LA 84-55

Westinghouse Electric Corporation

Water Reactor Divisions

Box 355
Pittsburgh Pennsylvania 15230

July 9, 1984

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Washington, D.C. 20555

Attention: Mr. Cecil O. Thomas, Chief
Standardization and Special Projects Branch

Gentlemen:

Subject: Submittal of Additional Information to Application for Renewal of License Number R-119 (Docket 50-87).

In response to your request for additional information dated May 4, 1984, the Westinghouse Electric Corporation hereby submits the enclosed additional information. This information updates the application for renewal submitted December 18, 1981 and amended by letter dated January 5, 1982. Attachment 1 to this letter provides a summary of the documents enclosed with this letter.

If you have any questions concerning this submittal, please contact me at the above address or by telephone on (412) 374-4652.

Very truly yours,

A Joseph Nardi

A. J. Nardi, Manager
NES License Administration

AJN/jh

Attachment

Copies Transmitted: 3 notarized & 19 conformed

COMMONWEALTH OF PENNSYLVANIA)
COUNTY OF ALLEGHENY) ss

A020
3/22

Sworn and subscribed before me this

9th day of July, 1984

Genevieve Kish
Notary Public

Commission Expires

GENEVIEVE KISH, NOTARY PUBLIC
MONROEVILLE BORO, ALLEGHENY COUNTY
MY COMMISSION EXPIRES SEPT. 3, 1984
Member, Pennsylvania Association of Notaries

ATTACHMENT 1 TO LETTER LA 84-55

The following presents a summary of the five enclosures transmitted with this letter.

Tab 1 - Reply to NRC License Review Committee Questions. These responses are keyed to the question numbers given in the NRC letter dated May 4, 1984.

Tab 2 - Revised FSAR including a summary of the changes made. This document replaces in its entirety the information included as Section III in the application for renewal dated December 18, 1981.

Tab 3 - Revised Proposed Technical Specifications including a summary of the changes made. This document replaces in its entirety the information included as Appendix A in the application for renewal dated December 18, 1981.

Tab 4 - Current Emergency Plan for the Facility. This updates the version of the Emergency Plan included as Appendix B in the application for renewal dated December 18, 1981.

Tab 5 - Current version of Operator Requalification Program. This document has no substantial changes, but has been revised to the N-24-S and the inclusion of previous appendices. This document was included as Appendix C in the application for renewal dated December 18, 1981.

REPLY TO NRC LICENSE
REVIEW COMMITTEE
QUESTIONS
FOR THE

WESTINGHOUSE NUCLEAR TRAINING REACTOR
FACILITY LICENSE NO. R-119
DOCKET NO. 50-87

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1. What are the reactivity worths at the individual control rods in the current core?

The reactivity worths of the control rods in the current N-24-S core is summarized below:

Rod Number	Integral Rod Worth (pcm)	Peak Differential Rod[2] Worth (pcm/.01 Turn)
1	10,060[1]	18.2
2	2,720[2]	7.5
3	2,775[2]	7.5
4	2,525[2]	7.1
5	2,450[2]	6.8

[1] Based on values determine experimentally with a subcritical reactor.

[2] Based on values reported in NTR Modified Core Test Program Results.

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2. What are the average operating power levels and total yearly use in kilowatt hours?

Below is a summary of the total integrated power and the average power as reported to the NRC in the Annual Operating Reports for 1979 through 1983.

Year	Integrated Power (KW-hrs)	Average Power Watts
1983	28.5	21.4
1982	56.2	23.3
1981	24.7	17.4
1980	50.9	25.6
1979	45.1	24.3

3. What is β for the WNTR reactor? How much excess reactivity is in the current core? How .s shutdown margin calculated?

The NTR is a small graphite reflected core with 93.5 percent U-235 fuel thus β_{eff} is:

$$\beta_{eff} = \beta \times I = .0069 \times 1.25$$

$$\beta_{eff} = .008$$

The excess reactivity available in the N-24-S core is 4400 pcm. This value is from the data reported in the NTR Modified Core Test Results.

Shutdown margin is the reactivity in the core or that can be rapidly inserted into the core below that required for criticality. The shutdown margin is thus calculated by summing the integral rod worth below the critical bank height (4.9T). Rod ←1 integral rod worth has been estimated below 3.5T by use of subcritical multiplication theory since the reactor cannot be made critical with rod ←1 below that height. The shutdown margin in the W-24-S core is 17.7 percent $\frac{\%K}{K}$.

4. How is nuclear power calibrated?

Power Calibration - During the initial core physics testing a map of the neutron flux level at a constant power level was made by irradiating gold foils. This data was compiled and used to calculate the average power and the Hot Channel Factor (Φ_{peak}/Φ_{avg}). Channel A is calibrated by placing the gold foils in this peak flux and comparing Channel A output with the irradiated gold foil results.

Channels E and F power calibration (watts/amp) are then determined by a linearity check with Channel A.

5. What is the normal fission product inventory of a fuel rod?

The average NTR power level in 1982 was 23.3 watts. This reference year accumulated the highest number of NTR operating hours to date. It operated only 2419 hours in the year or approximately 28 percent of the year.

Therefore, we can assume an average continuous power level of 6.43 watts for equilibrium conditions. Using 6 curies/watt as a fission product inventory rate yields a core total of 39 curies. Thus, the per element average is 1.63 curies.

6. Supply the analysis for the maximum step reactivity insertion accident.

The step reactivity addition to the core is limited to .80$ due to a common mode failure of an unsecured experiment.

A step insertion of .80$ of positive reactivity to a critical reactor will cause a prompt jump in reactor power of approximately 5 fold. This is determined by adding ρ=.0064 with β_{eff}=.008 to the equation:

$$\frac{\Phi_{(t)}}{\Phi_{(o)}} = \frac{\beta\ (1-\rho)}{\beta - \rho}$$

[Lemarsh, John R., Introduction to Nuclear Reactor Theory, Sept. 1972, Addison-Wesley Publishing Company p. 440.]

Under normal operations the reactor power would be maintained at less than 100 watts with an unsecured experiment in the core. Thus a failure of the unsecured experiment adding .80$ of positive reactivity would cause a prompt jump to less than 500 watts. The reactor would then assume a stable period of approximately 1 second.

The rate trip of τ = 3 sec would trip the reactor. If the rate trip failed the reactor would be tripped by overpower at 12 kw. Thus no significant power would be produced and there would be no danger to the public.

Assuming the highest steady state power limit of 10 kw at the time of the .80$ step insertion, power would go to 50 kw and the reactor would trip by either overpower or τ < 3 sec. This power is much less than that analyzed in the maximum credible accident and thus would not pose a danger to the reactor fuel or to the public.

7. Describe the administrative organization of the radiation protection program, including the authority and responsibility of each position identified.

The Radiation Safety Coordinator duties relate to the prevention and control of potential hazards associated with radiation and with the handling, use and disposal of radioactive materials. In cooperation with the Site Radiation Officer and the NTR Facility Manager, the Radiation Safety Coordinator ensures that special radiation safety procedures are performed and that proper radiation safety records are kept. The Radiation Safety Coordinator collects samples, performs analyses, makes measurements, maintains records and generally administers the technical aspects of the radiation protection program.

The Radiation Safety Coordinator reports to the WNTR Facility Manager who is responsible for the overall radiation safety and protection programs for the WNTR. The WNTR Facility Manager reports directly to the WNTC Manager who also serves as the Site Radiation Officer. The Site Radiation Officer is responsible for the radiation safety and protection program for the entire training center.

8. Describe any radiation protection training for the non-health physics staff. If possible, provide a topic outline of the courses and indicate the normal duration of each course or lecture.

The operating staff of the WNTR have completed the initial SRO Training Program prior to licensing. This program typically includes the following topics as they pertain to radiation safety:

Subject	Hours
Radiation Safety	3.0
Operations Manual Review Sec. 3.4	
Biological Effects of Radiation (NEP-213)	
Review 10CFR 20	
Measured Radiation Levels at NTR Facility Under Various Operating Conditions	
High Radiation Areas-Control of Access to	
NRC I.E. Information Notices 81-26 Part 3, 83-59 and 83-60	
Portable Survey Instruments	1.5
Radiation Detection Principles and Instruments	
Demonstrations and OJT (Including Calibration Procedure)	
Nuclear Instrumentation	3.5
Principles of Radiation Detection	
Nuclear Reactor Instrumentation	
Radioactive Materials and Fuel Handling	2.0
Emergency Procedures and Equipment	2.5

Fuel Handling (OJT) 4.0

Performing Radiation Surveys

Computing Stay Time

Moving Fuel

Performance of Surveillance Tests

Measurement of Reactor Water for Radioactive Contamination

Measurement of Reactor Room Air for Radioactive Contamination

Smearable Contamination Measurements

Use of Laboratory Counting Equipment

Most members of the staff have extensive backgrounds in radiation protection from industrial or military experience and also have completed the Nuclear Engineering Principles course using a text comparable with Radiation, Chemistry and Corrosion Considerations for Nuclear Power Plant Applications. Topics typically include:

- Radioactivity and the Laws of Radioactive Decay
- Radiation and its Interaction with Matter
- Biological Effects of Radiation
- Protection Against Radiation
- Radiation Detection Principles and Instruments

9. Describe your program to ensure that personnel radiation exposure and releases of radioactive material are maintained at a level that is "as low as reasonably achievable" (ALARA). Identify steps taken to implement the ALARA principle.

ALARA principles are utilized at the WNTR Facility. The radiation levels in the facility are kept at a minimum by operating the reactor at minimum power levels, usually less than 100 watts. They are also restricted by the safety limit on integrated power of 200 kw-hrs per year.

Low limits on the activity levels of the water and air as well as regular surveillances of the reactor room minimizes the contamination to extremely low levels.

Emergency procedures for radioactive material spills, reactor excursions and accidental criticality outlined in the Operating Manual have been established to minimize personnel exposure should an accident occur. Periodic drills with post drill critiques assure that the procedures are operable and staff members are well trained.

Reactor fuel handling procedures in the Operating Manual outline requirements for protective clothing, training, dosimetry and tool use. These procedures set forth those requirements which allow training with radioactive materials while minimizing exposure of personnel as low as reasonably achievable.

10. Identify the generic type, number, and operable range of each of the portable health physics instruments routinely available at the reactor installation. Specify the methods and frequency of calibration.

The generic type, number, and operable range of the portable HP instruments routinely available at the NTR are tabulated per Table 10.1. These instruments are calibrated on a semiannual basis.

Methods of calibration are those specified in WNTR Radiation Monitoring Instruments Calibration Procedure. This procedure directly reflects the methods of calibration as given in the instrument manufacturers' service manuals.

TABLE 10.1

PORTABLE HEALTH PHYSICS INSTRUMENTS

Detector	Generic Type	Operable Range	Number
$\beta-\gamma$	G-M	.5-50 mR/hr	1
$\beta-\gamma$	G-M	.2-250 mR/hr	1
$\beta-\gamma$	I-C	.1 mR/hr-1 KR/hr	1
γ	I-C	1 mR/hr-1.9 KR/hr	1
α	SC	0-500,000 cpm	1
n	BF_3	0-5 rem/hr	1

11. Describe your personnel monitoring program.

The WNTR Personnel Radiation Monitoring Program consists of the use by the NTR staff and students of self-reading pocket dosimeters, TLD's, and finger ring film badges.

The pocket dosimeters used by the staff and students are recorded and re-zeroed at weekly intervals by the Radiation Safety Coordinator as well as read daily and/or as required by the individual.

Visitors to the NTR are monitored by self-reading pocket dosimeters that are read and recorded on a per visit basis on the NTR Visitor's Log.

All TLD's and finger ring film badges are sent to and read by an outside vendor on monthly intervals. The vendor supplies the monthly and accumulated quarterly, yearly and lifetime doses in a report that is posted in the facility by the Radiation Safety Coordinator.

NTR staff members are also required to submit a yearly bio-assay sample.

12. Provide a summary of the reactor facility's annual personnel exposures [the number of persons receiving a total annual exposure within the designated exposure ranges, similar to the report described in 10CFR20.407(b)] for the last 5 years of operation.

A summary of the NTR annual personnel exposures for the previous five years of operations is provided in Table 40.1.

Table 42.1

Yearly Personnel Exposure

Whole-Body Exposure Range (REM)	Number of Individuals in Each Range 1979	1980	1981	1982	1983
No Measurable Exposure	236	332	383	408	416
Measurable Exposure < 0.1	21	7	18	27	19
0.1 to 0.25	2	4	4	0	0
0.25 to 0.5	0	0	0	0	0
> 0.5	0	0	0	0	0
Total Number of Individuals Monitored	259	343	405	435	435

13. Describe the Westinghouse Training Center fire protection programs.

The WNTC Fire Protection Program consists of system hardware, operational checks, employe training, and fire drills.

All employes working at the training center are instructed as to the use and location of detectors, pull stations, and extinguishers. They are made aware of the sound of the fire alarm and directed on how to correctly respond to the alarm. Employes are also told of their responsibilities to the training center, their students and themselves in the event of a fire.

The system hardware includes the following:

- Fire bell alarm system with battery back-up power supply
- Alarms initiated in the training center are also alarmed at the Zion Emergency Board located at the Zion Police Station
- Alarm pull stations in the hallways
- Heat sensing elements throughout the building
- Fire doors in the north-south hallway by the SNUPPS simulator and by the NTR. These doors are actuated by smoke detectors on both sides of the door.
- Smoke detectors in the ventilation lines of the EOF building
- CO_2 and ABC extinguishers throughout the facility
- Cardox CO_2 unit is installed in the Zion Computer Room
- Halon unit is installed in the SNUPPS Computer Room

Annual inspections and operational checks of all detectors, alarms, and extinguishers are conducted by an outside contractor. Monthly visual inspections and weighing of extinguishers is conducted by training center personnel.

13. Continued

Also, on a yearly basis a full scale fire drill is conducted with the full participation of all training center employes and the Zion Police and Fire Departments.

14. What is the minimum critical mass for the current core array with graphite reflection?

For the current core configuration of twenty graphitar reflectors in a hexagonal array and all control rods at 8.0 turns, twenty-one (21) standard fuel elements are required for a critical mass. The critical mass would be equal to

$$21 \text{ Standard Fuel Elements} \times \frac{200 \text{ grams U-235}}{\text{Standard Fuel Element}} = 4200 \text{ gms U-235}$$

With core positions 4-7, 5-8, and 6-8 empty and all control rods at 8.0 turns reactor power increases on a stable 57 second period for a K_{excess} of 121.8 pcm.

15. Technical Specifications No. 4 "Experiments", should limit the potential reactivity additions resulting from the failure of any and/or all unsecured experiments that may be in the core at any one time.

Section 4.4 of the Technical Specifications submitted for Facility License No. R-119 has been modified to read as follows:

- The maximum reactivity worth of any individual unsecured experiments shall be limited such that the failure of any experiment or associated equipment will not result in a positive reactivity addition greater than 0.80$. In addition the reactivity worth of all unsecured experiments is limited such that a common mode failure of all such experiments and their associated equipment will not result in a positive reactivity addition greater than 0.80$.

16. The terms "secured experiments", "unsecured experiments", and "movable experiments" should be clearly defined in the Technical Specifications if the terms are to be used in limiting the conditions of operations.

The terms have been defined in revised Technical Specifications in Section 1.0 "DEFINITIONS" as given below:

1.10	Movable Experiment	4	Definition added
1.20	Secured Experiment	6	Definition added
1.21	Unsecured Experiment	6	Definition added

17. Describe nearby transportation, heavy industry, and military facilities that by virtue of respective accidents could affect the safety of the Westinghouse Training Reactor.

The description of the local industries in the vicinity of the NTR are discussed in section 3.5.2 of the FSAR. These industries are all light industries posing no threat to the NTR.

Section 3.5.3 of the FSAR discusses the nearby transportation, heavy industry and military facilities and there risk to the NTR due to accidents at these facilities. These sections from the NTR are printed below.

3.5.3	Heavy Industry, Transportation, and Military Facilities	25	Added as required by NRC

3.5.3 Heavy Industry, Transportation and Military Facilities

The industries located in the vicinity of the reactor are discussed in section 3.5.2. These industries consist of only light industries.

