
FILE NO:
DATE: 05/97
CKD BY: J.A.N.

Figure 7-14



GASEOUS EFFLUENT
MONITORING SYSTEM
EFFLUENT
RELEASE
24" OD
FELT FILTERS
FLOW DIRECTION
PUMPING
STATION
ALARMING RATE METER
ORTEC MODEL 113
PREAMPLIFIER
ORTEC MODEL 484
SCALER
SHIELDING
SAMPLING
CHAMBER
2" x 2" NaI
DETECTOR
FLOW METER
TENNELEC TC214
LINEAR AMP &
ANALYZER

Figure 7-15

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Amendment No. 30 to Facility Operating License No. R-2 -- Penn State University, August 6, 1991.

NRC Generic Letter 95-02, Guidelines on the licensing of digital upgrades, in determining the acceptability of performing analog-to-digital replacements under 50.59, April 14, 1995.

Operation and Maintenance Manual for Victoreen Model 855 Area Monitoring System

Operation and Maintenance Manual for Victoreen Model 942A-200 Universal Digital Ratemeter

Criteria for the reactor safety system of research reactors, ANSI/ANS 15.15 -- 1978

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

The Nuclear Radiation Center, like all buildings on campus, has normal electrical power provided by the campus wide power grid. A three wire 4160 VAC power line on wooden poles runs along the south side of the road (Roundtop Road) out to the facility. This line stops across the road from the facility and feeds power underground from the line into the facility. In the facility the power feeds into a transformer vault, room [redacted] The transformer vault contains three 167 KVA transformers with 4160 VAC primaries and 120/208 VAC secondaries in a "Y" configuration. There is a three wire disconnect switch in the transformer vault between the power line and the transformers.

The transformers feed risers into the main breaker area, room 6, which has one 1000 amp and one 800 amp main disconnect. The main breakers feed six 225 amp individual circuit breakers. The first 225 amp breaker feeds three 100 amp secondary breakers. One of the 100 amp breakers feeds "Panel P" which has numerous small breakers including two 20 amp breakers that feed the reactor control system and reactor shop/control room area. That is, the reactor control power is fed from a single 20 amp breaker that does not power anything except the reactor control system. A second 225 amp breaker feeds power back to the pump room that provides power for the cooling tower and primary and secondary pumps. A third 225 amp breaker feeds power up to the penthouse which provides power to the entire building HVAC system and fume hood fans. The other 225 amp breaker feeds power to the rest of the Nuclear Radiation Center. A block diagram of the power system is shown in Figure 8-1.

Power to the campus grid is provided by Avista Utilities via a substation located off campus via a 4.16 KV 2000 amp line. Avista Utilities electrical power is highly reliable and power outages are infrequent and if and when they occur, it is usually in winter. A loss of power while the reactor is at full power will cause an immediate shut down of the reactor and a cessation of operation of the primary and secondary pumps as well as the cooling tower fan. The entire scram system functions by cutting off power to the trip amps which cut off power to the blade holding magnets which then fall into the reactor under the force of gravity. A power failure also cuts off power to the trip amps causing a scram (see Figure 7-9). The decay heat in the fuel will be dissipated by natural convection to the pool water and cause a very small increase in the pool water temperature. Even a loss of pool water as was demonstrated in section 13 of this SAR, would not precipitate a fuel cladding failure problem. Thus a loss of main electrical power does not create a safety problem, only an inconvenience. The HVAC system also shuts down and the dampers close on loss of power so there is no discharge of air from the facility during a power loss. That is, a power loss puts the HVAC system in the "Isolate" mode (see Table 7.5-1). If the power is off for an extended period of time, the humidity in the pool room will build up but this is not a safety problem, only an inconvenience.

The power feeding the reactor control system, as previously indicated, comes via one separate 20 amp circuit with no special power conditioning. All the modern electronics in the control system are essentially immune to small fluctuations in primary power and noise on the power line. The only time power line noise causes any type of problem is during the initial portion of start up and then it is only a inconvenience not a safety hazard sometimes precipitating

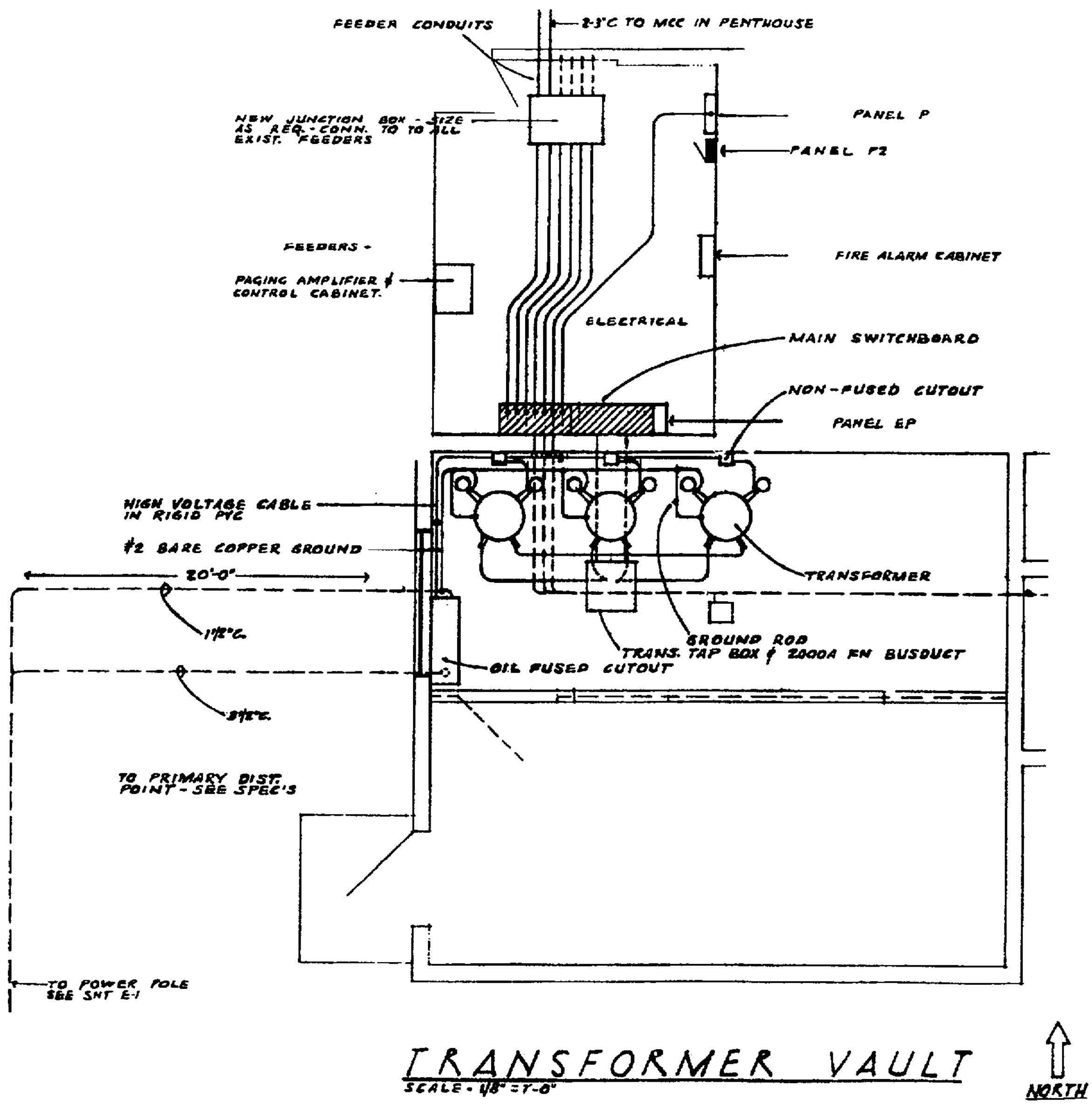
FEEDER CONDUITS
2-3"C TO MCC IN PENTHOUSE
NEW JUNCTION BOX - SIZE AS REQ. - CONN. TO TO ALL EXIST. FEEDERS
PANEL P
PANEL P2
FEEDERS -
PAGING AMPLIFIER & CONTROL CABINET.
FIRE ALARM CABINET
ELECTRICAL
MAIN SWITCHBOARD
NON-FUSED CUTOUT
PANEL EP
HIGH VOLTAGE CABLE IN RIGID PVC
#2 BARE COPPER GROUND
20'-0"
TRANSFORMER
1½"C.
GROUND ROD
TRANS. TAP BOX & 2000A FN BUSDUCT
OIL FUSED CUTOUT
3½"C.
TO PRIMARY DIST. POINT - SEE SPEC'S
TO POWER POLE SEE SHT E-1
TRANSFORMER VAULT
SCALE - 1/8" = 1'-0"
NORTH

Figure 8-1

spurious period trips. The WSU reactor has a period trip that is primarily used for training purposes though not required by the Technical Specifications.

8.2 Emergency Electrical Power Systems

Emergency power for critical instrumentation in the event of a power loss is provided by the Auxiliary Reactor Emergency Supply (ARIES) shown in Figure 7-12. ARIES contains a 24 vdc battery bank consisting of three 12 volt batteries that are routinely tested and serviced, two batteries in use and one in reserve. In accordance with the requirements of section 3.6 of the Technical Specifications, the ARIES system provides emergency power to the area monitoring system, building evacuation alarm system, pool level alarm, seismograph alarm, and the security system. The ARIES system is located in the pool room which is a controlled access area (CAA). The security system, fire alarm system, and digital telephone system each have separate battery backup (UPS) power units.

There are a number of self-contained emergency lighting units in the Nuclear Radiation Center that come on to provide lighting during a power loss. A large twin lamp unit is located in the pool room and smaller single lamp units in the control room and reactor shop. All hallways and stairways have one or more large twin lamp emergency lights.

9 AUXILIARY SYSTEMS

9.0 Design Basis

The primary auxiliary system of the WSU facility is the heating, ventilation, and air conditioning (HVAC) system which also plays an engineered safety feature role. Fuel elements and other special nuclear materials are protected by physical confinement and surveillance. The physical confinement serves to control the release of radioactive material during routine operation or potential accident conditions. Release of airborne radioactivity consists mostly of air activation products from routine operations or fission product materials from a non-routine fuel element failure. The effects of the failure of a fuel rod and the associated release is analyzed in detail in section 13 of this SAR. The purpose of the HVAC system is to provide for the comfort of the people operating the facility and to control exposure of operational personnel and the public in the event of a release of radioactive material into the pool room.

9.1 Heating, Ventilation, and Air Conditioning Systems

The Nuclear Radiation Center has two separate HVAC systems, one for the reactor related areas and one for the non-reactor portions of the building. The HVAC system for the reactor areas is shown in Figures 9-1 and 9-2. Electrical diagrams for the central system for the reactor area HVAC system are shown in Figures 9-3 and 9-4. The HVAC system may be controlled from the reactor console or from a special panel in the main office.

The main pool room exhaust fan, F-3, draws 4500 (2124 l/sec) cfm from the pool room and discharges that air into the atmosphere via a stack in normal or "auto" mode as shown in Figures 9-1, 9-2, and Table 9-1. In this mode of operation, fan F-4 supplies 4350 (2053 l/sec) cfm of treated air to the pool room. Since the supply rate is lower than the exhaust rate, the pool room operates at a slight negative pressure thus insuring that all air leakage is "inflow" into the pool room. Air from the beam room area is removed by fan F-2 and discharged into the stack and then into the atmosphere. The suction side of the beam room fan contains four (4) each 24 inch (61 cm) by 24 inch (61 cm) HEPA filters connected in parallel to remove all particulates from this air before it is discharged. There is no supply fan for beam room air so the beam room operates at a negative pressure drawing air from outside by leakage around doors, mainly the freight door. The HVAC system may also be operated in the "dilution" and "isolation" modes as indicated in Table 9-1.

Table 9-1
Reactor Ventilation System Operating Modes

CONTROL MODE	FAN STATUS			AUTO-DAMPER STATUS				
	F1	F3	F4	D1	D2	D3	D4	D5
AUTO	ON	OFF	ON	OPEN	CLOSED	CLOSED	OPEN	CLOSED
DILUTE	OFF	ON	OFF	CLOSED	OPEN	OPEN	CLOSED	OPEN
ISOLATE	OFF	OFF	OFF	CLOSED	CLOSED	CLOSED	CLOSED	CLOSED

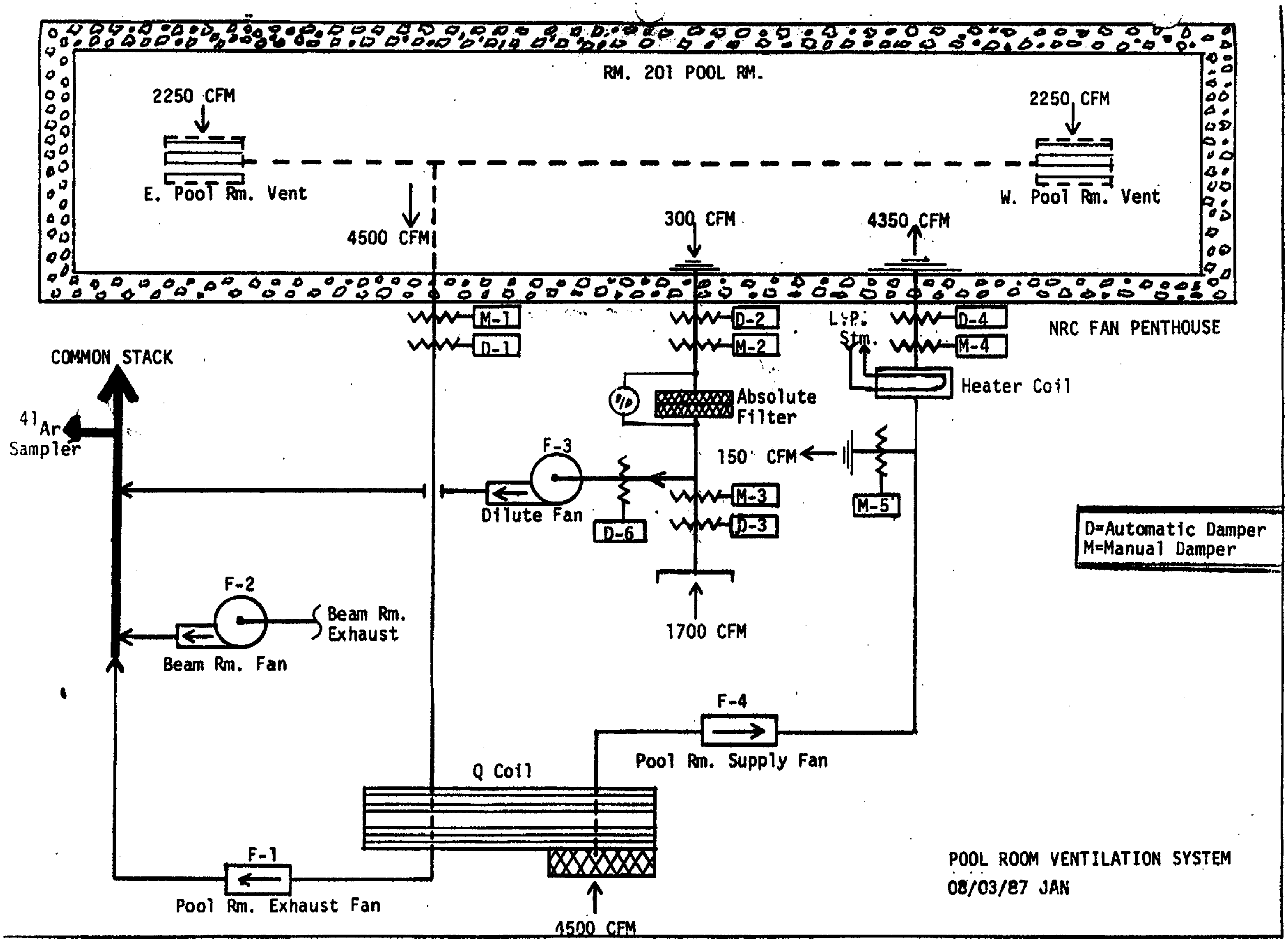
RM. 201 POOL RM.
2250 CFM
2250 CFM
E. Pool Rm. Vent
W. Pool Rm. Vent
4500 CFM
300 CFM
4350 CFM
M-1
D-1
D-2
M-2
L.P. Stm.
D-4
M-4
NRC FAN PENTHOUSE
COMMON STACK
41Ar Sampler
Absolute Filter
Heater Coil
F-3
150 CFM
M-3
D-3
M-5
Dilute Fan
D-6
D=Automatic Damper
M=Manual Damper
F-2
Beam Rm. Exhaust
Beam Rm. Fan
1700 CFM
F-4
Pool Rm. Supply Fan
Q Coil
F-1
Pool Rm. Exhaust Fan
4500 CFM
POOL ROOM VENTILATION SYSTEM
08/03/87 JAN

Figure 9-1

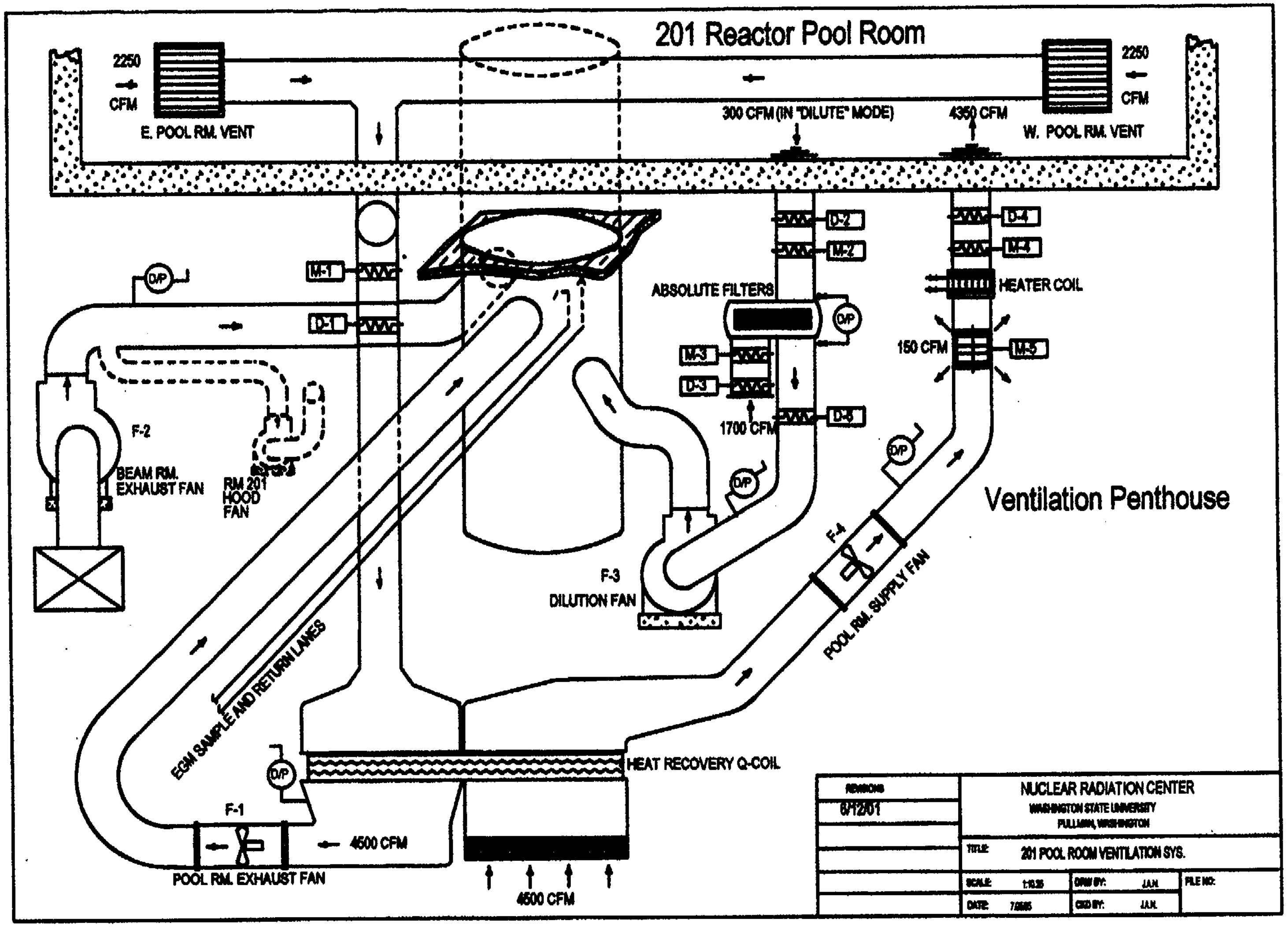
201 Reactor Pool Room
2250 CFM
E. POOL RM. VENT
2250 CFM
W. POOL RM. VENT
300 CFM (IN "DILUTE" MODE)
4350 CFM
M-1
D-1
D/P
F-2
BEAM RM. EXHAUST FAN
RM 201 HOOD FAN
D-2
M-2
ABSOLUTE FILTERS
M-3
D-3
1700 CFM
D-5
D-4
M-4
HEATER COIL
150 CFM
M-5
Ventilation Penthouse
F-3
DILUTION FAN
F-4
POOL RM. SUPPLY FAN
EGM SAMPLE AND RETURN LANES
HEAT RECOVERY Q-COIL
F-1
4500 CFM
POOL RM. EXHAUST FAN
4500 CFM
NUCLEAR RADIATION CENTER
TITLE: 201 POOL ROOM VENTILATION SYS.
6/12/01
DRW BY: J.A.N.
CKD BY: J.A.N.
FILE NO:

Figure 9-2

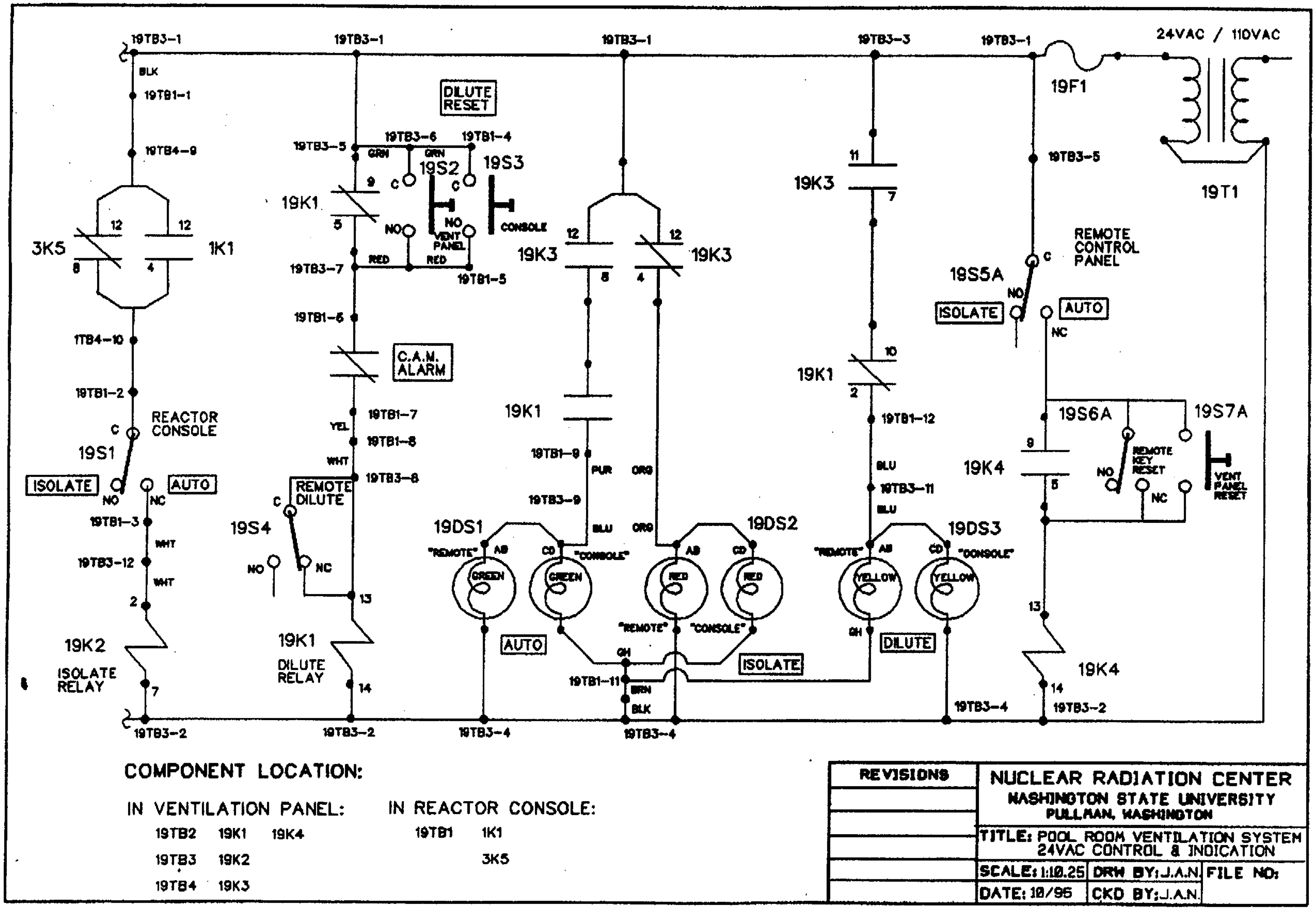
19TB3-1
24VAC / 110VAC
19F1
19T1
DILUTE RESET
19S2
19S3
CONSOLE
VENT PANEL
3K5
1K1
19K1
19K3
19K4
REMOTE CONTROL PANEL
19S5A
ISOLATE
AUTO
C.A.M. ALARM
REACTOR CONSOLE
19S1
19S6A
19S7A
REMOTE KEY RESET
VENT PANEL RESET
REMOTE DILUTE
19S4
19DS1
19DS2
19DS3
"REMOTE"
"CONSOLE"
GREEN
RED
YELLOW
DILUTE
19K2
ISOLATE RELAY
DILUTE RELAY
19TB3-2
19TB3-4
COMPONENT LOCATION:
IN VENTILATION PANEL:
19TB2 19K1 19K4
19TB3 19K2
19TB4 19K3
IN REACTOR CONSOLE:
19TB1 1K1
3K5
REVISIONS
NUCLEAR RADIATION CENTER
WASHINGTON STATE UNIVERSITY
PULLMAN, WASHINGTON
TITLE: POOL ROOM VENTILATION SYSTEM 24VAC CONTROL & INDICATION
SCALE: 1:10.25
DRW BY: J.A.N.
FILE NO:
DATE: 10/95
CKD BY: J.A.N.

Figure 9-3

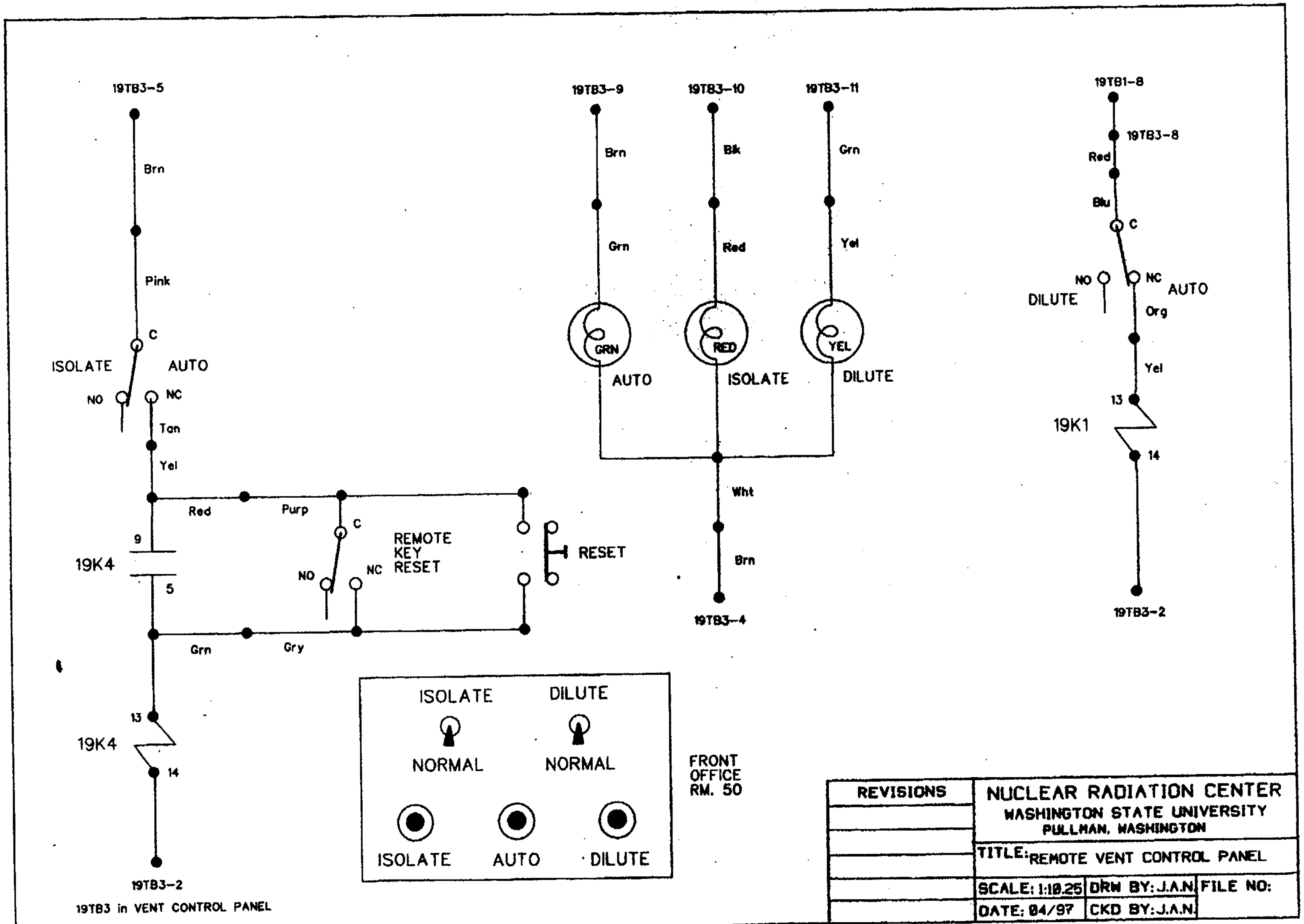
19TB3-5
Brn
Pink
C
ISOLATE
AUTO
NO
NC
Tan
Yel
Red
Purp
C
REMOTE
KEY
RESET
NO
NC
RESET
9
19K4
5
Grn
Gry
13
19K4
14
19TB3-2
19TB3 in VENT CONTROL PANEL
19TB3-9
19TB3-10
19TB3-11
Brn
Blk
Grn
Grn
Red
Yel
GRN
RED
YEL
AUTO
ISOLATE
DILUTE
Wht
Brn
19TB3-4
19TB1-8
19TB3-8
Red
Blu
C
NO
NC
DILUTE
AUTO
Org
Yel
13
19K1
14
19TB3-2
ISOLATE
DILUTE
NORMAL
NORMAL
ISOLATE
AUTO
DILUTE
FRONT
OFFICE
RM. 50
REVISIONS
NUCLEAR RADIATION CENTER
WASHINGTON STATE UNIVERSITY
PULLMAN, WASHINGTON
TITLE: REMOTE VENT CONTROL PANEL
SCALE: 1:10.25
DRW BY: J.A.N.
FILE NO:
DATE: 04/97
CKD BY: J.A.N.

Figure 9-4

The "dilute" mode is intended to be used to control the release of contaminated pool room air into the atmosphere, only 300 cfm of air is drawn from the pool room by fan F-3. The 300 cfm is first passed through a HEPA filter and mixed with 1700 cfm of outside air before being discharged to the atmosphere. The dilution factor is 6.67 and the HEPA filter will remove most if not all particulates and even iodine. Section 13 substantiates an insignificant exposure to individuals outside of the facility in the event of the MHA with the system operating in the dilution mode and the HEPA filter not functioning. In the isolation mode, no air is discharged from the facility and all dampers are closed.

The control system for the reactor area HVAC was designed and installed by Johnson Controls with a separate control panel in the reactor control room. The WSU Reactor Staff relocated the control system so that the control relays and control switches are located at the reactor console and main office. The reactor staff routinely tests and maintains the electrical portion of the reactor area HVAC system.

The actual fans, dampers, and other portions of the system are maintained by the Control Shop of the WSU Physical Plant. The flow rates are calibrated every four years or whenever the reactor staff discovers a problem with the operation of the HVAC system. The HEPA filter is checked and/or replaced when the pressure drop across the filter exceeds one inch of water or every two years, whichever is sooner.

9.2 Handling and Storage of Reactor Fuel

All movement of irradiated reactor fuel must be done as prescribed by SOP #7, "Standard Operating Procedure for Core Changes and Fuel Movement." This procedure requires that a detailed written and approved movement and/or change schedule be prepared prior to all core changes or fuel rod movement. It is inappropriate to include such procedures in a SAR since they change with time. The key points involved are:

1) Presence of a Senior Reactor Operator to supervise the operation.
2) Insuring that all reactor control systems are on and functioning properly.
3) The reactor will remain subcritical during the operation.
4) An accurate log of the operation is maintained.
5) Minimization of personnel radiation exposure.
6) Stored fuel maintained in a geometry with a multiplication factor of 0.8 or lower.
7) Resultant core meets all license and Technical Specification requirements.

Fuel not in the core is always stored in one of two storage racks that have been designed for "always safe geometry" with a K_{eff} of 0.8 or less when filled with fuel rods. All fuel rods and four rod clusters are inscribed with suitable identifying marks and an accurate log of the location of all fuel at the facility is maintained. Fuel movement tools are always kept locked and under the control of a Senior Reactor Operator. The facility maintains a computerized SNM record system that keeps track of fuel including burnup.

All aspects of the use and control of SNM at the WSU Radiation Center including receipt of new fuel and the shipment spent fuel is covered in the WSU Special Nuclear Materials Accountability Plans. The plan has been written to cover all the SNM related requirements in 10 CFR Parts 40, 70, and 150. The plan has been reviewed by the USNRC and covers:

1. Introduction
2. Definitions
3. Accountability Responsibility
4. Storage
5. Physical Inventory
6. Core Change Log
7. Fuel Rod Records
8. Material Status Reports
9. Transfer of SNM
10. Loss or Theft of SNM

9.3 Fire Protection Systems and Programs

Fire prevention is primarily a function of operation rather than of system design. The WSU Nuclear Radiation Center has an ADT fire alarm system installed with detectors in every room and space in the building. In the event of a fire alarm, the system automatically calls the WSU Fire Department who show up at the facility within 3 minutes along with the campus police. All fire prevention and control is the responsibility of the WSU Fire Department who are routinely trained on the unique features and hazards of the Nuclear Radiation Center. Also, a detailed procedure related to the action to be taken by the staff in the event of a fire or explosion is contained in the Nuclear Radiation Center Emergency Plan Implementing Procedures.

9.4 Communication Systems

The primary communication system for the Nuclear Radiation Center as a whole is the WSU Telephone system. The system is a digital system manufactured by INTECOM with many features including Voice Mail, Call Forward, Conference Calling, Call Waiting and Call Transfer. Each office in the Center has a separate number and telephone line including the reactor main office, reactor director's office, reactor supervisor's office and reactor console. There is one general line to the main office which the secretary answers and can transfer to the appropriate office or person. A special emergency telephone line and telephone of the analog type is installed in the room designated as the Emergency Control Center in the Emergency Plan.

Additional communication systems include an intercom system with the main unit located at the reactor console and satellite units located in the main office, reactor bridge, reactor control room door, radiochemistry laboratory, and beam room. Communication between the control console and satellite units may be initiated at either end but satellite units cannot communicate with each other. A paging system also exists with speakers throughout the Center and a microphone at the control console and main office. This system is used to announce changes in the reactor status to everyone in the Center.

[redacted]

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

Facility License R-76 of August 11, 1982 authorizes the possession of a maximum of 25 kilograms of uranium-235 of various enrichments and 32 grams of plutonium contained in a plutonium-beryllium neutron source. The State of Washington is an Agreement State and Washington State University has radioactive materials license WN-C003 with the State of Washington. The Nuclear Radiation Center takes the position that all by-product materials produced within the reactor are covered by the reactor license when such materials are within the reactor operating areas that include the reactor shop area, reactor pool room, radiochemistry laboratory, and beam room. When materials leave these areas they transfer to the State license and appropriate transfer documentation and control must be done as required by the State license. Control of the transferred material is the responsibility of the WSU Radiation Safety Office.

There is a cobalt-60 irradiation facility located in the reactor pool which is covered by the State license but the technical specification for license R-76 also contains restrictions to insure that such sources do not cause a problem with the reactor. The specific provisions are as follows:

(1) Sealed sources shall not at any time be stored or used closer than five (5) feet away from the face of an operating reactor core. The total activity of all sealed sources stored in the pool shall not exceed 100,000 curies. All sealed source configurations shall be designed so that a loss of pool water accident will not precipitate a sealed source encapsulation integrity problem and the sources shall be stored in an appropriate shield so as not to produce a significant radiation hazard in the event of a loss of reactor pool water accident.

(2) All storage of sealed sources greater than 100 curies in the reactor pool shall be considered as an experiment and shall be reviewed and approved by the Reactor Safeguards Committee. A written operating procedure for the storage and use of sealed sources in the reactor pool shall be in effect.

(3) The radionuclide content of the reactor pool water shall be monitored monthly at an interval not to exceed six (6) weeks in order to detect a significant leak in the sources stored in the reactor pool. If the specific radionuclide content of the pool water for radionuclides from a sealed source stored in the reactor pool exceeds one-third (1/3) the 10 CFR 20 Appendix B, Table 3 value, steps shall be taken to isolate the source of the activity and to mitigate the problem.

9.6 Other Auxiliary Systems

The WSU reactor facility does not have a closed primary system with cover gas and all the existing auxiliary systems have been described in the previous section of this section 9 of the SAR.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

The WSU Modified TRIGA reactor was initially constructed as a 100 kW open pool type reactor with MTR plate type fuel in the 1960s. The reactor was patterned after the design for open pool type research reactors developed at ORNL and built by General Electric. Thus, the WSU reactor has the experimental facilities associated with the open pool design rather than those of a traditional TRIGA reactor. These facilities are described in detail in the next section and include beam ports, a thermal column, and vertical in-core irradiation facilities.