The Waukegan Memorial Airport, located approximately 4 miles southwest of the reactor sight does not have regularly scheduled commercial air carrier service. It is utilized primarily for executive and small private air crafts. Chicago and Northwestern Railroad tracks are located approximately 0.4 miles to the west of the NTR location. This line is used for both passenger and freight transportati...

The major military installation is the Great Lakes Naval Training Center located approximately 10

miles south of the NIR site. Ordinance of this base is limited to a small arms practice range.

The staff concludes that there is virtually zero probability of risk of accident to the reactor from activities associated with military, heavy industry or transportation operations.

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SUMMARY OF NTR FSAR CHANGES

Section	Pages	Content/Reason
Title	1	Amended divisional title to reflect W reorganization
Table of Content	8	Updated content to agree with changes in MCA
List of Tables	9	Updated tables to agree with changes in MCA
List of Figures	10	Updates figures to agree with changes in MCA
1.0 Introduction	13	Revised years of operation
2.0 Background	14	Amended divisional title to reflect W reorganization
	15, 16	Revised years of operation
3.0 Site and Environment	17	Corrected mileage
	19	Removed seasonal population
	29	Corrected temperature
3.5.3 Heavy Industry, Transportation, and Military Facilities	25	Added as required by NRC

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Section		Pages	Content/Reason
4.0	Reactor Facility Description	47	Changed dimensions from decimal to fraction to agree with drawing
		51	Corrected dimensions to agree with drawing
		55	Corrected rod speeds
		57	Revised description of fill system
		58	Revised instrument range
		58, 59	Word change for clarity
		60	Changed table to agree with description
		61, 62, 63, 64, 65	Changed numerical order and wording changes for clarity
		67, 68	Revised main fuel storage to reflect current methods
		68, 71, 73	Wording changes for clarity
5.0	Organization and Administration of the Facility	87	Amended divisional title to reflect W reorganization
		90, 95, 98	Revised responsibility to reflect current NTR organization and procedures

Section		Pages	Content/Reason
6.0	Initial Reactor Testing and Reactor Operations	105, 106 108, 109	Revised responsibility to reflect current NTR organization and procedures
7.0	Safety Analysis	110	Revised years of operation
		117	Revised temperature rise to a more conservative value
		120	Unit designator change
		122	Revised dimension to agree with drawing
		129 - 137	Revised MCA

Westinghouse Electric Corporation

Nuclear Services Integration Division

Pittsburgh, Pennsylvania (15230)

Application for Renewal of

Facility License R-119, Docket 50-87

Section III

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(Final Safety Analysis Report)

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CHAPTER 1

INTRODUCTION

As required under Title 10, Code of Federal Regulations, Part 50, this document is compiled as the Safety Analysis Report for the Westinghouse Nuclear Training Reactor located at the Westinghouse Nuclear Training Center, in Zion, Lake County, Illinois. It is the general purpose of this Report to substantiate the safety aspects of the facility and to describe clearly the facility design and intended use. This Report includes sections which cover the design bases and safety analysis of the facility in accordance with the provisions of 10 CFR 50.35 b.

The reactor facility is housed in a 3200 square foot enclosure which makes up the south wing of the Westinghouse Nuclear Training Center. The boundaries of the facility are established to provide the proper radiation safety control and facility security. The reactor is installed in a separate reactor room and the reactor core is situated approximately eleven (11) feet below ground level. Adjacent rooms house the support facilities for the reactor including the console room. The reactor is a light water-moderated, graphite reflected and light water shielded, highly enriched uranium-aluminum, low power system. The reactor core, core support structure, moderator-shield water, graphite reflector rods and instrumentation are contained in an open eight (8) foot diameter,

aluminum tank. Access into the core is gained only from the top of the reactor tank at ground level.

The primary use of the reactor facility will be in support of nuclear training programs conducted by the Westinghouse Strategic Operation Division. the reactor is utilized to conduct demonstrations and to provide operating experience for (W) customer personnel in the areas of fundamental reactor physics and reactor operations. Other minor activities of the reactor facility include irradiation experiments and reactor instrumentation studies.

Because the reactor has been operated at Waltz Mill, Pennsylvania, under AEC Facility License CX-11 for a period of twelve years and at Zion, Illinois for a period of twelve years under NRC License R-119, its operating and safety characteristics are well known. Therefore, many of the safety considerations within this document have the added credibility of actual experimental and operational proof. This consideration will be utilized throughout this report.

CHAPTER 2

BACKGROUND

2.1 GENERAL DESCRIPTION OF NTR FACILITY IN THE WESTINHOUSE NUCLEAR TRAINING CENTER

The Nuclear Training Reactor (NTR) is owned and operated by the Westinghouse Electric Corporation. The Nuclear Training Operations Group, Nuclear Services Integration Division, Water Reactor Divisions will be responsible for the management of the reactor facility. The reactor is located in the City of Zion, Illinois, and is part of the Westinghouse Nuclear Training Center. The Nuclear Training Center (NTC) is a joint venture between Westinghouse and the Commonwealth Edison Company of Illinois (Commonwealth Edison). The NTC building and land is owned by Commonwealth Edison and the internal training and support equipment is owned by Westinghouse. The training programs performed at the NTC and their execution is the responsibility of Westinghouse and the NTC is operated solely by Westinghouse employees. The NTC consist of the reactor facility, two PWR power plant simulators, simulated PWR power plant console rooms, associated laboratories, classroom areas and office space, totaling over 40,000 square feet. Its primary function is the training of Westinghouse customer personnel in the operations, design and management of pressurized water reactor power plants.

The reactor facility is located in the isolated south wing of the NTC. The main entrance into the reactor facility is through the NTC building.

2.2 HISTORY OF THE REACTOR SYSTEM

The Nuclear Training Reactor (NTR) has been in existance as a reactor facility at Zion, Illinois for the past twelve years. The Reactor System achieved initial criticality under NRC Licesne R-119 in February 1972. In August of 1981, an NRC concurred 50.59 change was instituted that changed the normal core configuration from a 37 element, water reflected configuration to a twenty-four element, graphite reflected configuration. A complete and comprehensive analysis was conducted on the new core configuration that concluded that the change involved no unresolved safety questions and an amendment to the facility license was not required.

2.3 SUMMARY OF PREVIOUS REACTOR OPERATING EXPERIENCE AND REACTOR PERFORMANCE

The versatility of the Nuclear Training Reactor (NTR) as a critical assembly, experimental test reactor, irradiation facility and training reactor has been established throughout its existence. the basic reactor system has shown exceptional dependability and lack of mechanical problems.

Approximately eighty (80) percent of the operational time of the reactor has been used for training programs, fifteen (15) percent for maintenance operations and five (5) percent for irradiation experiments. During the past ten year period, training experiments and demonstrations have been developed, modified and improved. The results of this developmental work now constitute a formal operational training program in fundamental reactor physics and reactor operations. To date, over eight hundred individual customer trainees have received operational training on the reactor facility located in Zion, Illinois.

The general operating experience with the reactor systems with respect to equipment failure and malfunction has been very good over the past twelve years. No single major component failure has occurred and only normal mechanical and electrical problems have been apparent Frequent maintenance checks and inspections have been utilized throughout the existence of the reactor to detect deterioration of mechanical and electrical components and to prevent failure. The established scope and frequency of this maintenance have proven to be fully adequate to prevent major problems from arising and will continue to be utilized in the maintenance program for the reactor facility at NTC, Zion.

CHAPTER 3

SITE AND ENVIRONMENT

3.1 GENERAL

Information presented herein is based largely on the site analysis for Commonwealth Edison's Zion Station as presented in the Preliminary Safety Analysis Report (Docket No. 50-295 and 50-304). The NTR facility is located approximately 2400 feet from the Zion Station, adjacent to the Zion exclusion area as shown in Figure 3.2.3. The proximity of the two facilities makes the Zion site data valid for the NTR site in almost every detail.

The site is located in Northeast Illinois on the west shore of Lake Michigan about 40 miles north of Chicago, Illinois, and about 40 miles south of Milwaukee, Wisconsin, as shown in Figure 3.2.1.

The site is covered mainly by sandy soil with patches of peat and muck. The site is well ventilated and not subject to severe persistent inversion. While tornadoes occur in the region, none have been reported to effect the lake shore site directly. High winds (on the order of 70 mph) can be expected once in 50 years from storms.

3.2 LOCATION

The site is located in the extreme eastern portion of the city of Zion, Lake County, Illinois, on the west shore of Lake Michigan approximately 6 miles NNE of the center of the city of Waukegan, Illinois, and 8 miles south of the center of the city of Kenosha, Wisconsin. It is located at longitude 87° 48.1' W and latitude 42° 26.8'N.

The Zion site comprises approximately 250 acres which is owned by Commonwealth Edison. The site is traversed from west to east by Shiloh Boulevard near the northern property boundary and by no through ways from south to north. Figure 3.2.1 shows the general topography of the region within a 5 mile radius of the site. Figure 3.2.2 is an aerial photograph depicting the site boundaries and details of the site.

3.3 TOPOGRAPHY

The topography of the NTR site and its immediate environs is relatively flat with a mean elevation of approximately 586 feet which is 6 feet above the level of the lake. Approximately two miles west of Lake Michigan there is a topographical divide causing surface water drainage west of the divide to be away from the lake while to the east drainage is toward the Lake. The site itself has very little slope and is relatively marshy.

3.4 POPULATION DISTRIBUTION

Table 3.4.1 lists the population centers of over 25,000 people within a radius of 50 miles of the Zion site, their geographical relation to the Zion site and their populations. The distance to the closest boundary of the nearest such population center, Waukegan, Illinois, is 3.6 miles. These data are for the Zion Station and would be changed only slightly for the NTR site because of the 2400 ft. difference in location.

Additional population distribution data appears in the Zion Station Preliminary Safety Analysis Report, referenced in Section 3.1. The 1981 population estimates in Table 3.4.2 are based on aerial photographs an U.S. Census Bureau 1980 figure for number of people per household. Based on past trends in populations and probable future industrial, commercial, residential and recreational developments, the total projected population of Lake County, Illinois, is 669,000 in 1985; and 2,000,000 in 2000. The Illinois State Beach Park immediately to the south of the site maintains picnic and camping areas. Approximately one mile south of the site, within the park, there is also a 100 room lodge which is used year round.

TABLE 3.4.1

Population Centers of > 25,000 Inhabitants Within 50 Miles of the Zion Site.

Community	Distance From The Site (miles)	Direction	Population (1980 Census)
Waukegan, IL	7	SSW	67,653
North Chicago, IL	7	SSW	38,774
Kenosha, WI	9	N	77,685
Highland Park, IL	18	S	30,611
Racine, WI	22	N	85,725
Northbrook, IL	23	S	30,735
Palatine, IL	26	SSW	32,166
Glenview, IL	26	S	30,842
Wilmette, IL	26	S	28,229
Evanston, IL	27	S	73,706
DesPlaines, IL	28	SSW	53,568
Mount Prospect, IL	28	SSW	52,634
Skokie, IL	28	S	60,278
Niles, IL	29	SSW	30,363
Park Ridge, IL	30	S	38,704
Hoffman Estates, IL	32	SSW	38,258
Elk Grove Village, IL	32	SSW	28,907
Schaumburgh, IL	33	SSW	52,319

Hanover Park, IL	35	SSW	28,850
Greenfield, WI	37	NNW	31,467
Elgin, IL	37	SW	63,798
Oak Park, IL	37	S	54,887
Addison, IL	37	SSW	28,836
West Allis, WI	38	N	63,982
New Berlin, WI	38	NNW	30,529
Milwaukee, WI	40	N	636,212
Maywood, IL	40	S	27,998
Chicago, IL	40	S	3,005,072
Elmhurst, IL	40	SW	44,251
Berwyn, IL	41	S	46,849
Cicero, IL	42	S	61,232
Lombard, IL	42	SSW	37,295
Wheaton, IL	43	SSW	43,043
Janesville, WI	44	NW	51,071
Wauwatosa, WI	44	NW	51,308
Waukesha, WI	44	NW	50,319
Downers Grove, IL	45	SSW	39,274
Brookfield, WI	46	NNW	34,035
Naperville, IL	47	SSW	42,330
Burbank, IL	48	S	28,462
Menomonie Falls, WI	50	NNW	27,845

TABLE 4.2

SECTOR AND ZONE DESIGNATORS AND POPULATION DENSITY WITHIN 50 MILES

ZION STATION

SECTOR CENTERLINE IN DEGREES TRUE NORTH FROM FACILITY	22½° SECTOR	MILES FROM FACILITY 0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50	SECTOR TOTAL
		1980 POPULATION										
0+360	A	0	0	0	101	358	20,233	81,597	62,364	403,938	106,233	674,824
22½	B											
45	C											
67½	D											
90	E			LAKE MICHIGAN								
112½	F											
135	G											
157½	H								86,341	612,905	462,651	1,161,897
180	J	0	0	0	0	0	4,809	64,394	297,198	1,282,285	1,247,677	2,896,363
202½	K	0	432	1,481	473	2,030	108,616	39,958	261,583	365,143	344,392	1,124,108
225	L	0	1,043	1,389	797	1,605	13,538	31,875	61,285	114,033	34,518	260,083
247½	M	80	3,846	1,062	449	765	1,836	37,496	37,671	44,082	23,793	151,080
270	N	176	4,465	1,718	462	488	1,165	22,834	13,973	14,462	21,004	80,747
292½	P	33	4,465	2,730	518	107	748	21,477	32,478	12,402	12,870	87,828
315	Q	0	888	1,033	524	599	1,929	19,702	29,197	27,648	30,866	112,386
337½	R	0	78	3,263	1,138	647	42,063	38,327	20,844	475,496	239,360	821,216
RADIAL ZONE TOTAL		289	15,217	12,676	4,462	6,599	194,937	357,660	902,934	3,352,394	2,521,364	7,370,532

Notes: 1. 0-10 mile population estimates were derived by house counts from 1980 aerial photographs and the U.S. Census Bureau 1980 figure for number of people per household, also the 1980 population for all incorporated areas were visually proportioned to each sector.

2. 10-50 mile population data represents the incorporated areas for which 1980 Census Bureau information was available and estimates of unincorporated areas which were derived on the basis of 1980 County Population minus the 1980 population data for the incorporated areas. Both incorporated and unincorporated areas were visually proportioned into each sector.

3.5 LAND USE

3.5.1 Regional Land Use

The Waukegan-North Chicago-Gurnee area is predominantly an industrial region with approximately 100 manufacturing establishments employing some 30,000 individuals. The product of the largest of these manufacturing firms is pharmaceuticals and chemicals with the most predominant product of the remainder being in the metallurgical and fabricated metal products field. None of the industries listed by the Waukegan-North Chicago Chamber of Commerce will represent a limitation to the operation of the NTR. The areas starting some 4 miles inland from Lake Michigan are predominantly agricultural regions. The agricultural products are grain crops (mainly corn and wheat), cattle, horses, dairy products and furs (mink ranches). The number of farms in Lake County, Illinois, is approximately 800 utilizing about 40 percent of the county land area. The nearest areas in which there are dairy farms are 5 miles to the west and to the northwest of the site near the Illinois-Wisconsin state line.

3.5.2 Local Land Use

The Zion-Winthrop Harbor area is a small industrial region, though there is extensive farming to the west of the area as previously discussed. A major portion of this industry is located between the

western boundary of the site and the Chicago and Northwestern Railroad tracks approximately 0.4 miles due west of the NTR location. The industries in this area are:

Name	Products	Employment
Midway Heating and Air Conditioning	Heating, Air Cond. and Refrig. Supplies	40
Zion Industries, Inc.	Cookies	75
Burges, Anderson and Tate	Printing and Office Supplies	100

There is also a warehouse located in this industrial park. Other industries in the area produce candy, clothing, and printing.

The Zion site is bordered on the south by the Illinois State Beach Park and to the north by relatively open marshy land with some scattered residents located principally along the lake shoreline.

The centers of the communities of Zion and Winthrop Harbor are located about 1.2 and 2.5 miles, respectively, from the NTR location. The areas within the Zion and Winthrop Harbor city limits are approximately 6.5 and 3.9 square miles, respectively.