The primary experimental uses of the facility are for radioisotope production and for neutron activation analysis. In a typical year, NAA is performed on the order of 2000 samples. The samples are counted on Ge(Li) detectors so the activity level is very low in terms of radiation exposure to individuals. Beam type experiments are not done because the beam strengths available are too low for any significant beam type experiment. Thus 95% of all use involves sample irradiation in the vertical irradiation tubes. More recently, the thermal column has been converted into a Boron Neutron Capture Therapy (BNCT) facility discussion in section 16.2 of the SAR.

10.2 Experimental Facilities

The thermal column of the reactor as well as the beam ports as originally built are shown in Figure 10-1. The facility originally had four large beam tubes, H1 to H4, and six small beam tubes, E1 to E4 and T1 to T2. However, during the pool repair work in August of 1999 all beam tubes were removed in order to allow access to the pool wall. On reassembly new gaskets were installed on all the beam tubes and the small through tubes T1, T2, T3, and T4 were not put back. Tangential tubes E2 and E3 were however reinstalled. The thermal column facility originally consisted of a 5 ft by 5 ft by 6 ft parallel piped of graphite and a two inch thick lead gamma ray shield on the core side of the thermal column, however, the thermal column was converted in 2000 to a source unit for the Boron Neutron Capture Therapy (BNCT) facility. The in-core irradiation facilities consist of six rotator tubes marked "R" in Figure 10-2, a pneumatic rabbit tube that can be positioned in the location of the wet tube marked "W" in the drawings and the wet tube itself. The rabbit system is shown in Figure 10-3 and a typical rotator tube assembly in Figure 10-4. The most frequently used facilities are the six vertical rotator tubes used for isotope production and sample irradiation for NAA purposes as previously indicated.

The facility has been in operation for over 30 years as a Modified TRIGA reactor during which time no significant problem or excessive personnel exposure has occurred associated with the operation and use of the reactor and the associated experimental programs. The radiation monitoring systems described in section 7.7, and the Radiation Safety Program described in section 12 as well as the procedures described in section 10.3 have proven to be quite adequate to insure the safe usage of the facility. The highest exposure that has been recorded by any user or reactor operational personnel over the past 10 years has been 50 mr/year. The radiation monitoring program includes using film badges to measure the maximum possible exposure in a variety of locations around the facility. The hottest spot is on the bridge of the reactor which has an annual total exposure of 600 mr/year. Thus, the maximum possible exposure to any individual would be 600 mr if that individual spent 24 hours a day 365 days a year on the bridge.

All samples are monitored and the exposure rate recorded when they are removed from the reactor as required by SOP #1.

[redacted]

[redacted]

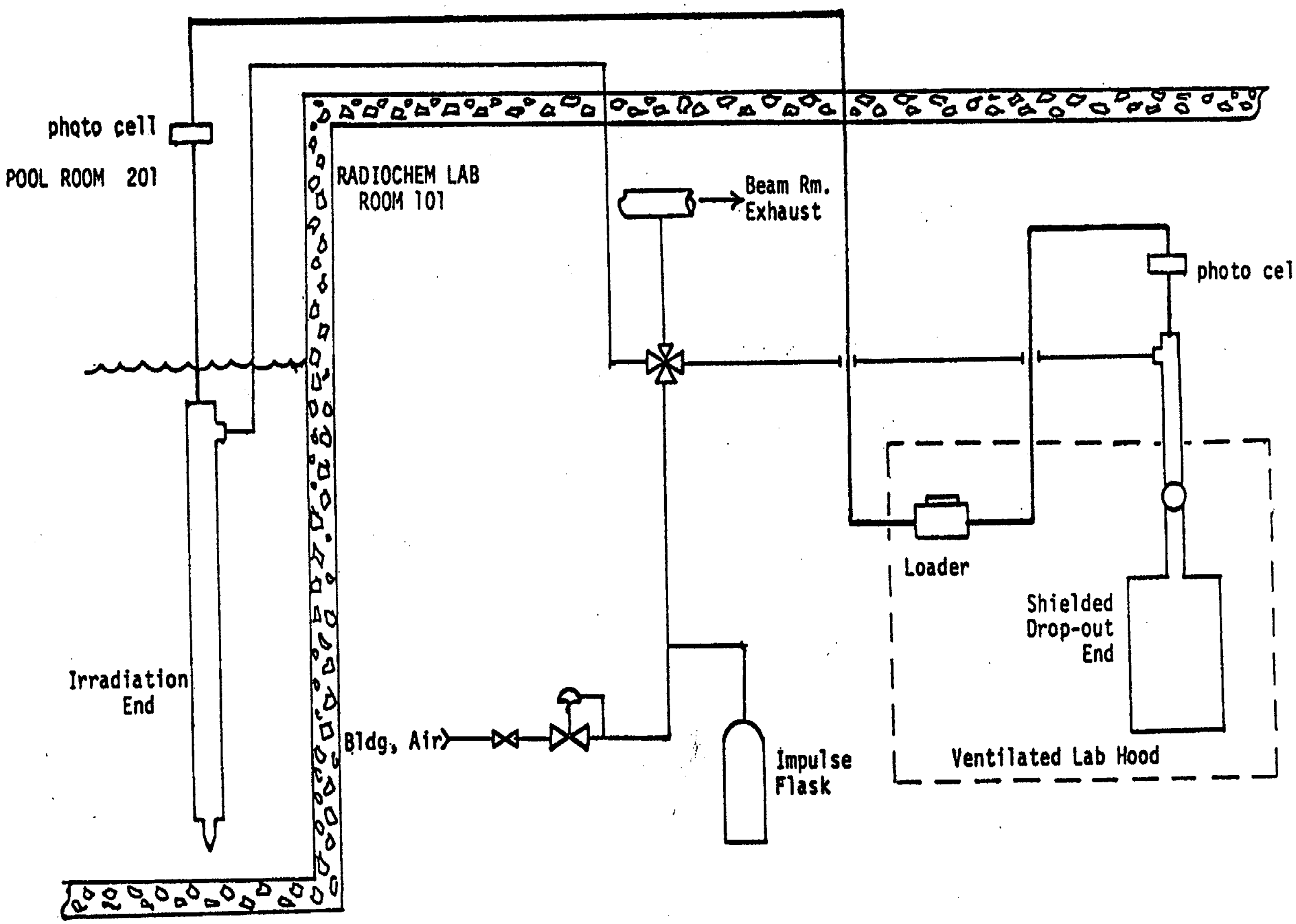

Figure 10-3
Pneumatic Transfer System

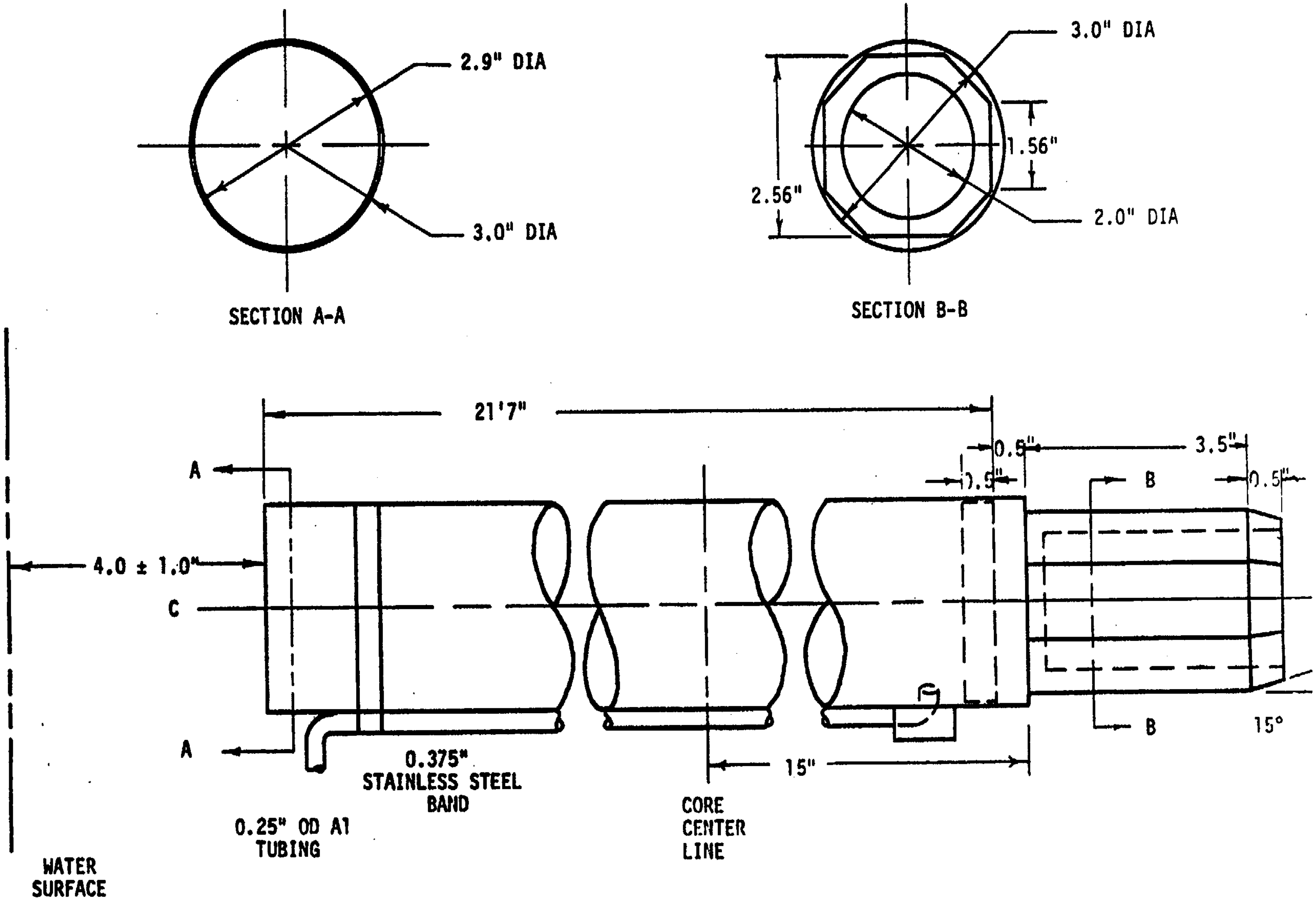

Figure 10-4
Rotator Tube Assembly

Recently the facility has converted the thermal column into a BNCT (Boron Neutron Capture Therapy) medical facility. The modifications and additional shielding associated with the BNCT facility is shown in Figure 10-5. Detailed information on the design, licensing and use of the BNCT facility is contained in Chapter 16 of this SAR.

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).

10.3.1 Reactor Staff Review

A number of detailed Standard Operating Procedures have been in place for many years to ensure the safe use of the WSU reactor. The specific SOPs related to reactor use and experiments are as follows: SOP #1 -- Standard Procedure for Use of the Reactor which includes details on reactor use if authorized, the required hazards analysis associated with all usage, the hazards review procedure, specific limits on the use of the reactor, and user-certification. The specific limitations on reactor usage are given in Appendix 10-A. The key criteria that the Reactor Supervisor must consider in approving the use of the reactor are:

1) No license limit will be exceeded during the proposed experiment.
2) No explosive material involved with the experiment.
3) Materials which could off-gas, sublime, volatilize or produce aerosols under a) normal operating conditions, b) credible accident conditions, or c) 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 pool room or atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the applicable limits of Appendix B of 10 CFR 20.

SOP #2 -- Standard Procedure for Performing Irradiations Using the Reactor which includes details on sample failure analysis and the following types of irradiations:

1) Procedure for in-core irradiations from the bridge
2) Procedure for pneumatic transfer system irradiations (rabbit)
3) Procedure for thermal and epithermal column irradiations

SOP #3 -- Standard Procedure for Performing Experiments Using the Reactor

1) Types of experiments
    a. Operational experiments
    b. Non-operational experiments
2) Authorization to perform experiments
    a. Operational experiments
    b. Non-operational experiments
3) Procedure for performing experiments
    a. Operational experiments
    b. Non-operational experiments

[redacted]

SOP #1, Standard Procedure for Use of the Reactor, requires all proposed usage of the reactor to be submitted on an "Irradiation Request Form" that must be reviewed and approved by the Reactor Supervisor and a second person qualified in health physics. This SOP requires a detailed hazard analysis of the proposed experiment including the effects of a sample failure. If it is a significant new type of irradiation or experiment, it also must be reviewed and approved by the Reactor Safeguards Committee.

10.3.2 Safeguards Committee Reviews

The responsibilities of the Reactor Safeguards Committee (RSC) are listed below and include a review of all experiments and changes done under 10 CFR 50.59.

A) Safety Review

The safety review responsibility of the RSC or designated Subcommittee thereof shall include, but is not limited to, the following:

1) Review and approval of all new experiments utilizing the reactor facilities,
2) Review and approval of all proposed changes to the facility, to the facility license by amendment, and to the Technical Specifications,
3) Review of the operation and operational records of the facility,
4) Review of unusual or abnormal occurrences encountered in the course of facility operations, and review of incidents which are reportable under Title 10, Code of Federal Regulations, Parts 20 and 50,
5) Review of abnormal performance of facility equipment and operating anomalies, and
6) Biennial review of all standard procedures, the facility emergency plan, and the facility security plan.

B) 10 CFR 50.59 Reviews

1) Review and approval of all determinations of whether a proposed change, test, or experiment would constitute a change in the Technical Specifications or an unreviewed safety question as defined in 10 CFR Part 50.

C) Audit

The audit responsibility of the RSC or designated Subcommittee thereof shall be audit of reactor operations semiannually, with intervals not to exceed eight months. The audit shall be conducted by voting committee members and shall include at least the following:

1) Review of the reactor operating records,
2) Inspection of the reactor operating areas and Room 101,
3) Review of unusual or abnormal occurrences, and
4) Review of new standard procedures or changes in existing standard procedures.

10.3.3 50.59 Reviews

An Administrative Procedure entitled "Standard Procedures for the Approval and Review of Facility Modifications and Special Tests or Experiments" covers the 50.59 review procedure in detail. A flow sheet used in the 50.59 review procedure is given below:

GUIDELINES FROM 50.59

Evaluation of Actions Involving Changes, Tests or Experiments

Does the action involve: FORM #7

1) A change in the Technical Specifications as incorporated in the license.

OR

2) A reduction in the margin of safety as defined in the basis for any technical specification.

OR

3) An increased probability of the <u>occurrence</u> of an accident analyzed in the Safety Analysis Report.

OR

4) An increase in the <u>consequences</u> of an accident analyzed in the Safety Analysis Report.

OR

5) The possibility of an accident or malfunction of a different type than any analyzed in the Safety Analysis Report.

YES →

FORM #9, 10, 11

Director and RSC

↓

Must file request for authorization of action or of change in technical specifications with NRC Division of Reactor Licensing. Request shall include an appropriate safety analysis report.

NO ↓

Does the action involve: FORM #7

1) A change in the facility as described in the Safety Analysis Report.

OR

2) A change in procedures as described in the Safety Analysis Report.

OR

3) A test or experiment <u>not</u> described in the Safety Analysis Report.

YES →

FORM #9, 10, 11

Director and RSC approval.

↓

Action may be performed. A record of the action including a written safety evaluation providing the bases for determination that the action does not involve an unreviewed safety question shall be made and placed in the facility files. A brief description of the action and a summary of the safety evaluation shall be submitted to the NRC with the Annual Report.

NO ↓

FORM #9

Director Approval

↓

Action may be performed and a record of the action shall be made and placed in the facility files.

→

Quarterly RSC audit review.

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 N401-1975/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.

Appendix 10-A
SPECIFIC LIMITATIONS ON THE USE OF THE REACTOR

The following license limits apply to all experiments and irradiations performed using the reactor. Any experiment or irradiation which presents a significant possibility of exceeding any of these limitations will not be performed.

1. The steady state power level of the reactor during an experiment or irradiation shall not exceed 1 MW.
2. The maximum power level to start a pulse shall not exceed 1 kW.
3. The reactor fuel temperature shall not exceed 500°C.
4. The worth of an individual pulse shall not exceed $2.50.
5. The reactivity worth of any individual experiment shall not exceed $2.00.
6. The total reactivity worth of all experiments and irradiations installed in the reactor at any one time shall not exceed $5.00. This includes potential reactivity which might result from malfunction, flooding or voiding, or removal or insertion of the experiment or irradiation.
7. The reactor shutdown margin including all experiments and irradiations and with the most reactive and regulating control elements fully withdrawn shall be $0.25 or greater.
8. The reactor pool level shall not fall below 15 feet above the top of the reactor core.
9. The core shall consist of standard fuel, FLIP fuel, or LEU fuel or a combination thereof provided that the FLIP fuel region contains at least 22 FLIP fuel rods or the LEU fuel region contains at least 34 LEU rods in a contiguous block in the central region of the core. Water holes in the FLIP region shall be limited to single rod holes. Vacant lattice positions in the standard fuel region shall be occupied with fixtures which will prevent installation of a fuel bundle.
10. Any experiment or irradiation having a reactivity worth greater than $1.00 shall not be moved during reactor operation and shall be securely fastened or located to prevent inadvertent movement.
11. The maximum quantity of explosive material irradiated at any one time shall not exceed 25 milligrams. Explosive materials must be encapsulated in a container that will not burst if the material were to detonate.
12. Materials which could off-gas, sublime, volatilize or produce aerosols under a) normal operating conditions, b) credible accident conditions, or c) 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 pool room or atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the applicable limits of Appendix B of 10 CFR 20.
13. Each fueled experiment or irradiation shall be controlled such that the total inventory of Iodine isotopes 131 through 135 in the experiment or irradiation is not greater than 1.5 Curies.

14. The maximum quantity of ^{235}U or other enriched fissionable material irradiated at any one time shall not exceed 8 milligrams. Fueled samples must be encapsulated in high purity quartz or other non-permeable material.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.0 Management Policy Statement

The Washington State University Nuclear Radiation Center will strive to maintain a safe and healthful work place and a clean environment. No activity or operation will be done at the facility unless it can be performed in a manner that protects employees, the public and the environment. Accomplishing these goals requires a team effort on the part of the facility staff, faculty, students and visitors.

The facility shall have a formal Radiation Protection Program that meets the requirements of ANSI/ANS 15.11 (1) and Part 20 Title 10 of the Code of Federal Regulations.(2)

11.1 Radiation Protection

The basic aspects of radiation protection at the facility are as follows:

a. Exposure Limits

The radiation exposure limits at the WSU TRIGA reactor facility shall not exceed the limits specified in the Code of Federal Regulations Title 10, Part 20 entitled "Standard for Protection Against Radiation." The important exposure limits are listed below:

1. Occupational Dose Limits - (10 CFR 20.1201)
   (a) Whole body total summation - 5 rems/year
   (b) Extremities excluding lens of eye - 50 rems/year
   (c) Lens of the eye - 15 rems/year
   (d) Skin (shallow dose) - 50 rems/year
   (e) Airborne exposure not to exceed limits of table 1, appendix B.
2. General Public Dose Limits - (10 CFR 20.1301/20.1302)
   (a) Total individual public limit - 0.1 rem/year
   (b) Unrestricted area maximum dose rate - .002 rem/hour
   (c) Unrestricted area dose limit - 0.05 rem/year
   (d) Radioactive material releases - do not exceed limits of table 2, appendix B.
3. Minor and Pregnant Women Dose Limits - (10 CFR 20.1502)
   (a) Minors and declared pregnant women - 10% of 20.1201 limits
   (b) Embryo/fetus - .5 rems

b. Surveys and Monitoring

1. A system of procedures shall be in place for routine surveys and monitoring of the radiation levels in the facility during reactor operations.
2. A permanent record system shall be maintained of all survey and monitoring activities.

c. Personnel Dosimetry

1. All personnel at the facility shall be assigned a personnel monitoring device that shall be worn at all times while at the facility.
2. The responsibility for processing personnel dosimetry devices data and maintaining a permanent record of all exposure shall be the responsibility of the Campus Radiation Safety Office.
3. Personnel exposure data shall be processed on a monthly basis.

11.1.1 Radiation Sources

a. General

The primary sources of radiation at the facility are the fission products built up in the reactor fuel rods when the reactor is operated. A typical four rod modified TRIGA fuel rod assembly if removed from the reactor has a dose rate ranging from about 100 to 1000 rems/hr at three feet from the fuel assembly in air. The dose rate is dependent upon the time since the fuel was removed from the operating reactor and the length of such reactor operations.

The most significant radiation exposure that could occur in relationship to irradiated fuel is described in detail in section 13.1.1 of this SAR under the Maximum Hypothetical Accident. The MHA for a TRIGA reactor is defined as the rupture of the fuel cladding of one TRIGA fuel rod in air and the release of the contained fission products into the pool room air. The most realistic estimate of the effect of the MHA are given in Tables 13-5 and 13-10. Table 13-5 indicates that the most realistic pool room exposures would be 376 mr/hr whole body and 5 rems/hr thyroid for a 1 hour exposure. In actual practice the exposure would be much less due to the limited stay time in the pool room in the event of a MHA.

Outside the facility, the most realistic MHA exposures would be a maximum immediately following the release decaying with time. For this case Table 13-6 indicates peak values are 1.3 mr/hr whole body and 17.4 mr/hr thyroid. These values are not very significant in relationship to the probability of such an accident occurring.

Other secondary sources of radiation are: 1) the startup source, 2) cobalt-60 exposure facility in the pool, 3) spent demineralized resin, 4) samples irradiated in the reactor for neutron activation analysis and radioisotope production, 5) nitrogen-16, and 6) argon-41.

b. Startup Source

The startup source is composed of a mixture of antimony and beryllium but is never removed from the pool so as to cause a radiation exposure hazard.

c. Cobalt-60 Exposure Facility

The east end of the pool contains a cobalt-60 exposure facility that contains 7000 curies of cobalt-60. Special conditions placed on this source are given in section 3.8.4 of the Technical Specifications. The primary limits are (1) sources must not be closer than 5 feet from an operating core, (2) total cobalt-60 activity is not to exceed 10,000 curies, and (3) the pool water must be monitored to detect a sealed source leak. The cobalt-60 exposure facility is designed to irradiate samples in an air filled tube under water so personnel exposure is minimal. Also, the sources do not move to the exposure position until a shield over the top of the exposure tube is in place.

d. Spent Demineralizer Resin

Spent demineralizer resin is not a significant radiation source and waste disposal is the primary consideration discussed in section 11.2 below.

e. Irradiated Samples

The potential for personnel radiation exposure from the radioactivity induced into samples for neutron activation analysis and isotope production represents the most significant radiation sources that are present at the facility. Very complete and detailed procedures are in place for the irradiation of samples including the estimation of the activity that will be produced, the exposure rate from the irradiated sample, and the monitoring required when the sample is removed from the reactor. Additional information on the control of radiation exposure from irradiated samples is covered in 11.1.4 and 11.1.5 of this section.

f. Nitrogen-16

Nitrogen-16 is produced in the cooling water of the reactor by the oxygen-16(n,p) nitrogen-16 reaction. The nitrogen-16 radionuclide has a half-life of 7.13 seconds and predominantly emits a 6.13 MeV gamma ray. The concentration of oxygen-16 at the surface of the pool has been calculated to be (8) about 0.1 μCi/μL or less which produces a bridge dose rate of 40 to 100 mrem/hr at a typical TRIGA reactor. At the WSU reactor the bridge dose rate from oxygen-16 with the diffuser off is about 50 mrem/hr which drops to 2 mrem/hr with the diffuser on. The diffuser directs a small flow of water downwards and across the top of the core area which significantly slows down the upward flow of heated water containing nitrogen-16 thereby reducing the bridge dose rate to a manageable level.

g. Argon-41

Argon-41 is induced in the air that flows through the reactor thermal column and is also evolved from the pool water surface. The core is cooled by natural convection circulation that causes the heated water to rise to the surface of the pool along with the air dissolved in that water. Some of the air containing Argon-41 escapes into the pool room air at the surface of the pool. The pool room exhaust air and thermal column air are combined and vented to the atmosphere after being monitored for the Argon-41 content by the system described in SAR section 7.7.3

Section 13.1.1(F) of this SAR substantiates a 4×10^{-3} dilution factor for Argon-41 released from the facility due to the atmospheric wake effect in the lee of the building. A more thorough analysis of the distribution of Argon-41 in the atmosphere about the site may be obtained by the use of equation F-1 of Appendix F of Regulatory Guide 1.109, "Calculation Reactor Effluents for the Purposes of Compliance with 10 CFR Part 50, Appendix I."

The annual release of Argon-41 from the WSU facility for the past ten years has averaged 10 Ci/year. Thus, the daily release is .027 Ci/day which is equivalent to a release rate of 3.17×10^{-7} Ci/sec. However, for the purposes of our calculations, we shall assume a 100 Ci/year release and a 3.17×10^{-6} Ci/sec release rate. Using a 100 Ci/year total release, the wind distribution data given in section 2.3.2 of this SAR and Equation (F-1), the Argon-41 concentration in the atmosphere about the site may be calculated. The results are shown in Figure 11-1 in terms of the percentage of the 10 CFR 20 Appendix B Table 2, Column 1 limit of 1×10^{-8} μCi/cm^3. The ground level Argon-41 concentration levels about the site for a 1000% normal release rate are significantly below the 10 CFR 20 limit as well as the ALARA criteria of 2% of the 10 CFR 20 limit. The closest occupied location to the site is 411 meters west and thus would be exposed to a 2.2×10^{-11} μCi/cm^3 annual average ^{41}Ar concentration for the postulated release. Accordingly, the Argon-41 released to the atmosphere by the operation of the WSU TRIGA reactor does not produce a significant radiation exposure hazard.

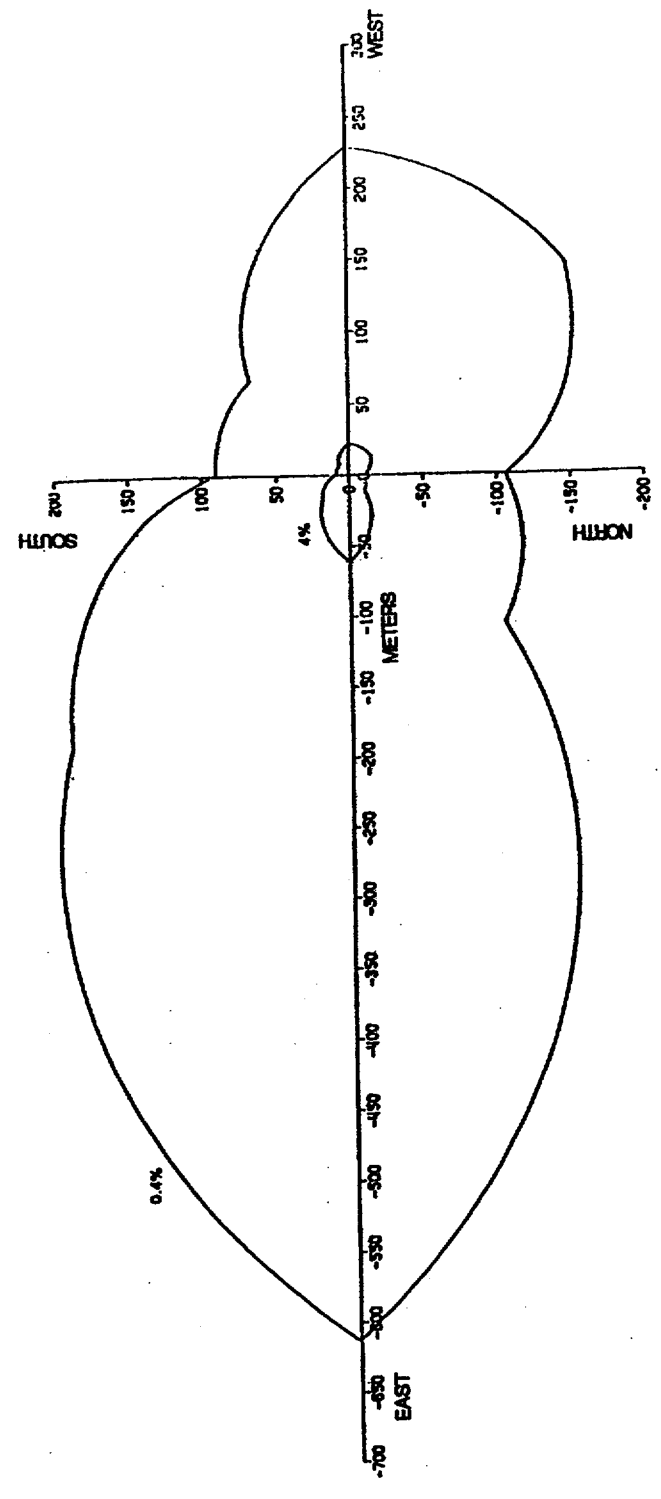

Figure 11-1
Annual Average Argon-41 Concentration Distribution in Atmosphere about Site in % of 10-CFR-20 Limits, Assuming a 100 Curie/Year Total Release

h. Historic Maximum Exposure Data

The facility has been in operation for over 20 years without a single incident of a significant personal radiation exposure. The reactor staff individuals have the highest annual radiation exposure values since they move irradiated fuel elements, insert and withdraw irradiated samples from the core, pack irradiated samples for shipment, and handle any radioactive waste. The typical staff exposure amounts to about 300 mrem/year. Other people at the facility receive a much smaller annual exposure including people who are involved with the Neutron Activation Analysis Program.

11.1.2 Radiation Protection Program

11.1.2.1 Radiation Control Administration

The State of Washington is an "Agreement State" in terms of the federal regulations regarding control of radioactive materials and of all radiation producing machines with the exception of nuclear reactors. The federal government thus has delegated the responsibility for the control of non-reactor related radiation control within the State to the State Government. The State of Washington in assuming this responsibility has promulgated appropriate control regulations entitled, "Rules and Regulations--Radiation Protection," which are contained in Chapter 402, Title 10, of the Washington Administrative Code.(3) The University, under the provisions of WAC 402-22-090, has a Type A broad scope radioactive materials license, #WN-C003-1.

The ultimate responsibility for all activities at Washington State University is vested in the President of the University. The responsibility and authority vested in the President related to the use and control of nuclear radiation, radioactive materials, and radiation producing machines is delegated to the Vice Provost for Research. The Vice Provost for Research, in turn, delegates this responsibility to the Radiation Safety Officer and Radiation Safety Committee for all non-reactor related radioactive materials and associated radiation exposure. In the case of the reactor related area, the responsibility is delegated to the Director of the Nuclear Radiation Center and to the Reactor Safeguards Committee.

The State regulations regarding Type A broad scope license requires the establishment of appropriate administrative controls including:

a) a radiation safety committee with purview over all aspects of non-reactor related radiation control, and
b) a radiation safety officer to supervise the acquisition, use, and disposal of radioactive materials under State jurisdiction. At Washington State University, the Radiation Safety Committee and Radiation Safety Officer report to the Vice Provost for Research, as depicted in Figure 11.2. The responsibilities of the Radiation Safety Committee are described in Section 11.B of the WSU Radiation Protection Program Manual.

Administratively, the Radiation Safety Office functions as the control agency at Washington State University for all non-reactor related radiation control. This includes all aspects of the procurement, distribution, use, and disposal of radioactive materials as well as the control and usage of radiation producing devices and machines. The Radiation Safety Committee functions as the policy making and review body for all aspects of non-reactor related radiation control including safety standards, facility standards, and training requirements as well as the use and disposition aspects of radioactive materials.

The TRIGA Reactor Facility at the Nuclear Radiation Center is operated under the requirements of federal regulations and Facility License R-76. The operation of the reactor and associated radioactive materials and radiation control in the reactor facility are governed by federal regulations and the provisions of Facility License R-76. Facility License R-76 provides for a Reactor Safeguards Committee which oversees the operation of the reactor facility as shown in the block diagram given in Figure 11-2. The day-to-day operation of the facility is under the direction and control of the reactor operating staff which is under the administrative control of the Director of the Nuclear Radiation Center.

Block Diagram of Radiation Control Administration at Washington State University

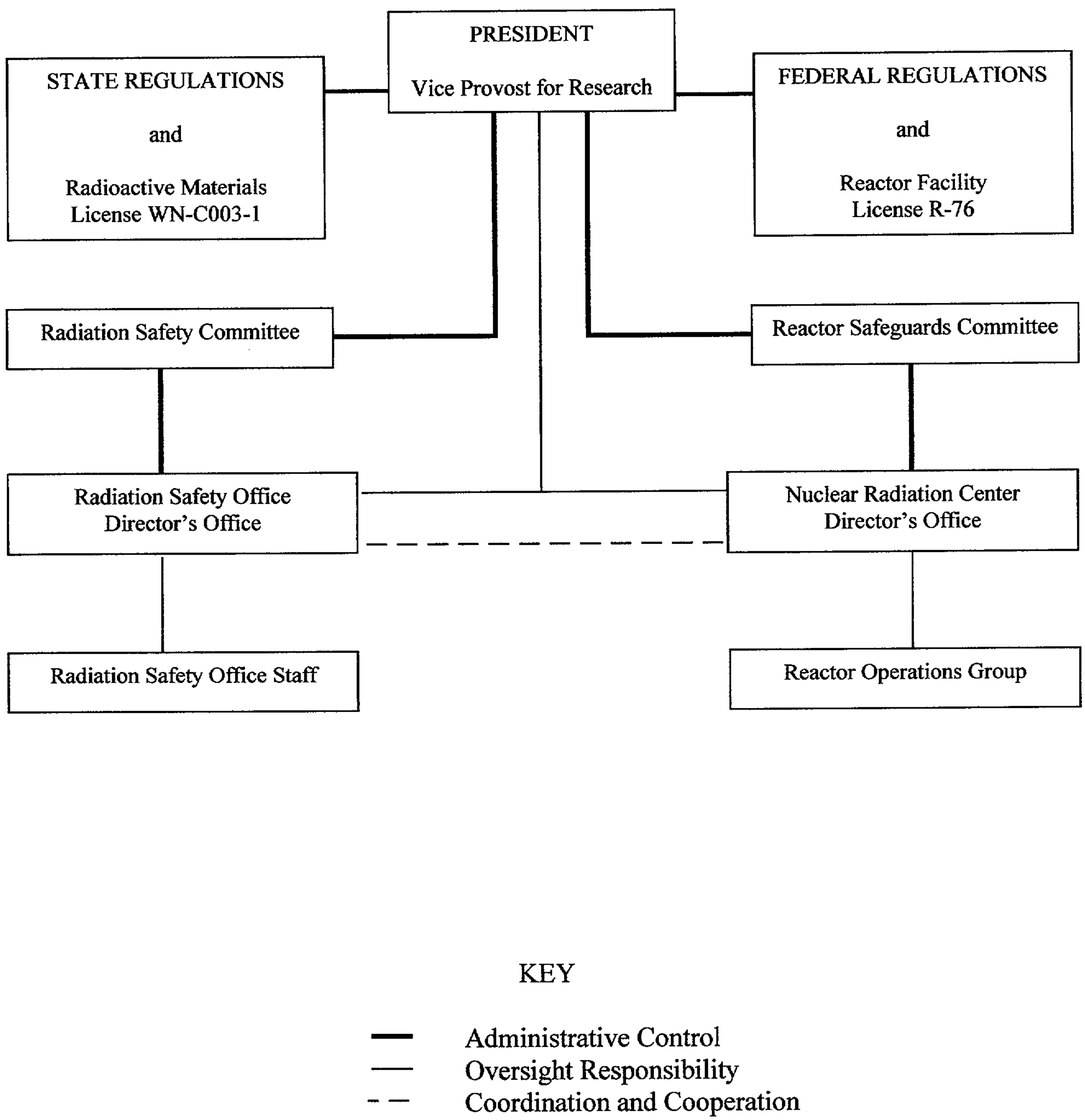

Figure 11.2

11.1.2.2 Program Basics

The basic aspects of the Radiation Protection Program at the WSU reactor facility are given in section 11.1 above.

11.1.2.3 Program Components

The Radiation Protection Program at the WSU TRIGA reactor is not a stand-alone program but is embedded into all the activities that occur at the facility. That is, radiation exposure is a vital consideration in every activity that is done at the facility. Listed below are the various components of activities that relate to radiation protection:

(a) Management commitment and worker responsibility
   1) Staff supervisors and faculty supervisors have the responsibility to
      (a) ensure that radiation doses to workers, visitors, and the public are kept ALARA;
      (b) identify radiation workers and ensure that they receive appropriate training.
   2) Every worker at the facility has a responsibility to
      (a) obey posted signs and instructions;
      (b) follow standard operation procedures (SOPs);
      (c) incorporate ALARA principles and apply concepts of time, distance, and shielding;
      (d) minimize the production of radioactive and mixed wastes;
      (e) receive training for your job assignments;
      (f) inform your supervisor of radiation hazards or potential problems;
      (g) stop work on any activity that poses danger to the health and safety of your co-workers or the public, or danger to the environment; and
      (h) ask questions when in doubt.

(b) Qualifications of personnel and adequacy of resources

The qualification requirements are addressed by administrative procedure 1 which outlines qualifications of the reactor operating staff. The Director of the facility has the responsibility to request adequate resources and the University Vice Provost for Research has the ultimate financial responsibility for providing the facility with adequate resources for the needed personnel and equipment.

(c) Adequacy of authority for responsible persons

Administrative Procedure 1 describes the authority of the various positions of the reactor operating staff.

(d) New staff training and continuing education for all personnel

New employees whose job requires them to use the facility on a regular basis are required to be badged by the Radiation Safety Office (RSO). Only those who successfully complete the WSU Radiation Safety Course are issued permanent TLD badges. New staff members who would like to become reactor operators are trained by the senior reactor operators. Continuing education is an integral part of the "Requalification Program."

(e) Radiological design is an integral aspect of facility and experiment design

SOP's 1-3 describe the authorizations required to do irradiations and experiments at the facility. These SOP's require all irradiations and experiments to have a radiological evaluation and review.

(f) Radiological planning as an integral aspect of operations planning

All operations are covered by SOP's that take into consideration radiation exposure.

(g) Performance reviews of designs and operations (lessons learned)

The Reactor Safeguards Committee audits the Reactor Operational records quarterly including all radiations and experiments.

(h) Analysis of personnel exposure patterns

The reactor supervisor is responsible for a quarterly evaluation and analysis of radiation badge exposure results.

(i) Periodic assessment and trend analysis of the radiological environment

The reactor supervisor records and summarizes the environmental radiation exposure data monthly and annually for each staff member. Any significant trends or anomalies would be readily recognized by the Reactor Supervisor.

(j) Periodic assessment and audits of the protection program

There is a quarterly audit of the records by the reactor safeguards committee and an annual review of the radiation protection program as specified in Technical Specifications section 6.5.5(4).

(k) Surveillance activities

1) Personnel exposure (dosimetry)

The personnel dosimetry is administered by the Radiation Safety Office which monitors the badges so as to conform to State and Federal laws, especially 10 CFR 20.