3.5.3 Heavy Industry, Transportation and Military Facilities

The industries located in the vicinity of the reactor are discussed in section 3.5.2. These industries consist of only light industries.

The Waukegan Memorial Airport, located approximately 4 miles southwest of the reactor sight does not have regularly scheduled commercial air carrier service. It is utilized primarily for executive and small private air crafts. Chicago and Northwestern Railroad tracks are located approximately 0.4 miles to the west of the NTR location. This line is used for both passenger and freight transportation.

The major military installation is the Great Lakes Naval Training Center located approximately 10 miles south of the NTR site. Ordinance of this base is limited to a small arms practice range. The staff concludes that there is virtually zero probability of risk of accident to the reactor from activities associated with military, heavy industry or transportation operations.

3.6 HYDROLOGY

3.6.1 General

3.6.1.1 Surface Water

The Lake County Public Water District has located a water intake about one mile north of the Zion Plant site and about 3000 feet out in the Lake. This action negated the use of wells.

3.6.1.2 Ground Water

Ordinarily there will be no potable uses of ground water in the Benton or Waukegan townships. There are wells in the communities of Zion and Winthrop Harbor (Benton Township) maintained on a standby basis to meet emergencies. However, of these wells the one with the highest yield is 1025 feet deep due to a 700 foot drop since 1864 in the artesian pressure of the deep aquifers. These wells are located near the southern edge of Shiloh Park about 1-1/2 miles west of the Zion plant location. Considering this location and the topographical divide which causes surface water to drain toward the east, any effects on these ground water supplies is very unlikely.

3.6.2 General Lake Hydrology

The normal water level in Lake Michigan is approximately 580 feet above MSL. The maximum recorded water level is 583.2 feet above MSL which occurred in 1886 and the minimum recorded to date in 1964 at 576.6 feet above MSL.

In the general vicinity of the site, the 30-foot depth contour of the Lake is 1.2 miles, and the 60-foot depth contour 2.0 miles from the shore.

3.7.2.1 Currents, Tides, Waves, and Littoral Drift

A detailed description of the currents, tides, waves, and littoral drift appears in the Zion Station Preliminary Safety Analysis Report. A maximum elevation of wave run-up and wind tide was estimated to be 6.7 ft. above the normal water level (at an occurrence frequency of once in 500 years). A maximum seiche level of five ft. above lake was considered for the Zion site.

Of the two phenomena, the seiche presents the greater potential hazard to the site. Although of greater height, the deep-watch wave will be quickly dissipated as it over-runs the shore and is therefore of little consequences to structures located at some distance from the shore line. However, the seichel-generated wave will encompass a much greater quantity of water, and the rise in level will endure for longer periods of time. Since the NTR facility is located 3250 ft. from the shoreline land at an elevation 10 ft. above the normal lake level, the site is not endangered by either of the wave phenomena.

3.6.2.2 Potable Water Sources

The subsurface water table of the area is sloped to the east towards the lake. The shallow aquifers are the sand and gravel overburden and the underlying dolomite formations. The deep aquifers are sandstone and dolomite formations with a strata of shale above them. The "free water" in the shallow aquifers over the six county northeast Illinois region is 4.72×10^{12} gallons and in the deep aquifers is 3.53×10^{14} gallons. However, the artesian pressure of the deep aquifers has dropped some 700 feet since 1864.

Since 1957, the cities of Zion, Winthrop Harbor, and the Illinois Beach State Park plus a number of retail establishments in unincorporated communities have obtained their water from Lake Michigan via the Lake County Public Water District (see 3.6.1.1). Two older wells in Zion are maintained on a standby basis to meet emergencies.

The next nearest potable water intake which utilizes surface water from Lake Michigan is 6 miles south of the site at Waukegan. Potable water supplies from Lake Michigan are also located at Kenosha, Wisconsin, and North Chicago, Illinois, ten miles north and south, respectively, of the site. Others are located farther up and down the lake shore.

3.7 METEOROLOGY AND CLIMATOLOGY

The climate of the region around the site is primarily continental, with characteristic cold winters and warm summers. There is no dry season; precipitation occurs with some uniformity throughout the year. Average annual precipitation is about 33 inches, average annual snowfall is about 40 inches, and the mean annual temperature in the area is near 50°F.

Extreme winds are not expected to exceed 70 miles per hour once every fifty years. Tornadoes occur with relatively high frequency in Illinois, but are mostly found in the southern half of the state.

Northern Illinois is well-ventilated, with infrequent periods of calms. Most frequent wind direction occurrences are southwest and northeast during the warm months of the year, and southwest and northwest during the cool months. The lake breeze effect is an important factor in wind direction during the summer months. The longest duration of uninterrupted winds blowing from one direction was 39 hours from the northwest.

Some extremes of meteorological variables are listed in Table 3.7.1, below:

Table 3.7.1

Meteorological Extremes

	Chicago	Milwaukee
Highest Temperature	105°F (July 1934)	105° (July 1934)
Lowest Temperature	-28°F (December 1983)	-25°F (Jan. 1875)
Greatest Monthly Precipitation	14.17" (Sept. 1961)	10.03" (June 1917)
Greatest 24-Hour Precipitation	6.24" (July 1957)	5.76" (June 1917)
Greatest Monthly Snowfall	42.5" (Jan. 1918)	52.6" (Jan. 1918)
Greatest 24-Hour Snowfall	23.0" (Jan. 1967)	20.3" (Feb. 1924)
Maximum Wind Velocity	NE 87 mph (Feb. 1894)	SW 73mph (March 1954)

Data and analyses in Section 3.7 are based on five years of hourly observations from Milwaukee, Wisconsin and Chicago (O'Hare Airport), wind summaries from Waukegan, Illinois, summaries of climatological data from Wisconsin and Illinois and other reference data of a more specifice nature.

Hourly observations over the past six years at the CWE Zion Nuclear Station located adjacent to the NTR facility have detected no extremes greater than those tabulated above.

3.8 GEOLOGY

3.8.1 General

The site is located on the shore of Lake Michigan in the extreme

eastern portion of the city of Zion, Illinois, and occupies portions of Sections 22, 23, 26, and 27 in Township 46 North, Range 23 East. Marshy depressions and sand ridges comprise the principal surface features. The uppermost soils at the site consist predominantly of granular lake deposits. These sediments are underlain by glacial drift which consists of till, outwash, and lake deposits. Beneath the glacial soils, Paleozoic sedimentary rocks extend for several thousand feet to the depth of the crystal line Precambrian basement rock.

There is no evidence of faulting closer than the Des Plaines disturbances located approximately 25 miles southwest of the site. Other inactive faults exist at a distance of about 45 miles to the northwest and 75 miles to the southwest.

3.8.2 Descriptive Geolgoy

3.8.2.1 Regional Geology

Bedrock in the region consists of Paleozoic sedimentary rocks which rest on the Precambrian basement rock. The thickness of the Paleozoic sedimentary rocks in northeastern Illinois is approximately 4,000 feet. The bedrock dips gently toward the east at a rate of about 10 feet per mile.

The bedrock surface in the northeastern Illinois region is covered by a thick mantle of glacial drift, formed when most of the Wisconsin, Illinois and adjacent areas were subjected to repeated glaciation during the Pleistocene epoch. The advancing glaciers scoured major stream valleys and formed the large depressions now occupied by the Great Lakes. The glacial drift deposited by the glaciers consisted of till, outwash, and lacustrine deposits. Recent deposits in the region consist of unconsolidated sand, silt and peat.

3.8.2.2 Local Geology

The site is located on a narrow strip of lake deposits which borders the Lake Michigan shoreline. Crossing the NTR and adjacent Zion site is a series of low, parallel, beach ridges separated by marshy depressions. The beach ridges are composed primarily of sand. In the depressions, organic materials have accumulated.

The subsurface conditions at the adjacent Zion site were investigated by drilling seven exploration test borings. The test borings revealed that the site is blanketed by granular lake deposits which range in thickness from 24 to 33 feet. The granular lake deposits consist of fine and fine to medium sand which contains variable amounts of coarse sand and gravel and occasional pockets of peak and organic material. The granular lake deposits are underlain by Pleistocene glacial till, glacial outwash, and glacial lacustrine deposits. The glacial deposits consist essentially of silty clays,

clayey silts, and silt; contain variable amounts of sand and gravel; contain pockets of granular outwash; and extend to depths ranging from approximately 102 to 116 feet below the existing ground surface. The glacial tills and glacial lacustrine deposits are firm to hard and are relatively impermeable. A detailed description of the subsurface conditions at the Zion site is presented in the Zion Preliminary Safety Analysis Report (Docket 50-295 and 50-304).

Ground water is near the surface over much of the site area. The beach ridges project slightly above the water table, and most of the intervening depressions are marshy and are at or slightly below the water table. A very slight ground water gradient trends to the east and south. A stagnant condition now generally prevails between the beach ridges.

3.9 SEISMOLOGY

3.9.1 Summary

The region within 100 miles of the site is considered an area of minor seismic activity and has experienced a few earthquake events of moderate magnitude during the last 150 to 200 years. The regional earthquake events are shown in Figure 3.9.1, Regional Earthquake Events, and are summarized in Table 3.9.1, Regional Earthquake Occurrences. Earthquake intensities are described in terms of the Modified Mercalli Intensity Scale of 1931 which is explained in Table 3.9.2.

TABLE 3.9.1

REGIONAL EARTHQUAKE OCCURRENCES

DATE	INTENSITY	LOCALITY	EPICENTER LOCATION N.LAT.	W.LONG.	AREA SQ. MILES
1804 Aug. 20	VII	Ft. Dearborn	4.20	87.8	30,000
1811 Dec. 16	XII (Felt throughout Illinois)	New Madrid, Missouri	36.6	89.6	2,000,000
1812 Jan. 23	XII (Felt throughout Illinois)	New Madrid, Missouri	36.6	89.6	2,000,000
1812 Feb. 7	VII (Felt throughout Illinois)	Mew Madrid, Missouri	36.6	89.6	2,000,000
1883 Feb. 4	VI	North of Michigan-Indiana Border	42.3	85.6	8,000
1886 Aug. 31	X (Felt in Chicago)	Charleston, S. C.	32.9	80.0	2,000,000
1895 Oct. 31	VIII (Felt through out Illinois and Wisconsin	Charleston Missouri	37.0	89.4	1,000,000

TABLE 3.9.1

REGIONAL EARTHQUAKE OCCURRENCES

DATE	INTENSITY	LOCALITY	EPICENTER LOCATION N.LAT.	EPICENTER LOCATION W.LONG.	AREA SQ. MILES
1905 March 13	V	Menominee, Michigan	45.0	87.7	
1909 May 26	VII (IV at Kenosha)	N.E. Illinois	42.5	89.0	500,000
1912 Jan. 2	VI	N.E. Illinois	41.5	88.5	40,000
1917	VI	E. Missouri	38.1	90.6	200,000
1923 Nov. 9	V	Cass County, Illinois			
1931 Oct. 18	VI	Madison, Wisconsin			
1933 Dec. 6	IV	Stoughton to Putland, Wisconsin			
1934 Nov. 12	VI	Rock Island, Illinois	41.5	91.5	
1935 Nov. 1	VI (Felt in Wisconsin)	Timiskaming, Canada	46.8	79.1	1,000,000

TABLE 3.9.1

REGIONAL EARTHQUAKE OCCURRENCES

DATE	INTENSITY	LOCALITY	EPICENTER LOCATION N.LAT.	W.LONG.	AREA SQ. MILES
1939 Nov. 23	V (III at Janesville, Wisconsin)	Southern Illinois			
1943 Feb. 9	II	Thunder Mt., Marinette Co., Wisconsin			
1947 May 6	V	S.E. Wisconsin			
1947 Aug. 9	VI	So. Central Michigan	42.0	85.0	50,000
1956 July 18	IV	Postburg, Wisconsin			
1956 Oct. 13	IV	Milwaukee-Racine, Wisconsin			
1968 Nov. 9	VII (III At Site Location)	Southern, Illinois	38.0	88.5	

The principal sources of data are given in the Zion PSAR (Docket 50-295 and 50-304).

3.9.2 Descriptive Seismology

Northeastern Illinois is considered an area of minor seismic activity. King's distribution of epicenters contours the area as having approximately three epicenters per 10,000 square kilometers, a figure near the lower levels of his classification. The Seismic Zone Map of the United States prepared by U.S. Department of Defense, dated 1966, also indicates that the area is a zone of minor seismic probability. The site itself is free of known seismic disturbance.

Since the beginning of the 19th century, two earthquakes with epicentral intensities of VII, Modified Mercalli Intensity Scale of 1931, and with epicenters within a distance of 60 miles of the site are known to have occurred. The first of these earthquakes, near Fort Dearborn, Illinois, occurred in 1808 at an epicentral distance of approximately 35 miles from the site. The second occurred in 1909 south of the Illinois-Wisconsin border near Beloit, Wisconsin, at an epicentral distance approximately 60 miles from the site. Including the earthquakes described above, three earthquakes are known to have occurred within a distance of 50 miles with epicentral intensities ranging from III to VII, and nine earthquakes have been recorded within 100 miles with epicentral intensities ranging from

II to VII. In addition to these, a few very great but distant earthquakes may have been felt at the site but with very low intensity.

TABLE 3.9.2

MODIFIED MERCALLI INTENSITY SCALE 1931

(Abridged)

I. Not felt except by a very few under especially favorable circumstances.

II. Felt only by a few persons at rest, especially on upper floors of buildings. Delicately suspended objects may swing.

III. Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing motor cars may rock slightly. Vibration like passing of truck. Duration estimated.

IV. During the day felt indoors by many, outdoors by few. At night some awakened. Dishes, windows, doors disturbed, walls make creaking sound. Sensation like heavy truck striking building. Standing motor cars rocked noticeably.

V. Felt by nearly everyone, many awakened. Some dishes, windows, etc., broken; a few instances of cracked plaster; unstable objects overturned. Disturbance of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop.

VI. Felt by all, many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight.

VII. Everybody runs outdoors. Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structure; considerable in poorly built or badly designed structures; some chimneys broken. Noticed by persons driving motor cars.

VIII. Damage slight in specially designed structures; considerable in ordinary substantial buildings with partial collapse, great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, wall. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well water. Disturbs persons driving motor cars.

IX. Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb; great in substantial buildings, with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken.

X. Some well-built wooden structures, destroyed; most masonary and frame structures destroyed with foundations; ground badly cracked. Rails bent. Landslides considerable from river banks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks.

XI. Few, if any, (masonary) structures remain standing. Bridges destroyed. Broad fissures in ground. Underground pipe lines completely out of service. Earth slumps and land slips in soft ground. Rails bend greatly.

XII. Damage total. Waves seen on ground surfaces. Lines of sight and level distorted. Objects thrown upward into the air.

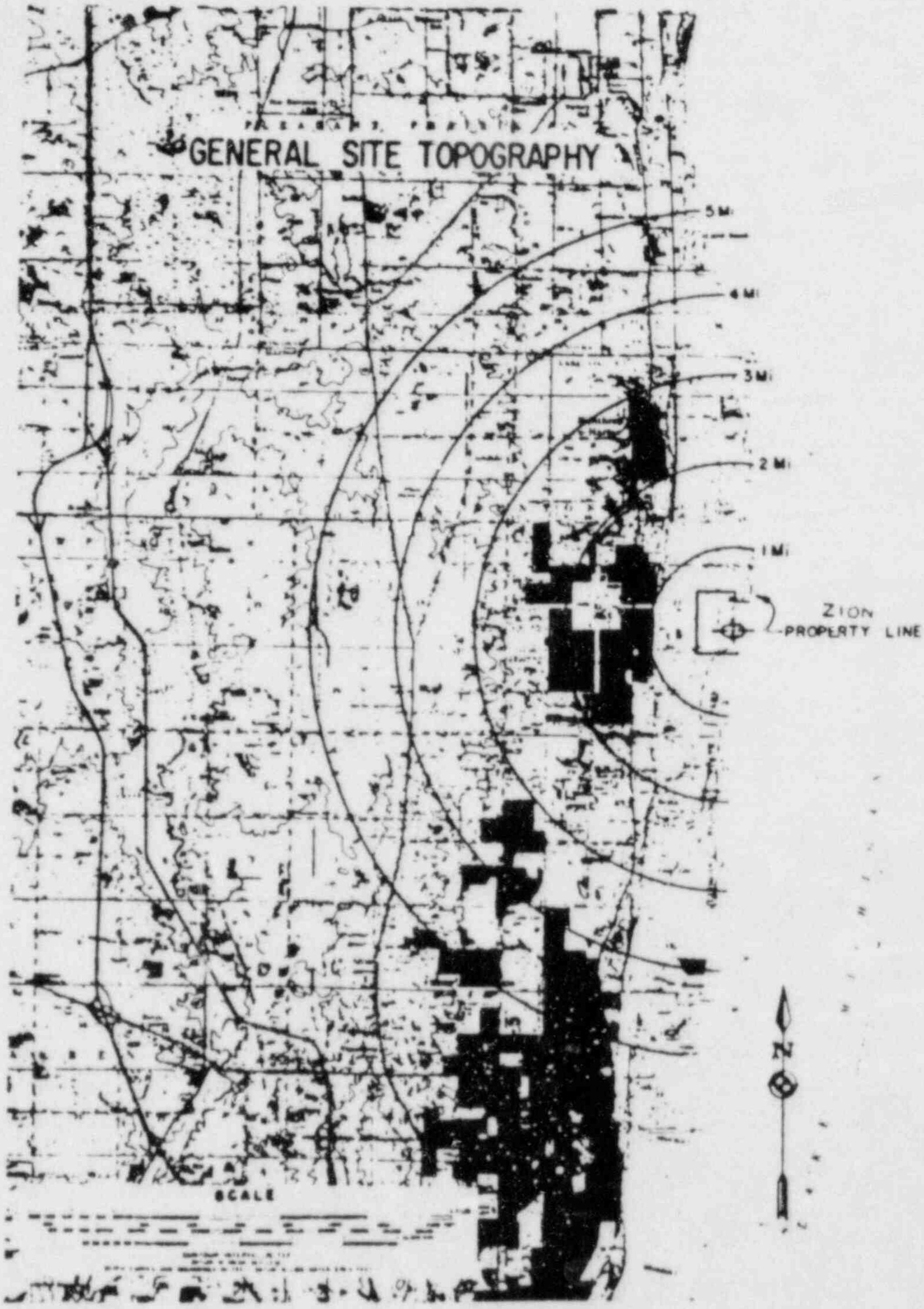


FIGURE 3.2.1: GENERAL SITE TOPOGRAPHY

FIGURE 3.2.2: SITE AERIAL PHOTOGRAPH

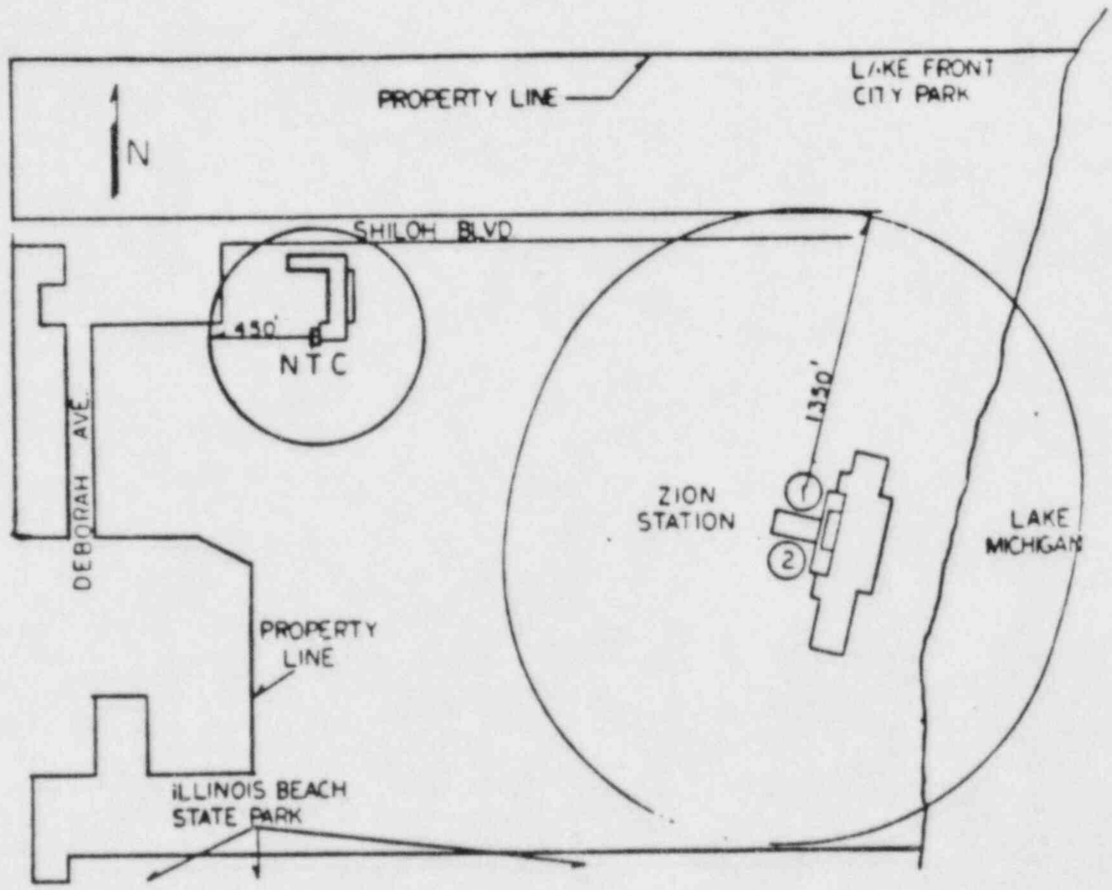


FIGURE 3.2.3: ZION SITE

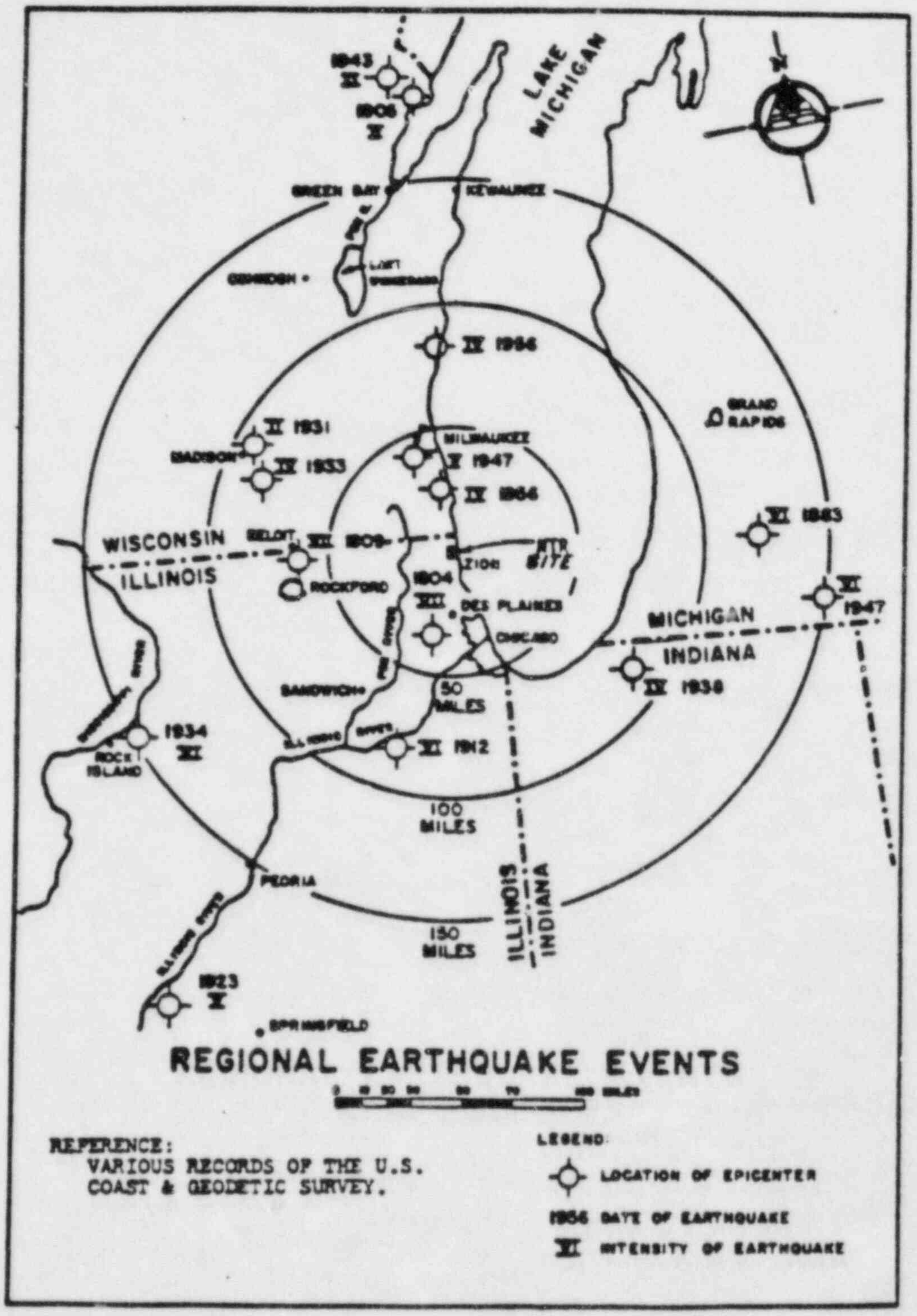


FIGURE 3.9.1: REGIONAL EARTHQUAKE EVENTS

CHAPTER 4

REACTOR FACILITY DESCRIPTION

4.1 REACTOR DESCRIPTION

The NTR is a highly enriched uranium, light water moderated and shielded, graphite reflected reactor. The aluminum-uranium alloyed fuel is contained in aluminum clad fuel tubes. A standard fuel element is an array of three concentric fuel tubes. The normal reactor core consists of nineteen (19) standard fuel elements and five (5) cadmium control rods with fuel element followers placed in the core structure in a hexagon configuration. The shape of the core is approximated by a cylinder with a height of 36 inches and a diameter of 16.08 inches. The total fuel loading is 4.8 kilograms of U-235 and the water to metal ratio is approximately 1.3:1. The core structure is located in the lower part of an 8 foot diameter, 19 foot deep aluminum reactor tank. The reactor tank is eccentrically positioned below ground level in a 12 foot diameter, 24 foot deep dump tank. Ten feet directly above the reactor tank is a platform to support the control rod drives and magnet carriages.

4.1.1. Reactor Components and Systems

4.1.1.1 Dump Tank

The reactor dump tank functions as a storage tank for the moderator-shield water. The tank is made of 3/8 inch thick aluminum, has an outside diameter of 12 feet and is 22 feet, 4-1/2 inches deep. The outside surface of the tank is coated with a Bitumastic coating to prevent corrosion. The tank itself is enclosed inside a cylindrical pit 24 feet, 1/2 inch deep. The pit is constructed of reinforced concrete with a minimum thickness of 30 inches for the base and 18 inches for the wall. The dump tank is recessed in the concrete pit, 20 inches below the floor level. the concrete pit provides the physical support for the dump tank and the suspended reactor tank (See Figure 4.1.1).

4.1.1.2 Reactor Tank

The reactor tank, which is constructed of 6061-T6 aluminum 3/8 inches thick, is 8 feet in diameter and 19 feet deep. The open ended tank is suspended within the dump tank near the top by a welded, six inch square, ring collar made of 1/2 inch thick aluminum, which rests on and is coupled to a horizontal I-beam structure. The I-beam structure consists of four (4), 12 inch beams welded into a square lattice resting on the concrete pit walls and embedded in the concrete floor. The reactor tank is positioned eccentrically in the dump tank so that its outer wall is 6 inches from the west side of the dump tank. This facilitates personnel and

equipment access into the dump tank on its east side. An additional collar is bolted against the reactor tank approximately 5 feet from its bottom. Extending from this collar to the dump tank are seven (7) horizontal support rods to prevent horizontal motion of the reactor tank.

The support for the core structure within the reactor tank is provided by an aluminum ring 1 inch thick and 26 inches wide supported by eight (8), 1/2 inch thick aluminum gussets. The ring and gussets are welded to the reactor tank 4 feet from the bottom of the tank. A 2 inch overlap is provided for the core structure whose base plate is 4 feet in diameter. Figure 4.1.1 illustrates these construction features.

4.1.1.3 Core Structure

The core is supported by the lower grid plate which is constructed of aluminum and is 48 inches in diameter. The lower grid plate is 2 inches thick at the edge and 6 inches thick in the center. This plate is directly coupled to the annular support ring in the reactor tank, thus providing support for the core and also preventing the core from shifting within the tank. There are one hundred and forty-four (144) positioning holes in the lower grid plate for fuel elements, control rods, associated shroud tubes and experimental equipment. The upper grid plate, constructed of 1 inch thick aluminum, has an elongated hexagonal shape and assures the proper

vertical alignment of fuel elements and control rods. The upper grid plate is located 44 inches above the lower grid plate and has ninety-three (93) positioning holes. The upper and lower grid plates are connected together by intervening aluminum core position guides or shroud tubes and held rigid by several tie rods.

Fuel elements are loaded downward through the upper grid plate into the shroud tubes. They are supported by the lower grid plate and are held in position by gravity. The control rod guide shrouds which are mounted on the upper grid plate serve as water dash-pots for the control rod shock absorbers and as support for the control rods when fully inserted in the core.

Due to the history of the reactor as an experimental facility, there exists in the core structure certain experimental positions. the center 7 shroud tubes have been made such that they are removable and an additional 7 positions are oversized such that an adapter sleeve is required to hold the standard fuel assemblies or graphite reflector rods in position. In addition, the outer fifty-one (51) positioning holes in the lower grid plate are available for experimental equipment. Figure 4.1.2 is a vertical section of the core support structure and Figure 4.1.3 is a top view of the core structure.

4.1.1.4 Fuel and Fuel Element

The standard fuel element consists of three (3) concentric tubes of aluminum-clad fuel alloy retained in configuration by web brackets at both ends as illustrated in Figure 4.1.4. Each standard fuel element contains approximately 200 grams of U-235 in its 36-inch long fuel-bearing region. The outside diameter of the standard fuel element is 2-1/2 inches and the length of a complete fuel element is 47-5/16 inches. The individual fuel tubes have a wall thickness of 1/8 inch. The fuel meat is made of an alloy of aluminum and uranium containing approximately 13 weight percent highly enriched (93.5%) uranium.

Other nominal dimensions and U-235 content for each of the three fueled tubes are given below:

	OD (in)	ID (in)	Clad (in)	Meat (in)	U-235 (gm)
Outside tube	2.50	2.25	0.0365	0.052	82
Middle tube	2.06	1.81	0.0365	0.052	67
Inner tube	1.62	1.37	0.0365	0.052	51

The center of each fuel element is available as a position for inserts. An aluminum thimble insert can be placed in position in the fuel. This thimble is supported by the upper web bracket of the fuel element.

The standard fuel element may be disassembled, reassembled, and loaded into the reactor in any combination of one, two, or three units (special fuel elements) provided that the maximum fuel loading per element does not exceed 200 grams of U-235. Fuel elements are normally loaded and unloaded from the reactor with a handling tool. This handling tool fits inside the fuel element top bracket and is positively actuated to grip the nozzle. The fuel may also be handled manually.

4.1.1.5 Graphite Reflector Rods

Twenty one (21) graphite reflector rods are available for use in the reactor. A graphite reflector rod consists of a 2.625 inch in diameter rod of type G83 graphite forty-eight inches long. The rod has a .500 inch hole bored axially the full length of the rod. A .500 inch aluminum support rod is inserted through each graphite rod with support spacers bolted on each end. A ball joint handling adapter is screwed and pinned to the top of each reflector rod support rod. Figure 4.1.5 indicates the structural support and construct features of a typical graphite reflector rod.