2) Radiation and contamination surveys

SOP #10, for "Health Physics Surveys" covers this section.

3) Environmental monitoring

SOP #21, for "Environmental Monitoring"

SOP #22, for "TLD Environmental Monitoring"

4) Effluent monitoring

Liquid Effluent: SOP #21, for "Environmental Monitoring" section C

Gaseous Effluent: SOP #21, for "Environmental Monitoring" section D

5) Warning and active protection systems functionality

SOP #4, for "Startup, Operation, and Shutdown of the Reactor"

Many of the functionality checks are done during the Reactor Start-up Checkoff, such as Area Radiation Monitor Operability Check

SOP #5, for "Performing Preventive Maintenance on the Reactor and Associated Equipment"

Pool Level Alarm Operability

Ventilation System Operability and Filter Check

Building Evacuation Alarm Operability Check
SOP #17, for "Checkout and Calibration of Area Radiation Monitors"
Operability, Alarm Set Point, Calibration Check
SOP #18, for "Calibration of Ar-41 Monitor"
Operability, Calibration, Set-point
SOP #26, for "Continuous Air Monitor Checkout and Calibration"
SOP #27, for "RM-14 Checkout and Calibration"

6) Operational limitation compliance

The first line of enforcement of compliance is the operator staff members who are trained to recognize and respond to abnormal situations. The Reactor Safeguards Committee reviews and audits all of the records associated with meeting our operational limits.

7) Engineered protective systems (shielding, ventilation, etc.)

AP#3*, "Approval and Review of Facility Modifications and Special Tests or Experiments" (10 CFR 50.59 changes)

8) Instrumentation

AP#3, "Approval and Review of Facility Modifications and Special Tests or Experiments" (10 CFR 50.59 changes)

9) Radioactive material accountability

AP#7, "Special Nuclear Materials Searches"
AP#9, "Special Nuclear Material Accountability Plan"

(l) Protective equipment (supply, Q.A.)

Included in monthly reactor maintenance item #12 is a radiac battery check and an inventory of emergency kits.

(m) Calibration and Q.A. programs

SOP #17, for "Checkout and Calibration of Area Radiation Monitors"
Operability, Alarm Set Point, Calibration Check
SOP #18, for "Calibration of Ar-41 Monitor"
Operability, Calibration, Set-point
SOP #23, for "Portable Survey Instrumentation Calibration"
SOP #26, for "Continuous Air Monitor Checkout and Calibration"
SOP #27, for "RM-14 Checkout and Calibration"

(n) Training

All reactor users are trained in the use of the reactor and must pass a written user certification exam. SOP #32, for "Security and Emergency Plan Training for Nuclear Radiation Center, Radiation Safety Office and Campus Police Personnel."

11.1.3 ALARA Program

ALARA policy statement: It is the policy of the facility administration to keep radiation exposures As Low as Reasonably Achievable (ALARA), which is also specified in facility Technical Specification section 3.12. The purpose of this policy is to reduce radiation exposure

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* Administrative Procedure

to as low a level that is socially, technically, and economically practical. Three general principles should be followed to reduce exposure:

1. **Minimize Time**. The less time spent near a radioactive source, the less the amount of exposure. One typical way this principle can be implemented is to carefully plan and practice a procedure before the actual implementation with radioactive sources.
2. **Maximize Distance**. When the distance between the body and the radioactive source is increased, the exposure decreases. Sometimes this can be effectively accomplished with tongs or other devices to hold a source away from the hands and body. Store and use radioactive materials far from locations used for other purposes.
3. **Use Proper Shielding**. When an appropriate shielding material is placed between the body and the radioactive source, the amount of radiation exposure is reduced. Use storage pigs and shielding blocks when possible.
4. **Control Contamination**. Minimize exposure by:
   - (a) Wearing lab coats, gloves, booties & safety glasses/goggles where appropriate
   - (b) Changing gloves frequently
   - (c) No eating, drinking, smoking, chewing, or application of cosmetics in radioactive work space
   - (d) No mouth pipetting
   - (e) Washing hands at completion of radioactive work
   - (f) Monitoring hands, clothes and work area regularly

11.1.4 Radiation Monitoring and Surveying

a. General

The facility shall have a comprehensive radiation monitoring and surveying program and explicit written procedures for carrying out the area and contamination monitoring program. The key components of the program must include the items listed below in this section.

In addition, the monitoring program conducted by the reactor staff, the Radiation Safety Office under the provisions of the campus-wide Laboratory Monitoring Program described in section IX of the WSU Radiation Protection Program Manual, monitor all areas of the Radiation Center. This monitoring program includes wipe tests for removable surface contamination as well as laboratory area radiation level monitoring.

b. Monitor Qualifications

Not just anyone should be allowed to perform radiation monitoring activities at the facility. AP #1 specifies that all facility monitoring shall be carried out by a licensed Reactor Operator who by virtue of his or her training and license is qualified to perform such activities.

c. Routine Surveys

SOP #10 entitled "Standard Procedure for Health Physics Surveys" covers the procedures and record keeping aspects of radiation surveys at the WSU facility.

d. Excessive Dose or Contamination Levels

SOP #10 explicitly lists the dose and removable contamination levels that are considered to be excessive and the action to be taken in the event that the levels are exceeded.

e. Equipment

The monitoring program shall include information on the existing equipment available and what is the appropriate equipment to be used for each type of activity.

11.1.5 Radiation Exposure Control and Dosimetry

a. Safe Uses of Sources of Ionizing Radiation

Since the guiding philosophy of the radiation safety program is characterized by ALARA, it is obvious that all sources of ionizing radiation must be used as safely as possible. In this section the focus will be requirements for safe procedures and practices in using radioactive materials, with peripheral attention devoted to various other hazardous materials encountered in laboratories.

b. Radioactive Materials

The rules for working with radioisotopes and radiation fields in a safe manner are governed by good judgment and common sense for safe laboratory practices and by a thorough knowledge of the nature of the experiment and the equipment being used. Experiments require careful planning from the first to last step. It is inevitable that certain steps in the experimental procedure, more accident-prone that other steps, result in spillage and spread of radioactive material. These problems must be anticipated in designing the experiments. Some "excessive" caution is necessary in dealing with radioisotopes. A set of guidelines is tabulated in the next subsection in order to minimize external radiation exposure, to minimize internal radiation exposure by avoiding ingestion, inhalation, and absorption of radioactive material, and to prevent the spread of contamination in the event of a spill or other accident.

c. General Laboratory Requirements

The laboratory requirements for the safe utilization of radioactive materials are not fundamentally different from those for use of other potentially hazardous materials. A properly designed laboratory gives due consideration to the movement of personnel and materials, the comfort and convenience of personnel, the required utilities, waste disposal, illumination, fire prevention and security, as well as to the minimization of potential hazards and to minimizing the probability of the creation of hazardous working conditions. The potential hazards that are unique to the utilization of radioactive materials in a laboratory are those of external radiation exposure, internal radiation exposure, and the spreading of radioactive contamination to other areas. Thus a radioisotope laboratory must, in addition to the usual safety considerations, provide due consideration for adequate shielding against external radiation, containment of volatile radioactive materials, minimization of contamination, and provision for ease of decontamination. The Radiation Safety Committee, in granting authorization to use radioactive materials, will consider the laboratory facilities in relation to the proposed use. The specific laboratory requirements are dependent upon the type of experiment, quantity of radioactive material to be used, and the hazard rating of the radionuclides being used. It is difficult to establish precise laboratory requirements for the wide varieties of utilization that occur at a university. However, radionuclides are classified into hazard groups, and a laboratory classification scheme for purposes of monitoring is predicated on these hazard groupings. Under this program the Radiation Safety Office monitors the Nuclear Radiation Center Laboratories as a cross check to the monitoring done by the reactor staff.

d. Basic Laboratory Practices

It is essential that all personnel who work with radioactive materials become familiar with the radiation protection program at WSU as set forth in this manual. (A radioactive materials laboratory is defined in Section XIV.) The guidelines for proper procedures as well as requirements in handling radioactive materials are summarized in this section. It is essential to point out that these guidelines pertain to all use of radioactive materials. It is expected that the individual will use the utmost care always to ensure safe use of radioisotopes in order to avoid endangering his or her colleagues in the laboratories. All experimenters must

1) Wear personal dosimeters (e.g., film badge, ring badge, or pocket dosimeter).
2) Wear protective clothing, such as lab coats, full-length slacks, overshoes, and safety glasses or goggles.
3) Protect the hands by wearing plastic gloves. (Consider the outer part of the gloves to be contaminated and limit the use of gloves to the immediate experimental area. Do not use gloves in the "inactive" regions of the laboratory, where it is normally allowed to use bare hands [e.g., doorknobs, light switches, fume hood doors, and telephones]).
4) Prohibit drinking, eating, smoking, and application of cosmetics in a radioactive materials laboratory. (Even if parts of the laboratory are "inactive," it is necessary to depart from the laboratory for drinking, eating, smoking, or application of cosmetics.) The presence of empty food or drink containers will be considered a violation of these regulations, since it will be inferred that consumption occurred on the premises. Food or drink may be transported (expeditiously) through a radioactive materials laboratory only if in a completely closed container.
5) Prohibit pipetting radioactive solutions using mouth, licking gummed labels, or combing hair in radioactive materials laboratory.
6) Monitor hands, feet, clothing, and shoes, before leaving the laboratory.
7) Use suitable monitoring equipment such as portable survey meters in laboratories. (These instruments give exposure rate to radiation in mR/hr, or the observed rate of decay of radioisotopes in counts/min.)
8) Survey the laboratory area before commencing an experiment using radioisotopes. (This precaution will ensure that the laboratory is uncontaminated when starting the work. Allocate a smaller portion of the surveyed area for experimental work. In case of an accident, it will be relatively easy to contain the radioactivity and to decontaminate that area.)
9) When using volatile materials, always work in fume hoods. (For extremely high activity levels, a glove box is preferred. Inasmuch as feasible, avoid open bench top experiments.)
10) Ensure that the fume hood is in satisfactory condition (e.g., strippable or washable paint on exposed area, proper air-flow, and unclogged drains). Physical Plant personnel are required to conduct an annual inspection of each fume hood to ensure proper operational characteristics. (It is preferable to use glove boxes with pressure inside the box slightly less than atmospheric pressure.)

11) Use a large porcelain or stainless steel tray lined with absorbent paper and carry out the experiments on top of this tray. (In case of an accident it is an easy matter to decontaminate the tray.)

12) Line adjacent porous surfaces with absorbent paper or equivalent material.

13) Store and transport radioactive material in closed containers. (Do not transport open containers from one part of the laboratory to another.)

14) Label all containers of radioactivity properly with date, radioisotope, quantity of radioactivity, and your name. (Regulations require that each container be clearly marked as to its contents.)

15) Use radiation shields if measured radiation levels at the body will result in a dose equivalent in excess of about 20 mrem (0.20 mSv). (Remember that the maximum permitted radiation exposure is 100 mrem/week (1.0mSv/week). In shielding samples, do not forget that the back or sides of the hood may face an adjacent laboratory; it will be necessary to consider exposure to this area as well.)

16) Survey the work area. Decontaminate the work area as necessary and clean up all equipment immediately after use. (Check the area with survey equipment to ensure the adequacy of the cleanup. Consult with the Radiation Safety Office if you are not able successfully to clean up the area.)

17) Properly post notices to designate areas containing radioactive materials. (Areas where radiation exposure rates would result in a dose equivalent in excess of 5 mrem (0.05 mSv) in one hour should be posted as a Radiation Area and those in excess of 0.1 rem (1 mSv) in one hour as a High Radiation Area. Signs are available for these designations. Remove all signs or markings when the hazard is removed.)

18) Rope off radiation areas and contaminated areas to restrict access and post signs to indicate the hazard. (The barriers should not be removed without prior consultation with the Radiation Safety Office.)

19) Report all accidents promptly to the Radiation Safety Office on the Radioactive Materials Incident and/or Accident Report (SPPM S.90.55.4). (Accidents can occur in the best-planned experiments.)

e. Security and Control

1) Stored radioactive materials must be secured from, or controlled in such a manner as to prevent, unauthorized removal from the place of storage.

2) Radioactive materials which are neither in storage nor in an unrestricted area must be tended under the constant surveillance and immediate control of the authorized user.

f. Safe Handling

The basic approach to safe handling of radioactive materials is to focus on avoidance of spills, escapes, or other avenues to contamination by the material being handled. Thus the container must be suitable for the material contained, both from the integrity standpoint and the shielding standpoint. In all types of handling, which usually involves a change in position or location of the radioactive materials, one must never allow his or her attention to wander from the procedure at hand. Moreover, because of the nature of radioactive materials and the attendant dangers of exposure or contamination, extra precautions for safe handling must be adopted. For example, in transporting radioactive materials from one laboratory space to another, even in the same building and on the same floor, the mode of transport must include (at least) double containment, so that there is a second barrier to dispersion should the first barrier fail.

11.1.6 Contamination Control

The rules for contamination control considerably overlap those for exposure control. Listed below are the basic components of contamination control.

a) Wear fully protective clothing, including gloves, a laboratory coat, wrist guards, full-length slacks, shoes (preferably overshoes) that cover the feet and possibly the ankles, and safety glasses or goggles.
b) Designate a specific area for work with radioactive materials.
c) Label all containers and tools properly.
d) Use trays and absorbent papers.
e) Prohibit smoking, drinking, eating, or application of cosmetics in the radioactive materials laboratory.
f) Change gloves frequently so as to avoid contaminating various laboratory articles, fixtures, and surfaces.
g) Use transfer pipettes and prohibit any mouth-pipetting.
h) Work with volatile compounds only in operational fume hoods.
i) Use traps to absorb volatiles. (Guidance on disposal of chemical traps should be obtained from the RSO.)
j) Provide for regular monitoring of clothing, shoes, and the work area.
k) Avoid all interruptions and distractions once the procedure has been commenced, and especially those which might cause contamination of laboratory articles or furniture (e.g., telephone calls).

At the Nuclear Radiation Center the main method of contamination control is embodied in the procedures used to irradiate samples and to handle irradiated samples. All samples are irradiated in a plastic container of some type and lowered down into a water containing irradiation tube of the reactor by one of the reactor staff or a certified experimenter. At the end of the irradiation the sample is pulled up in the irradiation tube and out of the core. Depending upon the expected dose rate from the sample, it may be immediately withdrawn from the core to the bridge level and the sample's exposure rate immediately measured or allowed to cool for a while before being removed from the top of the irradiation tube.

Samples removed from the core are passed down to a hood in the Radiochemistry Laboratory which is on the floor below the reactor operating area via a 4 inch plastic tube that leads to the hood. Documentation on the sample irradiation is then made out by the person removing the sample including the dose rate of the sample. Samples with a dose rate of over 100 mr/hr are not allowed to be transferred to the Radiochemistry Laboratory.

In the Radiochemistry Laboratory a person wearing plastic gloves processes the sample as needed and again monitors the sample or samples after they have been removed from the irradiation container. Only low level samples for Neutron Activation Analysis or irradiated samples in a shield are allowed to be removed from the Radiochemistry Laboratory. The Radiochemistry Laboratory has an exit monitor which will alarm if someone attempts to take a very hot sample through the door. Monitors are also located at the entrance of the office area and pool room door which will alarm if a hot sample passes through one of those doorways.

11.1.7 Environmental Monitoring

a. General

The facility has maintained a comprehensive environmental and facility monitoring program for over 15 years. This program has been very effective in quantifying the fact that the operation of the facility has had an insignificant impact on local environmental radiation levels and radiation exposure in and about the facility. The components of the program are listed below as described in SOP #21 that covers environmental monitoring.

b. Environmental Radiation Exposure Monitoring

The average quarterly radiation exposure in μRem in and about the facility shall be monitored using high sensitivity TLD type dosimeters. The monitoring locations shall include the unrestricted areas adjacent to the facility, the closest off-site point of continuous occupancy, and a number of off-site locations. The monitoring program shall include at least 20 sampling locations and the dosimeters shall be changed on a routine basis as specified in Standard Operating Procedure No. 22.

The exposure data shall be analyzed at least semiannually in order to insure compliance with 10 CFR 20.1301. If the exposure rate in an unrestricted area adjacent to the facility is found to be above the level specified in SOP #21 or greater above background, action should be taken to determine the cause of the exposure in the unrestricted area and to reduce the exposure to the area.

Annually the exposure to the closest off-site point of continuous occupancy shall be analyzed to insure compliance with the established ALARA criteria found in the facility's Technical Specifications, Section 3.12 (2).

c. Liquid Effluent Monitoring

All radioactive liquids discharged to the environment (sanitary sewer) shall be monitored before release. The total beta-gamma activity of the retention tank shall be monitored before release using the procedure specified in Standard Operating Procedure No. 11. The maximum activity of the liquids for direct discharge to the sewer system before dilution shall be 2×10^{-8} μCi/ml. For activities greater than this, further analysis is necessary as per SOP #11 to identify the isotopes. Discharge limits for the isotopes identified will be in accordance with 10 CFR 20 criteria for liquid effluents. Total annual activity of liquid effluent released shall not exceed 1 Curie per year.

d. Gaseous Effluent

The Argon-41 content of the reactor pool room exhaust shall be continuously monitored with the Argon-41 Exhaust Gas Monitoring System. Calibration of the system is covered in Standard Operating Procedure No. 18. Total annual discharge of Argon-41 into the environment shall not exceed 20 Curies per year in accordance with current ALARA criteria.

e. Pool Water Analysis

The specific radionuclide content of the reactor pool water shall be measured on a monthly basis using a 500 ml sample and a detector system as specified in Standard Operating Procedure No. 24.

f. Facility Exposure Rates

In accordance with the requirements of SOP #21, the exposure rate in each laboratory and in locations where individuals spend a significant number of hours per week shall be monitored using a permanently assigned film badge which shall be changed on a monthly basis.

11.2 Radioactive Waste Management

Radioactive waste and the management thereof is not a significant activity at the facility. The total volume of waste disposed of in a typical year is less than 50 cubic feet consisting mainly of spent demineralizer resin and old decayed neutron activation samples. SOP #25 covers the disposal of spent radioactive demineralizer resin. Radioactive waste for disposal is turned over to the campus Radiation Safety Office which manages Radioactive waste disposal for the entire campus.

11.2.1 Radioactive Waste Management Program

a. General

All radioactive waste or waste materials contaminated with radioactive materials may be disposed of only in accordance with the practices and procedures established by the Radiation Safety Committee and enforced by the Radiation Safety Office. The specific procedures, which may change with time, will include provision for handling the radioactive wastes as described below. All disposal (and use) will be conducted in a manner consistent with environmental monitoring requirements that are met by the Radiation Safety Office.

b. Routine Waste Collection

The Radiation Safety Office will routinely collect properly packaged and tagged solid and liquid materials from laboratories on campus including the Reactor Facility. It is the user's responsibility properly to tag the waste container and accurately to estimate the specific radionuclide content in the waste.

c. Radioactive waste storage

The Radiation Safety Office will accumulate and store properly tagged waste materials in an appropriate location prior to transfer to a permanent, licensed disposal site.

d. Disposal to Sanitary Sewer System

Authorized users on the Pullman Campus only may release to the sanitary sewer small quantities of non-alpha emitting radionuclides not to exceed the limits established in WAC 246-221-190 with the approval of the Radiation Safety Office. In order to ensure that total university releases do not exceed the appropriate limits, individual user release limits may not exceed those listed in the table below. (Hazardous [chemical] wastes with a radioactive component [officially labeled "Mixed wastes'] are subject to additional regulatory control. Proper disposition of these wastes must be determined by prior consultation with the Radiation Safety Office.)

User Release Limits

Radionuclide	Annual(mCi)	Monthly(μCi)	Daily(μCi)
H-3	2.77	231.0	10.5
C-14	0.08	6.9	0.3
P-32	0.56	46.8	2.4
S-35	1.56	130.0	5.9
Ca-45	2.08	173.3	7.9
Cr-51	0.31	26.0	1.2
I-125	1.56	130.0	5.9

Authorized users must keep an accurate record of the releases to the sanitary sewer by radionuclide and report them to the Radiation Safety Office on a monthly basis.

e. Release To The Atmosphere

Authorized users may release small quantities of radioactive gases, fumes and vapors to the atmosphere through hoods or directly in quantities not to exceed the limits established by WAC-246-221-070. Approval for such releases is required from the Radiation Safety Office and accurate records of the quantities and types of radionuclides released must be maintained by the user and communicated to the Radiation Safety Office on a monthly basis.

f. Radioactive Waste Incineration

Disposal of radioactive waste by incineration is governed by extensive and rigorous regulations of the State of Washington Departments of Health and Ecology. Incineration will be carried out at the WSU Incinerator only as planned and authorized by the Radiation Safety Office.

11.2.2 Radioactive Waste Packaging And Labeling

a. General

The Radiation Safety Office will collect and dispose of only packaged and labeled radioactive waste materials. The following paragraphs of this section outline the basic packaging and labeling requirements.

b. Dry Waste

Dry waste, such as paper, gloves, and plastics, should be placed in the standard Low Specific Activity box (LSA box) which has been lined with a plastic bag. No biological waste whatever is allowed in dry waste. Glass pipettes, broken glass, needles and any other sharp items should be placed in a strong inner package which is placed in the larger box. Damp material and other waste that will give off vapors or fumes should be contained in small, well-sealed plastic bags or containers before they are placed in the box. (Animal carcasses, blood and tissue, and larger amounts [10 grams] of waste that will putrefy should be frozen and disposed of according to animal waste procedures given below.)

c. Liquid Waste

Liquid waste that cannot be disposed of via the sewer system must be collected and stored in an appropriate container. Some liquid wastes may be incinerated and some must be absorbed on floor-dry and disposed of as solid waste. The Radiation Safety Office should be consulted for more specific assistance relating to liquid waste collection.

d. Liquid Scintillation Fluid Vials

The waste from liquid scintillation counting deserves special mention because of both the large volume and the fire and toxic-fume hazards from some of the solvents.

(1) When vials are reused: The spent cocktail should be emptied into a liquid waste container and treated as liquid (mixed) waste. Because of strong solvent fumes, this operation should be conducted in an operating fume hood.

(2) When vials are disposed of with contents: Liquid scintillation vials with their contents are generally disposed of by incineration. Vials should be collected in an "egg crate" carton or packed loose in a separate LSA box. Filled scintillation vials must not be mixed with dry waste. The Radiation Safety Office should be consulted for more specific assistance as required,

(3) When empty vials are disposed of: Empty vials should be placed in a dry waste box.

e. Animal Carcasses and Other Putrefiable Waste

Animal carcasses and other putrefiable waste should be sealed in two layers of plastic bags, labeled as radioactive, and stored in an appropriate freezer. The authorized user must coordinate directly with the RSO to schedule a waste pickup. Special arrangements should be made for large carcasses and large volumes of waste.

Certain carcasses and tissue waste may be contaminated with serious infectious and biohazardous micro-organisms. The normal handling practices with radioactive waste do not assure protection against biological contaminants. Special arrangements should be made to destroy the biological agents before the waste is transferred to Radiation Safety. In some instances, such infectious radioactive waste will be disposed of by incineration.

Carcasses and putrefiable waste must not be added to dry waste containers, nor can this type of waste be kept in the laboratory without freezing. These boxes are stored for several weeks before final disposal. Putrid material is extremely unpleasant. Chronic violators of these procedures may have "ripe" waste returned to them to be properly packaged. If not packed properly, putrid waste will not be collected.

f. Records and Labeling

The university license and state regulations require that inventory and control methods cover all aspects of work with radioactive material. Therefore, all packages and containers of radioactive waste must be labeled with a radiation symbol and a description of the contents. The label should indicate the radionuclide, the activity in kBq or MBq, and the name of the authorized user. Containers should be securely closed, both top and bottom, with strong tape or other appropriate devices. Flaps must not be tucked one under the other; this method is not as strong as properly folded and taped flaps. Radioactive label tape is not strong. Both the tops and bottoms of boxes must be secured. Plastic bags should be tied or taped closed.

It is especially important to avoid inadvertent collection of radioactive waste by custodians. This goal will be achieved by proper and prominent labeling of radioactive waste containers in the laboratory.

As waste is being accumulated in a container, a record of each addition should be made. The record sheet must be summarized and must accompany the package.

11.2.3 Release of Radioactive Waste

In addition to the solid waste created at the facility and turned over to Radiation Safety, liquid wastes from the hot drains at the facility are collected in a retention tank for future disposal. Disposal is by means of pumping to the sanitary sewer after appropriate analysis and dilution if necessary. A detailed procedure for analyzing and pumping of the contents of the retention tank is contained in SOP #11 which specifies a release limit of less than 1.0×10^{-7} μCi/ml.

11.3 Self-Protection Radiation Levels for TRIGA Fuel Rod

11.3.1 Regulatory Requirements

Non-power reactors which possess HEU fuel in quantities of 5000 grams or more are required to meet the requirements of 10 CFR 73.67 (a), (b), (c) and (d) if such fuel is not self-protecting. 10 CFR 73.60 defines self-protecting fuel as: "not readily separable material that has a dose rate in excess of 100 rem per hour at a distance of 3 feet from any accessible surface without intervening shielding." In the case of the WSU modified TRIGA reactor, the smallest separate unit of fuel is a 4-rod cluster.

11.3.2 TRIGA Fuel Exposure Rates

The beta and gamma rays emitted by fission products as a function of time after fission was studied extensively by a number of investigators in the late 1940's and early 1950's. The most well known relationship is that of Way and Wigner (4) that is based on theoretical nuclear physics only and fits experimental results reasonably well. Glasstone in his book (5) on nuclear engineering integrated the Way-Wigner decay function to yield a function that ESTIMATES the fission product emission of a reactor as a function of reactor operating power, reactor operating time, and decay time (time since shutdown). However, if one reads Glasstone's book carefully, this function was said to only be accurate within a factor of about two.

Another problem is the fact that a typical research reactor is not operated continuously and then shut down but rather on a cycle of a number of hours each day or week frequently with more off time than on time in a typical week. Thus the real world problem is to calculate the fission product energy emitted as a function of time after shutting down from a long cycle of intermittent short operations. Thus the exact exposure geometry calculation is not the critical factor but rather fission product energy emitted by the fuel for a reactor operated in a cyclical fashion.

Two TRIGA reactor facilities have developed extensive complex computer programs to calculate the dose rate from a TRIGA fuel rod in air at 3 feet when operated in a cyclical fashion. The facilities that undertook this task are the University of Wisconsin at Madison (6) and the Oregon State University in Corvallis.(7) They each have cross checked their calculations with actual experimental measurements that have been found to be reasonably accurate.

The bottom line is that the calculations done by these two facilities may be formalized into a function that may be used to simply calculate the dose rate at 3 feet from any TRIGA reactor fuel element. In the case of the Oregon State University data their results are applicable to the first 30 days after shut down and the University of Wisconsin data for longer time after shut down from 30 to 400 days.

The Oregon State University data fits the function:

$$D(t)\text{Rem} / \text{hr} = 13 \times \frac{\text{kW}}{\text{element}} \times \sqrt[5]{\text{Op}} \times t^{-.34}$$

where: D(t) is the dose rate in air in Rem/hr 3 ft from the fuel element;
kW/element is the power generated in the fuel element in kW;
Op is the number of hours per week that the reactor is operated on an extended cycle at full power prior to shut down; and
t is the time in days after shut down in the range of 1 to 30 days.

The University of Wisconsin data fits the function:

$$D(t)\text{Rem} / \text{hr} = 10 \times \frac{\text{kW}}{\text{element}} \sqrt[3]{\text{Op}} \times t^{-(0.2+.035\sqrt{t})}$$

where the symbols are as defined above and where t is the number of days after shut down in the range of 30 to 400 days. A table for the decay corrections involved with the Oregon State University and University of Wisconsin calculations is given below:

Table 11-1
Fission Product Decay Factor

t(days)	$t^{-.34}$	$t^{-(.20+.035\sqrt{t})}$
5	.58	
10	.46	
15	.40	
20	.36	
25	.33	
30	.31	
50		.174
70		.123
100		.079
120		.061
140		.048
160		.038
180		.031
200		.025
250		.016
300		.010
350		.0067
400		.0046

11.3.3 Self Protection Time for WSU TRIGA Fuel

The power level of the WSU TRIGA reactor is 1,000 kW and the present core configuration contains [redacted]
The dose rate in Rem/hr after shut down from a cyclic operation of the indicated hours per week and decay time in days after shut down is given in the table below.

Table 11-2
Dose Rate for 4-Rod Clusters

Time after Shut Down in Days	Dose Rate in Rem/hr at 3 ft in Air Operation Cycle, hours per week			
	4	8	16	32
10	301	338	380	426
30	203	228	256	287
50	109	139	193	221
60	91	114	144	182
70	78	98	124	156
90	57	73	92	116
100	50	63	79	100

11.4 References

1. Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation", U.S. Government Printing Office, Washington, DC.

2. American National Standards Institute /American Nuclear Society, ANSI/ANS 15.11, "Radiation Protection of Research Reactor Facilities," ANS LaGrange Park, IL 1993.

3. Washington Administrative Code, Chapter 402, Title 10, "Rules and Regulations – Radiation Protection," Olympia, WA.

4. Physics Rev., 70:115, E. Way and E.P. Wigner, "Radiation from Fission Products," (1946).

5. Principles of Nuclear Reactor Engineering, S. Glasstone, Section 2.182, Van Nostrand, 1955.

6. Private Communication from Richard Cashwell, Director of University of Wisconsin Reactor Laboratory.

7. Private Communication from Art Hall, Reactor Supervisor, Oregon State University Radiation Center.

8. "Safety Analysis Report for the Torrey Pines TRIGA Mark II Reactor," GA-9064, General Atomic, Inc., January 5, 1970.

12 CONDUCT OF OPERATIONS

The facility over the years has developed many "Standard Operating Procedures" (SOP's) that are contained in a SOP manual. At the present time the SOP manual comprises 35 procedures that specify exactly how the facility is to be operated. More information on procedures is given in section 12.3. The facility also has developed numerous "Administrative Procedures" that cover administrative type functions.

12.1 Organization

A block diagram of the facility organization is shown in Figure 1.1 on page 1-3 of this SAR. This organizational structure has functioned quite adequately for the operation of the facility for over 20 years and thus should continue to function many years into the future.

12.1.1 Structure

The ultimate responsibility for the operation of the WSU reactor facility is vested in the University Administration at the Vice Provost for Research level. However, that responsibility and the administration of the facility is delegated to the Director of the Nuclear Radiation Center. The actual practical hour by hour operation of the facility is under the supervision of the Reactor Supervisor who is a Licensed Senior Reactor Operator who reports to the Director of the facility. The reactor operating staff report to the Reactor Supervisor in all matters relating to the operation and maintenance of the reactor facility.

The reactor staff also performs the radiation monitoring function at the facility under the direction of the Reactor Supervisor. The Radiation Safety office on campus serves as an advisory and audit function.

12.1.2 Responsibility

In order to clearly define responsibility and other related administrative type functions, the facility has developed an Administrative Procedures Manual. The first procedure in this manual is entitled "Responsibility and Authority of Reactor Operating Staff." This procedure clearly defines the responsibilities of 1) the Director, 2) the Reactor Supervisor, 3) Reactor Technicians, and 4) Reactor Operators. This procedure and the entire administrative procedures manual has been reviewed many times by the NRC inspectors and found to be quite adequate. It is the facilities contention, however, that it is inappropriate to include any specific detailed procedures in the SAR since they would remove the facility's ability to modify the procedure as conditions and staff members change without a license or SAR change. Heretofore the Commission has agreed on this matter.

12.1.3 Staffing

The operations staff required to operate the WSU reactor is specified in procedure #4, "Standard Procedure for Startup, Operation, and Shutdown of the Reactor." This procedure meets all the applicable requirements of 10 CFR 50.54. The key requirements of this procedure are listed below:

1. A Licensed Reactor Operator must be in the Control Room whenever the reactor control power key is inserted in the key switch.

2. All manipulations of the reactor controls shall be done by a Licensed Operator or by a Trainee under the direct supervision of a Licensed Operator.
3. A Senior Reactor Operator must be present at the facility during:
    a. Reactor checkout involving control element withdrawal and all startups,
    b. Significant increases in power level changes,
    c. Recovery from an unplanned shutdown (including SCRAMS and emergency shutdowns), and
    d. All fuel handling operations.

    At other times during normal operations, the SRO must be readily on call but not necessarily at the facility. On call shall be taken to mean that the individual may be reached by telephone or by means of the campus "Page Boy" radio paging system (normal working hours only).
4. At least two people shall be present at the Nuclear Radiation Center at all times when the reactor is critical. The second individual need not be licensed but must be capable of following written instructions.

12.1.4 Selection and Training of Personnel

The facility maintains an informal training program for all licensed operators and trainees that involves a daily tutorial session when the operations schedule permits. In addition, there is a set of training manuals that trainees are expected to read on their own time. On a practical level, all trainees, once they learn the technical information, are required to startup, operate, and shutdown the reactor numerous times under the tutelage of a licensed individual.

The facility maintains a file of old NRC RO and SRO exams and trainees must successfully pass one of these exams before they are considered to be qualified to take an NRC license exam. Over the past 15 years only two trainees did not pass the entire written exam and had to take one section over again before they received their RO licenses.

Special facility requirements include that the Director must be a faculty person with considerable nuclear reactor related educational experiences and must obtain a SRO license. The Reactor Supervisor must be an individual with considerable reactor operational experience, have held an RO license for a number of years. In addition to the licensed RO/SRO training program, the facility has an experimenter certification program that includes basic health physics and reactor usage related questions. All reactor users must pass a written user certification exam before being allowed to use the reactor. This program meets the requirements of 10 CFR Part 19. The RO/SRO training program meets the requirements of 10 CFR Part 55 and ANSI/ANS 15.4.

12.1.5 Radiation Safety

The facility maintains a formal Radiation Protection Program that meets all the requirements of 10 CFR Part 20 and ANSI/ANS-15.11. Detailed information on this program is given in Chapter 11 of this SAR. This program includes:

1) Radiation Exposure Limits
2) Surveys, Monitoring and Records
3) Reports and Audits
4) ALARA criteria.

The program is under the direction of the reactor staff but is reviewed and audited by the Reactor Safeguards Committee with the advice and assistance of the Campus Radiation Safety Office.

12.2 Review and Audit Activities

The Technical Specifications for the facility include a "Reactor Safeguards Committee" (RSC) that is a formal review, audit, and approval group. The RSC's review, audit, and approval function includes:

1) Reactor Operation
2) Radiological Safety
3) General Safety
4) Testing and Experiments
5) Licensing and Reports
6) Quality Assurance

The RSC is required to audit the facility records semiannually not to exceed every 8 months.

12.2.1 Composition and Qualifications

The RSC shall be composed of at least five members knowledgeable in fields that relate to nuclear reactor safety. The members of the Committee shall include one facility Senior Reactor Operator and WSU faculty and staff members designated to serve on the Committee in accordance with the procedures specified by the WSU committee manual. The University's Radiation Safety Director shall be an ex officio member of the Committee.

12.2.2 Charter and Rules

The Technical Specification concerning the operation of the Reactor Safeguards Committee is as follows:

> The Reactor Safeguards Committee shall operate in accordance with a written charter, including provisions for:
>
> (1) meeting frequency: the full committee shall meet at least semiannually and a subcommittee thereof shall meet at least semiannually
> (2) voting rules
> (3) quorums: chairman or his designate and two members
> (4) method of submission and content of presentations to the committee
> (5) use of subcommittees
> (6) review, approval and dissemination of minutes.

12.2.3 Review Function

The Technical Specification concerning the review function of the Reactor Safeguards Committee is as follows:

> The responsibilities of the RSC or designated subcommittee thereof shall include, but are not limited to, the following:
>
> (1) review and approval of all new experiments utilizing the reactor facility

(2) review and approval of all proposed changes to the facility license by amendment, and to the Technical Specifications
(3) review of the operation and operational records of the facility
(4) review of significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety
(5) review and approval of all determinations of whether a proposed change, test, or experiment would constitute a change in the Technical Specifications or an unreviewed safety question as defined by 10 CFR 50
(6) review of reportable occurrences and the reports filed with the Commission for said occurrences
(7) review and approval of all standard operating procedures and changes thereto
(8) biennial review of all standard procedures, the facility emergency plan, and the facility security plan
(9) annual review of the radiation protection program.

12.2.4 Audit Function

The Technical Specification requirement concerning the audit function of the Reactor Safeguards Committee is as follows:

> The RSC or a subcommittee thereof shall audit reactor operations semiannually, but at intervals not to exceed 8 months. The semiannual audit shall include at least the following:
> (1) review of the reactor operating records
> (2) inspection of the reactor operating areas
> (3) review of unusual or abnormal occurrences
> (4) radiation exposures at the facility and adjacent environs.

12.3 Procedures

The facility over the years has developed many "Standard Operating Procedures" (SOP's) that are contained in a SOP manual. At the present time the SOP manual comprises 35 procedures that specify exactly how the facility is to be operated.

The Technical Specifications for the facility concerning operating procedures is as follows:

> Written operating procedures shall be adequate to ensure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect for the following items:
> (1) performing irradiations and experiments
> (2) startup, operation, and shutdown of the reactor
> (3) emergency situations including provisions for building evacuation, earthquake, radiation emergencies, fire or explosion, personal injury, civil disorder, and bomb threat
> (4) core changes and fuel movement
> (5) control element removal and replacement

(6) performing preventive maintenance and calibration tests on the reactor and associated equipment

(7) power calibration

Substantiative changes to the above procedures shall be made only with the approval of the licensed SRO directly in charge of the facility. Temporary changes to the procedures that do not change their original intent may be made by a licensed SRO. All such temporary changes shall be documented and subsequently reviewed by the licensed SRO directly in charge of the facility.

12.4 Required Actions

The Technical Specification requirements concerning the required action to be taken in the event that a Safety Limit is exceeded is as follows:

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 U.S. Nuclear Regulatory Commission (NRC).

(2) An immediate report of the occurrence shall be made to the Chairman of the Reactor Safeguards Committee, and reports shall be made to the NRC in accordance with Section 6.10 of these specifications.

(3) A report shall be prepared that 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 Safeguards Committee for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

(4) A report shall be made to the NRC in accordance with Section 6.10 of these specifications.

12.5 Reports

The Technical Specifications for the facility contain very detailed reporting requirements that meet or exceed all the reporting requirements of the applicable regulations. Please refer to Section 6.10 of the Technical Specifications for reporting requirement information.