The normal reflector for the reactor consists of twenty (20) graphite reflector rods loaded in core positions indicated on Figure 4.1.3.

The reflector rods are normally loaded and unloaded from the reactor with a handling tool. This handling tool fits over the ball joint handling adaptor and is positively actuated to grip the adaptor. The reflector rods may also be handled manually.

4.1.1.6 Control Rods, Control Rod Drives and Platform

Five (5) control rods are provided to control the reactor. A control rod consists of a magnet armature, connecting linkage rods, stop cap, a poison section, fuel element follower section, and a shock absorber. The poison section is a 2-1/2 inch diameter unit. The absorber material is a 1/8 inch thick, 2-3/8 inch O.D. cadmiun tube clad with 1/16 inch aluminum, and is 36 inches in length. The fuel follower is linked 2 inches below the cadmium tube. The complete unit is linked together by a stainless steel coaxial rod. For operation, the poison section is withdrawn from the core as needed. The attached fuel element follower, which is identical to a standard fuel element except that its fuel tubes are 44 inches in length, becomes part of the core as it is drawn upward. Aluminum guide shrouds extend above and below the core grid plates. The bottom of the shroud prevents the assembly from dropping through the core. Holes in the sides of the tubes are designed to give a water dash-pot, shock absorber action when a control rod is dropped. Figure 4.1.6 indicates the structural features of a typical control rod.

The control rod drive mechanisms are located above the reactor core. The drive mechanisms are a rack and pinion type driven by electric motors. They are attached to the control rod shafts by an electromagnet. A reactor trip is accomplished by de-energizing these magnets allowing the control rods to fall into the core under the force of gravity. Magnet release time is less than 125 milliseconds and integral drop time is less than 1.2 seconds. Rod withdrawal speed and multiple electrical coupling of control rods drives are limited to conform with the established reactivity ramp insertion limits. The control rod drive platform is supported by two horizontal I-beams which extend to the reactor room east/west walls where they are supported by vertical I-beams. The platform is located 10 feet above and directly over the reactor tank.

4.1.1.7 Moderator-Shield

The moderator as well as the shield is deionized light water with a resistivity above 200,000 ohm-cm and pH of 4.5 to 8.0. The water is pumped from the dump tank to the reactor tank by the moderator transfer pumps. The level in the reactor tank is normally maintained a minimum 5 feet above the core for critical operations. The water can be removed from the reactor tank through either of two water lines, one of which is the main 10 inch (nominal) pipe and valve dump line and the other a 1 inch drain line. A 10 inch pipe to which the dump valve is attached opens into the reactor tank about 28 inches below the bottom of the core. The 1 inch valve is

located at the bottom of the reactor tank and is used to completely drain the tank. For arrangement of the moderator-shield water system refer to Figure 4.1.7.

4.1.1.8 Neutron Source

A neutron source is included in the core assembly to provide a sensible shutdown neutron population. Different types of sources can be utilized, but normally a two Curie PuBe (α n) source is used. The neutron source can be placed into the core manually or remotely by a special drive assembly. The manual method consists of placing the source inside a capsule attached to the end of a long chain. This assembly can be placed manually in any desired fixed location in the core. The remote system is an electrically driven assembly, including an out-of-reactor shield-storage container. The neutron source can be remotely positioned vertically in and out of the reactor by controls located at the reactor console. The drive mechanism can be placed in any appropriate core shroud tube position. Interlocks are provided for both manual and remote source placements to prevent starting the reactor up without the source in the reactor.

4.1.2 Reactor Control Systems

4.1.2.1 Control Rod Drive System

Five (5) remotely operated control rods are installed in the core. See Figure 4.1.3 for control rod location. The inner rod is the safety rod and the remaining four outer rods are shim rods. The control rods are coupled to the drive system through an electromagnetic carriage which must be energized to permit the withdrawal of the rods. If power to the magnets is interrupted, the rods will fall into the core under the force of gravity. The control rod drive assemblies consist of a normal speed drive motor, a fast "cutback" speed drive motor, an electromagnetic friction clutch, a speed reducing gear train, a selsyn transmitter, rack and pinion gears, appropriate shafts and the electromagnet carriage. The actuating drive switch is spring-loaded in the "out" position to insure that positive action is necessary to add positive reactivity. In addition, power is not available to the "out" position until the required interlocks have been satisfied. There is always power available to drive the rods into the core. The normal speed motors drive the carriages up or down through the gear reducing train and clutch at speeds of 5.6 inches/minute for shim rods and 3.7 inches/minute for the safety rod. These speeds along with the interlock system prevents exceeding the maximum allowable positive reactivity addition rates. The fast speed "cutback" motor is used only for control rod insertions at 39.2 inches/minute (shim rods) and 26.3 inches/minute (safety rod). The selsyn transmitter mounted on the main drive shaft is electrically connected to a

receiver on the console to indicate control rod position within ± 0.02 inch. Relays installed in the system must be energized in order to add positive reactivity and failure of such a relay will prevent the positive addition of reactivity. A manual means of deenergizing the electromagnets is provided in the reactor room as well as on the reactor console. Figure 4.1.8 is a functional diagram of the control rod drive system.

4.1.2.2 Moderator-Shield System

The normal water level in the reactor, approximately 5 feet above the core, is initially achieved and held constant in all routine reactor operations. Only in approved special operations will the water level be below or above this level. The moderator dump system (auxiliary reactor trip system) consists of an electrically actuated, air-operated, 10 inch valve and piping from the reactor tank to the dump tank. The 10 inch valve is designed to cause negative reactivity to be inserted by loss of moderator within one minute of actuation. The "water dump" 10 inch line and valve are connected to the reactor tank approximately 28 inches below the core. The 10 inch valve will open upon loss of either air pressure to the valve operator or electric power to the solenoid, and the valve will remain open until both have been restored and the "reset" switch has been depressed. "Water Dump" can be initiated either at the console or in the reactor room. A 1 inch manual valve and pipe

are connected at the bottom of the reactor tank for the purpose of draining the lower part of the tank.

The water fill system includes one pump, two electrically actuated, air operated valves, appropriate manual valves and piping. The fill rate is controlled by two paralleled valves in the fill main. Slow fill is initiated by opening one of the valves which has a flow reducing orifice installed in line. Fast fill is initiated by opening both valves which increases the flow rate. Proper interlocks must be satisfied before the applicable fill switch on the reactor console will activate one or both valves. When the normal water level is reached, the float switch in the reactor tank automatically closes the valves.

The deionized water is maintained with a minimum resistivity of 200,000 ohm-cm and pH of 4.5 to 8.0. The water is maintained to these limits by the water purification system which is described in section 4.1.3.1.

4.1.2.3 Nuclear Instrumentation System

The nuclear instrumentation which is described is based on a minimum requirement consistent with safe operation. Additional instrumentation is normally available for operational purposes. Figure 4.1.9 is a block diagram of the minimum instrumentation required for reactor operation. When $k_{eff} < .99$, the minimum

operational nuclear instrumentation is two source-level-sensitive, neutron detectors and indicators including an effective reactor trip capability on one. When $k_{eff} > .99$, minimum operational nuclear instrumentation is two startup and operating level-sensitive, neutron detectors driving amplifier systems with the capability of flux level monitoring and including one rate of change of flux and two flux level reactor trips; and one gamma sensitive detector and indicator with a high gamma level reactor trip.

The nuclear instrumentation monitors the neutron flux level for a range of nine decades, from the source range through 12 kw. As part of the interlock system, one of the source range detector channels has a count rate cutout feature installed on it to limit reactivity insertion rates when the reactor is in certain conditions. Also, an audible beeper is installed on the source range instrumentation so the source level can be monitored by personnel in the reactor room. The neutron sensitive detectors of the nuclear instrumentation are located in water tight containers in the peripheral region of the core. The gamma sensitive detector is located above the upper water shield over the core.

4.1.2.4 Reactor Trip Systems

The two modes of reactor trip are: (1) Reactor trip which involves the loss of electrical current to the control rod magnets causing the control rods to fall to their down position in the reactor.

(2) Auxiliary reactor trip which involves the opening of the moderator-shield water dump valve and the loss of the reactor tank water into the dump tank. An automatic reactor trip can be initiated by any of the three level limits or the rate limit. Table 4.1.2 includes these instrumented systems, their range and set points. Additional "Manual" reactor trip controls are provided at the reactor console and in the reactor room. Also provided is a remote A.C. power breaker switch which when de-activated causes the electrical power to be lost to the reactor console thereby causing a reactor trip. The auxiliary reactor trip can be initiated manually with controls located in the reactor room and at the console. The auxiliary trip (water dump) also is initiated when electrical power is removed from the reactor console. All reactor trips and the auxiliary reactor trip are annunciated on the trip indicator panel. After the condition causing the trip has been corrected, the trip annunciator will require a manual reset.

CHANNEL TYPE	DETECTOR TYPE	RANGE	AUTOMATIC CONTROL	SET POINTS MIN.	MAX.
LOG-N	ION CHAMBER	SOURCE LEVEL TO > FULL POWER	PERIOD TRIP	3 SEC	
			LEVEL TRIP		12 KW
LINEAR-N	ION CHAMBER	SOURCE LEVEL TO > FULL POWER (MANUAL DECADING REQUIRED)	LEVEL TRIP		98% OF FULL SCALE
LINEAR-γ	SCINTILLATION OR ION CHAMBER	100 WATTS TO > FULL POWER	LEVEL TRIP		12 KW

TABLE 4.1.2: NUCLEAR INSTRUMENTATION SYSTEMS

4.1.2.5 Interlock System

The reactor interlock system is provided to insure that proper and safe reactor conditions exist and that the correct sequence of operations are performed. Figure 4.1.10 is a functional diagram of the interlock system.

4.1.2.5.1 Integrated Interlock Control

The interlock system is based on the principle of channeling power through its circuitry in a prescribed sequence. This interlock sequence is arranged to permit positive reactivity to be added by only one means at a time. The interlock system is not in itself sufficient to institute a reactivity change. The operator will be required to manipulate and maintain pressure on a specific switch in order to maintain a positive reactivity addition.

The interlocks and their corresponding "satisfied" conditions include:

1. CONSOLE MASTER KEY, "ON".
2. REACTOR ROOM DOOR, "CLOSED".
3. NEUTRON FLUX "UP". The linear amplifier neutron channel recorder is "ON" and indicating greater than 5 percent of full scale.
4. SAFETY ROD, "COCKED". The safety rod drive assembly is out approximately 17 inches.

5. WATER LEVEL "UP". The moderator-shield water level in the core tank has actuated the high level float switch which is approximately 5 feet above the top of the core.

6. REACTOR ROOM ACCESS KEY, "ON". The reactor room access key is in the console lock and turned on.

7. COUNT RATE CUTOUT, "HIGH" or "LOW". One of two variable limits on a source range neutron instrument channel has been reached.

8. MODE SELECTOR SWITCH. The MSS must be positioned in the proper position corresponding to the operation being performed.

The primary interlock system components and their functions include:

1. The CONSOLE MASTER KEY - With the key in the console lock and in the "ON" position electrical power is supplied to the control rod drive, the control rod drive magnet, and the moderator fill circuits.

2. The REACTOR ROOM DOOR - A microswitch actuated by the reactor room door removes power from all shim rod magnets and the safety rod magnet when it is not in the cocked position if the door is not closed. The door can only be locked from the outside with the REACTOR ROOM ACCESS KEY

which is required to actuate the console lock that permits control rod withdrawal.

3. The NEUTRON FLUX-UP - The linear neutron channel recorder which is used for the complete reactor range from source level to full power must be turned on and indicating greater than 5 percent of full scale in order to satisfy the interlock. If this interlock is not satisfied, no reactivity insertion modes can be used. This interlock insures that there is a sensible neutron population in the core and that it is, in fact, being monitored.

4. The SAFETY ROD COCKED - To satisfy the safety rod cocked interlock, the safety rod drive assembly must be withdrawn approximately 17 inches. Unless all the shim rods are completely in the core and their limit switches are energized, the safety rod cannot be cocked. The SAFETY ROD COCKED interlock will remain satisfied until the safety rod is returned to the bottom of the core. When the interlock is not satisfied, power is denied to the moderator fill switch and normal rod drive switches.

5. The WATER LEVEL - A float switch is installed in the reactor tank 5 feet above the core. When the float switch is not actuated, drive power is denied all control rods except to cock the safety rod.

6. The COUNT RATE CUTOUT - One of the source range neutron instrument channels has two variable cutouts installed on it, high and low level. When the count rate cutout is actuated, it denies power to the control modes which may give reactivity ramps greater than positive $0.035/second. It prevents "all shims" mode of control rod withdrawal and the "fast" moderator fill. The low level cutout is set to correspond to a minimum detector level of 2 cps to insure that high positive reactivity insertion rates cannot be achieved unless the channel is responding at an observable level. The high level cutout is normally set at 100 times the source level, or at the known level which approximately corresponds to a source multiplication of 100 or k_{eff} = 0.99.

7. The MODE SELECTOR SWITCH - The multi-position MSS must be placed in the required mode of operation. Because only one switch position may be selected at any given time, reactivity addition by more than one mode is prevented.

Console master key "On" reactor room entry is permitted when certain conditions are met (see Figure 4.1.10). The mode selector switch must be in the "DOOR RELEASE" position. Thus no reactivity could be added to the reactor. Magnet power would be denied to the shim rods and drive power would be denied to all control rods. The conditions which must be met for a console master key "On" room entry are:

1. The REACTOR ROOM DOOR interlock "Bypassed"
2. FLUX UP interlock "Satisfied" or "Bypassed"
3. SAFETY ROD at it's "Cocked" position
4. WATER LEVEL interlock "Satisfied" or "Bypassed"
5. DOOR RELEASE must be selected on the Mode Selector Switch
6. COUNT RATE CUTOUT interlock must not be "Satisfied"

When the above six conditions are met, magnet power to shim rods is automatically removed and an electric solenoid which locks the reactor room door is de-energized, making access into the room possible. The door key may then be removed from the console lock and the room may be unlocked for entry.

4.1.2.5.2 Interlock Bypass

Certain bypasses are required for preoperational checkout, and safety. Each bypass is displayed on the control console with appropriate annunciation. No bypass permits remote addition of reactivity when personnel are in the assembly area. Details of the bypass system are as follows:

a. DOOR Interlock Bypass
   1) Permits reactor room entry with safety rod cocked for allowed room activities.
   2) Permits operation of the reactor control rod system for maintenance purposes.

b. FLUX-UP Bypass

Permits operation of the reactor control rod system for maintenance purposes.

c. SAFETY ROD Cocked Bypass

Permits operation of the reactor control rod system for maintenance purposes.

d. WATER LEVEL UP Bypass

1) Permits operation of the reactor control system for maintenance purposes.

2) Permits drive system operation in partial water height experiments.

4.1.3 Reactor Auxiliary Systems

4.1.3.1 Water Purification System

The purpose of the water purification system is to provide demineralized water for the reactor system and to maintain the water specific resistivity above 200,000 ohm-cm and the water pH between 4.5 and 8.0. Demineralized water is obtained by passing Zion City water through the demineralizer and into the dump tank. The water in the dump tank can also be recirculated through the demineralizer to maintain the water above minimum specifications. The system utilizes either one or two ion exchange columns as is necessary. A conductivity probe measures the conductivity of the demineralized water and indicates whether or not the water resistivity at the

demineralizer is at least 500,000 ohm-cm. All valves and controls in the system are manually operated and the overall system is completely separate from the reactor moderator-shield water control system. Figure 4.1.11 is a drawing of the water purification system.

4.1.3.2 Waste Disposal System

The operation of the NTR involves the disposal of contaminated wastes only on a very intermittent basis. The disposal of reactor water will comply with the requirements established in Title 10 Code of Federal Regulations, Part 20 and other Federal, state and local regulations. The disposal of facility materials and equipment will be in accordance with Reg. Guide 1.86. All materials found to exceed the above requirements and regulations will be packaged properly and shipped off site to an authorized receiver of radioactive materials for storage, reprocessing or disposal.