12.6 Records

The Technical Specification requirements for the facility concerning records are as follows:

In addition to the requirements of applicable regulations, and in no way substituting for those requirements, records and logs shall be prepared for at least the following items and retained for a period of at least 5 years for items (1) through (6) and indefinitely for items (7) through (11).

(1) normal reactor operation

(2) principal maintenance activities

(3) abnormal occurrences

(4) equipment and component surveillance activities required by the Technical Specifications

(5) experiments performed with the reactor
(6) gaseous and liquid radioactive effluents released to the environs
(7) off-site inventories and transfers
(8) fuel inventories and transfers
(9) facility radiation and contamination surveys
(10) radiation exposures for all personnel
(11) updated, corrected, and as-built drawings of the facility

12.7 Emergency Planning

The facility has an existing Emergency Plan approved by the Commission and a set of implementing procedures that meet all the requirements of the applicable regulations. If and when the facility shifts from HEU to LEU fuel, no significant changes will be required.

12.8 Security Planning

The facility has an existing Security Plan approved by the Commission that meets all the applicable requirements of 10 CFR Part 73 for a facility with HEU fuel. If and when the facility shifts from HEU to LEU fuel, some of the requirements of the present Security Plan can be removed, including:

1) Insuring that the HEU fuel in the reactor is self-protecting.
2) [redacted]

12.9 Quality Assurance

The Technical Specification requirements concerning Quality Assurance are as follows:

In accordance with Regulatory Guide 2.5 and ANSI 402, "Quality Assurance Program Requirements for Research Reactors," Section 2.17, the "facility shall not be required to prepare quality assurance documentation for the as-built facility." Quality Assurance (QA) requirements will still be limited to those specified in Section 2.17 as follows:

"All replacements, modification, and changes to systems having a safety related function shall be subjected to a QA review. Insofar as possible, the replacement, modification, or change shall be documented as meeting the requirements of the original system or component and have equal or better performance or reliability."

"The required audit function shall be performed by the RSC specified in Section 6.5."

12.10 Operator Training and Requalification

The facility has had a Commission approved Training and Requalification Plan for many years. The current plan's most recent revision was approved on August 16, 1995, meets all the applicable requirements of 10 CFR Part 55 as well as ANSI/ANS-15.4, and includes the following features:

1. biennial written requalification exam
2. annual operations test
3. biennial operations requirements
4. quarterly operations certification
5. facility and safety reviews
6. training records.

12.11 Startup Plan

The facility has been operating for over 20 years and thus a Startup Plan is not required.

12.12 Environmental Reports

12.12.1 General

This Environmental Report for the continued operation of the Washington State University Modified TRIGA Reactor is submitted to enable the Commission to comply with the National Environmental Protection Act of 1969 and the requirements of 10 CFR 51 in the renewal of Facility License R-76. On January 23, 1974 the Commission staff concluded in the 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." Thus no formal EIA is required for the extension of the operating license of Facility R-76 for the WSU TRIGA 1 MWt Research Reactor.

12.12.2 Location of Facility

The WSU TRIGA reactor is located in the Nuclear Radiation Center on the campus of Washington State University in Pullman, Washington. Pullman is a small town in the southeast corner of the State of Washington as described more fully in section 1.3.1 of this SAR and has a total population, including the university of 23,500. The Palouse region surrounding the town is a rural agricultural area devoted to dry land farming.

The actual reactor site is east of Pullman and east of the main portion of the WSU campus and the site is surrounded by university property used mainly for grazing livestock and growing animal feed. The closest occupied dwelling is 411 meters west of the facility.

12.12.3 Physical Characteristics Of The Facility

The WSU reactor is a modified TRIGA reactor and operates with a core of mixed Standard and FLIP or LEU fuels. The reactor was originally designed to use MTR plate-type fuel but was converted to TRIGA fuel in 1967 by replacing the MTR fuel elements with 4-rod clusters of TRIGA fuel. The reactor is housed in the WSU Nuclear Radiation Center which is a 1200 square meter laboratory devoted to nuclear related research and educational activities. The core of the reactor is situated in a 242,000 liter water pool which functions as shield, moderator, and coolant.

The WSU modified TRIGA reactor, like all TRIGA type reactors, has a very large prompt negative temperature coefficient, thus making the reactor inherently very safe. The kinetic behavior of TRIGA reactors permits them to be safely pulsed to very high power levels for a short duration. The pulse is automatically terminated by the effects of the large negative temperature coefficient. The WSU reactor operates at a maximum continuous steady state power level of 1 MWt and may be pulsed with a $2.25 insertion. The peak power during a pulse is on the order of 1200 MW.

12.12.4 Environment in the Area

The reactor site lies approximately 3.2 kilometers east of the center of the town of Pullman and 1.6 kilometers east of the center of the WSU campus. The land surrounding the site for at least 400 meters in all directions is uninhabited grass land owned by the university and used for the grazing of livestock. Geologically the site is located at an elevation of 808 meters on the south slope of a typical Palouse formation hill.

Pullman is situated near the eastern margin of the Columbia Plateau and the associated lava flows. The site is thus underlaid with basaltic rock produced by horizontal lava flows. The bedrock was capped with silt and clay deposited during the Pleistocene Age to form the present topsoil of the Palouse Loess with their characteristic rolling hill topography. The Palouse formation (topsoil) at the reactor site is approximately 30 meters thick.

Pullman is located approximately 480 kilometers inland from the Pacific Ocean. The Cascade Mountains, which average more than two kilometers in height, separate the region from the coast. The combined effect of the distance from the ocean and the extensive mountain barrier produces a climate that is continental in character. However, because the prevailing winds blow inland from the Pacific Ocean, winters are somewhat warmer than might be expected 480 kilometers inland at a latitude of 47° north. Winters in Pullman are characterized by cloudy skies and frequent snowstorms. On the average, the sun shines only about 30% of the time during the winter months.

During the summer months, the westerly winds weaken, and continental climatic conditions prevail. This causes rainfall, cloud cover, and relative humidity to be at their minimum; the daily mean temperature and daily temperature variation are at their maximum. Summers in Pullman are characterized by warm clear days and cool nights. On the average, the sun shines in Pullman about 80% of the time during the summer months.

One of the characteristics of the Palouse region is that of being a rather windy area. The average annual wind velocity is of the order of 16 km/hr. For the most part, winds peak in January averaging about 21 km/hr and the low occurs in July averaging about 11 km/hr. The wind velocity is greater than 5 km/hr 94% of the time and greater than 8 km/hr 76% of the time. The wind is from a westerly direction of the order of 60% of the time and an easterly direction 30% of the time.

The annual precipitation in the Pullman area is 50 centimeters and the annual average temperature is 8.7°C. The highest precipitation month is January with 6.8 centimeters and the lowest is July with 1 centimeter. The daily mean temperature peaks in July at 20°C. The mean daily minimum-to-maximum for the two extremes is 15.7°C and 5.7°C respectively.

There are no unique environmental or natural characteristics of the reactor site or archaeological or historical sites located within close proximity of the reactor site. The site is in a very low population density region and east of the main population concentrations of both the town of Pullman and the WSU campus. The population centers are also upwind of the site over 60% of the time.

12.12.5 Environmental Effects of Construction

No modifications to the Facility or the site will be required for the continued operation of the WSU reactor. There are no exterior conduits, pipelines, electrical or mechanical structures or transmission lines attached to the reactor facility other than utilities services which are required for other structures and laboratories on campus. Thus there will be no significant effects upon the terrain, vegetation, wildlife, nearby waters, or aquatic life due to construction-type activities.

12.12.6 Environmental Effects of Facility Operation

(a) Water Use Consumption

Make-up water for both the reactor pool and wet cooling tower are required for operation of the reactor. The WSU campus has its own water system with water derived from wells independent of the Pullman water system. Pool make-up amounts to 20,000 liters per month on the average and the cooling tower requires 2100 liters per hour of reactor operation at full power. The total water consumption of the reactor cooling system is approximately 195,000 liters per month.

The required make-up water is readily available from the WSU campus water system and thus will have no impact on the Pullman water supply system.

(b) Heat Dissipation

The WSU TRIGA reactor has a maximum steady state power output of 1 MWt. The 1 MWt of heat generated by the reactor is dissipated by an evaporative mechanical draft cooling tower located on the north side of the facility. In the summer of 1999 a new cooling system was installed at the facility that includes a new high efficiency cooling tower that uses less water than the original cooling tower. The evaporative cooling system cools the reactor pool and dissipates the heat generated by the reactor to the atmosphere through the latent heat of vaporization of water. On the average, the facility generates 1000 MW-h per year and thus approximately 1.5×10^6 kilograms of water vapor are added to the local atmosphere per year by the operation of the facility. In other words, an average of about 4000 liters of water per day are added to the atmosphere at the site.

Evaporative cooling towers have the potential for creating visible plumes of water vapor under certain atmospheric conditions. The plume is a region of air with a higher temperature and higher water content than the ambient air. The climatic and atmospheric conditions at the site and the small amount of water involved preclude the development of a plume by the WSU reactor cooling tower during the summer months. However, during the winter months a very small plume is sometimes produced that rises of the order of 30 meters into the air above the cooling tower. Fogging and icing conditions at the site are not affected by the operation of the cooling tower. The amount of water added to the local atmosphere annually by the cooling tower is really insignificant compared to the 50 centimeters annual precipitation in Pullman. Thus the water added to the atmosphere by the operation of the facility have in the past and will continue in the future to have a minimal effect on the environment.

(c) Chemical Discharges (non-radioactive)

No chemical discharges are generated directly from the operation of the reactor. The chemical discharges into the sanitary waste system at the Nuclear Radiation Center are related to conventional chemical laboratory operations at the site and are not different than those of other laboratories on campus.

The blow-down of the cooling tower also discharges into the sanitary sewer system. The blow-down discharge amounts to 9300 liters per month on the average which contains an increased amount of total dissolved solids (TDS) than the input potable water. The concentration factor will be less than 10 and thus the increased TDS is not significant.

The cooling tower and associated heat exchanger, like all boilers and other water cooling systems on campus, are maintained by the WSU water treatment group. The standard campus water treatment involves the use of MOGUL WS164 water treatment liquid at the rate of 40 ppm plus 22 ppm of algicide. The incremental increase in the discharge of treated water by the operation of the reactor is, however, insignificant compared to the total campus discharge of such water into the sanitary sewage system. Thus the environmental effects related to chemical discharges created by the operation of the reactor are not significant.

(d) Radioactive Discharges

(1) Gaseous

The ventilation system of the reactor discharges 2.12 m^3/sec of air from the pool room into the atmosphere. The principal radionuclide contained in the discharge air is Argon-41 which is produced by the activation of argon contained in air. The Argon-41 content of reactor pool room exhaust is continuously monitored with a special gamma-ray spectrometer set to detect Argon-41. Over the past 5 years the total average quantity of Argon-41 discharged from the facility amounted to 20.7% of the Technical Specification limit. On a concentration basis, taking into account the dilution of the atmospheric wake effect in the lee of the building, the 5 year average release concentration of Argon-41 was 2.1×10^{-10} $\mu Ci/cm^3$. The release concentration for Argon-41 given by the EPA for reactor facilities in 40 CFR 61, subpart I is 1.7×10^{-9} $\mu Ci/cm^3$ which is 58 times lower than the new 10 CFR 20 Appendix B, Table II, Column 1 limit for Argon-41 of 1.0×10^{-8} $\mu Ci/cm^3$. The actual release concentration over the past 5 years amounted to 2.1% of the 10 CFR 20 limit and 12% of the EPA limit. A small amount of tritium is produced in the pool water through neutron capture in the deuterium present in the pool water. Measurements of the ^3H level in the pool water of a number of TRIGA reactors including the WSU reactor are reported on Page 170 of the August, 1976 issue of Health Physics. Measurements made by the WSU Radiation Safety Office agree with the reported value for the WSU reactor of .045 μCi/l. The pool evaporation rate amounts to 560 liters per day and the pool room exhaust discharge is 1.834×10^{11} cm^3 per day. If we make the conservative assumption that the ^3H content of the pool water and evaporated water are the same, then the pool room exhaust would contain 1.37×10^{-10} $\mu Ci/cm^3$ of tritium. This is significantly below the applicable limit in

10 CFR 20 of 1 x 10^{-7} µCi/cm^3 and the EPA limit of 1.5 x 10^{-9}. No other significant quantity of gaseous radioactive material or particulate radioactive material with a half-life greater than eight days has been released by the facility during the past 10 years.

In the event of a Loss of Coolant Accident (LOCA) or the Maximum Hypothetical Accident (MHA), the 1995 review analysis of this postulated accident has shown the gaseous radioactive discharges to be minimal. The worst case whole body dose from a cloud of fission products discharged from the facility as a result of the MHA is only 1.28 mrem/hr. The worst case maximum thyroid dose outside the facility for a 3% halogen release was found to be 17.4 mrem/hr. Thus no realistic hazard to the general public would result from the MHA or a LOCA.

(2) Liquids

No radioactive liquids are generated by the operation of the reactor in and by itself. However, the nuclear research and educational activities at the Nuclear Radiation Center generate radioactive liquids from radiochemistry experiments and from activation analysis activities. All hot drains from the laboratory flow into a holdup tank system which is monitored and diluted as necessary before being discharged into the sanitary sewer. Over the past 3 years the radioactive liquid released from the holdup tanks, on the average, contained 4 x 10^{-8} µCi/cm^3 or about 10% of the applicable release limit and amounts to about .5 µCi/month.

The Radiation Safety Office at WSU has, for over 10 years, monitored the Radiochemistry level in the waters in the vicinity of WSU including the South Fork of the Palouse River, local tap water, and sewage treatment plant effluent. An increase in the activity levels attributable to the operation of the WSU TRIGA reactor has never been detected.

(3) Solids

The only solid radioactive waste generated directly by the operation of the reactor is spent ion exchange resin. Approximately .3 cubic meters of spent resin is disposed of each year. It is estimated that the long-lived components of the activity in the spent resin amounts to about .1 Ci/yr.

The entire WSU campus generates of the order of 8 cubic meters of solid radioactive waste annually containing approximately .5 curies of activity. This solid waste is predominantly generated by research activities in university laboratories other than the Nuclear Radiation Center utilizing long-lived purchased radionuclides. Thus the incremental increase in solid wastes generated by the operation of the reactor is minimal. All solid wastes are transferred to the Nuclear Engineering Company of Richland, Washington for disposal.

(e) Radiation Levels

An extensive Environmental Radiation Monitoring Program was instituted at the WSU Nuclear Radiation Center in July of 1974. The program involves measuring the integrated radiation exposure for a period of three months at 40 points at the site and associated environs. Commercially available thermoluminescent dosimeters (TLD's) of the $CaSO_4$:Dy type provided and processed by the Radiation Detection Company, Sunnyvale, California are utilized.

Table 12-1 lists the average exposure rate above ambient background per megawatt hour of reactor operation for a number of locations at the site. The two highest exposure points are on the roof directly above the pool and at the freight door to the pool room. The maximum possible on-site exposure at a readily accessible location would be to an individual standing at the pool room freight door for the 1000 hours per year that the reactor operates. The total maximum annual exposure at this on-site point would be 87 mrem/year.

The exposure rates at points from 50 meters to 24 kilometers from the Nuclear Radiation Center have also been monitored quarterly since 1974. The average exposure rate at the 24 locations involved is 188 ± 30 μR per day. No statistically significant variations in the above background exposure rates at the sample locations have been observed or any exposure attributable to the operation of the WSU reactor. In addition, the average exposure rates at these locations which are 50 meters from the site are not statistically different on a quarterly basis than the average of the background exposure rates at 17 locations in the State of Washington monitored by the State of Washington Department of Emergency Services. Thus no significant effect on the radiation levels in the environment surrounding the facility has been observed to date.

12.12.7 Alternatives To Continued Operation Of The Facility

There are no suitable or more economical alternatives which can accomplish both the educational and research objectives of the facility. These objectives include but are not limited to: the training of students in the operation of nuclear reactors; the training of students in the use of radioisotopic tracer techniques; the production of radioisotopes for use in numerous areas of the physical, biological, and animal sciences; he training of students and research applications of trace element analysis by neutron activation analysis; and also a demonstration tool to familiarize the student body and general public with nuclear reactors and their operation.

In addition, the WSU Reactor facility is in the process of establishing a Neutron Capture Therapy facility for preclinical NCT research. The NCT method can only be done at a nuclear reactor facility and thus there is no alternative to this new, important cancer treatment methodology.

12.12.8 Short-Term Effects Versus Long-Term Gain Of Facility Operation

One of the chief objectives of any institution of higher education is to increase the body of knowledge available to mankind and to impart that knowledge to individuals. Accordingly, it is very difficult to compare the long-term gains from the operation of a research reactor in relation to the short-term environmental effects. However, the total environmental effects of the WSU TRIGA reactor and associated Nuclear Radiation Center are not significantly different from other research laboratories at a typical university. For the most part, the cumulative long-term benefits of university research activities far outweigh the environmental effects of such activities. This would also be true for the continued operation of the WSU reactor.

12.12.9 Cost Benefit Analysis

The facilities at the Nuclear Radiation Center represent an investment of the order of $2 million dollars. If the facility were shut down, the benefits derived from this investment would drop to zero. On the other hand, continued operation would allow the continuation of 10 ongoing research programs and the completion of about 8 graduate thesis research projects per year. The benefits also include the educational objectives mentioned in Section 6.0 and the new NCT cancer therapy project being undertaken. Considering the minimal environmental effects of the continued operation of the rector as previously cited in this report, the environmental cost effects are very small compared to the benefits to be derived from continued operation.

Table 12-1
Median Exposure Rates per Megawatt Hour of Reactor Operation
in Close Proximity to the Nuclear Radiation Center

Location	(Adjacent to Room)	Exposure (μR/MW-Hr)
Front Entrance	50V	32
Pool Room Freight Door	201	87
North Side of Building	201B	10
Roof above Control Room	201B	16
Roof above Pool	201	152
Roof above Laboratory Area	214	0
West Side Door at Beam Room	2X	14
Storage Building	217A	21
Lower Loading Dock	123A	17

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. WSU Nuclear Radiation Center Reactor Operations Procedure Manual.

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, LaGrange Park, Illinois, 1995.

13 ACCIDENT ANALYSES

13.1 Accident-Initiating Events and Scenarios

In normal operation a TRIGA type non-power reactor does not in and of itself constitute a threat to the health and safety of the facility staff or the general public. However, in this section of the Safety Analysis Report we will postulate various types of hypothetically possible non-normal events and the associated consequences of these events. The core of the reactor produces a significant amount of radiation during operation which is shielded by the pool water. Fission products are built up over time in the fuel, but are contained within the fuel and again shielded by the pool water. Any event that could cause these normal conditions to be compromised have a theoretical potential to produce a non-safe set of conditions.

In the following subsections of this section a number of potential accident initiating events will be considered along with the consequences of each event. The first analysis will be for the Maximum Hypothetical Accident (MHA) which for a TRIGA reactor is defined as the loss of the integrity of the fuel cladding of one fuel rod in air. The next event that will be considered is that of the insertion of excessive reactivity into the core. This may be precipitated by an accidental fuel addition or the accidental ejection of the transient rod. The next event that will be analyzed is the complete loss of coolant accident. Finally, a number of other minor events will be considered.

13.1.1 Maximum Hypothetical Accident

A <u>INTRODUCTION</u>

The Maximum Hypothetical Accident (MHA) for a TRIGA reactor is defined as the loss of the integrity of the fuel cladding of one fuel rod in air. The hazard associated with this hypothetical accident is thus the effects of the postulated fission product release within the facility and to the surrounding environment. The MHA for the WSU TRIGA reactor was originally analyzed in the Safety Analysis for converting the WSU TRIGA reactor to FLIP fuel of May, 1974 (1, 2). This revised analysis uses the same basic data as used in the previous analysis but the effects are calculated using more recent analysis methods and guide lines published by the Federal Government (3).

B. <u>FISSION PRODUCT INVENTORY</u>

The fission product release fraction for TRIGA-type reactor fuel has been measured experimentally (4) and documented before the AEC hearings on the Columbia reactor as being [redacted] release fraction, FR, is, however, a function of the fuel temperature, T, in °C given by the relationship (4):

[redacted] by the above relationship. A release fraction of 1.2×10^{-4} will be used in the calculations for the MHA.

A power density of 30 kW per fuel rod and an infinite irradiation time will also be assumed for the MHA. Under these conditions the fission product inventory for one TRIGA fuel rod and the associated released fission products are tabulated in Tables 13-1 and 13-2. This tabulation was derived from the basic data of Perkins and King (5) along with the documented fact (4) that only the gaseous fission products escape when the cladding of a TRIGA fuel rod ruptures. The data in Tables 13-1 and 13-2 are comparable to fission product inventory recently calculated for the Bangladesh TRIGA reactor by G.A. after correcting for reactor power level.

C. WHOLE BODY RADIATION EXPOSURE IN POOL ROOM

The whole body exposure for each fission product radionuclide released as a result of a postulated single fuel element cladding failure is given in Table 13-2. This Table contains all the parameters involved with the dose calculation including the DCF (Dose Conversion Factor) for each radionuclide taken from the 1992 EPA document: "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents" (3). The DCF's include the sum of the dose from external radiation exposure from an infinite radioactive hemispherical cloud and exposure due to inhalation of each airborne radionuclide.

The total dose to an individual in the Pool Room of the WSU TRIGA reactor for 5 minute and 1 hour exposure times are given in Tables 13-4 and 13-5. Table 13-4 assumes a 100% noble gas release and a 25% halogen release. Table 13-5 assumes a 100% noble gas release and a 3% halogen release (most realistic case).

D. THYROID RADIATION EXPOSURE IN POOL ROOM

The thyroid radiation exposure for each iodine radionuclide in a single fuel element and 100% escape into the pool room is given in Table 13-3. The total thyroid dose for 5 minute and 1 hour exposure times is given in Tables 13-4 and 13-5. Table 13-4 assumes a 25% halogen release and Table 13-5 assumes a 3% halogen release.

Table 13-1
SOLUBLE GASEOUS FISSION PRODUCTS
CONTAINED IN AND RELEASED FROM A SINGLE TRIGA FUEL ROD*

Isotope	Saturated Inventory	Released Activity	Half-Life
Br-82	[redacted]		35.3 hr
83			2.3 hr
84			31.8 min
85			3.0 min
87			55.0 sec
Total Br			
I-130m			9.2 min
131			8.1 days
132			2.3 hr
133			21.0 hr
134			54.0 min
135			6.8 hr
136			86.0 sec
Total I			
Kr-83m			1.9 hr
85m			4.4 hr
85			10.7 yr
87			78.0 min
88			2.8 hr
89			3.2 min
Total Kr			
Xe-131m			12.0 days
133m			2.3 days
133			5.3 days
135m			15.0 min
135			9.0 hr
137			3.9 min
138			17.0 min
Total Xe			

Total Released Soluble Gaseous Fission Products = [redacted]
Total Gamma Emitters =
Total Beta Emitters =

*Power Density = 30 kW/rod, fuel temperature = 500°C, release fraction 1.2 x 10^{-4}

Table 13-2
POOL ROOM FISSION PRODUCT CONCENTRATIONS
AND ASSOCIATED EXPOSURE RATES FOR
SINGLE FUEL ELEMENT CLADDING FAILURE (a)

Isotope	Activity Released in mCi (b)	Release Concentration in $\mu Ci/cm^3$ (c)	DCF in rem per $\mu Ci/cm^3/hr$ (d)	Dose Rate in mrem/hr (b)
Br-82	[redacted]			
83				
84				
85				
87				
Total Brom[redacted]				
I-131				
132				
133				
134				
135				
136				
Total Iodin[redacted]				
Kr-83m				
85m				
85				
87				
88				
89				
Total krypt[redacted]				
Xe-131m				
133m				
133				
135m				
135				
137				
138				
Total xeno[redacted]				

(a) Fracture and release of fission products in a TRIGA Fuel Element with a 1.2×10^{-4} release fraction.
(b) Averaged over a 15 minute period.
(c) Pool Room volume = $1 \times 10^9 cm^3$.
(d) Values from EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents. Includes both inhalation and external exposure effects.
(e) In actual fact, only about 3% of the Bromine and Iodine will escape from the pool water.

Table 13-3
SINGLE FUEL ELEMENT FAILURE
Total Iodine Release Thyroid Exposure (a)

Isotope	Activity Released in mCi (b)	Release Concentration in μCi/cm³ (c)	DCF in rem per μCi/cm³/hr (d)	Dose Rate in rem/hr
I-131	[redacted]			
132				
133				
134				
135				
Total thyroid d				

(a) Fracture and release of fission products in our TRIGA Fuel Element with a 1.2×10^{-4} release fraction.
(b) Averaged over a 15 minute period.
(c) Pool Room volume = $1 \times 10^9 cm^3$.
(d) Values from EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents. Includes both inhalation and external exposure effects.

Table 13-4
WORST CASE DOSE TO PERSONNEL IN THE REACTOR POOL ROOM:
SINGLE FUEL ELEMENT FAILURE WITH POOL WATER LOSS.
RELEASE: 100% NOBLE GASES, 25% HALOGENS

Exposure Time	Whole Body (mrem)	Thyroid (mrem)
5 min	46	3,470
1 hr	555	41,650

Table 13-5
WORST CASE DOSE TO PERSONNEL IN THE REACTOR POOL ROOM:
SINGLE FUEL ELEMENT FAILURE WITHOUT POOL WATER LOSS.
RELEASE: 100% NOBLE GASES, 3% HALOGENS

Exposure Time	Whole Body (mrem)	Thyroid (mrem)
5 min	31	417
1 hr	376	4,998

E. DISCHARGE OF THE FISSION PRODUCTS INTO THE ENVIRONMENT

The rate at which fission products from the pool room are released into the environment in a MHA condition is dependent upon the rate of removal of pool room air by the pool room ventilation system. In the normal operation mode, air is exhausted from the pool room at the rate of 4500 cfm or 2.12×10^6 cm^3/sec. In the dilution mode, 300 cfm of air from the pool room is passed through a HEPA filter system, diluted with 1700 cfm of outside air and discharged into the atmosphere. In the dilution mode, 2000 cfm of air is discharged or 9.44×10^5 cm^3/sec with a dilution factor of 6.67. If the ventilation system is off, the release to the environment would only be by leakage from a sealed building which is estimated to be of the order of 100 cfm or 4.72×10^4 cm^3/sec. In the dilution mode the HEPA filter would remove at least 90% of the iodine from the exhaust air.

The activity of each radionuclide exhausted from the facility, X_i in μCi per cm^3 at any time after t = 0 is given by the equation

$$X_i = A_i e^{-(\lambda_i + \ell/V)t} .$$

where A_i = the concentration of the i^{th} isotope in the pool room at t = 0 in μCi/cm^3
ℓ = the building exhaust rate in cm^3/sec
V = the volume of pool room in cm^3
λ_i = the decay constant of the i^{th} isotope in sec^{-1}
t = time after t = 0 in seconds.

F. DILUTION OF DISCHARGE IN THE LEE OF THE BUILDING

The gaseous radioactive material discharged from the facility ventilation system will be diluted by atmospheric air in the lee of the building due to turbulent wake effects. The dilution is proportional to the product of the cross sectional area of the building times the wind speed. That is,

ϕ = dilution factor $\phi = 1/CAu$ (sec/cm^3)
C = constant (.5 to 2), select 1 (cm^3/m^3)
where A = building cross-sectional area in square meters
u = wind speed in meters/sec.

Thus for a nominal 2m/sec wind velocity (4.4 mph)* and a 17.07m (56 ft) x 8.53m (28 ft) building, the dilution factor is $\phi = 3.4 \times 10^{-3}$.

G.1. WHOLE BODY RADIATION EXPOSURE AND THYROID EXPOSURE OUTSIDE THE FACILITY

The activity discharged into the atmosphere as a result of the MHA under two modes of operation of the ventilation system are given in Tables 13-6 and 13-7 along with the thyroid and whole body exposure to a person outside the facility in both cases. The activities reported include a correction for the atmosphere dilution factor in the lee of the building and in the case of the dilution mode, for the dilution factor for this mode of operation. The activity discharged under the third mode of ventilation system operation, the isolation mode, is assumed to be more

* Average annual wind speed is in excess of 5 mph (2).

conservative than the normal mode since its release to the environment is at a rate 20 times less than for the normal mode.

Table 13-6
ENVIRONMENTAL FISSION PRODUCT CONCENTRATIONS
AND ASSOCIATED EXPOSURE RATES FOR
SINGLE FUEL ELEMENT CLADDING FAILURE
(Ventilation system on, no radionuclide decay)[a]

Isotope	Pool Room Conc. in $\mu Ci/cm^3$	Environmental Conc. in $\mu Ci/cm^3$	DCF in rem per $\mu Ci/cm^3/hr$	Dose Rate in mrem/hr
Br-82	[redacted]			
83				
84				
85				
87				
Total Bromi				
I-131				
132				
133				
134				
135				
136				
Total Iodine				
Kr-83m				
85m				
85				
87				
88				
89				
Total krypto				
Xe-131m				
133m				
133				
135m				
135				
137				
138				
Total xenon				

(a) Worst case assuming the ventilation systems is ON and discharge equals pool room concentration and only dilution effect is that of wake effect in the lee of the reactor building.

Table 13-6A
SINGLE FUEL ELEMENT FAILURE
Environmental Thyroid Exposure for Total Iodine Release*

Isotope	Environmental Release Concentration in $\mu Ci/cm^3$	DCF in rem per $\mu Ci/cm^3/hr$	Dose Rate in rem/hr
I-131	[redacted]		
132			
133			
134			
135			
Total thyroid dose rat			

Table 13-6B
TOTAL ENVIRONMENTAL WORST CASE EXPOSURE:*
Single Fuel Element Failure with Pool Water Loss.
Ventilation System in Normal Exhaust Mode
Release: 100% Noble Gases, 25% Halogens

Exposure Time	Whole Body (mrem)	Thyroid (mrem)
5 min	.157	12
1 hr	1.89	145

Table 13-6C
TOTAL ENVIRONMENTAL WORST CASE EXPOSURE:*
Single Fuel Element Failure with Pool Water Loss.
Ventilation System in Normal Exhaust Mode
Release: 100% Noble Gases, 3% Halogens

Exposure Time	Whole Body (mrem)	Thyroid (mrem)
5 min	.106	1.45
1 hr	1.28	17.4

*Single fuel element failure, ventilation system in normal exhaust mode, no radioactive decay correction, only dilution effect is that of wake effect in the lee of the reactor building.

Table 13-7
ENVIRONMENTAL FISSION PRODUCT CONCENTRATIONS
AND ASSOCIATED EXPOSURE RATES FOR
SINGLE FUEL ELEMENT CLADDING FAILURE
(Ventilation system in dilution mode, no radionuclide decay)[a]

Isotope	Pool Room Conc. in $\mu Ci/cm^3$	Environmental Conc. in $\mu Ci/cm^3$	DCF in rem per $\mu Ci/cm^3/hr$	Dose Rate in mrem/hr
Br-82	[redacted]			
83				
84				
85				
Total Bromin				
I-131				
132				
133				
134				
135				
136				
Total Iodine				
Kr-83m				
85m				
85				
87				
88				
89				
Total krypton				
Xe-131m				
133m				
133				
135m				
135				
137				
138				
Total xenon				

(a) Worst case dilution mode assuming the ventilation systems is in dilution mode and discharge equals initial pool room concentration diluted by the wake effect in the lee of the reactor building and the effects of the dilution mode but no radioactive decay correction or time dependent exhaust effect.

Table 13-7A
SINGLE FUEL ELEMENT FAILURE
Environmental Thyroid Exposure for Total Iodine Release,
Ventilation System in Dilution Mode, HEPA Filter Not Functioning*

Isotope	Environmental Release Concentration in $\mu Ci/cm^3$	DCF in rem per $\mu Ci/cm^3/hr$	Dose Rate in rem/hr
I-131	[redacted]		
132			
133			
134			
135			
Total thyroid dose rat			

Table 13-7B
TOTAL ENVIRONMENTAL WORST CASE EXPOSURE:*
Single Fuel Element Failure with Pool Water Loss.
Ventilation System in Dilution Mode
Release: 100% Noble Gases, 25% Halogens

Exposure Time	Whole Body (mrem)	Thyroid (mrem)
5 min	.024	1.8
1 hr	.28	21.7

Table 13-7C
TOTAL ENVIRONMENTAL WORST CASE DILUTION MODE EXPOSURE:*
Single Fuel Element Failure without Pool Water Loss.
Ventilation System in Dilution Mode
Release: 100% Noble Gases, 3% Halogens

Exposure Time	Whole Body (mrem)	Thyroid (mrem)
5 min	.016	.22
1 hr	.19	2.60

*No radioactive decay corrections, HEPA filter not working. Only corrections are t = 0 dilution effects.

G.2. DECAY CORRECTED ENVIRONMENTAL EXPOSURE ESTIMATE

In order to simplify the calculation of the effects of fission product release into the environment from the pool room, including radioactive decay of the isotopes involved, the released fission products have been lumped into four groups of isotopes with similar half-lives. The mean weighted half-life of each group along with the mean weighted DCF for each group and the total quantity of radioactive material in curies was calculated for each group. The group data has then been used to calculate the environmental radiation exposure as a function of time given in Tables 13-8 and 13-9.

a. Group 1. Half-life = 0 to 30 min
Effective half-life = 14.1 min
Weighted mean DCF = 521
Total group activity = 353 mCi at t = 0

b. Group 2. Half-life 31 min to 3.0 hr
Effective half-life = 153 min
Weighted mean DCF = 1200
Total group activity = 176 mCi at t = 0

c. Group 3. Half-life = 3.1 hr to 10 hr
Effective half-life = 8.5 hr
Weighted mean DCF = 258
Total group activity = 233 mCi at t = 0

d. Group 4. Half-life = 11 hr to ∞
Effective half-life = 5.38 days
Weighted mean DCF = 182
Total group activity = 205 mCi

Table 13-8
Pool Room Dose at t = 0 Using Four Group Data

Group	Activity mCi	Pool Room $\mu Ci/cm^3$	DCF in rem per $\mu Ci/cm^3/hr$	Dose in mrem/hr
1	[redacted]			182
2				211
3				60
4				37
	Total			490

The previous result shown in Table 13-5 is 376 mrem/hr so at least at t = 0 the group method is conservative.

H. ENVIRONMENTAL DOSE RATES AS A FUNCTION OF TIME

(a) Normal Ventilation System Operation Mode.

In the case that the ventilation system is left in the normal operation mode, radioactive decay is an insignificant consideration because the volume of the air in the pool room is exhausted over seven times in one hour. The only significant effect is the dilution by the exhaust system. In this case the environmental dose rate outside the facility as a function of time is given by $D(t) = D(t{=}0)\ e^{-(l/v)t}$ where D(t=0) is the dose rate at t = 0, l = exhaust rate, v = pool room volume, and t is the time.

Table 13-9
ENVIRONMENTAL DOSE RATES AS A FUNCTION OF TIME*
VENTILATION SYSTEM IN NORMAL OPERATIONAL MODE

Time (min)	(l/v)t	D(t)/D(t=0)	D(t) in mrem/hr Thyroid	D(t) in mrem/hr Whole Body
0	1	1	17.4	1.28
10	1.28	.28	4.9	.36
20	2.56	.08	1.4	.10
30	3.84	.021	.04	.003
60	7.68	.00046	.08	0
120	15.76	2.13×10^{-7}	0	0

*Assume 100% nobel gas release and 3% halogen release.

(b) Dilution Mode Operation

In the dilution mode, Pool Room air is diluted with air drawing from outside the facility by a factor of 6.67 before being discharged into the environment. Also the air is passed through a HEPA filter which will remove all particulate matter as well as most of the halogens. However, for worst case calculational purposes, it will be assumed that the dilution effect is the only mitigating effect.

(c) Isolation Mode Operation

In the isolation mode, Pool Room air loss should be less than 100 cfm and therefore environmental exposures should be less than for the normal mode for which the air loss is 20 times greater.

Table 13-10
WORST CASE ENVIRONMENTAL WHOLE BODY DOSE
AS A FUNCTION OF TIME WITH THE VENTILATION SYSTEM OPERATING
IN THE DILUTE MODE, HEPA FILTER NOT FUNCTIONING

Time (hrs)	$(\lambda + l/v)t$	D(t)/D(t=0)	D(t) in mrem/hr Whole Body
0	1	1	.19
1	.0132	.986	.19
4	.0527	.949	.18
8	.1055	.90	.17
24	.3164	.729	.14
48	.6327	.531	.11
96	1.268	.282	.05
144	1.898	.15	.03
320	4.218	.015	.03

Table 13-11
ONE GROUP IODINE ENVIRONMENTAL EXPOSURE RESULTS FOR
3% IODINE RADIONUCLIDE ESCAPE FROM ONE FUEL ELEMENT.
VENTILATION SYSTEM OPERATING IN DILUTION MODE,
HEPA FILTER NOT WORKING[a]

Time (hrs)	$(\lambda + l/v)t$[b]	D(t)/D(t=0)	Thyroid Exposure Rate mrem/hr
0	1	1	2.7
.5	.2603	.771	2.08
1	.521	.594	1.60
2	1.042	.353	.95
4	2.984	.124	.33
8	4.168	.015	.041
12	6.252	.0019	.0051
24	12.50	0	0

(a) Total dilution effect at t = 0 is 1962.
(b) $(\lambda + l/v) = .521\ hr^{-1}$

I. SUMMARY OF RESULTS OF MHA

The preceding calculations on the consequences of the Maximum Hypothetical Accident indicate that the only significant worst case radiation exposure is the thyroid dose to a person in the pool room. The conditions necessary to produce this exposure are the failure of the cladding of one fuel rod along with a complete loss of pool water. The maximum possible radiation exposure to an individual outside the facility under the postulated conditions is minimal. The exposures are significantly below the generally acceptable accident results for nonpower reactors of not more than 5 rem whole body and 30 rem thyroid for occupational exposure and not more than .5 rem whole body and 3 rem thyroid for members of the general public. In addition, the calculated accident exposures are well below the maximum values established in 20.1201 for occupational exposure and 20.1301 for public exposure. Thus, no realistic hazard to the staff at the reactor as well as the general public would result from the MHA.

13.1.2 Insertion of Excess Reactivity

The two possible events that would create a sudden insertion of a large amount of reactivity are: 1) the accidental addition of fuel to a reactor while operating, and 2) the accidental ejection of the most reactive rod (pulse rod) at full power. Both of these events are very unlikely since in order to create the conditions assumed, a number of procedures would have to be violated and the staff would have to purposely create the event.