4.1.3.3 Fuel Storage

The main fuel storage area is located against the west wall of the reactor room. Facilities are provided to store at least 31 fuel elements. The geometrical configuration of the array of fuel element storage locations is a slab. High density concrete surrounds the storage area to afford the proper radiation shielding. A gamma sensitive criticality monitor is provided near the main fuel storage area which causes the facility criticality

alarm to sound if a high level of gamma radiation is detected. This monitor also serves as an area monitor.

Up to 6 fuel elements may also be stored in the reactor tank in individual storage tubes mounted on the wall of the tank. The top of the storage tubes are located 9 feet below the top of the tank.

A rack is provided in the reactor room, shielded by high density concrete, on which the control rods may be hung when they are removed from the core. There are five positions on the rack with a minimum center-to-center spacing of 5-3/4 inches.

4.1.3.4 Radiation Safety Equipment

A minimum of three stationary radiation area monitors are present in the facility. Two of the monitors, which can also be used for personnel contamination monitoring, are located near the main entrance to the facility and in the control room. The third area monitor is located inside the reactor room and causes an alarm to be actuated in the control room if the gamma radiation level in the room reaches a predetermined value. The third monitor also serves as the criticality monitor required for the storage of special nuclear material.

The facility emergency alarm is a continuously sounding horn which can be heard throughout the Training Center. The purpose of this

alarm is to alert personnel of a possible danger from the reactor system. It is actuated manually at the north control room door and at the main entrance door of the NTR Facility.

Portable radiation monitoring instruments capable of measuring all expected radiation types and levels are located in the facility. The minimum portable instruments which must be present in the facility are:

Detector	Range
β-γ Detector	0-20 mR/hr
β-γ Detector	0-5 R/hr
γ Detector	$\geq$ 100 R/hr
n Detector	0-5 rem/hr
α Detector	$\geq$ 100,000 CPM

A small radiation safety laboratory exists in the facility in which routine analysis of air, water, and smear surveys are conducted. Laboratory radiation detector systems are available for measuring α, β, and γ radiation. In addition, analysis of the reactor water for resistivity and pH are conducted in the laboratory.

Each member of the operating staff is issued a neutron and β-γ sensitive thermoluminescent dosimeter (TLD) and a direct readout ion chamber dosimeter which must be worn whenever they are in the facility. The TLD's are worn for no longer than one month and then sent to a commercial processing agency for reading. Visitors in the

facility are either issued TLD's or pocket dosimeters, or they are provided with an escort having personnel monitoring devices.

An emergency cabinet is maintained which contains a minimum of two (2) complete sets of anti-contamination clothing, two (2) full face filter-type respirators, and one (1) self-contained breathing apparatus.

Proper storage containers are kept on hand to store suspect radioactive waste materials until they are shipped off for disposal.

4.2 FACILITY AND SUPPORT SYSTEMS

4.2.1 NTR Facility and Controlled Area

The NTR Facility is located in the south wing of the Nuclear Training Center building. The main construction of this building consists of poured concrete, concrete block and steel supports. Figure 4.2.1 describes the Facility layout and also defines the controlled area of the Facility. The Facility building consists of the reactor room, console room, equipment room and other support areas. The Facility building enclosure is approximately 70 feet x 45 feet.

The controlled area which serves as a restricted area, is approximately 90 feet x 100 feet. This area is enclosed either by a

10 foot cyclone fence with barbed wire on the top or by building walls. The minimum distance from the controlled area boundary to the reactor centerline is 44 feet. Entrance to the controlled area can be normally gained through the main entrance door in the south corridor of the Training Center or through either of two normally locked, 20 foot, fence gates. The main entrance door has a standard key lock on the outside and a manual and electrical solenoid lock on the inside.

The 27 foot square, 27 foot high reactor room is constructed of 8 inch concrete block walls and has a 5 inch thick concrete floor. The reactor dump tank and concrete pit are centered in the room. Two access doors lead into the reactor room. The 3 x 9-5/6 foot exterior door, which is set in 10 x 7-1/6 foot opening can only be unlocked from the inside and may be used as an emergency exit. The panels supporting this door and closing the remainder of the 10 x 7-1/6 foot opening can be removed from the inside to serve as a large equipment entrance. The interior door, which has a 3 x 9-5/6 foot opening, provides the main access into and out of the reactor room. This door is controlled from the console room during reactor operations and is normally locked. The reactor room represents a limited access area within the facility and cannot be entered unless the entry is made under the direction of an Authorized Individual.

4.2.2 Utilities

4.2.2.1 Water and Sewer Service

The water and sanitary sewer service is supplied by the Lake County Public Water District through the City of Zion system.

4.2.2.2 Electrical Service

Electric service is provided by Commonwealth Edison. A substation with a 1000 kVA transformer is located on the Nuclear Training Center, Zion Site. A main switchboard distributes power at 440 volts, 3 phase, 60 hertz to the site. Power to the reactor system is stepped down through a transformer which is independent of the rest of the site. A schematic diagram of the system is given in Figure 4.2.2.

4.2.3 Ventilation

The NTR Facility is heated by an all electric system which also heats the complete Nuclear Training Center. The heating control center for the south wing and the reactor facility is located in the upper level of the equipment room. There is an auxiliary 9 kW heater fan in the reactor room which can be used as necessary.

A separate air-conditioning unit, which is independent of the air conditioning unit for the Training Center, exists for the Facility and is also located in the upper level of the equipment room. The air in the reactor room is not recirculated through the air conditioning system, but rather is exhausted through a fan in the roof of the room. This fan and the air supply to the room can be secured from the equipment room. Both the heating control center and the air conditioning unit can be secured either in the facility or at the main switchboard in the Training Center Mechanical Equipment Room.

4.2.4 Compressed Air

Compressed air is supplied from an air compressor system installed in the facility equipment room. Compressed air is required to hold the 10 inch moderator-shield water dump valve closed. The release of the air pressure from the valve piston causes the valve to open.

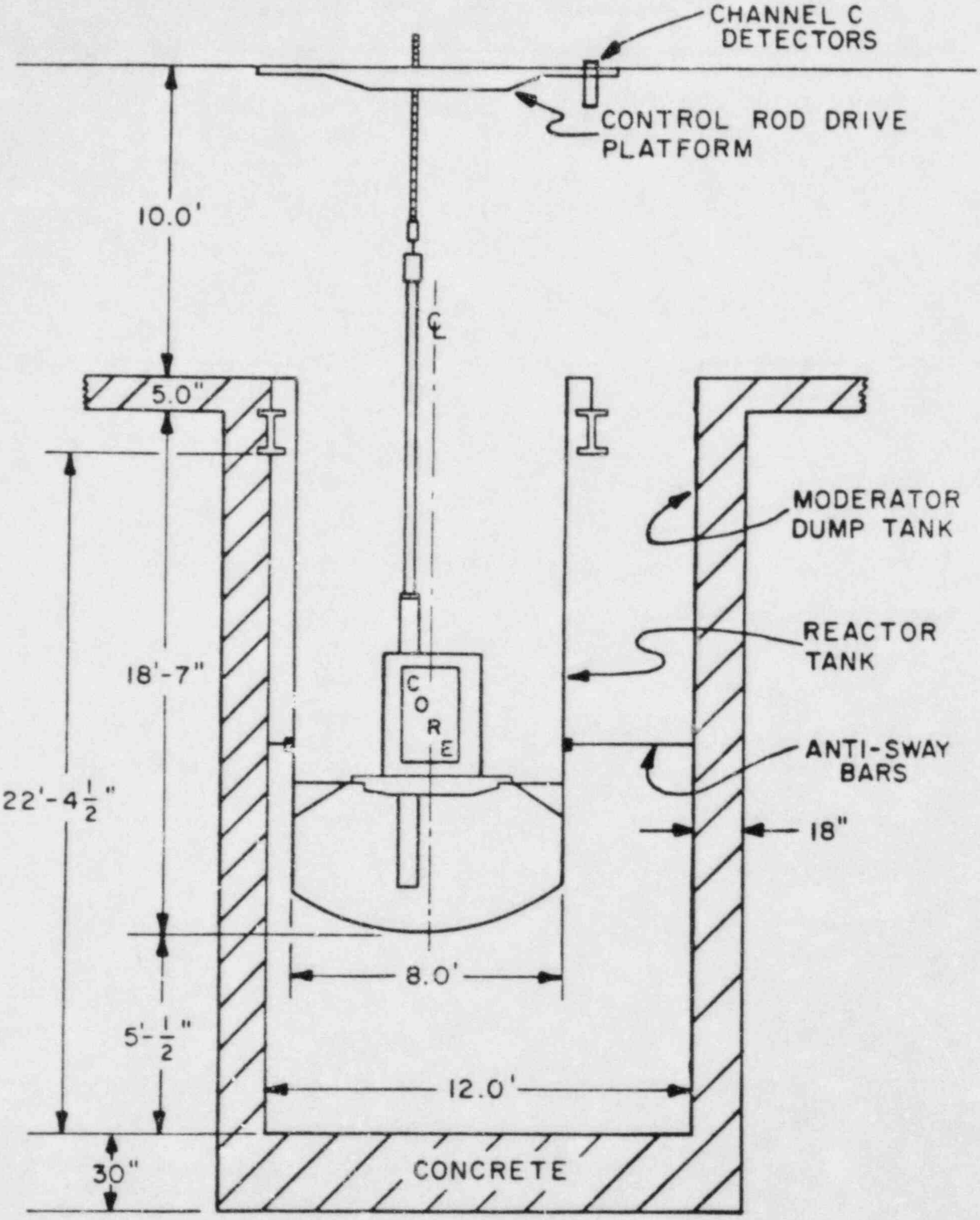


FIGURE 4.1.1: REACTOR ASSEMBLY

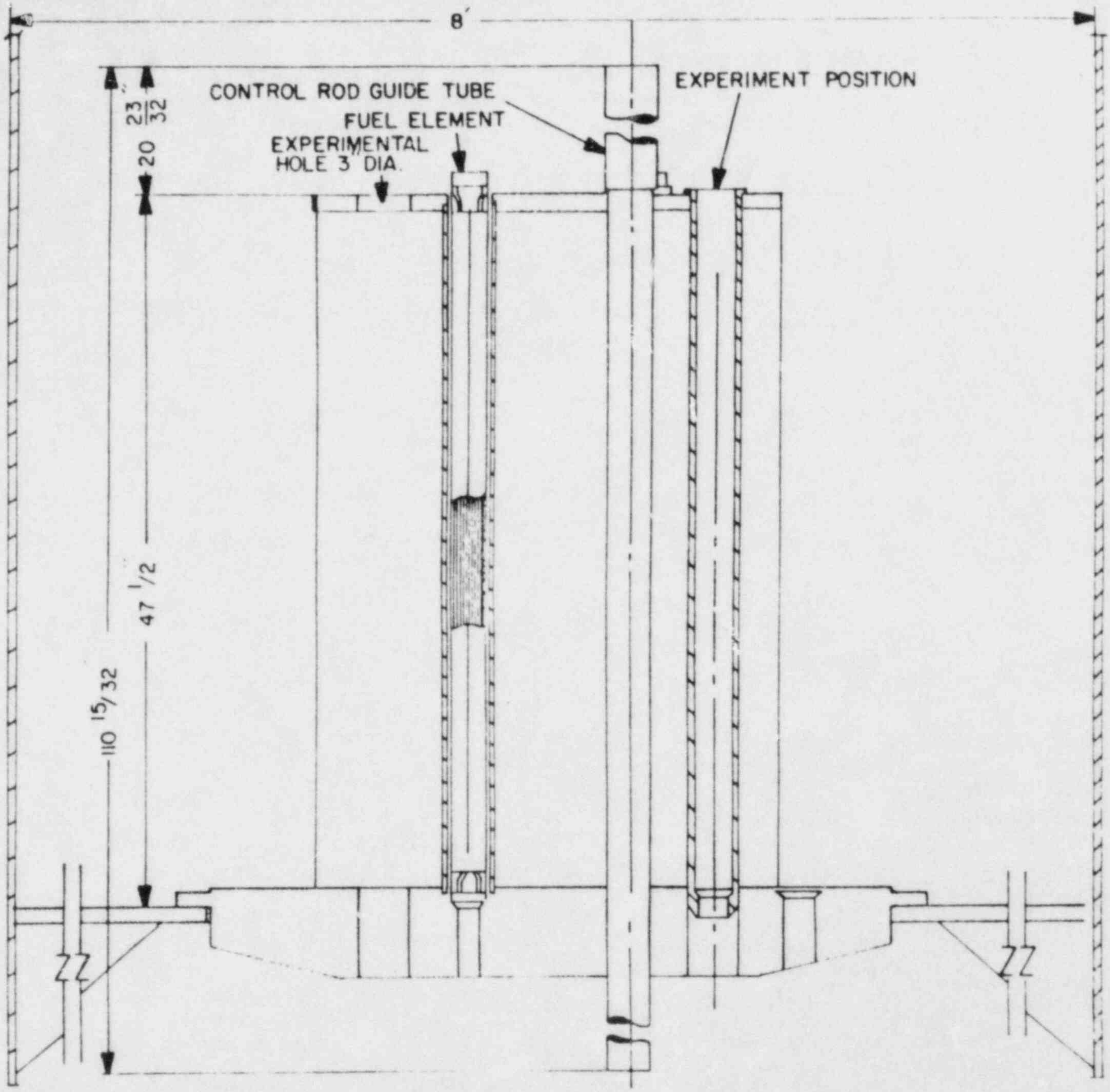


FIGURE 4.1.2: VERTICAL SECTION OF THE NTR CORE SUPPORT STRUCTURE SHOWING TYPICAL FUEL ELEMENT, CONTROL ROD GUIDE TUBE AND EXPERIMENTAL POSITION

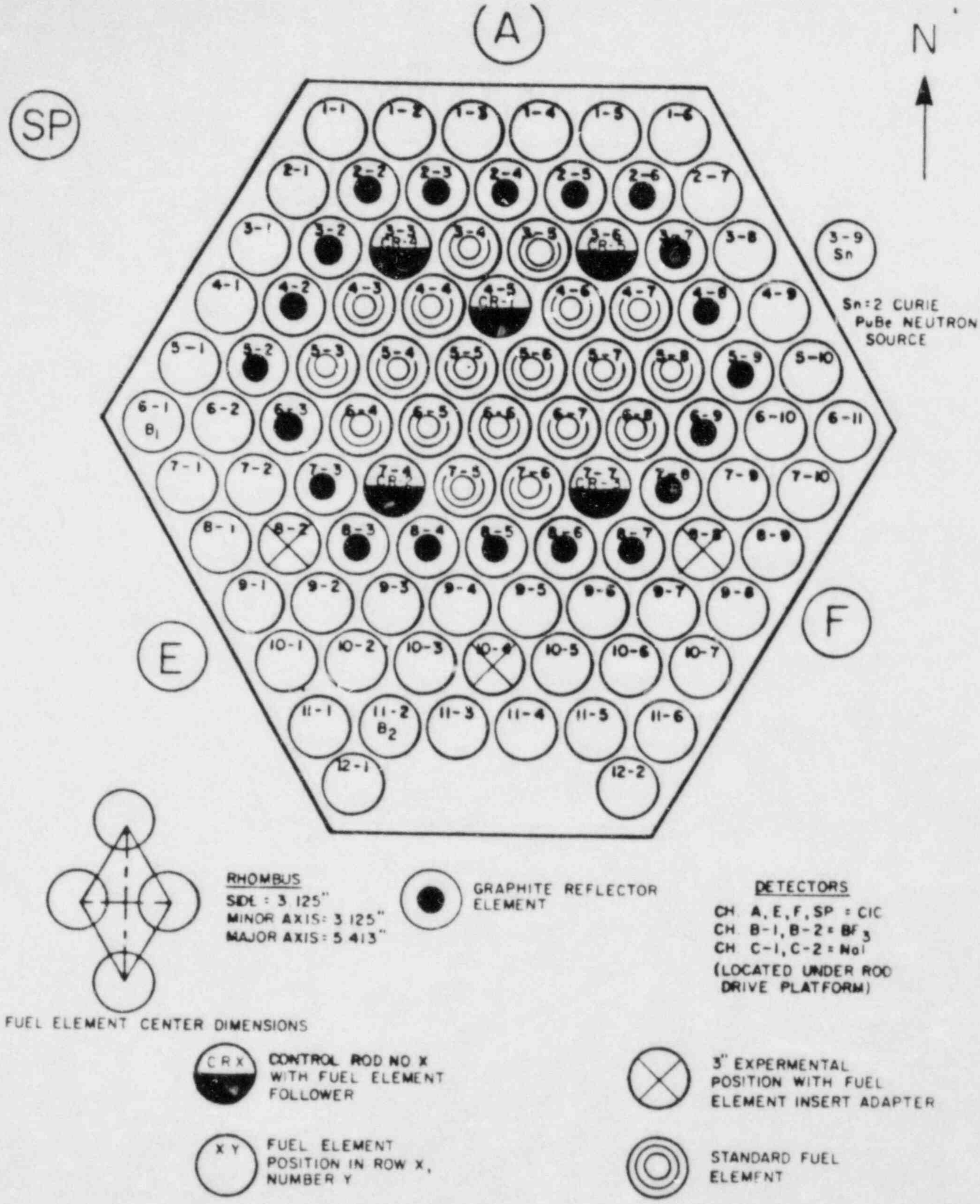


FIGURE 4.1.3: CORE STRUCTURE (TOP VIEW)