The conditions for the first postulated reactivity addition accident are the accidental addition of a FLIP four(4)-rod fuel bundle to the central region of the core with the reactor operating at full power. In actual practice, the Technical Specifications prohibit operation of the reactor with a vacant grid position. The maximum worth of a FLIP four(4)-rod cluster in the central region of the WSU TRIGA core is calculated to be less than \$3.75. Thus, the maximum possible reactivity insertion for an accidental fuel addition would be less than \$3.75.

A step increase of \$3.75 in a typical mixed FLIP-Standard core would produce an average core temperature increase of approximately 422°C. Adding this average core temperature increase to the average full power core temperature of 265°C and multiplying by a 1.67 peaking factor yields a maximum possible peak fuel temperature of 1142°C. This is below the 1150°C safety limit for FLIP fuel. The high temperature scram would, in this accident, limit the maximum fuel temperature to below the value calculated above and thus increase the margin of safety. Thus, no additional hazard is caused by the addition of a four(4)-rod FLIP fuel cluster to the core during full power operation.

The conditions for the second postulated reactivity added accident are the accidental ejection of the full worth of the transient rod with the reactor operating at full power. During normal operation, pulsing will be administratively limited to the maximum value established in Section 4.5.3 of this SAR. However, for the purposes of this accident it will be assumed that the operator deliberately violates the established limit and also bypasses the interlock that inhibits pulsing above 1 kW. Thus, the total maximum worth of the pulse rod of \$3.75 would be added to the core at full power.

The effects of a \$3.75 reactivity addition at full power were shown not to exceed the safety limit in the paragraph above. Thus, no additional hazard is caused by the ejection of the pulse rod at full power.

13.1.3 Loss of Coolant

The conditions for this postulated accident are that the reactor has been operating at 1 MW for essentially an infinite length of time and then a sudden complete loss of coolant (pool water) occurs. The loss of water will shut down the reactor; however, the decay heat from the fission products will continue to produce heat in the fuel elements. In order to insure the safety of the reactor in the event of this postulated accident, the fuel cladding temperature in air must be maintained below the point where a cladding failure could occur.

The strength of the fuel element cladding is a function of cladding temperature which is a function of the fuel element temperature. The conservative assumption is made for purposes of this analysis that the fuel and cladding are at the same temperature. The stress imposed on the cladding by hydrogen disassociation within the fuel is a function of the fuel temperature, the fuel burnup and the free gas volume with the fuel rod. Figure 13-1 (1, 3) shows the stress imposed upon the cladding by TRIGA reactor fuels with H-Zr ratios of 1.6 and 1.7 as a function of fuel temperature. This Figure also shows the yield and ultimate strength of the cladding as a function of temperature.

An examination of Figure 13-2 indicates that the maximum temperature that Standard fuel (H-Zr 1.7) can tolerate in air without damage to the cladding and subsequent release of fission products is 900°C; that is, the point at which the hydrogen pressure equals the yield strength of the cladding. This point for FLIP fuel (H-Zr 1.6) is 940°C. Thus, the operating conditions of the reactor must be limited such that in the event of a loss of water accident the fuel cladding temperature does not exceed the above limiting values.

The maximum fuel cladding temperature after a loss of pool water as a function of fuel rod power density is shown in Figure 13-2. This curve was generated from data calculated with a two-dimensional transient transport computer code TAC developed by Gulf General Atomic. It was assumed that the fuel has been operated for 7700 MW-days, and that the reactor is shut down 15 minutes prior to the loss of coolant. The 15-minute delay was selected because it would take approximately 15 minutes for the pool to drain down from a point where a low level alarm would occur to the point where the core was uncovered from the catastrophic failure of a 10-inch beam port.

The results of the TAC code calculations summarized in Figure 13-2 indicate that Standard and FLIP fuel rods operated at power densities up to 22.3 and 23.5 kW/rod, respectively, would not fail in the event of a loss of coolant accident. That is, the loss of coolant immediately after shutdown from 21 continuous years of operation at full power would not produce a fuel cladding failure and subsequent release of fission products.

An examination of the power density data given in Section 4.5.3 of this report indicates that all the analyzed core configurations have fuel rod power densities below the above-mentioned limiting values. Thus, a loss of coolant accident would not precipitate a fuel cladding failure and the fission product decay heat would be removed primarily by natural convection of air.

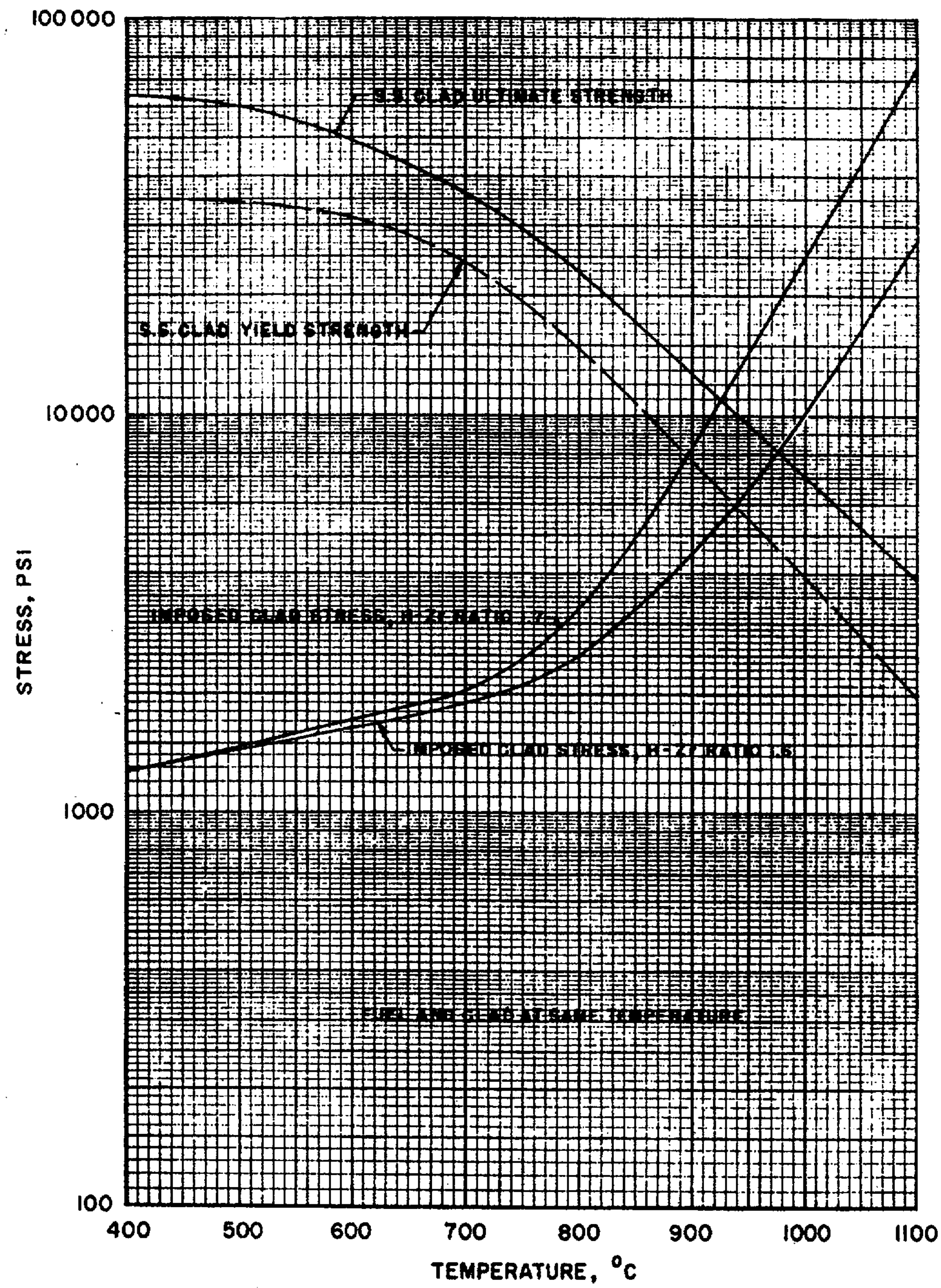

Figure 13-1: Strength and Applied Stress as a Function of Temperature for 1.7 and 1.6 H-Zr TRIGA Fuel

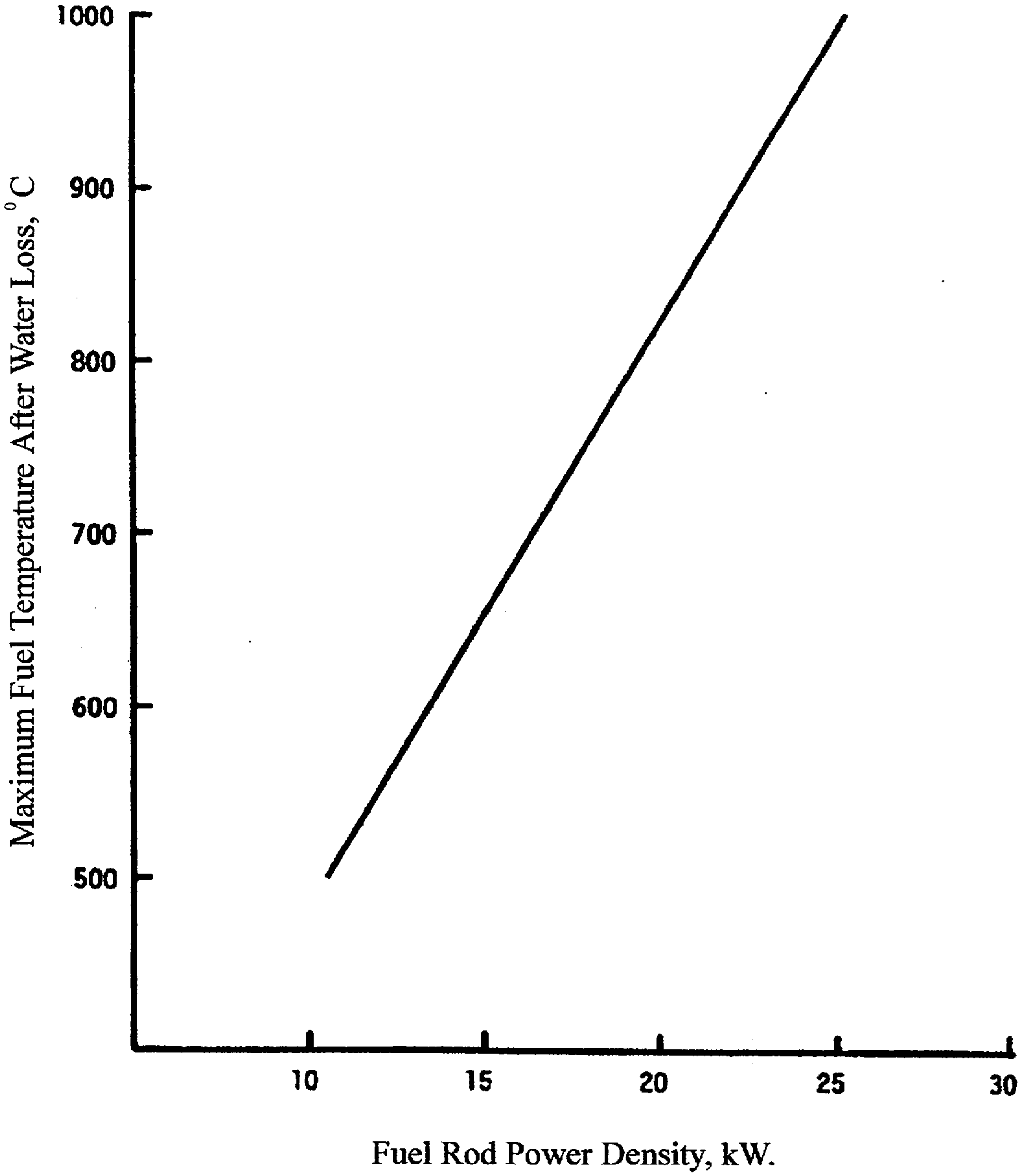

Figure 13-2: Maximum Fuel Rod Temperature as a Function of Fuel Rod Power Density for a Loss of Coolant 15 Minutes After Shut Down from 7700 MW-Days of Operation

Even though the possibility of a loss of coolant which is also shielding water is very remote, calculations have been performed to evaluate the radiological hazards associated with this type of accident. The radiation dose rates calculated are given in Table 13-12 and are based on the assumption that the reactor has been operating for a very long time at a power level of 1 MW prior to the loss of all the shielding water. The times listed in the Table are after shutdown of the reactor from full power. The first location is directly on top of the reactor at the bridge level which is 23.5 feet above the top of the actual fueled portion of the core. The second location is at the pool room floor level at the freight door at the east end of the pool room. This second location is shielded from direct radiation from the core but subjected to scattered radiation from the ceiling of the pool room which is 21.8 feet above the top of the normal water level of the pool. The ceiling is assumed to be thick concrete yielding the maximum possible reflected radiation dose which is thus a conservative estimate, since the actual roof structure would yield much less back scattered radiation.

Table 13-12

Calculated Radiation Exposure Rates in the
Event of a Loss of Pool Water Accident

Time after Shutdown	Reduction Factor Due to Decay	Direct Radiation rem/hr	Scattered Radiation rem/hr
10 sec	1	7.35×10^3	0.25
1 hr	2.70	2.72×10^3	.093
1 day	8.67	8.48×10^2	.025
1 week	18.84	3.96×10^2	.015
1 month	72.2	102	.0035

The data given in Table 13-12 was calculated assuming that the bare unshielded core is a cylindrical source of 1 MeV photons with a uniform source distribution. The dimensions of the cylinder were taken equal to the active core lattice which has a radius of 29.1 cm, height of 38.1 cm, and a volume of 5.75×10^4 cm^3. The source strength as a function of time was determined from Perkins and King's data (4) on fission product decay. No accounting was made for sources other than fission product decay gammas or for attenuation through the fuel rod end pieces, core support structure, or bridge deck plate. It is also assumed that no buildup occurs in the core. The sum total effect of these simplifying assumptions is a conservative (over estimation) of the dose rates.

The direct dose rate at distance above the core large compared to the core height may be approximated by (6,7):

$$D = \frac{S_v\ r^2}{4\mu\ a^2\ K}\left(1 - e^{-\mu h}\right)$$

where: Sv = source strength of (1 MeV) photons, photons/cm^3-sec

r = core radius, cm
μ = core attenuation coefficient (0.207 cm^{-1})
h = core height, cm
a = distance from surface to dose point, cm
K = flux-to-dose conversion factor = 5.77 x 10^5 photons/cm^3-sec/rem/hr.
At $t = 0$, $Sv = 3.9 \times 10^{12}$ photons/cm^3-sec.

The methodology described above and that for calculating the scattered dose rate are described in detail in SARs that have been submitted and approved by the Commission (6,7).

13.1.4 Loss of Coolant Flow

The WSU Modified TRIGA reactor is cooled by natural convection cooling and thus there is no primary coolant system that could fail to cause a loss of coolant flow type accident. A loss of secondary coolant flow that cools the reactor pool would not cause a reactor accident but only limit the number of hours that the reactor could be operated without pool cooling. A siphon break in the primary coolant line also prevents a primary coolant break from draining the pool.

13.1.5 Mishandling or Malfunction of Fuel

Over the years there have been a number of TRIGA fuel rods damaged during fuel movement and fuel rod inspection. The most frequent event is the dropping of a single fuel rod out of the fuel handling device and subsequent damage of the fuel rod end upon striking the floor. Dropped fuel rods should not be used until they have been inspected for damage. The following analysis of fuel rod inspection is designed to insure fuel rod integrity but limit mishandling accidents.

The rapid increase in power and the resultant increase in fuel temperature in a TRIGA reactor during pulsing subjects the fuel rod cladding to stress and to thermal cycling effects. In order to insure that the fuel rod cladding integrity has not significantly deteriorated, it is customary to inspect the fuel rods periodically. This inspection at some specified interval of time involves checking the transverse bending and elongation of TRIGA fuel rods.

In order to inspect the fuel rods, they must be removed from the reactor core and placed in a jig or fixture in the reactor pool. This operation involves a considerable amount of manipulation of the fuel rods using underwater handling tools. During the manipulation there is a possibility that physical damage to the fuel rod may result from mishandling. A few fuel rods have even been dropped during such operations at some facilities. While it is important that adequate inspection frequency be maintained to guard against possible pulsing induced damage, it is also important to minimize the number of inspections in order to reduce the possibility of physical damage to the fuel rods.

The strain produced during the pulsing transient results from the stress of internal pressure in the heated rod and the differential expansion of the fuel-moderator rod and cladding. The increased pressure results from the increased temperature of the air in the cladding gap, the fission products released from the fuel, and the hydrogen released from the partial disassociation of the zirconium hydride. Actual measurements made by General Atomic on specially instrumented fuel rods during pulsing reveal equilibrium pulsing pressure increase to be only about 20 psia (8).

A .25-inch gap is provided in a TRIGA fuel rod between the lateral end of the graphite reflector and the end piece welded onto the cladding. This gap reduces to about half this value (9) at a fuel rod temperature of 1200°C and a cladding temperature of 200°C. Because of this gap, differential expansion during pulsing will not produce a significant amount of lateral strain on the fuel rod cladding.

The predominant and most significant effect that pulsing has on the cladding is that of radial differential expansion in new, unpulsed rods. Near the middle region of the fuel rod the uranium-zirconium hydride is in close contact with the stainless steel cladding. The effects of differential expansion between the fuel rod and the cladding is greatest in the middle region. Assuming a temperature increase of 750°C for the fuel-moderator rods (which is three times the expected average value for a $2.50 pulse and 200°C for the cladding, the amount of strain which is equal to the fractional increase in the cladding circumference due to the fuel rod expansion is calculated as follows:

1. Change in circumference of cladding due to increase in temperature above nominal 25°C

   ΔC_c = Circumference x Cladding Linear Coefficient of Expansion x Temperature Change = $1.41 \times \pi \times 17 \times 10^{-6} (200 - 25) = .0132$ in.

2. Increased area of fuel rod due to increase in temperature above nominal 25°C

   A_t = Area (1 + 2 x Fuel Linear Coefficient of Expansion** x Temperature Change)

   $$A_t = \left(\frac{1.36}{2}\right)^2 \times \pi\left[1 + 2 \times 14.2 \times 10^{-6} \times 750\right]$$

   $A_t = 1.452672 \times 1.0213 = 1.483614 \text{ in.}^2$

3. Increase in circumference of fuel moderator rod

   $$\Delta C_r = \pi\left(D_r - D_c\right) = 2\pi\left(r_r - r_c\right) = 2\pi\left[\left(\frac{A_t}{\pi}\right)^{1/2} - \left(\frac{1.36}{2}\right)\right]$$

   $\Delta C_r = 2\pi(.68577 - .6800) = .0453$ in.

4. Difference between circumference of heated cladding and heated fuel rods = strain

   $\Delta C_r - \Delta C_c = .0453 - .0132 = .0321$ in.

5. Radial strain on cladding due to differential radial thermal expansion

   $$\text{Strain} = \text{Fractional Deformation} = \frac{\text{Increase in Cladding Circumference}}{\text{Cladding Circumference}}$$

   Strain = .0321/1.41 = .0072 or 72% strain.

**$\alpha = 14.2 \times 10^{-6}$/°C over range 200°C to 850°C.

Cracks may start to appear at about 10% of the cycles at which fracture occurs. Applying the 10% factor to the above calculation, a conservative estimate of 3,600 cycles of pulsing will occur before the onset of possible cracking that could cause a fission product leak.

The above amount of strain would cause some permanent deformation in the cladding but is well within the safety limit for the expansion of type 304 stainless steel. The fact that some permanent deformation is produced is substantiated by the fact that the heat transfer between the fuel-moderator rod and the cladding of a TRIGA fuel rod decreases in new fuel after pulsing. In other words, the time required for new instrumented TRIGA fuel rods to cool down from a given temperature to ambient increases after pulsing. This factor is indicative of an increase in the space between the fuel rod and the cladding due to permanent stretching of the cladding by pulsing.

The amount of strain produced during extended pulsing will in actual fact be significantly lower than that calculated above. The permanent deformation of the cladding will obviously reduce the value of the strain. The actual fuel rod temperature will also be below that assumed in the calculation. Furthermore, during pulsing, film boiling causes an increase in the fuel cladding temperature somewhat above the steady-state case. All of these factors will decrease the magnitude of the strain so that the value calculated above is conservative.

Studies made for NASA (10) on low-cycle fatigue indicate that the cladding could receive over 36,000 cycles of the postulated strain previous to the appearance of cracks that could allow fission product gases to escape. This value was obtained by the following consideration

$$N = \left(K / S_p\right)^2 = \left(.69 / \frac{.0072}{2}\right)^2 = 36{,}737$$

where N = number of cycles previous to failure

S_p = the plastic strain (less one-half of the total strain on the cladding)

K = a measured constant for a given material which is related to fracture (for type 304 stainless steel, K has been measured to be .69).

The deleterious effects on FLIP TRIGA fuel of frequent pulsing to a high power level became clearly evident when a FLIP TRIGA fuel problem developed at the Texas A&M modified TRIGA reactor in 1976. The report of the damaged FLIP TRIGA fuel rods of November 1, 1976 was filed by Texas A&M with the Atomic Energy Commission, the predecessor to the U.S. Nuclear Regulatory Commission. That report indicates that the damaged fuel rods had not leaked fission products but indicates that "bulges in the cladding and unevenly spaced, dark rings are clearly evident, with the most severe bulges occurring in the center of the elements and the rings being spaced closest there. . ." That is, there was significant visual evidence of rod damage without a cladding failure occurring or even any fission products leaking from the damaged rods. A subsequent reevaluation of the characteristics of TRIGA fuel by General Atomics entitled "The U-ZrHx Alloy: Its Properties and Use in TRIGA Fuel, E-117-833, by M.T. Simnad, of February 1980", again established the maximum safe fuel operating temperature for FLIP fuel of 1150°F. However, as a result of the lessons learned from the Texas A&M fuel rod damage, the temperature of FLIP rods during pulsing should not exceed 830°F. This limit is used at the WSU reactor and is covered in more detail in section 4.8 of this SAR.

The WSU reactor is similar to the Texas A&M facility so on 10/18/76 the two 4-rod clusters adjacent to the pulse rod where the damage had occurred in the Texas A&M reactor were removed and inspected under water with a video system. These clusters, did not, however, show any of the obvious damage experienced at the Texas A&M facility. The pulsing of the WSU reactor was much less than done at the Texas A&M facility and to lower power levels. Never the less, all subsequent pulsing of the WSU facility was limited to reactivity insertion that would not produce a fuel rod temperature above 830°F.

The new pulsing limit for the WSU TRIGA reactor fueled with a mixed core is $2.20. If the core was pulsed 3,600 times with the maximum allowable pulse, this limit would amount to a total of about $8,000 worth of pulses. As a prudent rule one could set an inspection frequency at 50% of the expected pulsing life or a total of $4,000 worth of $2.20 pulses. This minimum inspection frequency limit is established for the WSU TRIGA reactor in which all the fuel is inspected.

In actual operation a variety of sizes of pulses up to the maximum allowable value are shot. Only those pulses above $1.50 will produce a fuel temperature higher than that attained during normal steady-state operation. Thus, in making the tabulation for fuel rod inspection, only those pulses over $1.50 should be summed for inspection purposes.

It is evident from the Texas A&M fuel rod damage problem that significant fuel rod damage can occur without a cladding rupture with subsequent release of fission products. Accordingly, it is prudent to visually inspect the fuel rods adjacent to the pulse rod more frequently than the previous failure analysis dictates. The entire core is to be visually inspected at a $3,500 pulsing worth frequency.

13.1.6 Experiment Malfunction

The limits placed on experiments and irradiations in sections 3.10 and 3.11 of the Technical Specifications have been formulated to insure that a serious reactor accident will not be precipitated by the failure of an experiment or irradiation. The key limits are:

1. Nonsecured experiments shall have reactivity worths less than 1.00$.
2. The reactivity worth of any single experiment shall not exceed 2.00$.
3. Total worth of all experiments will not exceed 5.00$.
4. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 mg shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 mg may be irradiated in the reactor or experimental facilities, 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.
5. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (a) normal operating conditions of the experiment or reactor, (b) credible accident conditions in the reactor, or (c) possible accident conditions in the experiment, shall be limited in activity so 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 applicable limits of Appendix B of 10 CFR 20.

These limiting conditions placed on experiments and irradiations are based on the following considerations:

1. This first specification is intended to provide assurance that the worth of a single unsecured experiment will be limited to such a value that the safety limit will not be exceeded if the positive worth of the experiment were to be suddenly inserted.
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 an experiment 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 so that the reactor protective systems would act to prevent power levels from exceeding the safety limits.
3. The total worth of all experiments is limited to ensure that the reactor will remain subcritical in the event of a simultaneous removal of all of the experiments with one safety control element withdrawn.
4. Specification 4 is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials.
5. The last specification is intended to reduce the likelihood that radioactive airborne particles in excess of the limits of Appendix B of 10 CFR 20 will be released to the atmosphere outside the facility.

13.1.7 Loss of Normal Electrical Power

Over the years electrical power to the facility and reactor has been lost a number of times while the reactor has been operating. That is, power distribution problems on campus or with the electrical supplier to the campus have cutoff power to the facility. In these instances, the reactor just automatically shuts down (scrams) without incident. Thus a power loss is only an inconvenience and does not and can not precipitate any type of accident.

13.1.8 External Events

The only external events that could impact the operation of the reactor is a significant earthquake which was considered in section 2.5 of this SAR and the crashing of a commercial aircraft into the west end of the reactor control room which is analyzed in Appendix C. Though the likelihood of a significant earthquake is very remote, the reactor control system contains a seismic switch which will automatically shut down the reactor in the event of a significant earthquake. The slosh effect in the WSU reactor pool as a consequence of a significant earthquake was calculated by the method given in section 11 of IAEA-TECDOC-348. Using the worst case, the period of the slosh would be about 5 seconds and the wave height would be about 3 cm. That is, the dam in the center of the pool structure inhibits the development of a significant slosh effect. The slosh effect thus is an insignificant hazard to the reactor bridge structure in the event of a significant earthquake. The worst thing that a large earthquake could cause is the total loss of pool water which has been analyzed in section 13.1.3 of this SAR. Appendix C substantiates the fact that an aircraft crash precipitated significant reactor accident is not credible.

13.1.9 Mishandling or Malfunction of Equipment

The entire control system of the reactor is designed for "fail safe operation" as previously discussed in section 7 of this SAR. Over the years numerous items of equipment have failed that only resulted in shutting down of the reactor until the malfunction was repaired. All conceivable malfunctions only cause an inconvenience and inability of the reactor to be operated until repairs are completed. No conceivable equipment malfunction would precipitate a reactor accident of a type that has not already been analyzed.

13.2 Accident Analysis and Determination of Consequences

The only credible accidents that have any significant consequences for a TRIGA type reactor are the MHA and LOCA analyzed in section 13.1.1 and 13.1.4 of this report. In order to precipitate the MHA a sudden complete loss of pool water must occur which as shown in section 13.1.3 would not precipitate a fuel rod cladding failure. However, for the purposes of the MHA, the failure of one fuel rod is assumed to occur with the subsequent release of the fission products contained in that fuel rod. The radionuclides released under the postulated accident are given in Table 13-1. The only likely event that could possibly precipitate the MHA is a severe earthquake causing a rupture of the reactor pool. Such an event is extremely unlikely as shown in section 3.4 of this SAR. An earthquake of sufficient magnitude to fracture the pool could also do significant damage to the reactor building.

The worst case condition for an individual in the pool room when the accident occurs is given in Table 13-4. Assuming a 5 minute exposure before evacuation, the whole body exposure would be .05 rem and the thyroid exposure would be 3.5 rem which are well below the acceptable limits stated at the end of section 13.3. The worst case exposure outside the facility for the case where the ventilation system is left in the normal mode is given in Table 13-6B. In this case a one hour exposure would result in a whole body exposure of 1.9 mrem and thyroid exposure of 145 mrem. This would be the maximum possible exposure since some type of mitigating action would be taken such as limiting access to the reactor facility area, etc.

In actual practice, a LOCA with the complete uncovering of the core could constitute a greater potential radiological hazard to the reactor staff than the MHA. The exposure rates at the bridge level and at the least shielded portion of the reactor facility are given in Table 13-12. The exposure values given are similar to those for all of the existing 60 TRIGA type reactors. A one minute exposure at the bridge level immediately following a complete loss of water with the reactor having been operated at full power for a very long time would be of the order of 100 rem. However, the scattered radiation to individuals outside the facility could not exceed about .15 rem in one hour whole body.

13.3 Summary and Conclusions

The preceding paragraphs of this section of the SAR concerning accident analysis clearly demonstrate that the relicensing of the WSU modified TRIGA reactor and the associated operation of that reactor with Standard and FLIP fuel does not in any conceivable manner create a significant threat to the health and safety of the reactor staff and general public. The maximum possible radiation exposure to an individual outside the facility under the postulated conditions is minimal. The exposures are significantly below the generally acceptable accident results for non-power reactors of not more than 5 rem whole body and 30 rem thyroid for occupational exposure and not more than .5 rem whole body and 3 rem thyroid for members of the general public. In addition, the calculated accident exposures are well below the maximum values established in 20.1201 for occupational exposure and 20.1301 for public exposure. Thus, no realistic hazard to the staff at the reactor as well as the general public would result from any postulated accident event. Furthermore, the analysis provided for FLIP type HEU fuel also applies to LEU fuel if and when the reactor switches to LEU fuel. That is, the two key parameters for the fuel which are the U-ZrH ratio and U-235 content are essentially equal for FLIP and LEU fuel as given in Table 1-1.

13.4 References

1. "Amendment 1 to Safety Analysis Report of October, 1966" for the use of FLIP fuel in the WSU reactor, Washington State University, May 1974.

2. "Safety Analysis Report for the Washington State University Modified TRIGA Nuclear Reactor," Washington State University, May 1979.

3. "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA-400-R-92-001, U.S. Environmental Protection Agency, May 1992.

4. "Summary of TRIGA Fuel Fission Product Release Experiments," Gulf-EES-A10801, September 1971.

5. "Energy Release from the Decay of Fission Products," Nuclear Science and Engineering, Vol. 3, Pg. 726, 1968, Perkins, J.F. and King, R.W.

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APPENDIX A

FACILITY LICENSE NO. R-76

TECHNICAL SPECIFICATIONS

AND BASES

FOR THE

WASHINGTON STATE UNIVERSITY

MODIFIED TRIGA REACTOR

DOCKET NO. 50-27

Updated Version of April 26, 2002

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TECHNICAL SPECIFICATIONS AND BASES FOR THE
WASHINGTON STATE UNIVERSITY MODIFIED TRIGA REACTOR

This document constitutes the Technical Specifications for Facility License No. R-76 and supersedes all prior Technical Specifications. Included in these Technical Specifications are the "Bases" to support the selection and significance of the specification. These bases are included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere. Furthermore, the dimensions, measurements, and other numerical values given in these specifications may differ slightly from actual values because of normal construction and manufacturing tolerances, or normal degree of accuracy or instrumentation.

1.0 DEFINITIONS

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

1.1 Reactor Operating Conditions

Abnormal Occurrence: An abnormal occurrence is defined for the purposes of the reporting requirements of Section 208 of the Energy Reorganization Act of 1974 (PL 93-438) as an unscheduled incident or event which the Nuclear Regulatory Commission determines is significant from the standpoint of public health or safety.

Cold Critical: The reactor is in the cold critical condition when it is critical with the fuel and bulk water temperature both below 40°C.

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

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

Reactor Secured: The reactor is secured when all the following conditions are satisfied:

(1) The reactor is shut down.

(2) 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.

(3) 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.

Reactor Shutdown: The reactor is shut down when the reactor is subcritical by at least 1.00$ of reactivity.

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 Section 2.2, "Limiting Safety System Settings"

(2) operation in violation of a limiting condition of operation listed in Section 3.0

(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 1.00$

(5) an observed inadequacy in the implementation of either administrative or procedural controls, to such degree 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) release of fission products into the environment

Shutdown Margin: Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that (1) the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating conditions, and (2) the reactor will remain subcritical without further operator action.

Steady-State Mode: Steady-state mode operation shall mean any operation of the reactor with the mode selector switch in the steady-state position.

1.2 Reactor Experiments and Irradiations

Experiment: Experiment shall mean: (1) any apparatus, device or material which is not a normal part of the core or experimental facilities, but which is inserted into these facilities or is in line with a beam of radiation originating from the reactor, or (2) any operation designed to measure reactor parameters or characteristics.

Experimental Facilities: Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, in-core irradiation baskets or tubes, 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 irradiation facility so that the device or material is exposed to a significant amount of the radiation available in that irradiation facility.

Irradiation Facilities: Any in-pool experimental facility that is not a normal part of the core and that is used to irradiate devices and materials.

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 components

1.3 Reactor Component

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, a burnable poison.

Fuel Bundle: A fuel bundle is a cluster of three or four fuel rods fastened together in a square array by a top handle and bottom grid plate adapter.

Fuel Rod: A fuel rod is a single TRIGA-type fuel rod of either Standard or FLIP-type fuel.

Instrumented Fuel Rod: An instrumented fuel rod is a special fuel rod in which thermocouples have been embedded for the purpose of measuring the fuel temperatures during reactor operation.

Mixed Core: A mixed core is a core arrangement containing Standard and FLIP-type fuels with at least 22 FLIP fuel rods located in the central positions in the core.

Operational Core: An operational core is any arrangement of TRIGA fuel that is capable of operating at the maximum licensed power level and that satisfies all the requirements of the Technical Specifications.

Regulating Control Element: Regulating control element shall mean a low worth control element that may be positioned either manually or automatically by means of an electric motor-operated positioning system and that need not have a scram capability.

Standard Control Element: Standard control element shall mean any control element that has a scram capability, that is utilized to vary the reactivity of the core, and that is positioned by means of an electric motor-operated positioning system.

Standard Core: A standard core is any arrangement of all-Standard fuel.

Standard Fuel: Standard fuel is TRIGA fuel that contains a nominal 8.5 weight percent of uranium with a ^{235}U enrichment of less than 20%.

Transient Control Element: Transient control element shall mean any control element that has the capability of being rapidly withdrawn from the reactor core by means of a pneumatic drive, that is capable of being positioned by means of an electric motor-operated positioning system, and that has scram capabilities.

1.4 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 variables.

Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification may include comparison with independent channels measuring the same variable or other measurements of the variables.

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, that are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information that requires manual protective action to be initiated.

Limiting Safety Systems Setting: Limiting safety systems settings are the 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 devices that 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.

PTR (Peak-to-Measured-Fuel Temperature Ratio): The PTR is defined as the ratio between the maximum calculated fuel temperature in a given core arrangement to that measured by the instrumented fuel element.

Reactor Safety Systems: Reactor safety systems are those systems, including their associated input circuits, designed to initiate a scram for the primary purpose of protecting the reactor or to provide information that requires 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 that are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Fuel Element Temperature

Applicability: This specification applies to the temperature of the reactor fuel.

Objective: The objective is to define the maximum fuel temperature that can be permitted with confidence that a fuel cladding failure will not occur.

Specifications:

(1) The maximum temperature in a Standard TRIGA fuel rod shall not exceed 1000°C under any condition of operation.

(2) The maximum temperature in a FLIP-type TRIGA fuel rod shall not exceed 1150°C under any condition of operation.

Bases: The important parameter for a TRIGA reactor is the fuel rod temperature. This parameter is well-suited as a single specification, especially since it can be measured. A loss in the integrity of the fuel rod 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 the presence of air, fission product gases, and hydrogen from the disassociation of the hydrogen and zirconium in 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 is based on data that indicate that the stress in the cladding because of the hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided the temperature of the fuel does not exceed 1150°C and the fuel cladding is water cooled.* The safety limit for the Standard TRIGA fuel is based on data, including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding because of hydrogen pressure from the disassociation 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.*

2.2 Limiting Safety System Settings

Applicability: This specification applies to the settings that prevent the safety limit from being reached.

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

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*GA-9064, Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, submitted under Docket No. 50-227.

Specifications: The limiting safety system settings shall be 500°C as measured in an instrumented fuel rod located in the central region of the core. For a mixed core, the instrumented rod shall be located in the region of the core containing the FLIP-type fuel rods.

Bases: The limiting safety system setting is the measured instrumented fuel rod temperature that, if exceeded, shall initiate a scram to prevent the fuel temperature safety limit from being exceeded. Section 5.4 of the FLIP conversion safety analysis report for the Washington State University (WSU) TRIGA reactor indicated that a 500°C safety system setting would limit the maximum possible steady-state fuel temperature in the FLIP fuel region to less than 800°C. This setting provides at least a 350°C margin of safety for FLIP fuel and at least a 200°C margin of safety for Standard fuel.

In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will not limit the peak power generated during the pulse because of the relatively long response time of the temperature channel as compared with the width of a pulse. On the other hand, the temperature scram would limit the total amount of energy generated in a pulse by cutting off the "tail" of the energy transient in the event that the fuel temperature limit is exceeded. Thus, the fuel temperature scram provides an additional degree of safety in the pulse mode of operation to protect the fuel in the event of such conditions as sticking of the transient control element in the withdrawn position after a pulse.

3.0 LIMITING CONDITIONS OF OPERATION

3.1 Steady-State Operation

Applicability: This specification applies to the energy generated in the reactor during steady-state operation.

Objective: The objective is to ensure that the fuel temperature safety limit will not be exceeded during steady-state operation.

Specifications: The reactor power level shall not exceed 1.3 MW under any condition of operation.

Basis: Thermal and hydraulic calculations performed by the vendor indicate that TRIGA fuel may be safely operated up to power levels of at least 2.0 MW with natural convection cooling.

3.2 Reactivity Limitations

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worth of control elements and experiments. They apply for all modes of operation.

Objective: The objective is to ensure that the reactor can be shut down at all times and to ensure that the fuel temperature safety limit will not be exceeded.

Specifications: The reactor shall not be operated unless the shutdown margin provided by control elements shall be 0.25$ or greater with:

(1) the highest worth nonsecured experiment in its most reactive state

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

(3) the reactor in the cold critical condition without xenon

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

3.3 Pulse Mode Operation

Applicability: This specification applies to the peak fuel temperature in the reactor as a result of a pulse insertion of reactivity.

Objective: The objective is to ensure that fuel element damage does not occur in any fuel rod during pulsing.

Specifications: The maximum reactivity inserted during pulse mode operation shall be such that the peak fuel temperature in any fuel rod in the core does not exceed 830°C. The maximum safe allowable reactivity insertion shall be calculated annually for an existing core and prior to pulsing a new or modified core arrangement.