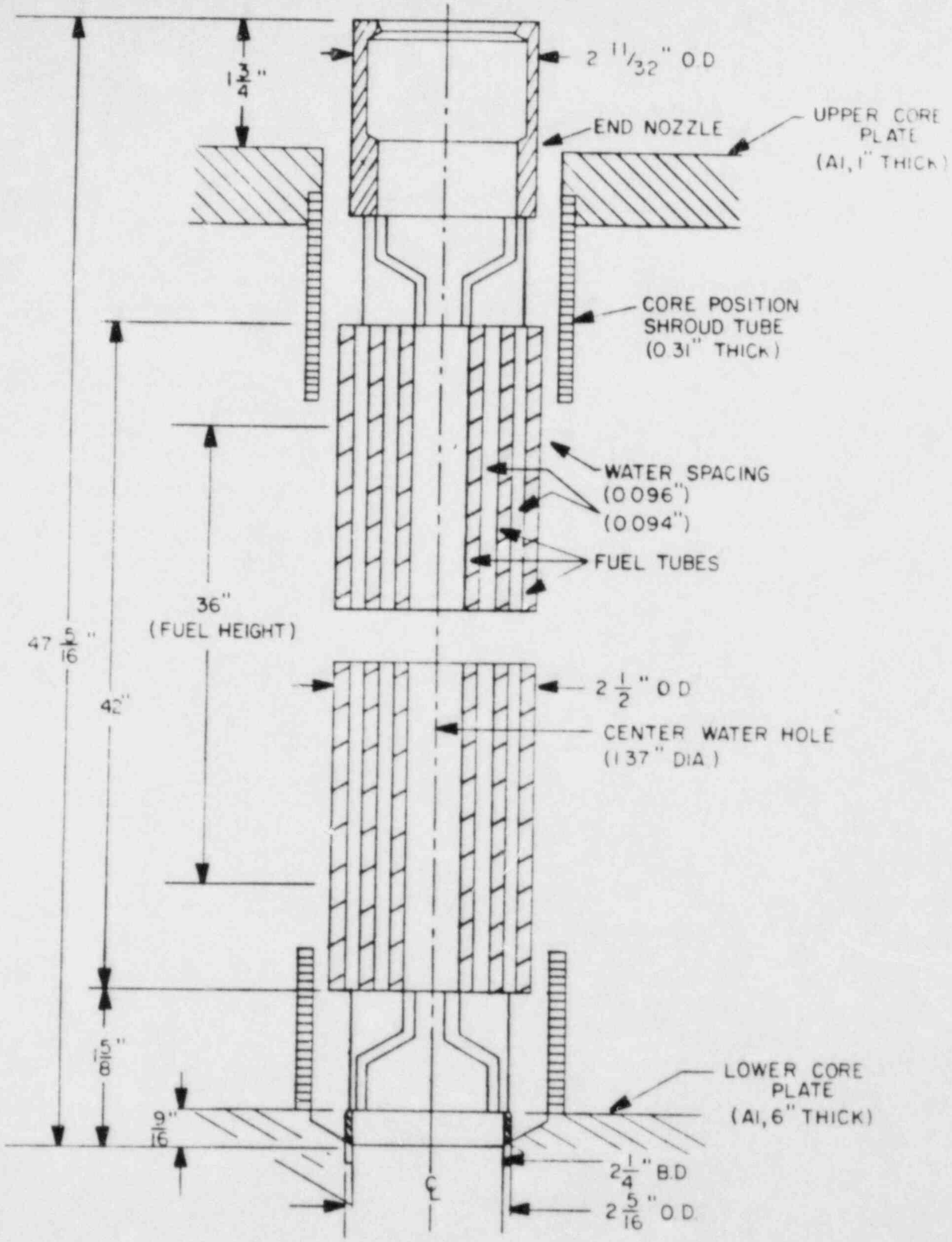


FIGURE 4.1.4: STANDARD FUEL ELEMENT

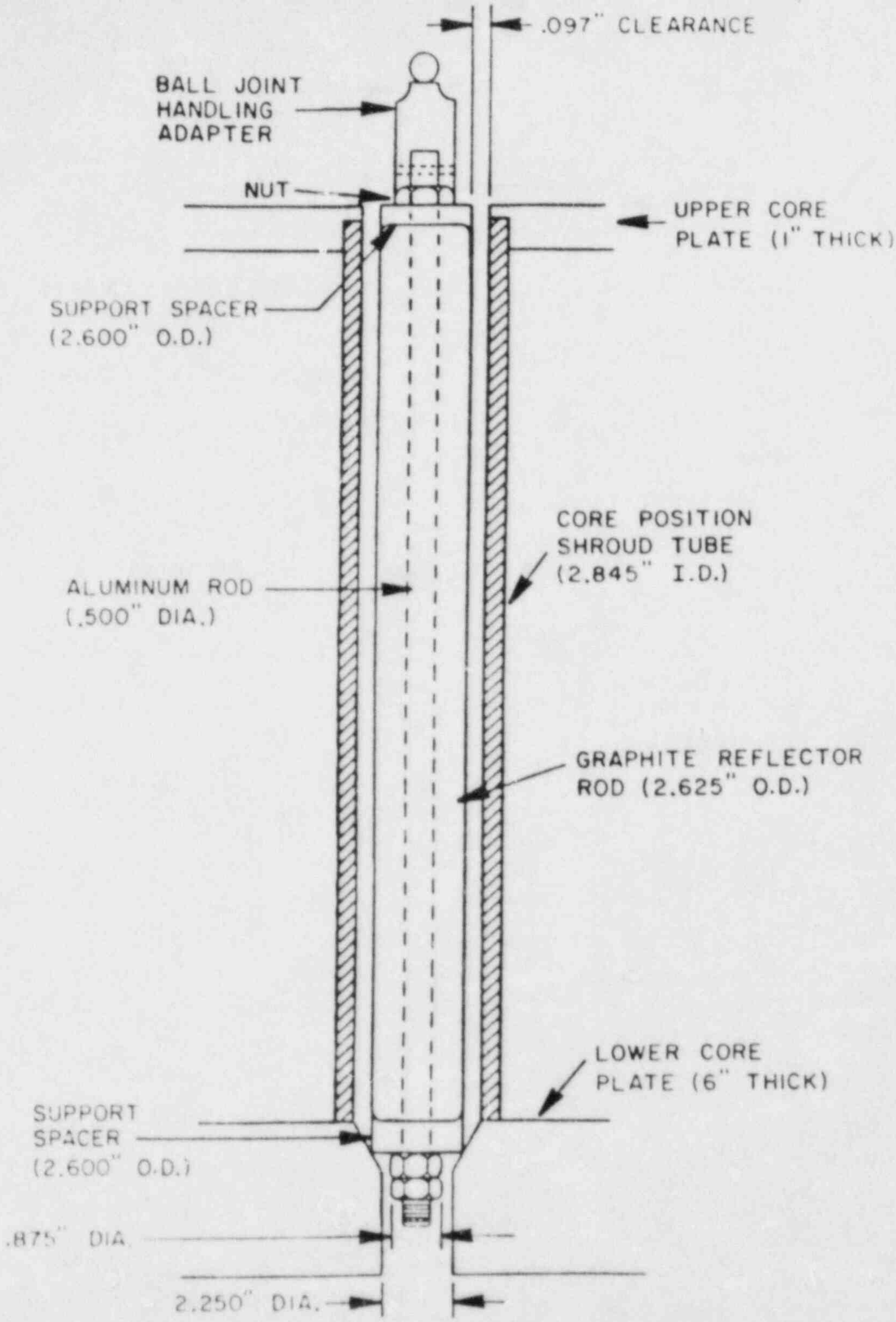


FIGURE 4.1.5: GRAPHITE REFLECTOR ROD

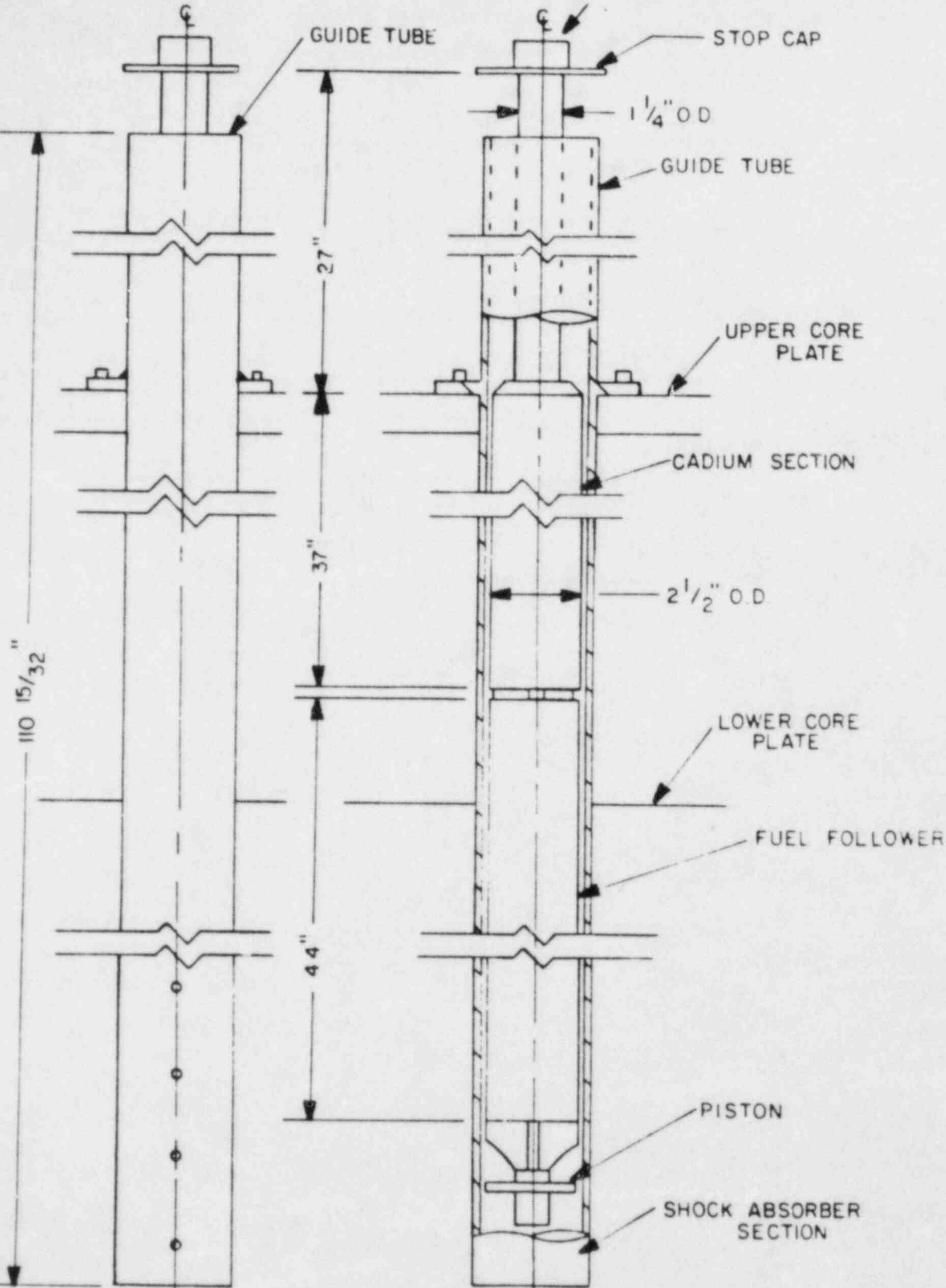


FIGURE 4.1.6: CONTROL ROD IN GUIDE TUBE (PARTIALLY REMOVED)