Basis: TRIGA fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 for FLIP fuel and 1.65 for Standard. This yields delta phase zirconium hydride which has a high creep strength and undergoes no phase changes at temperatures over 1000°C. However, after extensive steady-state operation at 1 MW, the hydrogen will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions. When the fuel is pulsed, the instantaneous temperature distribution is such that the highest values occur at the surface of the element and the lowest values occur at the center. The higher temperatures in the outer regions occur in fuel with a hydrogen to zirconium ratio that has now substantially increased above the nominal value. This produces hydrogen gas pressures considerably in excess of that expected for $ZrH_{1.6}$. If the pulse insertion is such that the temperature of the fuel exceeds 874°C, then the pressure will be sufficient to cause expansion of microscopic holes in the fuel that grow larger with each pulse. The expansion of the fuel stresses and distorts the fuel rod material which, in turn, can cause overall swelling and distortion of the cladding and entire fuel rod. The pulsing limit of 830°C is obtained by examining the equilibrium hydrogen pressure of zirconium hydride as a function of temperature. The decrease in temperature from 874°C to 830°C reduces hydrogen pressure by a factor of two, which provides an acceptable safety factor. This phenomenon does not alter the steady-state safety limit since the total hydrogen in a fuel element does not change. Thus, the pressure exerted on the clad will not be significantly affected by the distribution of hydrogen within the element.

3.4 Maximum Excess Reactivity

Applicability: This specification applies to the maximum excess reactivity, above cold critical, which may be loaded into the reactor core at any time.

Objective: The objective is to ensure that the core analyzed in the safety analysis report approximates the operational core within reasonable limits.

Specifications: The maximum reactivity in excess of cold, xenon-free critical shall not exceed 5.6% $\Delta k/k$ (8.00$).

Basis: Although maintaining a minimum shutdown margin at all times ensures that the reactor can be shut down, that specification does not address the total reactivity available within the core. This specification, although over-constraining the reactor system, helps ensure that the licensee's operational power densities, fuel temperatures, and temperature peaks are maintained within the evaluated safety limits. The specified excess reactivity allows for power coefficients of reactivity, xenon poisoning, most experiments, and operational flexibility.

3.5 Core Configuration Limitation

Applicability: This specification applies to mixed cores of FLIP and Standard types of fuel.

Objective: The objective is to ensure that the fuel temperature safety limit will not be exceeded as a result of power peaking effects in a mixed core.

Specifications:

(1) The FLIP-fueled region in a mixed core shall contain at least 22 FLIP fuel rods in a contiguous block of fuel in the central region of the reactor core. Water holes in the FLIP region shall be limited to nonadjacent single-rod holes.

(2) The PTR as defined in Section 1.4 and as calculated by the method used in the FLIP conversion safety analysis report shall not exceed 1.5 for an operational core.

Bases: The limitation on the allowable core configuration as set forth in Section 4.1 of the FLIP fuel conversion safety analysis report limits power peaking effects. The limitation on power peaking effects ensures that the fuel temperature safety limit will not be exceeded in a mixed core.

A 500°C safety system setting and a 1.5 PTR limit the maximum possible steady-state fuel temperature in the FLIP region to less than 800°C.

3.6 Control and Safety System

3.6.1 Scram Time

Applicability: This specification applies to the time required for the scrammable control rods 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 from the instant that a safety system setting is exceeded to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds. For purposes of this section, the above specification shall be considered to be satisfied when the sum of the response time of the slowest responding safety channel, plus the fall time of the slowest scrammable control rod, is less than or equal to 2 seconds.

Basis: This specification ensures 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 ensure the safety of the reactor.

3.6.2. Reactor Control System

Applicability: This specification applies to the information that 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 ensure safe operation of the reactor.

Specifications: The reactor shall not be operated in the specified mode of operation unless the measuring channels listed in Table 3.1 are operable.

Table 3.1 Measuring Channels

Measuring Channel	Min. no. operable	Effective mode	
		SS	pulse
Fuel element temperature	1	X	X
Linear power level	1	X	
Log power level	1	X	
Integrated pulse power	1		X

Note: SS = steady-state

Bases: Fuel temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit. The power level monitors ensure that the reactor power level is adequately monitored for both steady-state and pulsing modes of operation. The specifications on reactor power level indication are included in this section since the power level is related to the fuel temperature.

3.6.3 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 system channels that must be operable for safe operation.

Specifications: The reactor shall not be operated unless the safety channels described in Table 3.2 are operable.

Table 3.2 Minimum Reactor Safety Channels

		Number operable in specified mode	
Safety Channel	Function	SS	Pulse
Fuel temperature	Scram if fuel temperature exceeds 500°C	1	1
Power level	Scram if power level exceeds 125% of full licensed power	1	
Manual scram	Manually initiated scram	1	1
Wide range	Prevent initiation of a pulse above 1 kW		1
	Prevent control element withdrawal when neutron count is less than 2 cps	1	
High-voltage monitor	Scram on loss of high voltage to power channels	1	1
Pulse-mode switch	Prevent withdrawal of standard control and regulation elements in pulse mode		1
Preset timer	Transient rod scram 15 seconds or less after pulse		1
Pool level	Alarm if pool level falls below 16 ft over the core	1	1
Transient rod control	Prevent application of air unless fully inserted	1	

Note: SS = steady-state

Bases: The fuel temperature and power level scrams provide protection to ensure that the reactor can be shut down before the safety limit on the fuel element temperature will be exceeded. The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs. In the event of failure of the power supply for the safety chambers, operation of the reactor without adequate instrumentation is prevented. The preset timer ensures that the reactor power level will reduce to a low level after pulsing. The interlock to prevent startup of the reactor with less than 2 cps ensures that sufficient neutrons are available for proper startup.

The interlock to prevent the initiation of a pulse above 1 kW is to ensure that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded. The interlock to prevent withdrawal of the standard or regulating control elements in the pulse mode is to prevent the reactor from being pulsed while on a positive period. The pool level alarm is intended to alert the operator to any significant decrease in the pool level.

3.7 Radiation Monitoring System

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 ensure that sufficient radiation monitoring is available to the operator to ensure safe operation of the reactor.

Specifications: The reactor shall not be operated unless the radiation monitoring channels listed in Table 3.3 are operable. Each channel shall have a readout in the control room and be capable of sounding an audible alarm that can be heard in the reactor control room.

Basis: The radiation monitors inform operating personnel about 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.

Table 3.3 Minimum Monitoring Channels

Channel*	Function	No.
Area radiation monitor	Monitor radiation level on the bridge	1
Area radiation monitor	Monitor radiation level in the beam room	1
Continuous air monitor	Monitor the activity of the pool room air	1
Exhaust gas monitor	Monitor the Argon-41 activity in the exhaust	1

*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 that shall be kept under visual observation.

3.8 Argon-41 Discharge Limit

Applicability: This specification applies to the concentration of ^{41}Ar that may be discharged from the WSU TRIGA reactor facility.

Objective: To ensure that the health and safety of the public are not endangered by the discharge of ^{41}Ar from the WSU TRIGA reactor facility.

Specification: The concentration of ^{41}Ar 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: The maximum allowable concentration of ^{41}Ar in air in unrestricted areas as specified in Appendix B, Table II of 10 CFR 20 is 1×10^{-8} μCi/ml. Section 6.5 of the safety analysis report for conversion of the WSU TRIGA reactor to FLIP fuel substantiates a 3.4×10^{-3} atmospheric dilution factor for a 4.4 mph wind speed. A somewhat more conservative value of 4×10^{-3} has been selected for the calculation of ^{41}Ar dilution.

3.9 Engineered Safety Feature - Ventilation System

Applicability: This specification applies to the operation of the facility ventilation system.

Objective: The objective is to ensure 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 facility ventilation system is operable, except for periods of time not to exceed 48 hours to permit repair or testing of the ventilation system. In the event of a substantial release of airborne radioactivity within the facility, the ventilation system will be secured or operated in the dilution mode to prevent the release of a significant quantity of airborne radioactivity from the facility.

Basis: During normal operation of the reactor and the ventilation system, the concentration of ^{41}Ar and other airborne radionuclides discharged from the facility is below the applicable maximum air effluent concentration (AEC) values. In the event of a substantial release of airborne radioactivity within the facility, the ventilation system will be secured or operated in a dilution mode as appropriate. This action will permit minimizing the concentration of airborne radioactive materials discharged to the environment until it is within the appropriate AEC value. In addition, operation of the reactor with the ventilation system shut down for short periods of time to make system repairs or tests does not compromise the control over the release of airborne radioactive materials. Moreover, radiation monitors within the building, independent of the ventilation system, will give warning of high levels of radiation that might occur during operation with the ventilation system secured.

3.10 Limitations on Experiments

Applicability: This specification applies to experiments installed in the reactor and its experimental facilities (defined in Section 1.2).

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

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

(1) Nonsecured experiments shall have reactivity worths less than 1.00$.

(2) The reactivity worth of any single experiment shall not exceed 2.00$.

(3) Total worth of all experiments will not exceed 5.00$.

(4) Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 mg shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 mg may be irradiated in the reactor or experimental facilities, 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.

(5) Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (a) normal operating conditions of the experiment or reactor, (b) credible accident conditions in the reactor, or (c) possible accident conditions in the experiment, shall be limited in activity so 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 applicable limits of Appendix B of 10 CFR 20.

In calculations pursuant to item 5 above, the following assumptions shall be used:

- 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.

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

- For materials whose boiling point is above 60°C and in cases 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.

- An atmospheric dilution factor of 4×10^{-3} for gaseous discharges from the facility.

(6) Each fueled experiment shall be controlled so that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 Ci.

(7) If a capsule fails and releases material that could damage the reactor fuel or structure by corrosion or other means, that material shall be removed and physically inspected to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the senior operator responsible for the operation and must be determined to be satisfactory before operation of the reactor is resumed.

Bases:

(1) This specification is intended to provide assurance that the worth of a single unsecured experiment will be limited to such a value that the safety limit will not be exceeded if the positive worth of the experiment were to be suddenly inserted.

(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 an experiment 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 so that the reactor protective systems would act to prevent power levels from exceeding the safety limits.

(3) The total worth of all experiments is limited to ensure that the reactor will remain subcritical in the event of a simultaneous removal of all of the experiments with one safety control element withdrawn.

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

(5) This specification is intended to reduce the likelihood that radioactive airborne particles in excess of the limits of Appendix B of 10 CFR 20 will be released to the atmosphere outside the facility.

(6) The 1.5-Ci limitation on iodine isotopes 131 through 135 ensures 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 20 for an unrestricted area.

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

3.11 Limitations on Irradiations

Applicability: This specification applies to irradiations performed in the irradiation facilities contained in the reactor pool as defined in Section 1.2, "Irradiation Facilities." Irradiations are a subclass of experiments that falls within the specifications hereinafter stated in this section. The surveillance requirements for irradiations are given in Section 4.3.5(2).

Objective: The objective is to prevent damage to the reactor, excessive release of radio-active materials, or excessive personnel radiation exposure during the performance of an irradiation.

Specifications: A device or material shall not be irradiated in an irradiation facility under the classification of an irradiation unless all the following conditions exist:

(1) The irradiation meets all the specifications of Section 3.10 for an experiment.

(2) The expected radiation field produced in air by the device or sample upon removal from the reactor pool is not more than 10 rem/hr beta and gamma equivalent at 1 ft; otherwise, it shall be classed as an experiment.

(3) The device or material is encapsulated in a suitable container.

(4) The reactivity worth of the device or material is 0.25$ or less; otherwise, it shall be classed as an experiment.

(5) The device or material does not remain in the reactor for more than a 15-day period; otherwise, it shall be classed as an experiment.

Basis: This specification is intended to provide assurance that the special class of experi-ments called irradiations will be performed in a manner that will not permit any safety limit to be exceeded.

3.12 As Low As Reasonably Achievable (ALARA) Radioactive Effluent Releases

Applicability: This specification applies to the measures required to ensure that the radio-active effluents released from the facility are in accordance with ALARA criteria.

Objective: The objective is to limit the annual population radiation exposure owing to the operation of the WSU TRIGA reactor to a small percentage of the normal local background exposure.

Specifications:

(1) In addition to the radiation monitoring specified in Section 5.4, an environmental radiation monitoring program shall be conducted to measure the integrated radiation exposure in and around the environs of the facility on a quarterly basis.

(2) The annual radiation exposure due to reactor operation, at the closest off-site point of extended occupancy, shall not, on an annual basis, exceed the average local off-site background radiation by more than 20%.

(3) Whenever practicable, the reactor shall be operated 4 in. or more from the thermal column in order to minimize the production of ^{41}Ar.

(4) The total annual discharge of ^{41}Ar into the environment shall not exceed 20 Ci per year.

(5) In the event of a significant fission product leak from a fuel rod or a significant airborne radioactive release from a sample being irradiated, as detected by the continuous air monitor, the reactor shall be shut down until the source of the leak is located and eliminated. However, the reactor may continue to be operated on a short-term basis as needed to assist in determining the source of the leakage.

(6) Before discharge, the facility liquid effluents collected in the holdup tanks shall be analyzed for their beta-gamma activity content. The total annual quantity of liquid effluents released (above background) shall not exceed 1 Ci per year.

Basis: The simplest and most reliable method of ensuring that ALARA release limits are accomplishing their objective of minimal facility-caused radiation exposure to the general public is to actually measure the integrated radiation exposure in the environment on and off the site.

3.13 Primary Coolant Conditions

Applicability: This specification applies to the quality of the primary coolant in contact with the fuel cladding.

Objectives: The objectives are (1) to minimize the possibility for corrosion of the cladding on the fuel elements, and (2) to minimize neutron activation of dissolved materials.

Specifications:

(1) Conductivity of the pool water shall be no higher than 5×10^{-6} mhos/cm.

(2) The pH of the pool water shall be between 5.0 and 7.5.

Basis: A small rate of corrosion continuously occurs in a water-metal system. In order to limit this rate and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limits provides acceptable control.

By limiting the concentrations of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the ALARA principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposures during maintenance and operations.

3.14 Sealed Sources in the Reactor Pool

Applicability: This specification applies to any and all sealed sources stored or used in the reactor pool.

Objective: The objectives of this requirement are to ensure that: 1) any sealed source or sources that are stored or used in the pool do not constitute any type of significant hazard to the operation of the reactor, 2) any such sealed source or sources do not create a significant environmental or personal radiation exposure hazard, and 3) any such sealed source or sources do not compromise the ALARA criteria of the facility.

Specifications:

(1) Sealed sources shall not at any time be stored or used closer than five (5) feet away from the face of an operating reactor core. The total activity of all sealed sources stored in the pool shall not exceed 100,000 curies. All sealed source configurations shall be designed so that a loss of pool water accident will not precipitate a sealed source incapsulation integrity problem and the sources shall be stored in an appropriate shield so as not to produce a significant radiation hazard in the event of a loss of reactor pool water accident.

(2) All storage of sealed sources greater than 100 curies in the reactor pool shall be considered as an experiment and shall be reviewed and approved by the Reactor

Safeguards Committee. A written operating procedure for the storage and use of sealed sources in the reactor pool shall be in effect.

(3) The radionuclide content of the reactor pool water shall be monitored monthly at an interval not to exceed six (6) weeks in order to detect a significant leak in the sources stored in the reactor pool. If the specific radionuclide content of the pool water for radionuclides from a sealed source stored in the reactor pool exceeds one-third (1/3) the 10 CFR 20 Appendix B, Table 3 value, steps shall be taken to isolate the source of the activity and to mitigate the problem.

Basis:

(1) Limiting the proximity of sealed sources to five (5) or more feet away from the surface of the reactor core minimizes the effect of such sources on the reactor and the operation of the reactor upon the sources. The neutron flux at a distance of five (5) feet from the core surface is insignificant and thus could not cause activation of the sources and any associated shielding. The presence of the sources in the pool would have no impact upon the D.B.A. which is the rupture of the cladding on one fuel element. However, the presence of sources in the pool could contribute to the radiation hazard associated with a loss of pool water accident. The dose rate 25 feet above an unshielded core in the event of a loss of pool water accident would only be increased by less than 2% with the presence of 100,000 curies of ^{60}Co stored in the irradiation unit in the reactor pool.

(2) Classifying the storage of sealed sources in the reactor pool as an experiment mandates that such storage be reviewed by the Reactor Safeguards Committee.

(3) The 10 CFR 20 Appendix B, Table 3 limit for ^{60}Co is 3×10^{-5} μCi/mℓ. At this limit the entire pool could be dumped into the WSU sewage system without taking advantage of the dilution factor associated with the discharge volume of the WSU sewage system. The detection limit for ^{60}Co in the reactor pool water depends upon the system used but in the worst case would be at least 1×10^{-7} μCi/mℓ, or 100 pCi/ℓ, or about one-three-hundredth of the 10 CFR 20 limit stated above. Setting a limit of 100 times the detection limit and one-third the discharge limits provides the facility with ample time to take corrective action in the event the limit is exceeded and does not compromise ALARA considerations.

3.15 Generation of Boron Neutron Capture Facility Beam

Definitions:

(1) For the purpose of this technical specification, the term "BNC facility" shall refer to the boron neutron capture facility which includes the beam, bridge moving system, beam monitoring equipment, beam shielding room, access gate and experimental area viewing equipment. The experimental bench, positioning equipment, and other equipment used for the beam targets are not considered part of the BNC facility for purposes of this provision, except insofar as radiation safety (i.e., activation and/or contamination) is concerned.

(2) The term "BNC experiment" shall refer to a boron neutron capture experiment involving the neutron irradiation of biological cells enriched with boron.

(3) The term "calibration check" refers to the process of checking the beam intensity and quality via one or more of the following: foil activation; use of a fission chamber; use of an ion chamber; or an equivalent process. The purpose of a calibration check is to ensure that the beam has not changed in a significant way (e.g., energy spectrum or intensity) from the beam that was characterized.

(4) The term "functional check of the beam monitors" shall consist of verifying that system output is consistent (± 10%) with previously measured values upon normalization to a common reactor neutronic power level.

(5) The term "characterization" refers to the process of obtaining the dose-versus-depth profile in phantoms. The dose-versus depth profile from the surface of the phantom to a depth at least equivalent to the total thickness of the target volume to be irradiated on a central axis is deemed adequate for a characterization. Fast neutron, thermal neutron, and gamma ray components are determined in a characterization and monitors are normalized by this characterization.

(6) The term "calibration of the beam monitors" refers to the process whereby the beam monitors are calibrated against instruments that measure dose including a tissue equivalent chamber and a graphite or magnesium wall ionization chamber (or the equivalent to any of these three) that have in turn been calibrated by a secondary calibration laboratory.

(7) The term "design modification" as applied to the BNC facility beam refers (a) to a change that is shown to alter the dose-versus-depth profile of the fast neutrons, thermal neutrons, or gamma rays in the beam as sensed by the calibration check and (b) to a change that has the potential to increase significantly the amount of activation products in the BNC facility.

(8) The term "radiation fluence" means the total fluence of neutrons and gamma radiation that is emitted in the BNC facility beam. The determination of the ratios of gamma, fast neutron, and thermal neutron fluences is part of the beam characterization. Knowledge of these ratios allows the total radiation fluence to be monitored by the on-line detectors, which are neutron sensitive. Compliance with the limits specified on radiation fluence by this specification is determined by reference to the fluence monitored by these detectors.

Applicability: This specification applies solely to the generation of the BNC facility beam for BNC experiments. It does not apply to any other use of the BNC facility and/or its beam. Surveillances listed in this specification are required only if BNC experiments are planned for the interval of the surveillance. However, in the event of a hiatus in the scheduled performance of any given surveillance, that surveillance shall be performed prior to the initiation of BNC experiments during the interval in question.

Objective: To acquire testing and operational experience in use of a facility developed specifically for Boron Neutron Capture Technology.

Specifications:

(1) It shall be possible to initiate a scram of the reactor from a control panel located in the BNC facility area. In the event that the BNC facility scram is inoperable, it shall be acceptable to use one of the control room scrams via communication with the reactor operator as a temporary means of satisfying this provision. Use of this temporary provision is limited to seven consecutive working days.

(2) Access to the BNC facility shall be controlled by means of the access gate located at its entrance.

(3) The following features and/or interlocks shall be operable:

  (a) An interlock shall prevent moving the bridge from the retracted position unless the BNC facility"s access gate is closed.

  (b) The reactor shall scram and the bridge shall move to the retracted position automatically upon opening the treatment room"s access gate.

  (c) The bridge shall be designed to move to the retracted position automatically upon failure of facility electric power or low voltage on the backup batteries that power the bridge motor.

  (d) Bridge movement that controls beam delivery shall be designed for manual movement to the retracted position.

  (e) It shall be possible to move the bridge to the retracted position from within the BNC facility.

  (f) A BNC facility lockdown near the access gate shall inhibit blade withdrawal when the key is not inserted and turned to the locked position.

(4) Bridge shall be equipped with a position readout that indicates the status of the bridge. A bridge position readout shall be visible at the BNC facility"s local control panel. In the event of a bridge position readout malfunction, it shall be acceptable to use an alternate means of verifying position such as a video camera in the pool room providing a signal to a monitor at the BNC facility"s local control panel. Use of this alternate means of bridge position verification is limited to seven consecutive working days.

(5) The BNC facility shall be equipped with a read out display of the reactor log-power and the linear power on the BNC facility control console just outside of the shielding.

(6) The BNC facility shall be equipped with a monitor that provides a visual indication of the radiation level within the facility, that indicates both within the facility and at the local

control panel, and that provides an audible alarm both within the facility and at the local control panel.

(a) This radiation monitor shall be equipped with a backup power supply such as the reactor emergency power system or a battery.

(b) This radiation monitor shall be checked for proper operation by means of a check source on the calendar day of and prior to any BNC experimentation.

(c) This radiation monitor shall be calibrated quarterly.

(d) The audible alarm shall be set at or below 50 mR/hr. This monitor and/or its alarm may be disabled once the BNC room has been searched and secured, such as is done immediately prior to initiation of BNC experimentation. If this is done, the monitor and/or its alarm shall be interlocked so that they become functional upon opening of the BNC facility access gate.

(e) In the event that this monitor is inoperable, personnel entering the BNC facility shall use either portable survey instruments or audible alarm personal dosimeters as a temporary means of satisfying this provision. These instruments/dosimeters shall be in calibration as defined by the WSU Research Reactor"s radiation protection program and shall be source-checked daily prior to use on any day that they are used to satisfy this provision. Use of these instruments/dosimeters as a temporary means of satisfying this provision is limited to seven consecutive working days.

(7) An intercom or other means of two-way communication shall be operable both between the BNC facility control panel and the reactor control room, and also between the BNC facility control panel and the interior of the BNC facility shielding.

(8) It shall be possible for personnel monitoring a BNC experiment to open the BNC facility access gate manually.

(9) [redacted]

(10) The following interlocks or channels shall be tested at least monthly and prior to a BNC experiment if the interlock or channel has been repaired or deenergized:

	Interlock or Channel	Surveillance
a)	The reactor scrams and the bridge retracts upon BNC facility scram	Scram test
b)	Bridge will not move from the retracted position unless access gate is closed	Operational test
c)	Upon opening the BNC room"s access gate the reactor scrams and the bridge moves to the retracted position	Operational test
d)	The bridge moves toward the retracted position on loss of electrical power and low voltage on the bridge motor batteries	Operational test
e)	Manual movement of bridge	Operational test
f)	Bridge can be moved manually by someone standing on the reactor bridge	Operational test
g)	Bridge position indicator and status lights	Operational test
h)	Radiation monitor alarm	Operational test
i)	Radiation monitor and/or alarm enabled upon opening of shield door	Operational test
j)	Intercoms	Operational test
k)	BNC facility TV cameras, monitors and its power backup	Operational test
l)	BNC facility emergency lighting	Operational test
m)	BNC facility lockdown blade inhibit	Operational test

In addition to the above, the BNC facility scram shall be tested prior to reactor startup if the reactor has been shut down for more than sixteen hours.

(11) Manual operation of the BNC facility"s access gate in which the door is opened fully shall be verified semi-annually.

(12) Use of the BNC facility beam shall be subject to the following:

(a) A calibration check of the beam and a functional check of the beam monitors shall be made weekly for any week that the beam will be used for BNC experiments. These checks shall be made prior to any BNC experiment for a given week. In addition, a calibration check shall be performed prior to any BNC experiment in the event that any component of a given beam design has been replaced. Finally, a calibration and a functional check shall be performed prior to any BNC experiment in the event of a design modification.

(b) A characterization of the beam shall be performed every six months for any six-month interval that the beam will be used for BNC experiments. This six-month characterization shall be made prior to any BNC experiment for a given six-month interval. A characterization shall also be performed prior to any BNC experiment in the event of a design modification. As part of the characterization process, the proper response of the beam monitors shall be verified.

(c) A calibration of the beam monitors shall be performed at least once every two years for any two-year interval that the beam will be used for BNC experimentation. The two-year calibration shall be made prior to any BNC experimentation during any given two-year interval.

(d) A scram from full power initiated when the reactor is positioned against the BNC facility filter shall be performed every six months or in the event of a design modification. The BNC room radiation monitor reading shall not exceed 50 mR/hr, 30 seconds after the initiation of the scram and bridge retraction.

(13) Maintenance, repair, and modification of the BNC facility shall be performed under the supervision of a senior reactor operator who is licensed by the U.S. Nuclear Regulatory Commission to operate the WSU Research Reactor. All modifications will be reviewed pursuant to the requirements of 10 CFR 50.59.

(14) Personnel who are not licensed to operate the WSU Research Reactor but who are responsible for either the BNC or the beam"s design including construction and/or modification may operate the controls for the BNC facility beam provided that:

(a) Training has been provided and proficiency satisfactorily demonstrated on the design of the facility, its controls, and the use of those controls. Proficiency shall be demonstrated annually.

(b) Instructions are posted at the BNC facility"s local control panel that specify the procedure to be followed:

(i) to ensure that only the appropriate target is in the irradiation facility before turning the primary beam of radiation on to begin an irradiation;

(ii) if the operator is unable to turn the primary beam of radiation off with controls outside the BNC facility, or if any other abnormal condition occurs. A directive shall be included with these instructions to notify the reactor console operator in the event of any abnormality.

(c) In the event that bridge movement affects reactivity, personnel who are not licensed on the WSU Research Reactor but who have been trained under this provision may initiate bridge movement provided that verbal permission is requested and received from the reactor console operator immediately prior to such action. Emergency scrams causing a bridge retraction are an exception and may be made without first requesting permission.

Records of the training provided under subparagraph (a) above shall be retained in accordance with the WSU Research Reactor"s training program or at least for three years. A list of personnel so qualified shall be maintained in the reactor control room.

Basis: The requirement that it be possible to initiate a scram from a control panel located in the BNC facility area assures the experimenter of the capability to terminate the irradiation immediately should the need arise. The provision that access to the BNC facility be limited to a single gate ensures that there will be no inadvertent entries. The various interlocks for the bridge movement system that controls beam delivery ensure that exposure levels in the BNC facility will be minimal prior to entry by personnel. The bridge position indicator and status lights serve to notify personnel of the beam"s status. The provision for a radiation monitor ensures that personnel will have information available on radiation levels in the BNC facility prior to entry. The purpose of this monitor"s audible alarm is to alert personnel to the presence of elevated radiation levels. This monitor and/or its alarm may be disabled once the BNC facility has been searched and secured so that it will not distract attending personnel. The monitor and/or its alarm are interlocked with the access gate so that they are made functional upon opening that gate, and hence prior to any possible entry to the BNC facility. One intercom provides a means for the prompt exchange of information between the experimenter(s) and the reactor operator(s).

The provision for manual operation of the BNC facility"s access gate ensures access to the experimental area in the event of a loss of electrical power. The presence of the closed-circuit TV cameras provide the experimenter(s) with the opportunity to monitor the target area visually as well as through the use of various instruments. The emergency lighting and the backup power for a TV camera and monitor will permit visual surveillance of the target area in the event of a power failure.

The surveillance requirements for beam calibration checks and characterizations provide a mechanism for ensuring that the BNC facility and its beam will perform as originally designed. Similarly, the surveillance requirements on the beam monitors ensure that these instruments are calibrated by a means traceable to the National Institute of Standards and Technology. The chambers specified (tissue-equivalent, and graphite or magnesium-wall) were chosen because they measure dose as opposed to fluence.

The specifications on maintenance and repair of the BNC facility ensures that all such activities are performed under the supervision of personnel cognizant of quality assurance and other requirements such as radiation safety. The provision on the training and proficiency of non-licensed personnel ensures that all such personnel will receive instruction equivalent to that given to licensed reactor operators as regards use of the BNC facility beam. (Note: Licensed reactor operators may, of course, operate the BNC facility beam.) Also, this provision provides for the posting of instructions to be followed in the event of an abnormality.

References

6.5-1 MITR Staff, “Safety Analysis Report for the MIT Research Reactor (MITR-II),” Report No. MITNE-115, 22 Oct. 1970, Section 10.1.3.

6.5-2 Choi, R.J., “Development and Characterization of an Epithermal Beam for Boron Neutron Capture Therapy at the MITR-II Research Reactor,” Ph.D. Thesis, Nuclear Engineering Department, Massachusetts Institute of Technology, April 1991.

4.0 SURVEILLANCE REQUIREMENTS

4.1 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.

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 control element 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 Safeguards Committee. A system shall not be considered operable until after it has been successfully tested.

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

4.2 Safety Limit - Fuel Element Temperature

Applicability: This specification applies to the surveillance requirements of the fuel element temperature measuring channel.

Objective: The objective is to ensure that the fuel element temperatures are properly monitored.

Specifications:

(1) Whenever a reactor scram caused by high fuel element temperature occurs, the peak indicated fuel temperature shall be examined to determine whether the fuel element temperature safety limit was exceeded.

(2) The fuel element temperature measuring channel shall be calibrated semiannually or at an interval not to exceed 8 months by the substitution of a thermocouple simula-tor in place of the instrumented fuel element thermocouple.

(3) A channel check of the fuel element measuring channel shall be made each time the reactor is operated by comparing the indicated instrumented fuel element tempera-ture with previous values for the core configuration and power level.

Basis: Operational experience over the past 5 years with the TRIGA system gives assurance that the thermocouple measurements of fuel element temperature have been sufficiently reliable to ensure accurate indication of this parameter.

4.3 Limiting Conditions for Operation

4.3.1 Reactivity Requirements

Applicability: These specifications apply to the surveillance requirements for reactivity control of experiments and systems.

Objective: The objective is to measure and verify the worth, performance, and operability of those systems affecting the reactivity of the reactor.

Specifications:

(1) The reactivity worth of each control rod and the shutdown margin shall be determined annually but at intervals not to exceed 15 months.

(2) The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.

(3) The control rods shall be visually inspected for deterioration at intervals not to exceed 2 years.

(4) The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated as necessary semiannually at intervals not to exceed 7.5 months.

(5) The reactor shall be pulsed semiannually to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity.

Basis: The reactivity worth of the control rods is measured to ensure that the required shut-down margin is available and to provide an accurate means for determining the reactivity worths of experiments inserted in the core. Past experience with TRIGA reactors gives assurance that measurement of the reactivity worth on an annual basis is adequate to ensure no significant changes in the shutdown margin. The visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation in the reactor. The reactor is pulsed at suitable intervals and a comparison is made with previous similar pulses to determine if changes in fuel or core characteristics are taking place.

4.3.2 Control and Safety System

Applicability: These specifications apply to the surveillance requirements for measure-ments, 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) The scram time shall be measured annually but at intervals not to exceed 15 months.

(2) A channel check of each of the reactor safety system channels for the intended mode of operation shall be performed before each day's operation or before each opera-tion extending more than 1 day, except for the pool level channel which shall be tested monthly.

(3) A channel calibration shall be made of the power level monitoring channels by the calorimetric method annually, but at intervals not to exceed 15 months.

(4) A channel test of each item in Table 3.2, other than measuring channels, shall be performed semiannually, but at intervals not to exceed 7.5 months.

Basis: 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. The channel tests will ensure that the safety system channels are operable on a daily basis or before an extended run. The power level channel calibration will ensure that the reactor will be operated at the proper power levels. Transient control element checks and semiannual maintenance ensure proper operation of this control element.

4.3.3 Radiation Monitoring System

Applicability: This specification applies to the surveillance monitoring for the area monitoring equipment, Argon-41 monitoring system, and continuous air monitoring system.

Objectives: The objectives are to ensure that the radiation monitoring equipment is operating properly and capable of performing its intended function, and that the alarm points are set correctly.

Specifications: All radiation monitoring systems shall be verified to be operable at least monthly at an interval not to exceed 45 days. In addition, the following surveillance activities shall be performed on an annual basis at intervals not to exceed 15 months: 1) the area radiation monitoring system shall be calibrated using a certified source; 2) a calibration of the Ar-41 system shall be done using at least two different calibrated gamma-ray sources; 3) a calibration shall be performed on the CAM in terms of counts per unit time per unit of activity using calibrated beta sources.

Basis: Experience has shown that monthly verification of Radiation Monitoring Systems' operability in conjunction with an annual more thorough surveillance is adequate to correct for any variations in the systems caused by a change of operating characteristics over a long timespan.

4.3.4 Ventilation System

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

Objective: The objective is to ensure the proper operation of the pool room ventilation system in the isolation and dilution modes, which would be used in controlling the release of radioactive material to the uncontrolled environment in the event of an emergency.

Specifications: The operation of the pool room system shall be checked monthly (at intervals not to exceed 6 weeks) by cycling the system from the "normal" to the "isolate" and "dilution" modes of operation. The positions of the associated dampers, indicator display, and fan operation shall be visually checked to ensure correspondence between the device performance and selected mode of operation. The pressure drop across the absolute filter in the pool ventilation system shall be measured at least twice a year. The absolute filter shall be changed whenever the pressure drop across the filter increases by 1 in. of water.

Basis: Experience has shown that the only reliable method of testing the ventilation is to cycle the system into the various modes and visually check each portion of the system for proper operation in that mode.

4.3.5 Experiment and Irradiation Limits

Applicability: This specification applies to the surveillance requirements for experiments installed in the reactor and its experimental facilities and for irradiations performed in the irradiation facilities.

Specifications:

(1) A new experiment shall not be installed in the reactor or its experimental facilities until a hazards analysis has been performed and reviewed for compliance with "Limitations on Experiments," Section 3.10, by the Reactor Safeguards Committee. Minor modifications to a reviewed and approved experiment may be made at the discretion of the senior operator responsible for the operation, provided that the hazards associated with the modifications have been reviewed and a determination has been made and documented that the modifications do not create a significantly different, a new, or a greater hazard than the original approved experiment.

(2) An irradiation of a new type of device or material shall not be performed until an analysis of the irradiation has been performed and reviewed for compliance with "Limitations on Irradiations," Section 3.11, by a licensed senior operator qualified in health physics, or a licensed senior operator and a person qualified in health physics.

Basis: It has been demonstrated over a number of years that experiments and irradiations reviewed by the reactor staff and the Reactor Safeguards Committee, as appropriate, can be conducted without endangering the safety of the reactor or exceeding the limits in the Technical Specifications.

4.4 Reactor Fuel Elements

Applicability: This specification applies to the surveillance requirements for the fuel elements.

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

Specifications: All fuel elements shall be inspected visually for damage or deterioration and measured for length and bend at intervals not to exceed the sum of 3,500.00$ in pulse reactivity. The reactor shall not be operated with 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 exceeds 0.125 in. over the length of the cladding

(2) in measuring the elongation, its length exceeds its original length by 0.125 in.

(3) a clad defect exists as indicated by release of fission products

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

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 ensure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to ensure adequate coolant flow.

4.5 Primary Coolant Conditions

Applicability: This specification applies to the surveillance of primary water quality.

Objective: The objective is to ensure that water quality does not deteriorate over extended periods of time if the reactor is not operated.

Specification: The conductivity and pH of the primary coolant water shall be measured at least once every 2 weeks, and shall be as follows:

(1) conductivity $\leq 5 \times 10^{-6}$ mhos/cm

(2) pH between 5.0 and 7.5

Basis: Section 3.3 ensures that the water quality is adequate during reactor operation. Section 4.5 ensures that water quality is not permitted to deteriorate over extended periods of time even if the reactor does not operate.

5.0 DESIGN FEATURES

5.1 Reactor Fuel

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

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

Specifications:

(1) TRIGA-FLIP Fuel - The individual unirradiated FLIP fuel elements shall have the following characteristics:

- uranium content: maximum of 9 wt% enriched to nominal 70% ^{235}U
- hydrogen-to-zirconium ratio (in the ZrH_x): between 1.5 and 1.7
- natural erbium content (homogeneously distributed): between 1.1 and 1.6 wt%
- cladding: 304 stainless steel, nominal 0.020 in. thick
- identification: top pieces of FLIP elements will have characteristic markings to allow visual identification of FLIP elements employed in mixed cores

(2) Standard TRIGA Fuel - The individual unirradiated Standard TRIGA fuel elements shall have the following characteristics:

- uranium content: maximum of 9.0 wt% enriched to less than 20% ^{235}U
- hydrogen-to-zirconium atom ratio (in the ZrH_x): between 1.5 and 1.8
- cladding: 304 stainless steel, nominal 0.020 in. thick

Basis: A maximum uranium content of 9.25 wt% in TRIGA-FLIP elements is about 6% greater than the design value of 8.5 wt%. Such an increase in loading would result in an increase in power density of about 2%. Similarly, a minimum erbium content of 1.1 wt% in an element is about 30% less than the design value. This variation would result in an increase in power density of only about 6%. An increase in local power density of 6% reduces the safety margin by at most 10%. The maximum hydrogen-to-zirconium ratio of 1.75 could result in a maximum stress under accident conditions in the fuel element clad about a factor of 2 greater than the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad.

A maximum uranium content of 9 wt% for the standard TRIGA elements is about 6% greater than the design value of 8.5 wt%. Such an increase in loading would result in an increase in power density of 6% and reduces the safety margin by at most 10%. The maximum hydrogen-to-zirconium ratio of 1.8 could result in a maximum stress under accident conditions in the fuel element clad about a factor of 2 greater than the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad.

5.2 Reactor Core

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

Objective: The objective is to ensure 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 composed of Standard fuel, FLIP fuel, or a combination thereof (mixed cores) provided that the FLIP fuel region contains at least 22 FLIP fuel rods located in a contiguous block in the central region of the core.

(3) The reactor fueled with a mixture of fuel types shall not be operated with a core lattice position vacant in the FLIP fuel region. Water holes in the FLIP region shall be limited to single-rod holes. Vacant lattice positions in the core fuel region shall be occupied with fixtures that will prevent the installation of a fuel bundle.

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

Basis: Standard TRIGA cores have been used for years and their characteristics are well-documented. Mixed cores of FLIP and Standard fuel have been tested by General Atomics Co. and operated at a number of university reactors. Calculations, as well as measured performance of mixed cores in the WSU reactor, the Texas A&M reactor, and the University of Wisconsin reactor, have shown that such cores may be safely operated.

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.

Vacant core lattice positions in the Standard fuel region will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. Vacant core positions are not permitted in the FLIP fuel region as specified by Section 3.5.

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.

5.3 Control Elements

Applicability: This specification applies to the control elements used in the reactor core.

Objective: The objective is to ensure that the control elements are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications:

(1) The standard control element shall have scram capability and contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding.

(2) The regulation control element need not have scram capability and shall be a stainless steel element or contain the materials as specified for standard control elements.

(3) The transient control element shall have scram capability and contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The transient element shall have an adjustable upper limit to allow a variation of reactivity insertions. This element may incorporate a nonfueled follower.

Basis: The poison requirements for the control elements are satisfied by using neutron-absorbing borated graphite, B_4C powder, or boron and its compounds. Since the regulating element normally is a low worth element, its function could be satisfied by using solid stainless steel. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to ensure mechanical stability during movement and to isolate the poison from the pool water environment. Scram capabilities are provided for rapid inser-tion of the control element which is the primary safety feature of the reactor. The transient control element is assigned for a reactor pulse. The nuclear behavior of the nonfueled follower which may be incorporated into the transient element is similar to a void.

5.4 Radiation Monitoring System

Applicability: This specification describes the functions and essential components of the area radiation monitoring equipment and the system for continuously monitoring airborne radioactivity.

Objective: The objective is to describe the radiation monitoring equipment that is available to the operator to ensure safe operation of the reactor.

Specifications:

(1) Function of Area Radiation Monitor (gamma-sensitive instruments): Monitor radiation fields in key locations, alarm and readout at control console.

(2) Function of Continuous Air Radiation Monitor (beta-, gamma-sensitive detector with particulate collection capability): Monitor radioactive particulate activity in the pool room air, alarm and readout at control console.

(3) Function of Argon-41 Stack Monitor (gamma-sensitive detector): Monitor ^{41}Ar content in reactor exhaust air, alarm and readout at console.

Basis: The radiation monitoring system is intended to 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 radioacti-vity to the surroundings.

5.5 Fuel Storage

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

Objective: The objective is to ensure that fuel that is being stored will not become critical and will not reach an unsafe temperature.

Specifications:

(1) All fuel elements shall be stored in a geometrical array where the k_{eff} 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, so that the fuel element or fueled device temperature will not exceed design values.

Basis: The limits imposed by Specifications 5.5(1) and 5.5(2) are conservative and ensure safe storage.

5.6 Reactor Building and Ventilation System

Applicability: This specification applies to the building that houses the reactor.

Objective: The objective is to ensure that provisions are made to restrict the amount of radioactivity released into the environment.

Specifications:

(1) The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall be 10^9 cm^3.

(2) The reactor building shall be equipped with a ventilation system designed to filter and exhaust air or other gases from the reactor building and release them from a stack at a minimum of 20 ft from ground level.

(3) Emergency shutdown controls for the ventilation system shall be located outside the pool and control room areas and the system shall be designed to shut down in the event of a substantial release of airborne radioactivity within the facility.

(4) The pool room ventilation system shall have a dilution mode of operation in which air from the pool room is mixed and diluted with outside air before being discharged from the facility.

Basis: The facility is designed so that the ventilation system will normally maintain a negative pressure with respect to the atmosphere to minimize uncontrollable leakage to the environment. The free air volume within the reactor building is confined when there is an emergency shutdown of the ventilation system. Emergency controls for startup, isolation, dilution, and normal operation of the ventilation system are located external to the control and pool rooms. Proper handling of airborne radioactive materials (in emergency situations) can be effected with a minimum of exposure to operating personnel.

5.7 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 ensure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications:

(1) The reactor core shall be cooled by natural convection water flow.

(2) All piping extending more than 5 ft below the surface of the pool shall have ade-quate provisions to prevent inadvertent siphoning of the pool.

(3) A pool level alarm shall be provided to indicate a loss of coolant if the pool level drops more than 2 ft below the normal level.

(4) The reactor shall not be operated with less than 15 ft of water above the top of the core.

Basis: 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 2700 kW with natural convection flow of the coolant water. A comparison between operation of the TRIGA-FLIP and standard TRIGA MARK III has shown them to be safe for the above power level. Thermal and hydraulic characteristics of mixed cores are essentially the same as those for TRIGA-FLIP and standard cores.

In the event of accidental siphoning of pool water through system pipes, the pool water level will drop no more than 5 ft from the top of the pool.

Loss of coolant alarm after 2 ft of loss requires corrective action. This alarm is observed in the reactor control room, at the office, and at the campus police station.

5.8 Physical Security

The Licensee shall maintain in effect and fully implement all provisions of the NRC staff-approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.70, collectively titled, "Washington State University, Pullman, Washington TRIGA Reactor Security Plan."

6.0 ADMINISTRATIVE CONTROL

6.1 Responsibility

The facility shall be under the direct control of a licensed Senior Reactor Operator (SRO) designated by the Director of the WSU Nuclear Radiation Center. The SRO shall be responsible to the Director for the overall facility operation including the safe operation and maintenance of the facility and associated equipment. The SRO shall also be responsible for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, Federal and State regulations, and requirements of the Reactor Safeguards Committee.

6.2 Organization

(1) The reactor facility shall be an integral part of the Nuclear Radiation Center of Washington State University. The organization of the facility management and operation shall be as shown in Figure 6.1. The responsibilities and authority of each member of the operating staff shall be defined in writing.

(2) When the reactor is not secured, the minimum staff shall consist of:

- (a) Reactor Operator (RO) at the controls (may be the SRO)
- (b) Senior Reactor Operator (SRO) on call but not necessarily on site
- (c) another person present at the facility complex who is able to carry out prescribed written instructions

6.3 Facility Staff Qualifications

Each member of the facility staff shall meet or exceed the minimum qualifications of ANS 15.4, "Standard for the Selection and Training of Personnel for Research Reactors," for comparable positions.

6.4 Training

The licensed Senior Reactor Operator designated by the Director as being responsible for the facility also shall be responsible for the facility's Requalification Training Program and Operator Training Program.

6.5 Reactor Safeguards Committee (RSC)

6.5.1 Function

The RSC shall function to provide an independent review and audit of the facility's activities including:

(1) reactor operations

(2) radiological safety
(3) general safety
(4) testing and experiments
(5) licensing and reports
(6) quality assurance

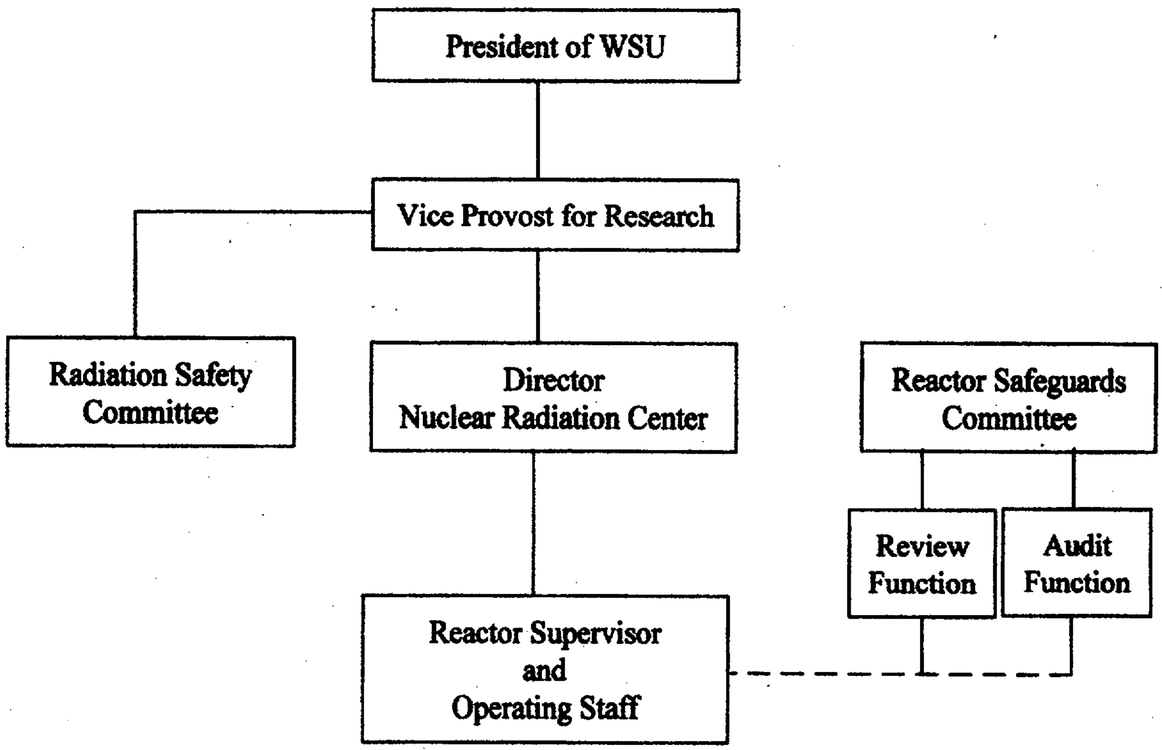


Figure 6.1 Facility organization

6.5.2 Composition and Qualifications

The RSC shall be composed of at least five members knowledgeable in fields that relate to nuclear reactor safety. The members of the Committee shall include one facility Senior Reactor Operator and WSU faculty and staff members designated to serve on the Committee in accordance with the procedures specified by the WSU committee manual. The University's Radiation Safety Director shall be an ex officio member of the Committee.

6.5.3 Operation

The Reactor Safeguards Committee shall operate in accordance with a written charter, including provisions for:

(1) meeting frequency: the full committee shall meet at least semiannually and a subcommittee thereof shall meet at least semiannually

(2) voting rules

(3) quorums: chairman or his designate and two members

(4) method of submission and content of presentations to the committee

(5) use of subcommittees

(6) review, approval and dissemination of minutes

6.5.4 Reviews

The responsibilities of the RSC or designated subcommittee thereof shall include, but are not limited to, the following:

(1) review and approval of all new experiments utilizing the reactor facility

(2) review and approval of all proposed changes to the facility license by amendment, and to the Technical Specifications

(3) review of the operation and operational records of the facility

(4) review of significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety

(5) review and approval of all determinations of whether a proposed change, test, or experiment would constitute a change in the Technical Specifications or an unreviewed safety question as defined by 10 CFR 50

(6) review of reportable occurrences and the reports filed with the Commissions for said occurrences

(7) review and approval of all standard operating procedures and changes thereto

(8) biennial review of all standard procedures, the facility emergency plan, and the facility security plan

(9) annual review of the radiation protection program

6.5.5 Audits

The RSC or a subcommittee thereof shall audit reactor operations semiannually, but at intervals not to exceed 8 months. The semiannual audit shall include at least the following:

(1) review of the reactor operating records
(2) inspection of the reactor operating areas
(3) review of unusual or abnormal occurrences
(4) radiation exposures at the facility and adjacent environs

6.5.6 Records

The activities of the RSC shall be documented by the secretary of the Committee and distributed as follows:

(1) A written report of all audits performed under Section 6.5.5 shall be prepared and forwarded within 30 days to the Dean of the Graduate School and Facility Director.

(2) A written report of all reviews performed under Section 6.5.4 shall be prepared and forwarded to the Facility Director within 30 days following the completion of the review.

(3) The secretary of the RSC shall maintain a file of the minutes of all meetings.

6.6 Quality Assurance

In accordance with Regulatory Guide 2.5 and ANSI 402, "Quality Assurance Program Requirements for Research Reactors," Section 2.17, the "facility shall not be required to prepare quality assurance documentation for the as-built facility." Quality Assurance (QA) requirements will still be limited to those specified in Section 2.17 as follows:

> "All replacements, modification, and changes to systems having a safety related function shall be subjected to a QA review. Insofar as possible, the replacement, modification, or change shall be documented as meeting the requirements of the original system or component and have equal or better performance or reliability."
>
> "The required audit function shall be performed by the RSC specified in Section 6.5."

6.7 Action To Be Taken in the Event a Safety Limit Is Exceeded

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 U.S. Nuclear Regulatory Commission (NRC).

(2) An immediate report of the occurrence shall be made to the Chairman of the Reactor Safeguards Committee, and reports shall be made to the NRC in accordance with Section 6.10 of these specifications.

(3) A report shall be prepared that 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 Safeguards Committee for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

(4) A report shall be made to the NRC in accordance with Section 6.10 of these specifications.

6.8 Operating Procedures

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

(1) performing irradiations and experiments

(2) startup, operation, and shutdown of the reactor

(3) emergency situations including provisions for building evacuation, earthquake, radiation emergencies, fire or explosion, personal injury, civil disorder, and bomb threat

(4) core changes and fuel movement

(5) control element removal and replacement

(6) performing preventive maintenance and calibration tests on the reactor and associated equipment

(7) power calibration

Substantiative changes to the above procedures shall be made only with the approval of the licensed SRO directly in charge of the facility. Temporary changes to the procedures that do not change their original intent may be made by a licensed SRO. All such temporary changes shall be documented and subsequently reviewed by the licensed SRO directly in charge of the facility.

6.9 Facility Operating Records

In addition to the requirements of applicable regulations, and in no way substituting for those requirements, records and logs shall be prepared for at least the following items and retained for a period of at least 5 years for items (1) through (6) and indefinitely for items (7) through (11).

(1) normal reactor operation
(2) principal maintenance activities
(3) abnormal occurrences
(4) equipment and component surveillance activities required by the Technical Specifications
(5) experiments performed with the reactor
(6) gaseous and liquid radioactive effluents released to the environs

(7) off-site inventories and transfers
(8) fuel inventories and transfers
(9) facility radiation and contamination surveys
(10) radiation exposures for all personnel
(11) updated, corrected, and as-built drawings of the facility

6.10 Reporting Requirements

In addition to the requirements of applicable regulations, and in no way substituting for those requirements, reports shall be made to the Nuclear Regulatory Commission as follows:

(1) A report within 24 hours by telephone to the NRC Operations Center, of

(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 the safety limit;

(c) Any reportable occurrence as defined in Section 1.1, "Reportable Occurrence," of these specifications.

(2) A report within 10 days in writing to USNRC Document Control Desk, Washington, D.C. 20555, of

(a) Any accidental release or radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure. The written report (and, to the extent possible, the preliminary telephone or telegraph report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event;

(b) Any violation of a safety limit;

(c) Any reportable occurrence as defined in Section 1.1, "Reportable Occurrence," of these specifications.

(3) A report within 30 days in writing to the USNRC Document Control Desk, Washington, D.C. 20555, of

(a) Any significant variation of measured values from a corresponding predicted or previously measured value of safety-connected operating characteristics occurring during operation of the reactor;

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

(c) Any significant changes in facility organization;

(d) Any observed inadequacies in the implementation of administrative or procedural controls.

(4) A report within 60 days after completion of startup testing of the reactor (in writing to the USNRC 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 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.

(5) An annual report within 60 days following the 30th of June of each year in writing to the USNRC Document Control Desk, Washington, D.C. 20555, providing the following information:

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

(b) Tabulation of the energy output (in megawatt-days) of the reactor, hours reactor was critical, the cumulative total energy output since initial criticality, and number of pulses greater than 1.00$;

(c) The number of emergency shutdowns and inadvertent scrams, including reasons for them;

(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 procedures and of tests and experiments carried out pursuant to 10 CFR 50.59;

(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 before the point of such release or discharge:

Liquid Waste (summarized on a monthly basis)

(i) monthly radioactivity discharged

- total estimated quantity of radioactivity released (in curies),
- an estimation of the specific quantity for each detectable radionuclide in the monthly release,
- fraction of 10 CFR 20 table 3, appendix B limit for each detectable radionuclide taking into account the dilution factor from the total volume of sewage released by the licensee into the sewage system,
- sum of the fractions for each radionuclide reported above.

(ii) total quantity of radioactive material released by the facility into the sewage system during the year period of the report

Gaseous Waste (summarized on a monthly basis)

(i) radioactivity discharged during the reporting period (in curies)

- total estimated quantity of radioactivity released (in curies) determined by an appropriate sampling and counting method,
- total estimated quantity of ^{41}Ar released (in curies) during the reporting period based on data from an appropriate monitoring system,
- estimated average atmospheric diluted concentration of ^{41}Ar released during the reporting period in terms of µCi/ml and fraction of the applicable DAC value,
- total estimated quantity of radioactivity in particulate form with half-lives greater than 8 days (in curies) released during the reporting period as determined by an appropriate particulate monitoring system,
- average concentration of radioactive particulates with half-lives greater than 8 days released in µCi/ml during the reporting period, and
- an estimate of the average concentration of other significant radionuclides present in the gaseous waste discharge in terms of µCi/ml and fraction of the applicable DAC value for the reporting period if the estimated release is greater than 20% of the applicable DAC.

Solid Waste (summarized on an annual basis)

(i) total amount of solid waste packaged (in cubic feet),

(ii) total activity in solid waste (in curies),

(iii) the dates of shipment and disposition (if shipped off-site).

(g) An annual summary of the radiation exposure received by facility personnel and visitors in terms of the average radiation exposure per individual and greater exposure per individual in the two groups. Each significant exposure in excess of the limits of 10 CFR 20 should be reported, including the time and date of the exposure as well as the circumstances that led up to the exposure;

(h) An annual summary of the radiation levels of contamination observed during routine surveys performed at the facility in terms of the average and highest levels;

(i) An annual summary of any environmental surveys performed outside the facility.

6.11 Written Communications

All written communications with the Nuclear Regulatory Commission shall be made in accordance with the requirements of 10 CFR 50.4 “Written Communications.”

15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Construct a Non-Power Reactor

The WSU TRIGA Reactor has already been constructed and thus the University's ability to construct the facility has already been demonstrated.

15.2 Financial Ability to Operate a Non-Power Reactor

Washington State University has been funding the operation of the WSU TRIGA Reactor Facility for over forty years and the University Administration is committed to funding the continued operation of the facility.

The basic financial considerations related to the operation of the facility are listed below:

(1) Budget Information

a. Washington State University is a land grant educational institution in the State of Washington funded directly by State appropriations approved by the Legislature of the State. Funding is appropriated to the University on a biannual basis and appropriation is $183,361,041 for 2000. A five year financial review is shown in the attached data. A complete copy of the WSU Financial Report can be viewed at: http://wsu.edu/~genacct/finstat.htm.

b. The current annual budget for the Nuclear Radiation Center in which the Facility is located is $330,098.35. The Nuclear Radiation Center budget covers a variety of activities including the TRIGA reactor facility.

(2) Operating Costs

The cost of operating the WSU TRIGA reactor facility and attendant research projects during the current year is $180,000. The funds come from Program 10D of the university budget entitled "Other Organized Research." Since all funding for WSU is by action of the State Legislature, it is not possible to guarantee funding for any program within the university. However, the State of Washington is an Agreement-State and has in the past chosen to comply with all Federal regulations and commitments along with the costs thereof. It is thus deemed that it would be incumbent upon the State to continue to provide the necessary funding for operation of the facility.

(3) Commercial Operations

At the present time and at any time in the foreseeable future significantly less than 50% of all utilization of the WSU TRIGA Reactor and income associated with such work is non-commercial in nature. The primary utilization of the reactor is for research associated with graduate education. However, the facility is in the process of installing a facility for BNCT cancer therapy which will initially be a research project but may eventually lead to a small amount of commercial utilization for medical purposes.

15.3 Financial Ability to Decommission the Facility

In 1989 the Nuclear Division of Westinghouse Electric Corporation made a detailed technical and cost proposal for the decommissioning of the WSU Modified TRIGA Reactor. The 2000 inflation adjusted decommissioning costs are estimated to be $4,994,615. If the facility is shut down by action of the university or termination of the facility license, the funds required to decommission the facility would be provided by appropriate sources within the university and the State of Washington.

Five Years in Review

MEMORANDUM ONLY

Fund Balances	2000	1999	1998	1997	1996
Current-Unrestricted	35,808,394	30,037,612	18,383,899	20,615,314	22,137,552
Current-Restricted	28,975,017	23,250,045	22,108,237	24,417,308	21,514,329
Loan	22,704,490	22,441,268	23,242,018	21,641,722	21,185,704
Endowment	272,469,732	275,916,539	259,764,529	222,572,667	206,772,664
Plant	740,127,224	629,227,895	581,825,180	527,484,788	520,201,327
Fund Balance Total	1,100,084,857	980,873,359	905,323,863	816,731,799	791,811,576
Major Sources of Operating Revenue					
Tuition and Fees	98,834,698	95,773,810	91,238,786	90,044,368	83,727,462
Federal Appropriations	9,283,842	8,660,299	8,249,671	8,657,445	9,535,906
State Appropriations	183,361,061	171,299,147	170,765,542	160,848,357	151,097,924
Federal Grants & Contracts	65,171,570	58,931,904	57,822,469	60,474,405	62,252,534
State Grants & Contracts	27,289,909	26,577,498	23,298,323	23,478,148	19,909,044
Private Grants, Gifts, & Contracts	34,480,675	32,255,297	27,829,925	26,759,570	24,986,032
Sales & Services of Educational Activities	10,310,971	8,501,708	8,880,469	9,031,788	8,440,571
Sales & Services of Auxiliary Enterprises	57,008,414	56,619,716	55,363,986	50,745,899	48,691,303
Other Sources	16,216,403	14,995,402	12,383,927	15,391,316	13,077,286
Operating Revenue Total	501,957,543	473,614,781	455,833,098	445,431,296	421,718,062
Major Operating Expenditures					
Instruction	125,706,574	130,684,544	129,439,962	124,851,670	116,261,481
Research	76,088,604	70,200,635	65,040,447	61,381,393	63,751,567
Public Service	39,127,051	35,793,656	32,898,212	33,843,093	28,690,608
Academic Support	68,784,265	55,658,539	58,824,709	63,813,104	62,657,625
Student Services	14,097,606	13,153,071	13,953,404	13,933,812	12,862,865
Institutional Support*	27,614,878	28,572,219	28,657,676	28,319,878	27,295,426
Operation & Maintenance of Plant	31,046,684	28,366,594	28,499,798	26,508,892	24,904,578
Scholarships & Fellowships*	46,499,245	42,549,005	37,483,809	35,144,799	30,990,444
Retirement of Indebtedness			120,000	120,000	115,000
Interest on Indebtedness			2,340	6,900	11,075
Mandatory Transfer for Tuition	968,000	974,882	1,047,899	1,048,011	1,042,553
Auxiliary Enterprises	63,017,672	53,669,261	59,880,966	54,996,740	54,246,815
Operating Expenditure Total	492,950,579	459,622,406	455,849,222	443,968,292	422,830,037
Net Student Loans Outstanding	19,447,880	19,814,946	19,814,289	19,025,707	18,718,083
WSU Regents/Foundation Endowment Fund, Fair Value	184,694,004	168,552,267	152,526,875	122,385,171	97,272,534
Land-Grant Endowment Fund, Fair Value	274,587,373	249,313,144	232,842,856	208,573,602	192,506,216
Plant in Use, Net of Accumulated Depreciation	785,405,906	695,515,391	640,371,621	611,002,566	563,172,016
Plant Liabilities	163,627,570	140,913,254	99,642,170	105,582,783	97,692,963

Major Construction Report as of June 30, 2000

MEMORANDUM ONLY

Construction Projects Completed in Fiscal Year 2000	Approved Budget	Expenditures To Date	SOURCES OF FUNDING: State Capital Funds	WSU Building Fund	Federal Grants	Local Funds
Animal Disease Biotechnical Facility	22,615,837	22,615,837		764,284	21,851,553	
Chemical Waste Collection Sites	3,327,000	3,327,000		3,327,000		
South Campus Electrical Upgrade	2,588,729	2,588,729	2,588,729			
Thompson Hall Renovation	11,068,804	11,068,804	10,984,336	84,468		
Total Projects Completed	39,600,370	39,600,370	13,573,065	4,175,752	21,851,553	0

Construction in Progress	Approved Budget	Expenditures To Date	State Capital Funds	WSU Building Fund	Federal Grants	Local Funds
AMID/Landscape Architecture Facility	30,728,000	583,200	30,630,000	98,000		
Bohler Gym Renovation	20,432,369	18,996,374	19,382,133	1,050,236		
Campus Infrastructure Project	11,842,000	7,083,698	8,292,000	3,550,000		
Children's Center Development Lab	3,100,000	244,419		3,100,000		
Cleveland Hall Addition	11,750,000	471,548	11,610,000	140,000		
Creamery Warehouse	1,671,537	1,568,772				1,671,537
Hazardous Waste Incinerator	4,790,200	4,739,513	4,790,200			
Hazardous Waste Projects	19,711,000	966,615	15,000,000	4,711,000		
Johnson Hall Addition	49,000,000	177,857	38,700,000	300,000	10,000,000	
Kimbrough Addition and Remodel	11,733,000	11,412,287	11,292,700	440,300		
McCroskey Hall Renovation	4,999,800	83,280				4,999,800
Murrow Hall Renovation/Addition	12,665,000	266,068	12,560,000	105,000		
Power Plant Boiler Renewal	3,600,000	1,280,062	3,600,000			
Shock Physics Building	12,400,000	351,175	12,400,000			
Spokane Health Sciences Building	39,061,222	8,701,522	39,061,222			
Student Recreation Center	37,065,000	27,074,094				37,065,000
Teaching and Learning Center	41,574,500	11,982,980	30,870,175	704,325		10,000,000
Vancouver Campus Circulation	4,500,000	2,486,613	4,500,000			
Vancouver Engineering and Life Science	29,470,650	15,323,720	29,470,650			
Vancouver Media/Electronic Communication	18,500,000	675,683	18,500,000			
Vancouver Physical Plant Maintenance Shop	3,700,000	3,213,425	3,700,000			
White Hall Renovation	15,300,000	1,498,099	5,000,000			10,300,000
Total Construction in Progress	387,594,278	119,181,004	299,359,080	14,198,861	10,000,000	64,036,337

16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

The entire WSU Reactor Facility is composed of prior used components since this SAR is for license renewal purposes. Over the years it has been the policy of the WSU Reactor Facility to incrementally update and improve major portions of the system by replacement with newer more modern components. Thus the entire electronic portion of the reactor instrumentation and the control and monitoring systems are all relatively new and highly reliable. In other words, all the electronic portions of the control and monitoring systems will reliably continue to perform their safety related functions for the period of the license renewal.

The only control system components that have not been upgraded and/or replaced are the actual control rods and their associated drives. The entire reactor system is under a very comprehensive preventative maintenance program that ensures that all portions of the system function properly including the control rods and their drive motors. During check-out of the reactor, the rod drive systems are functionally tested to insure that they all operate properly for that day's reactor operation. Historically there have been very few problems with TRIGA control rod systems. Thus the control rods and their drive systems will be able to perform their safety related functions for the period of the license renewal.

The entire cooling system was replaced with a completely new system in 1999 as described in section 5 of this SAR. Thus the new cooling system will be able to function reliably for the period of the license renewal.

In the summer of 1999 the leak that had developed in the reactor pool wall over the years was professionally repaired as described in the appendix of this section. This repair job was quite effective and thus no further pool leak problems are anticipated for the period of the license renewal.

The TRIGA reactor fuel in the WSU modified TRIGA reactor was purchased new by the facility from General Atomics and has been in use for a number of years. However, the Megawatt Days of usage on the fuel is quite small compared to the tests run at GA or usage at other TRIGA facilities. The facility has never experienced a fission product leak problem with any TRIGA fuel and has instituted a program to insure that pulsing will not cause fuel problems like those that occurred at the Texas A&M TRIGA facility. Thus the existing TRIGA fuel in the WSU reactor should continue to operate adequately and safely for the period of the license renewal.

Over the years the WSU Reactor Facility has developed an extensive set of Standard Operating Procedures (SOPs) which include a comprehensive Preventative Maintenance Program (SOP #5). This set of SOPs has proven to be quite effective and has precluded any significant system problems or failures for over 20 years. All test and maintenance periods and procedures recommended by component manufacturers are followed where appropriate and/or the maintenance period shortened to insure increased reliability. Thus the SOPs and Maintenance Program as it exists will continue to adequately perform their safety related functions during the period of the license renewal. An index of the WSU SOPs is given at the end of this SAR section.

16.2 Medical Use of Non-Power Reactors

Attached is a copy of the changes to the WSU facility Technical Specifications that have been negotiated with the Commission for the BNCT project at the WSU Nuclear Radiation Center.

16.3 HEU to LEU Conversion

Due to a lack of DOE funding and the changing world political climate, it is very unlikely that the WSU facility will be converted from HEU to LEU fuel. However, the technical aspects of the use of LEU fuel in TRIGA type reactors have been investigated by G.A. Technologies and appropriate reports developed under DOE funding. If conversion does take place during the period of this renewed license, appropriate documentation required for the conversion will be written using the existing technical information on the use of LEU fuel in a TRIGA reactor.

W.S.U. NUCLEAR RADIATION CENTER
STANDARD OPERATING PROCEDURE INDEX

SOP NO.

1	Standard Procedure for Use of the Reactor.
2	Standard Procedure for Performing Irradiations Using the Reactor.
3	Standard Procedure for Performing Experiments Using the Reactor.
4	Standard Procedure for Startup, Operation, and Shutdown of the Reactor.
5	Standard Procedure for Performing Preventive Maintenance on the Reactor and Associated Equipment.
6	Standard Procedure in the Event of an Emergency Situation.
7	Standard Procedure for Core Changes and Fuel Movement.
8	Standard Procedure for Control Element Maintenance, Removal and Replacement.
9	Standard Procedure for Maintenance of Reactor Pool Facilities Using a Diver.
10	Standard Procedure for Health Physics Surveys.
11	Standard Procedure for Analysis of Liquid Waste Samples.
12	Standard Procedure for Specific Activity - Dose Rate Calculations and Sample Failure Analysis.
13	Standard Procedure for Performing Power Calibrations.
14	Standard Procedure for Calibration of Pulse Instrumentation.
15	Standard Procedure for Alignment of the Fuel Temperature System.
16	Standard Procedure for Control Element Calibration.
17	Standard Procedure for Checkout and Calibration of the Area Radiation Monitors.
18	Standard Procedure for ^{41}Ar Monitor Checkout and Calibration.
19	Standard Procedure for Action in the Event of an Alarm.
20	Standard Procedure for Fuel Burn-up Calculations.
21	Standard Procedure for Environmental Monitoring.
22	Standard Procedure for T.L.D. Environmental Monitoring Program.
23	Standard Procedure for Portable Survey Instrumentation Check and Calibration.
24	Standard Procedure for Pool Water Analysis.
25	Standard Procedure for Purification System Resin and Filter Change.
26	Standard Procedure for Continuous Air Monitor Check and Calibration.
27	Standard Procedure for RM-14 Check and Calibration.
28	Standard Procedure for Removal and Installation of the Reactor Pool Room Ventilation System Absolute Filters.
29	Standard Procedure for Continuous Air Monitor Filter Analysis.
30	Standard Procedure for Security System Check.
31	[Unused]
32	Standard Procedure for Security and Emergency Plan Training for Nuclear Radiation Center, Radiation Safety Office and Campus Police Personnel.
33	Standard Procedure for Off-Site Shipment of Radioactive Material.
34	Standard Procedure for the Transfer of Non-fuel Devices and Experimental Apparatus into and out of the Reactor Pool.

35 Standard Procedure for Receiving and Opening Packages Containing Licensed Materials

Appendix 16A
GENERATION OF BORON NEUTRON CAPTURE FACILITY BEAM

<u>Definitions</u>:

1. For the purpose of this technical specification, the term 'BNC facility' shall refer to the boron neutron capture facility which includes the beam, bridge moving system, beam monitoring equipment, beam shielding room, access gate and experimental area viewing equipment. The experimental bench, positioning equipment, and other equipment used for the beam targets are not considered part of the BNC facility for purposes of this provision, except insofar as radiation safety (i.e., activation and/or contamination) is concerned.
2. The term 'BNC experiment' shall refer to a boron neutron capture experiment involving the neutron irradiation of biological cells enriched with boron.
3. The term 'calibration check' refers to the process of checking the beam intensity and quality via one or more of the following: foil activation; use of a fission chamber; use of an ion chamber; or an equivalent process. The purpose of a calibration check is to ensure that the beam has not changed in a significant way (e.g., energy spectrum or intensity) from the beam that was characterized.
4. The term 'functional check of the beam monitors' shall consist of verifying that system output is consistent (± 10%) with previously measured values upon normalization to a common reactor neutronic power level.
5. The term 'characterization' refers to the process of obtaining the dose-versus-depth profile in phantoms. The dose-versus depth profile from the surface of the phantom to a depth at least equivalent to the total thickness of the target volume to be irradiated on a central axis is deemed adequate for a characterization. Fast neutron, thermal neutron, and gamma ray components are determined in a characterization and monitors are normalized by this characterization.
6. The term 'calibration of the beam monitors' refers to the process whereby the beam monitors are calibrated against instruments that measure dose including a tissue equivalent chamber and a graphite or magnesium wall ionization chamber (or the equivalent to any of these three) that have in turn been calibrated by a secondary calibration laboratory.
7. The term 'design modification' as applied to the BNC facility beam refers (a) to a change that is shown to alter the dose-versus-depth profile of the fast neutrons, thermal neutrons, or gamma rays in the beam as sensed by the calibration check and (b) to a change that has the potential to increase significantly the amount of activation products in the BNC facility.
8. The term 'radiation fluence' means the total fluence of neutrons and gamma radiation that is emitted in the BNC facility beam. The determination of the ratios of gamma, fast neutron, and thermal neutron fluences is part of the beam characterization. Knowledge of these ratios allows the total radiation fluence to be monitored by the on-line detectors, which are neutron sensitive. Compliance with the limits specified on radiation fluence by this specification is determined by reference to the fluence monitored by these detectors.

Applicability:

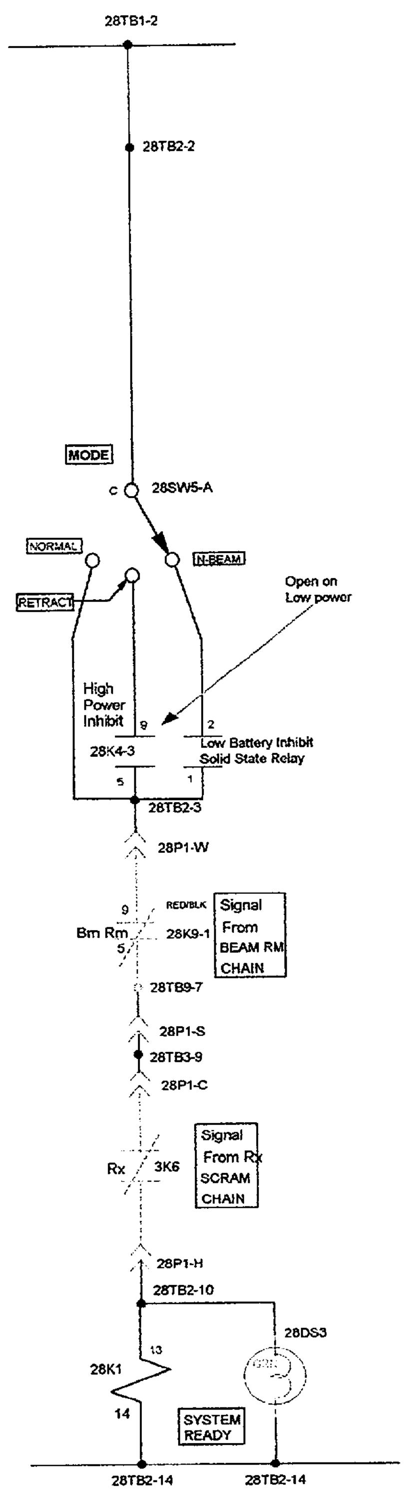
This specification applies solely to the generation of the BNC facility beam for BNC experiments. It does not apply to any other use of the BNC facility and/or its beam. Surveillances listed in this specification are required only if BNC experiments are planned for the interval of the surveillance. However, in the event of a hiatus in the scheduled performance of any given surveillance, that surveillance shall be performed prior to the initiation of BNC experiments during the interval in question.

Objective:

To acquire testing and operational experience in use of a facility developed specifically for Boron Neutron Capture Technology.

Specifications:

1. It shall be possible to initiate a scram of the reactor from a control panel located in the BNC facility area. In the event that the BNC facility scram is inoperable, it shall be acceptable to use one of the control room scrams via communication with the reactor operator as a temporary means of satisfying this provision. Use of this temporary provision is limited to seven consecutive working days.
2. Access to the BNC facility shall be controlled by means of the access gate located at its entrance.
3. The following features and/or interlocks shall be operable:
   (a) An interlock shall prevent moving the bridge from the retracted position unless the BNC facility's access gate is closed.
   (b) The reactor shall scram and the bridge shall move to the retracted position automatically upon opening the treatment room's access gate.
   (c) The bridge shall be designed to move to the retracted position automatically upon failure of facility electric power or low voltage on the backup batteries that power the bridge motor.
   (d) Bridge movement that controls beam delivery shall be designed for manual movement to the retracted position.
   (e) It shall be possible to move the bridge to the retracted position from within the BNC facility.
   (f) A BNC facility lockdown near the access gate shall inhibit blade withdrawal when the key is not inserted and turned to the locked position.
4. Bridge shall be equipped with a position readout that indicates the status of the bridge. A bridge position readout shall be visible at the BNC facility's local control panel. In the event of a bridge position readout malfunction, it shall be acceptable to use an alternate means of verifying position such as a video camera in the pool room providing a signal to a monitor at the BNC facility's local control panel. Use of this alternate means of bridge position verification is limited to seven consecutive working days. The BNCT/Reactor Bridge Movement Control System circuit diagram is shown in figures BNCT-A through BNCT-E.



The System is ready when the following conditions are met:
- Reactor Scrams are cleared
- Beam Room Scrams are cleared

Also:

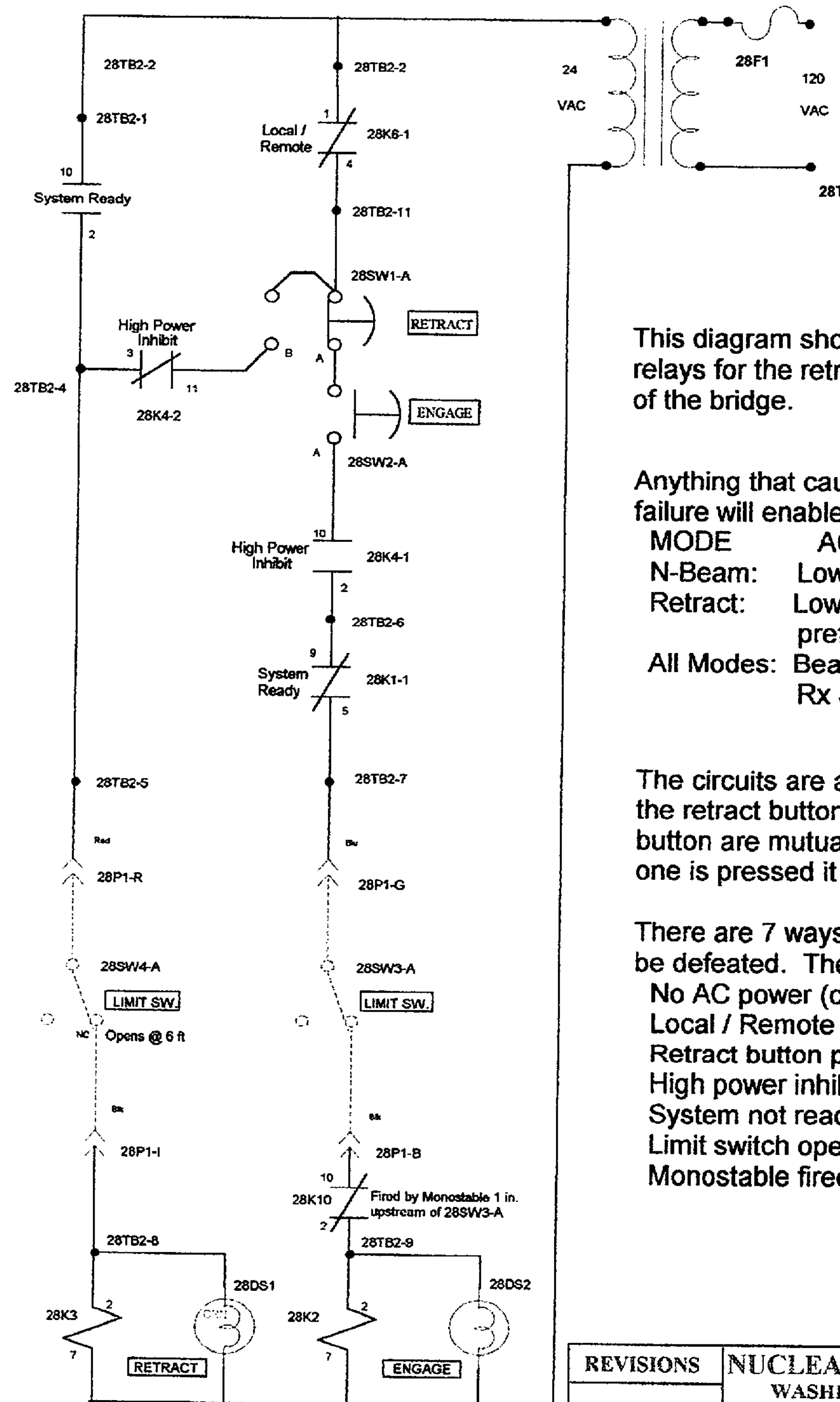
In Retract Mode, when Rx power falls below 1 Kw, contacts 5 & 9 on 28K4 open causing the system to be "not ready". This will retract the bridge.

In N-BEAM Mode UPS Battery must not be low

REMOTE COMPONENT

REVISIONS	NUCLEAR RADIATION CENTER WASHINGTON STATE UNIVERSITY PULLMAN, WASHINGTON		
	TITLE: BNCT / SYSTEM READY		
	SCALE: N.A.	DWG BY: G.E.T.	FILE NO:
	DATE: 07/31/2001	CHK BY: G.E.T.	

BNCT-A



This diagram shows the main control relays for the retraction and engagement of the bridge.

Anything that causes a "System Ready" failure will enable the retract relay, e.g.

MODE	ACTION
N-Beam:	Low UPS Battery
Retract:	Low Reactor Power, i.e. prefer retraction below 1KW
All Modes:	Beam Rm Scram Rx Scram

The circuits are arranged so that the retract button and the engage button are mutually exclusive. If one is pressed it disables the other.

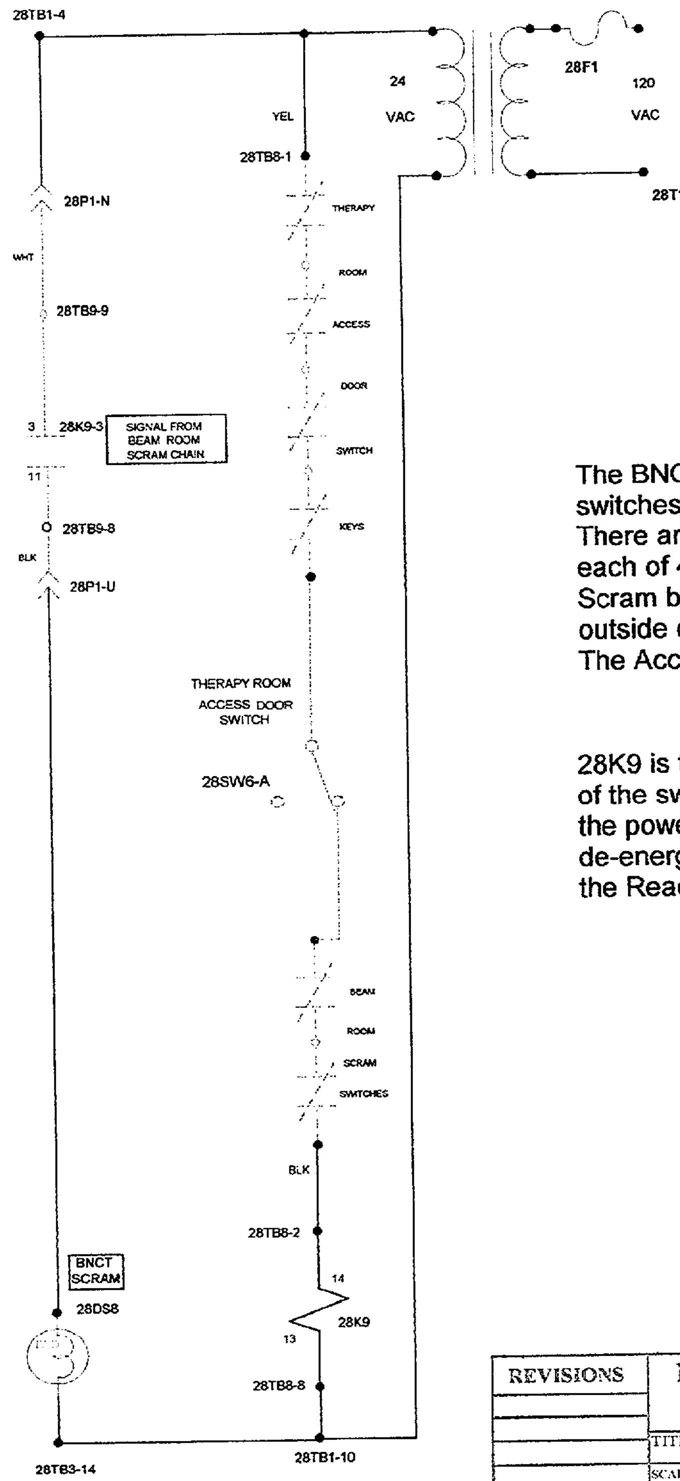
There are 7 ways the engage can be defeated. These are:

- No AC power (on the bus)
- Local / Remote set to remote
- Retract button pressed or open
- High power inhibit
- System not ready
- Limit switch open
- Monostable fired or ON

REMOTE COMPONENT

REVISIONS	NUCLEAR RADIATION CENTER WASHINGTON STATE UNIVERSITY PULLMAN, WASHINGTON		
	TITLE: RETRACT- ENGAGE		
	SCALE: N.A.	DWG BY: G.E.T.	FILE NO:
	DATE: 07/30/2001	CHK BY: G.E.T.	

BNCT-B



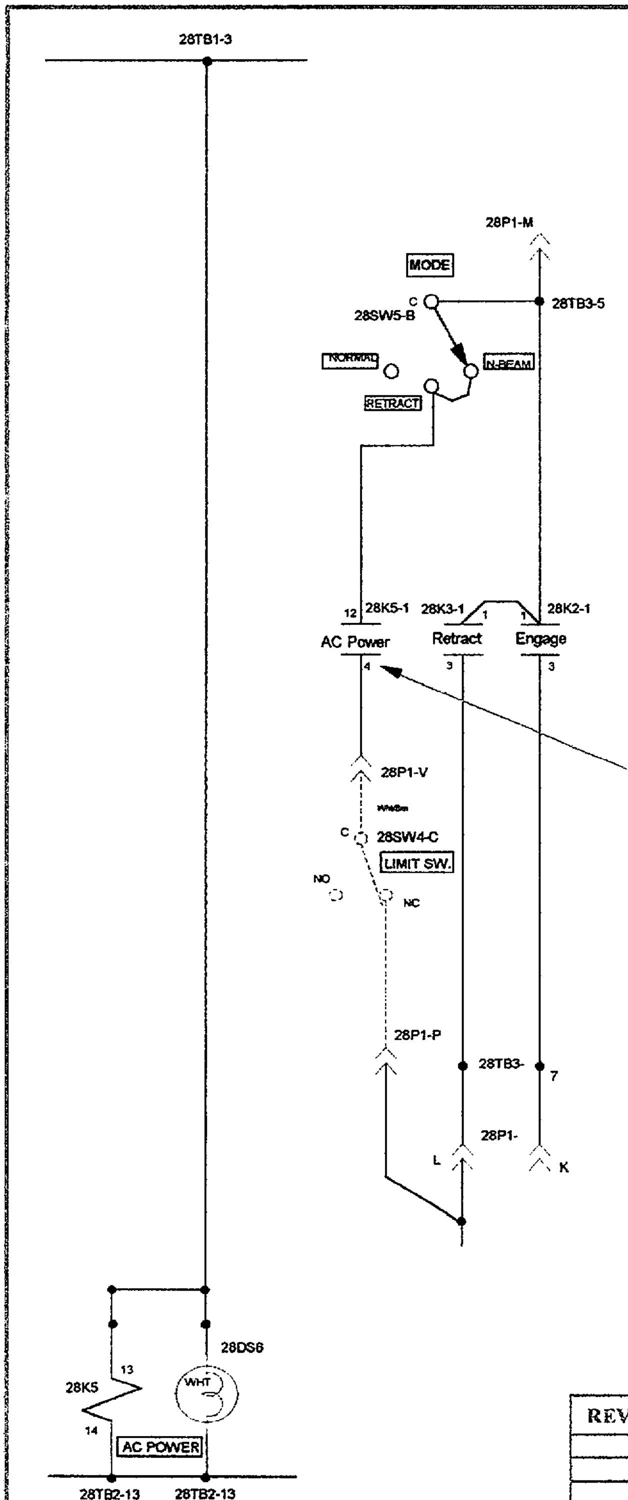
The BNCT Scram Chain consists of 7 switches in the beam room area. There are 4 switches associated with each of 4 access keys. There are 2 Scram buttons, one inside and one outside of the treatment room. The Access Door has the 7th switch.

28K9 is the BNCT Scram Relay. If any of the switches in this chain open or the power to this relay fails, this relay de-energizes and opens a contact in the Reactor's Main Scram Chain.

REMOTE COMPONENT

REVISIONS	NUCLEAR RADIATION CENTER WASHINGTON STATE UNIVERSITY PULLMAN, WASHINGTON		
	TITLE: BEAM ROOM SCRAM CHAIN		
	SCALE: N.A.	DWG BY: G.E.T.	FILE NO: 5
	DATE: 07/30/2001	CHK BY: G.E.T.	

BNCT-C



The Bridge Motor Direction is controlled by making a connection between 28P1-M with either 28P1-K (this moves the bridge towards the beam port) or with 28P1-L (this retracts the the bridge)

Ordinarily the movement will be determined by the status of the Engage and Retract relays. However, if there is a loss of power, relay 28K5 will lose power and close contacts 12 and 4 of that relay. This will retract the bridge until switch 28SW4 opens.

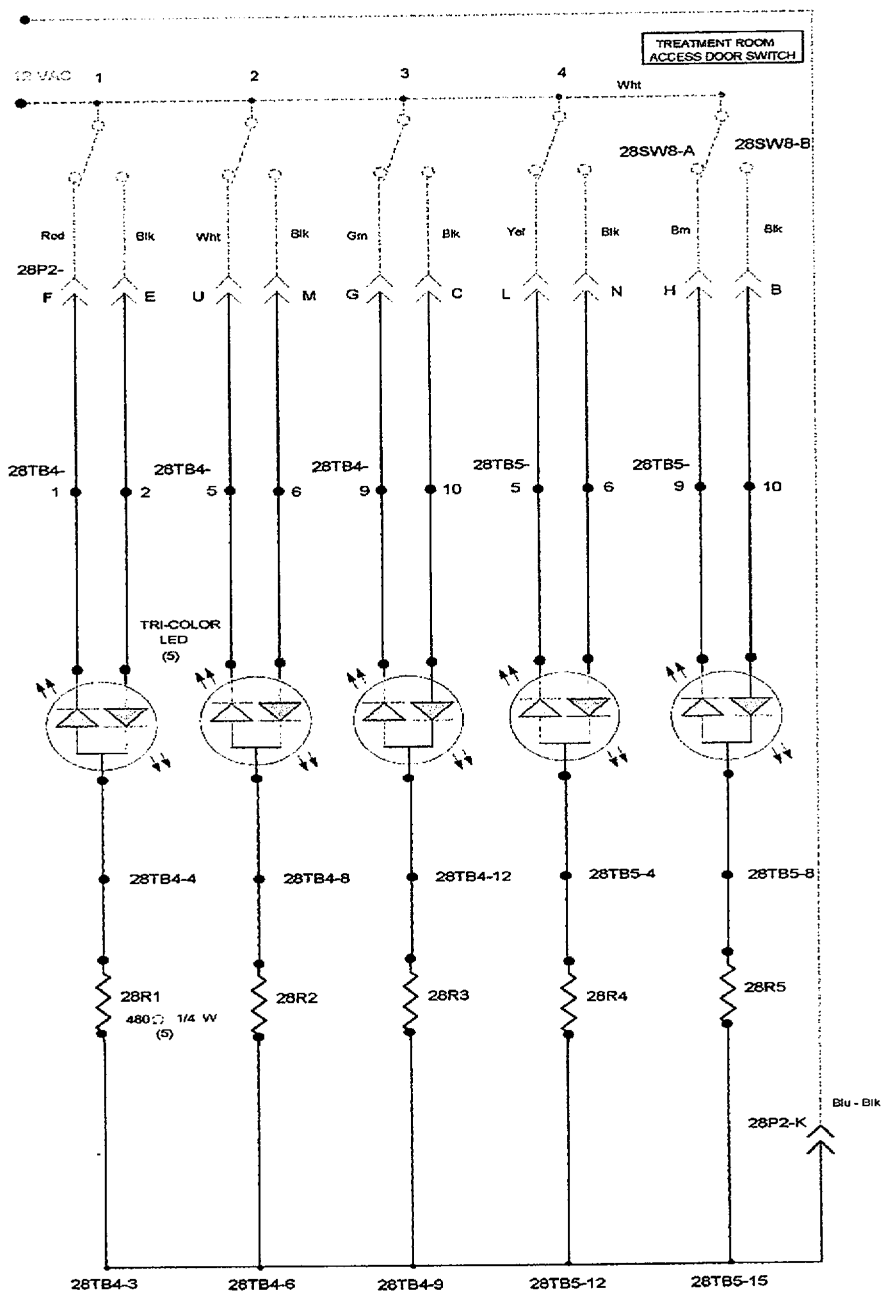
If there is a loss of AC Power, relay 28K5 serves the function of the Retract relay that would not be able operate the flash & sound or the Retract relay during a loss of AC Power.

REMOTE COMPONENT

REVISIONS	NUCLEAR RADIATION CENTER WASHINGTON STATE UNIVERSITY PULLMAN, WASHINGTON		
	TITLE: BNCT / REACTOR BRIDGE CONTROL		
	SCALE: N.A.	DWG BY: G.E.T.	FILE NO: 4
	DATE: 07/30/2001	CHK BY: G.E.T.	

BNCT-D

BEAM ROOM KEY-BANK & ACCESS GATE INTERLOCK SWITCHES



REVISIONS	NUCLEAR RADIATION CENTER WASHINGTON STATE UNIVERSITY PULLMAN, WASHINGTON		
05/21/2000	TITLE: BNCT / RX BRIDGE CONTROL PANEL		
	SCALE: N.A.	DWG BY: J.A.N.	FILE NO:
	DATE: 05/03/1999	CHK BY: J.A.N.	2 of 2

BNCT-E

5. The BNC facility shall be equipped with a read out display of the reactor log-power and the linear power on the BNC facility control console just outside of the shielding.
6. The BNC facility shall be equipped with a monitor that provides a visual indication of the radiation level within the facility, that indicates both within the facility and at the local control panel, and that provides an audible alarm both within the facility and at the local control panel.
   (a) This radiation monitor shall be equipped with a backup power supply such as the reactor emergency power system or a battery.
   (b) This radiation monitor shall be checked for proper operation by means of a check source on the calendar day of and prior to any BNC experimentation.
   (c) This radiation monitor shall be calibrated quarterly.
   (d) The audible alarm shall be set at or below 50 mR/hr. This monitor and/or its alarm may be disabled once the BNC room has been searched and secured, such as is done immediately prior to initiation of BNC experimentation. If this is done, the monitor and/or its alarm shall be interlocked so that they become functional upon opening of the BNC facility access gate.
   (e) In the event that this monitor is inoperable, personnel entering the BNC facility shall use either portable survey instruments or audible alarm personal dosimeters as a temporary means of satisfying this provision. These instruments/dosimeters shall be in calibration as defined by the WSU Research Reactor's radiation protection program and shall be source-checked daily prior to use on any day that they are used to satisfy this provision. Use of these instruments/dosimeters as a temporary means of satisfying this provision is limited to seven consecutive working days.
7. An intercom or other means of two-way communication shall be operable both between the BNC facility control panel and the reactor control room, and also between the BNC facility control panel and the interior of the BNC facility shielding.
8. It shall be possible for personnel monitoring a BNC experiment to open the BNC facility access gate manually.
9. It shall be possible to observe the BNC experiment by means of two independent closed-circuit TV cameras. Both cameras providing visualization shall be operable at the outset of any BNC experiment. Should either fail during the irradiation, the experiment may be continued at the discretion of the experimenter. Adequate lighting to permit such viewing shall be assured by the provision of emergency lighting and backup power for one TV camera and monitor.
10. The following interlocks or channels shall be tested at least monthly and prior to a BNC experiment if the interlock or channel has been repaired or de-energized:

	Interlock or Channel	Surveillance
a)	The reactor scrams and the bridge retracts upon BNC facility scram	Scram test
b)	Bridge will not move from the retracted position unless access gate is closed	Operational test
c)	Upon opening the BNC room's access	Operational test

	gate the reactor scrams and the bridge moves to the retracted position	
d)	The bridge moves toward the retracted position on loss of electrical power and low voltage on the bridge motor batteries	Operational test
e	Manual movement of bridge	Operational test
f)	Bridge can be moved manually by someone standing on the reactor bridge	Operational test
g)	Bridge position indicator and status lights	Operational test
h)	Radiation monitor alarm	Operational test
i)	Radiation monitor and/or alarm enabled upon opening of shield door	Operational test
j)	Intercoms	Operational test
k)	BNC facility TV cameras, monitors and its power backup	Operational test
l)	BNC facility emergency lighting	Operational test
m)	BNC facility lockdown blade inhibit	Operational test

In addition to the above, the BNC facility scram shall be tested prior to reactor startup if the reactor has been shut down for more than sixteen hours.

11. Manual operation of the BNC facility's access gate in which the door is opened fully shall be verified semi-annually.

12. Use of the BNC facility beam shall be subject to the following:
    a) A calibration check of the beam and a functional check of the beam monitors shall be made weekly for any week that the beam will be used for BNC experiments. These checks shall be made prior to any BNC experiment for a given week. In addition, a calibration check shall be performed prior to any BNC experiment in the event that any component of a given beam design has been replaced. Finally, a calibration and a functional check shall be performed prior to any BNC experiment in the event of a design modification.
    b) A characterization of the beam shall be performed every six months for any six-month interval that the beam will be used for BNC experiments. This six-month characterization shall be made prior to any BNC experiment for a given six-month interval. A characterization shall also be performed prior to any BNC experiment in the event of a design modification. As part of the characterization process, the proper response of the beam monitors shall be verified.
    c) A calibration of the beam monitors shall be performed at least once every two years for any two-year interval that the beam will be used for BNC experimentation. The two-year calibration shall be made prior to any BNC experimentation during any given two-year interval.

d) A scram from full power initiated when the reactor is positioned against the BNC facility filter shall be performed every six months or in the event of a design modification. The BNC room radiation monitor reading shall not exceed 50 mR/hr, 30 seconds after the initiation of the scram and bridge retraction.

13. Maintenance, repair, and modification of the BNC facility shall be performed under the supervision of a senior reactor operator who is licensed by the U.S. Nuclear Regulatory Commission to operate the WSU Research Reactor. All modifications will be reviewed pursuant to the requirements of 10 CFR 50.59.

14. Personnel who are not licensed to operate the WSU Research Reactor but who are responsible for either the BNC or the beam's design including construction and/or modification may operate the controls for the BNC facility beam provided that:

(a) Training has been provided and proficiency satisfactorily demonstrated on the design of the facility, its controls, and the use of those controls. Proficiency shall be demonstrated annually.

(b) Instructions are posted at the BNC facility's local control panel that specify the procedure to be followed:

(i) to ensure that only the appropriate target is in the irradiation facility before turning the primary beam of radiation on to begin an irradiation;

(ii) if the operator is unable to turn the primary beam of radiation off with controls outside the BNC facility, or if any other abnormal condition occurs. A directive shall be included with these instructions to notify the reactor console operator in the event of any abnormality.

(c) In the event that bridge movement affects reactivity, personnel who are not licensed on the WSU Research Reactor but who have been trained under this provision may initiate bridge movement provided that verbal permission is requested and received from the reactor console operator immediately prior to such action. Emergency scrams causing a bridge retraction are an exception and may be made without first requesting permission.

Records of the training provided under subparagraph (a) above shall be retained in accordance with the WSU Research Reactor's training program or at least for three years. A list of personnel so qualified shall be maintained in the reactor control room.

Basis

The requirement that it be possible to initiate a scram from a control panel located in the BNC facility area assures the experimenter of the capability to terminate the irradiation immediately should the need arise. The provision that access to the BNC facility be limited to a single gate ensures that there will be no inadvertent entries. The various interlocks for the bridge movement system that controls beam delivery ensure that exposure levels in the BNC facility will be minimal prior to entry by personnel. The bridge position indicator and status lights serve to notify personnel of the beam's status. The provision for a radiation monitor ensures that personnel will have information available on radiation levels in the BNC facility prior to entry. The purpose of this monitor's audible alarm is to alert personnel to the presence of elevated radiation levels. This monitor and/or its alarm may be disabled once the BNC facility has been searched and secured so that it will not distract attending personnel. The monitor and/or its alarm are interlocked with the access gate so that they are made functional upon opening that gate, and hence prior to any possible entry to the BNC facility. One intercom provides a means for the prompt exchange of information between the experimenter(s) and the reactor operator(s).

The provision for manual operation of the BNC facility's access gate ensures access to the experimental area in the event of a loss of electrical power. The presence of the closed-circuit TV cameras provide the experimenter(s) with the opportunity to monitor the target area visually as well as through the use of various instruments. The emergency lighting and the backup power for a TV camera and monitor will permit visual surveillance of the target area in the event of a power failure.

The surveillance requirements for beam calibration checks and characterizations provide a mechanism for ensuring that the BNC facility and its beam will perform as originally designed. Similarly, the surveillance requirements on the beam monitors ensure that these instruments are calibrated by a means traceable to the National Institute of Standards and Technology. The chambers specified (tissue-equivalent, and graphite or magnesium-wall) were chosen because they measure dose as opposed to fluence.

The specifications on maintenance and repair of the BNC facility ensures that all such activities are performed under the supervision of personnel cognizant of quality assurance and other requirements such as radiation safety. The provision on the training and proficiency of non-licensed personnel ensures that all such personnel will receive instruction equivalent to that given to licensed reactor operators as regards use of the BNC facility beam. (Note: Licensed reactor operators may, of course, operate the BNC facility beam.) Also, this provision provides for the posting of instructions to be followed in the event of an abnormality.

References

6.5-1 MITR Staff, "Safety Analysis Report for the MIT Research Reactor (MITR-II)," Report No. MITNE-115, 22 Oct. 1970, Section 10.1.3.

6.5-2 Choi, R.J., "Development and Characterization of an Epithermal Beam for Boron Neutron Capture Therapy at the MITR-II Research Reactor," Ph.D. Thesis, Nuclear Engineering Department, Massachusetts Institute of Technology, April 1991.

Appendix 16B
REACTOR CONCRETE WATER TANK REPAIR PROJECT
AUGUST 1998-SEPTEMBER1999

Soon after commissioning the facility in the late '50's it was noted that water was leaking from the tank in some small quantity. While it was deemed a nuisance, the leaking water appeared to pose no other problems and it continued for the next forty years. Over time a slow rate of increase of the leak was observed and dealt with by directing flow way from electrical and other components. In July and August of 1998 the leaking began to increase weekly at an intolerable rate. The leaking reached an unacceptable rate of about 8000 gallons a week by April of 1999 and concerns for the structural integrity of the tank were heightened.

In October of 1998 a firm experienced in assessment and repair of similar facilities was contracted to examine, evaluate and make recommendation for repair of the tank. A "hardhat" diver was placed in the pool to examine the condition of the interior walls information was transmitted to the surface via video camera. Other methods of evaluation included Ground Imaging Radar, Thermal Imaging and Hammer Toning. The visual inspection identified a crack, which extended around the perimeter of the concrete tank walls at a level not visible from the surface at about 12 feet below the surface of the pool. While other cracks had been observed from the surface, it became evident that this was the primary source of loss of water from the pool. In retrospect, the leak was probably due to poor bonding of separate pours on the north side of the pool wall during original construction. A temporary "patch" of an epoxy coated fabric was placed over the crack while work began in securing a permanent repair method. Over the next few months an emergency declaration was issued by university president, Sam Smith, which allowed for a fast track approach to remediation. Options were reviewed, funding obtained, and a repair contract was signed with Oceaneering to effect permanent repairs of the facility.

Engineered repairs began in June 1999 and continued over the next three months to complete in early September. Repairs included excavation, removal and repair of deteriorated concrete, crack repair by epoxy injection, and removal and replacement of old epoxy pool coating. The epoxy coating was selected because of desirable properties regarding resistance to gamma radiation and the ability to apply underwater or to wet surfaces. In conjunction with the repairs to the concrete tank a second project was underway to replace the aging cooling tower and heat exchanger which served to cool the pool water.

Repairs on the tank were completed in early September of 1999 and the deionized water was returned to the tank. The contractor demobilized from the site and reactor startup awaited completion of the Cooling Tower. After the cooling tower was complete and it was determined that all systems were operational the reactor was brought back on line after having been off line all summer. There were some immediate conductivity problems with the water, requiring additional deionizing capacity and resin replacement. After about a week the conductivity returned to within operational limits. The conductivity is believed to have climbed due to non-catalyzed epoxy molecules being knocked loose by Gamma radiation which has now completed it's curing cycle. Operations have continued successfully since re-commissioning.

17 DECOMMISSIONING

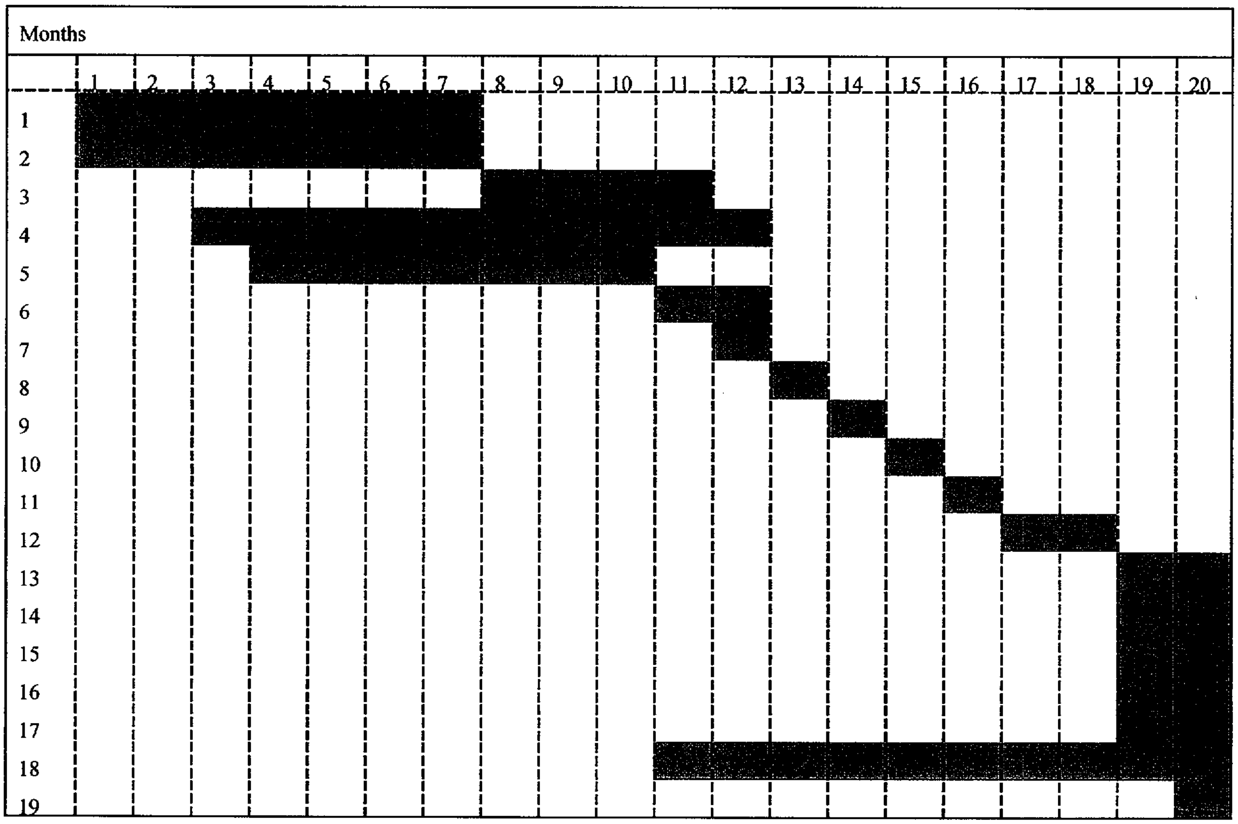
On April 19, 1990 the facility submitted to the Commission the decommissioning information required under 10 CFR 50.75 and 10 CFR 50.82. At the present time the University has no intention of decommissioning the facility for a number of years, but is in the process of renewing the license for the facility including the development of this SAR. Facility license R-7 of August 11, 1982 expires on August 16, 2002 and a renewal license will extend the license period until 2022. The required updated decommissioning information is as follows:

1) The estimated cost for the complete decommissioning of the WSU TRIGA reactor facility for 1990 was $3.84 million. This cost estimate was prepared by the Installation and Construction Services Manager of the Nuclear and Advanced Technology Division of Westinghouse Electric Corporation in Pittsburgh, Pennsylvania, assuming that Westinghouse would provide all the people and services needed to perform the decommissioning operation. A schedule is attached indicating the time required each phase of the decommissioning operation.
2) Washington State University is a state institution and thus, according to the provisions of 50.75(e) (2) (iv), the funds needed for decommissioning will be requested from the Washington State Legislature if and when a decision to decommission the WSU reactor facility is made.
3) The cost estimate for decommissioning the WSU TRIGA reactor facility for years 1991 and beyond will be adjusted for inflation by the consumer price index and the new estimate kept on file at the facility.

The new Facility License R-76 for the WSU modified TRIGA reactor expires in about 2022. In accordance with the requirements of 50.82(a), WSU will either submit an application for renewal of the license or a formal decommissioning plan five years or more prior to this date.

FIGURE 1

WASHINGTON STATE UNIVERSITY
DECOMMISSIONING SCHEDULE



1. Prepare Decommissioning Plan
2. Prepare License Document
3. Regulatory Review
4. Detail Work Packages
5. Equipment and Services
6. Staffing and Training
7. Install Temporary Equipment
8. Defueling
9. Radiation Survey
10. General Cleanup
11. Remove Reactor Comp.
12. Remove Concrete Pool
13. Remove Mechanical Equipment
14. Remove Piping
15. Remove I & C Systems
16. Remove Electrical Systems
17. Remove HVAC Systems
18. Pack and Ship Wastes
19. Final Radiation Survey

18 HIGHLY ENRICHED TO LOW-ENRICHED URANIUM CONVERSION

18.1 Introduction

On February 25, 1986, the U.S. Nuclear Regulatory Commission (NRC) promulgated a final rule in 10 CFR 50.64 of its regulations limiting the use of high-enriched uranium (HEU) fuel in domestic research and test reactors (non-power rectors). The rule, which became effective on March 27, 1986, required that an existing non-power reactor licensee replace HEU fuel with low-enriched uranium(LEU) fuel acceptable to the NRC: (1) unless the NRC has determined that the reactor has a unique purpose and (2) contingent upon Federal Government funding for conversion-related costs. The rule is intended to promote the common defense and security by reducing the risk of theft and diversion of HEU fuel used in non-power reactors and the adverse consequences to public health and safety and the environment from such theft or diversion.

10 CFR 50.64(c) (2) (i) of the rule, among other things, requires each non-power reactor licensee authorized to possess and use HEU fuel to develop and to submit to the Director of the Office of Nuclear Reactor Regulation by March 27,1987, and at 12-month intervals thereafter, a written proposal for meeting the rule's requirements. 10 CFR 50.64(c) (2) (i) also requires the licensee to include as part of the proposal (1) a certification that Federal Government funding for conversion is available through the Department of Energy (DOE), and (2) a schedule of conversion, based upon availability of fuel acceptable to the NRC for the specific reactor and upon consideration of other factors such as the availability of shipping casks, implementation of arrangements for the available financial support, and reactor usage.

10 CFR 50.64(c) (2) (iii) requires the licensee's proposal to include, to the extent required to effect conversion, all necessary changes to the license, to the facility, and to the licensee's procedures. This paragraph also requires the licensee to provide supporting safety analyses so as to meet the schedule established for conversion.

Prior to 1992, the Washington State University Modified TRIGA Reactor met these regulatory requirements by first determining if funding for conversion from HEU to LEU was available. Then, after being informed that money was not available, Washington State University (WSU) notified the NRC of that fact.

On February 24, 1992, WSU received notification from the DOE that funding had become available to begin converting the WSU Reactor to LEU fuel. As a result of this, WSU submitted a proposal on February 10, 1992 to solicit funding from the DOE to support the relicensing part of the fuel conversion project. Notification was received on June 18, 1992 that the proposal was accepted and that funding would be available for the budget period August 15, 1992 through August 14, 1993.

A safety analysis for conversion to LEU fuel was subsequently submitted to the Commission but Congress did not provide DOE with the expected funding level and the conversion did not take place. Subsequent funding to DOE has not included funds for conversion. The remainder of this section of the SAR considers the safety aspects of conversion under the new SAR should funding for conversion eventually become available.

18.2 Summary and Conclusions of Principal Safety Consideration

The principal safety consideration for the use of LEU fuel in TRIGA type reactors is the performance of the LEU fuel itself as compared to Standard and HEU TRIGA fuels. Numerous studies have been done with DOE funding and at University TRIGA Reactors (1,2,14,15, & 16). The unique safety feature of all TRIGA reactors is the prompt negative temperature coefficient of reactivity of the uranium-zirconium hydride (U-ZrH) fuel-moderator material. This characteristic allows sudden large insertions of reactivity in which the power level increases many thousand times on periods of less than 2.0 msec. The control is based on the prompt negative temperature coefficient of reactivity, which decreases the power level to normal operating values in a fraction of a second. The same characteristic also restricts the upper steady-state thermal power level that may be obtained with a given amount of fuel. Thus, both transient and steady-state operations have inherent safeguards which do not require manual, electronic, or mechanical control. This self-actuating temperature coefficient allows great freedom in operation, because the effect on temperature and power of accidental reactivity changes is greatly suppressed. The prompt shutdown mechanism has been demonstrated extensively in many thousands of transient tests. The above mentioned studies have conclusively shown that LEU fuel will perform identically to the previous Standard and HEU TRIGA fuels in the critical negative temperature coefficient behavior.

There are no significant changes needed in the new SAR to allow the use of LEU fuel in the WSU Reactor. The new SAR includes operation with LEU fuel as well as Standard and HEU fuels.

18.3 Summary of Reactor Facility Changes

No changes to the WSU facility are required for conversion to LEU fuel.

18.4 Summary of Operating License, Technical Specification, and Procedure Changes

No changes are required in these areas since the new SAR includes the use of LEU fuel in the WSU reactor.

18.5 Comparison with Similar Facilities Already Converted

No similar facilities to the WSU reactor have been converted to LEU fuel. However, a study done at the University of Wisconsin facility (15) concludes that the neutron flux and power distribution in that facility which is identical to the WSU facility will be essentially identical before and after conversion to LEU fuel.

18.6 Site Consideration

There are no site considerations related to the conversion to LEU fuel. The new SAR covers all the site aspects of operation of the WSU TRIGA reactor.

18.7 Design of Structures, Systems, and Components

No needed changes, new SAR covers all aspects of these subjects.

18.8 Reactor Facility

No changes needed to new SAR for use of LEU fuel including Core, Fuel Elements, Control Rods, Neutron Reflector, Neutron Source, In-Core Experimental Facilities, Reactor Materials, Reactor Tank and Biological Shield, Core Support Structure, Dynamic Design, Excess

Reactivity, Shutdown Margins, Other Core Physics Parameters, Reactor Control System, Thermal-Hydraulic Characteristics and Reactor Coolant System.

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