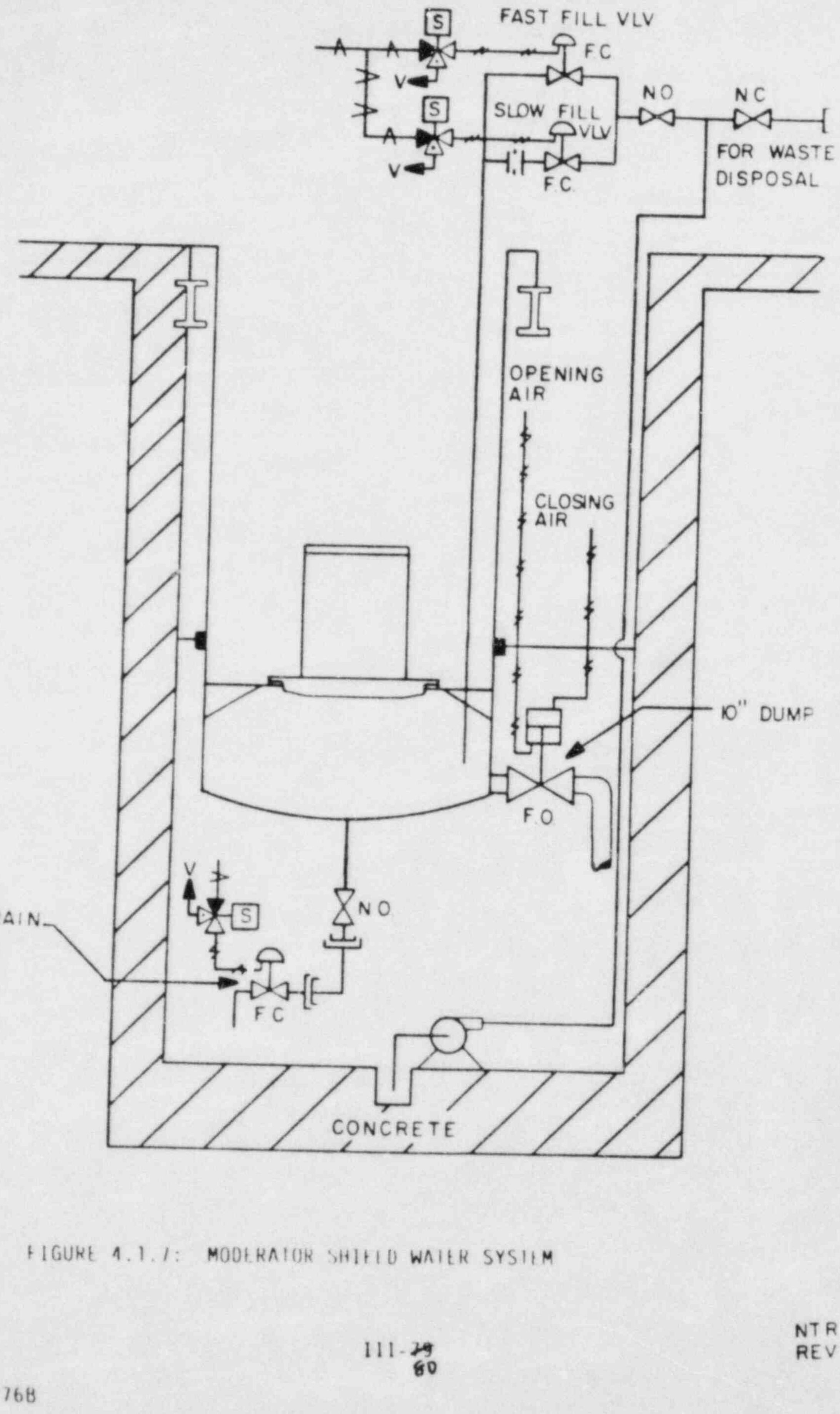


FIGURE 4.1.7: MODERATOR SHIELD WATER SYSTEM

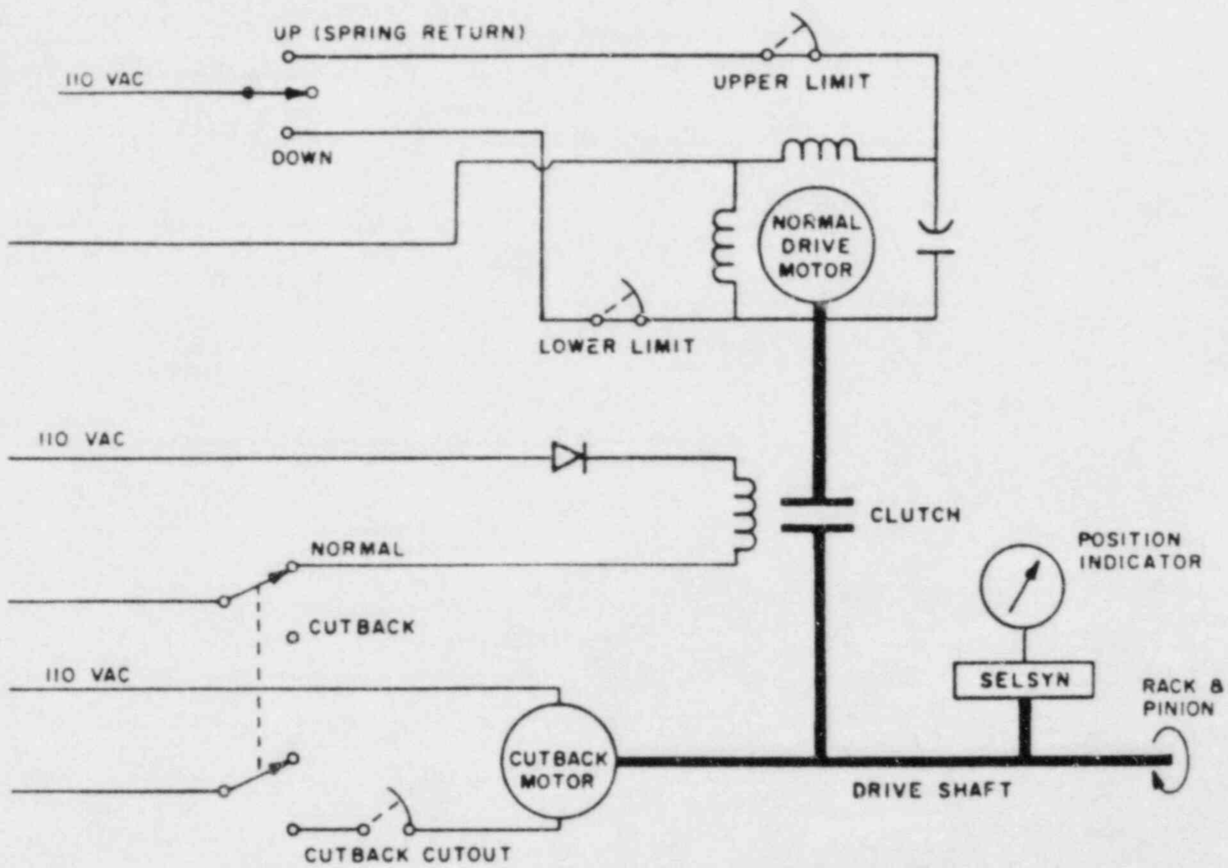


FIGURE 4.1.8: NTR ROD DRIVE SIMPLIFIED SCHEMATIC

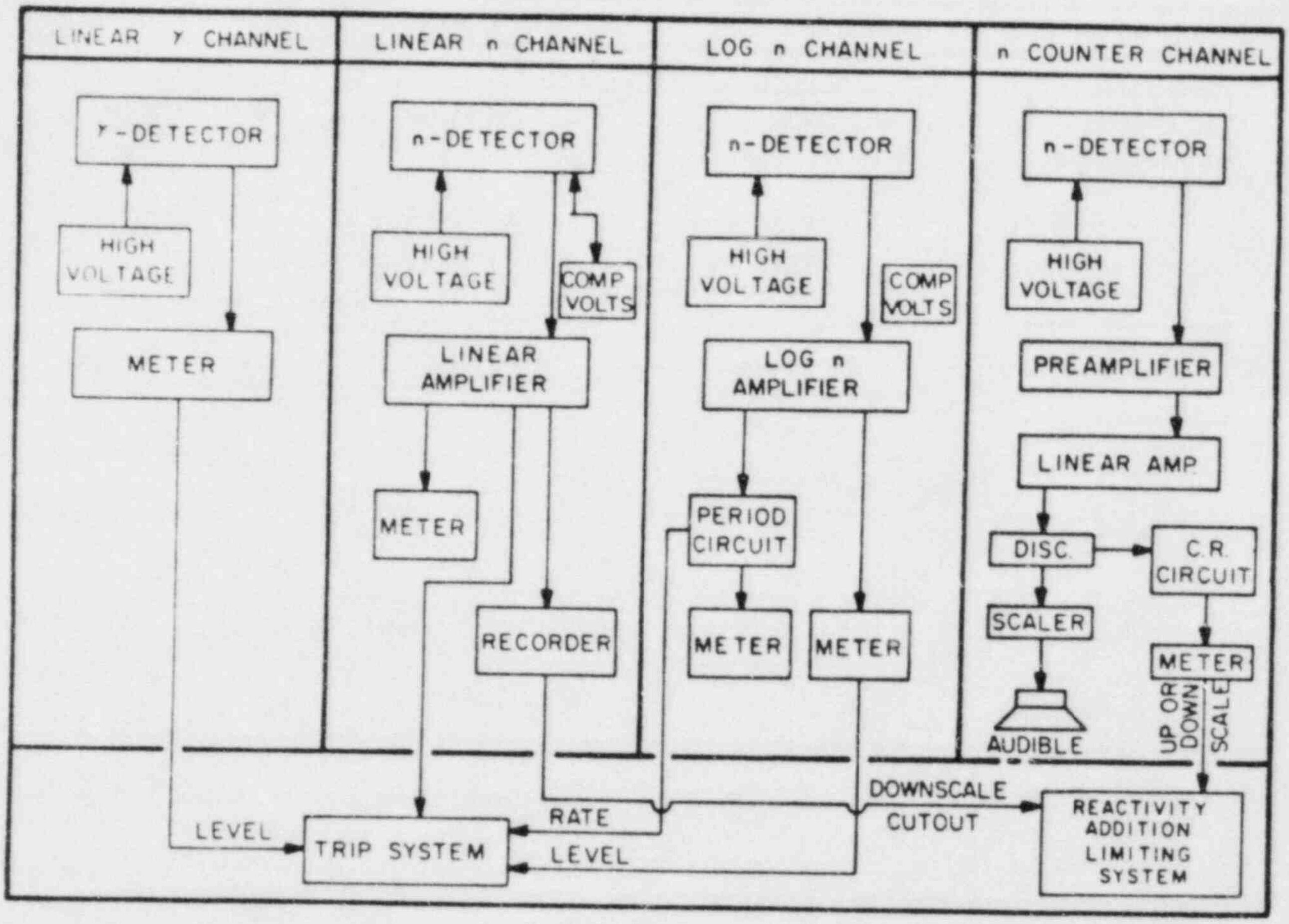
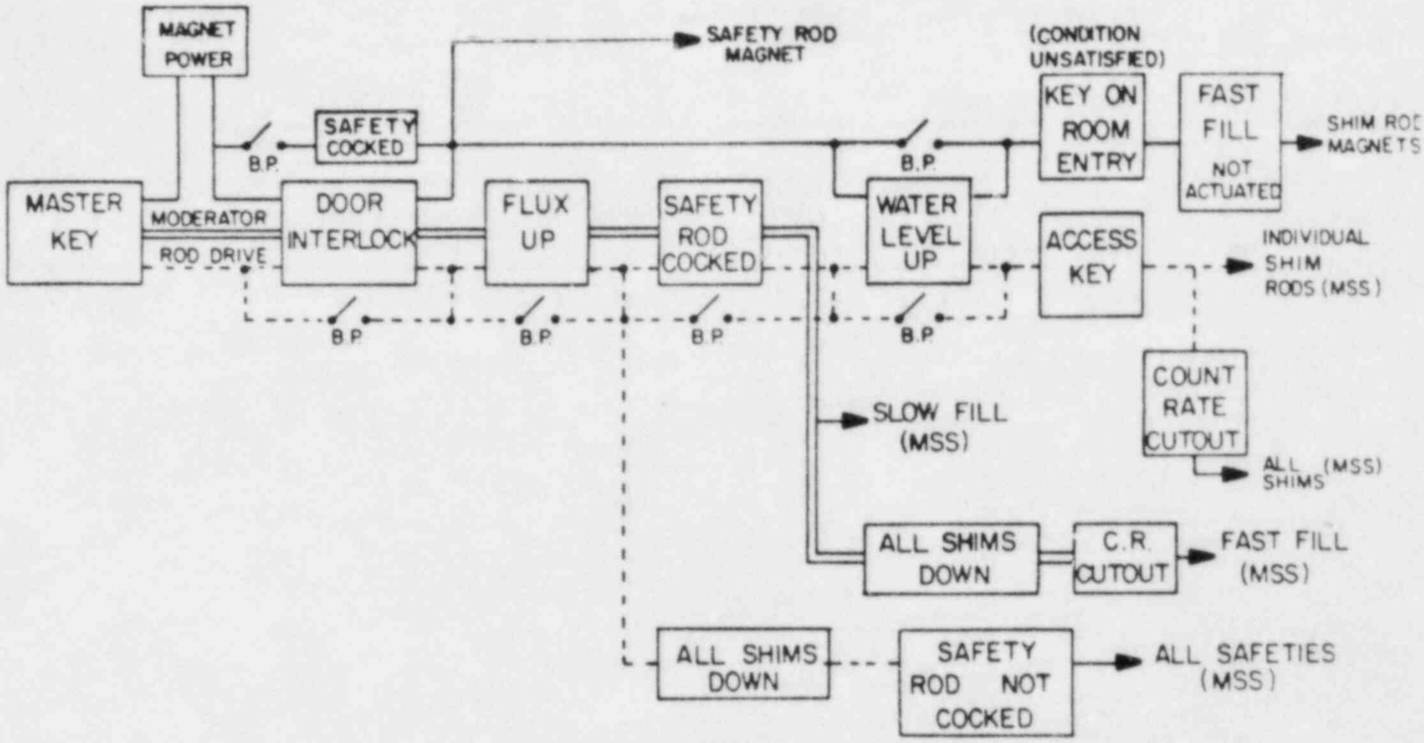


FIGURE 4.1.9: MINIMUM INSTRUMENTATION



KEY ON ROOM ENTRY INTERLOCKS

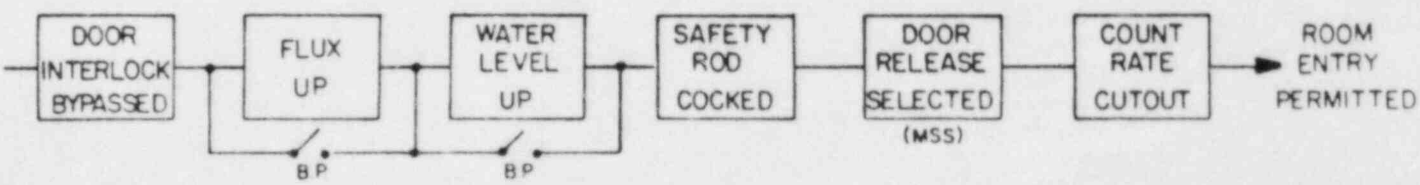


FIGURE 4.1.10: INTERLOCK SYSTEM

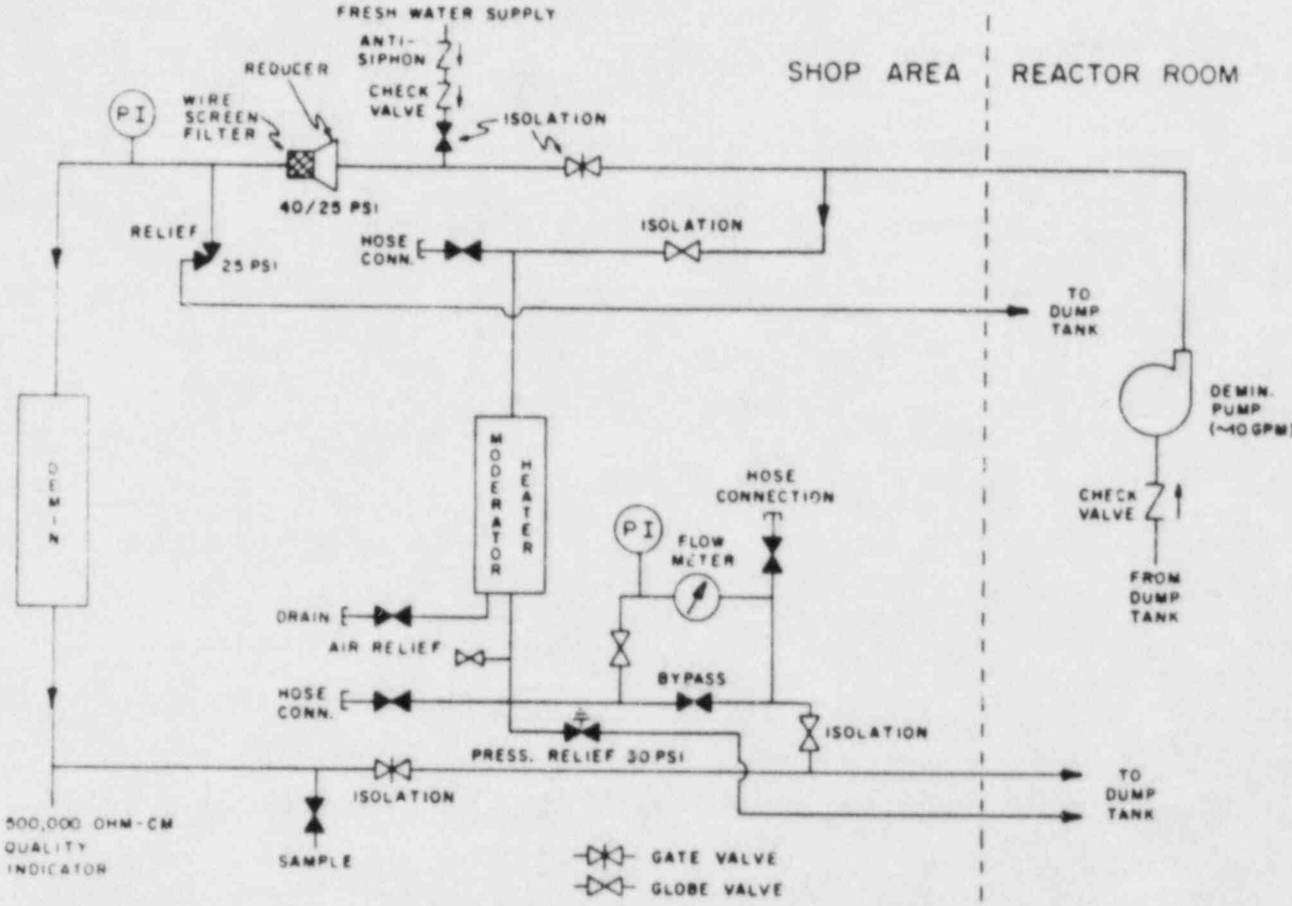


FIGURE 4.1.11: WATER PURIFICATION SYSTEM

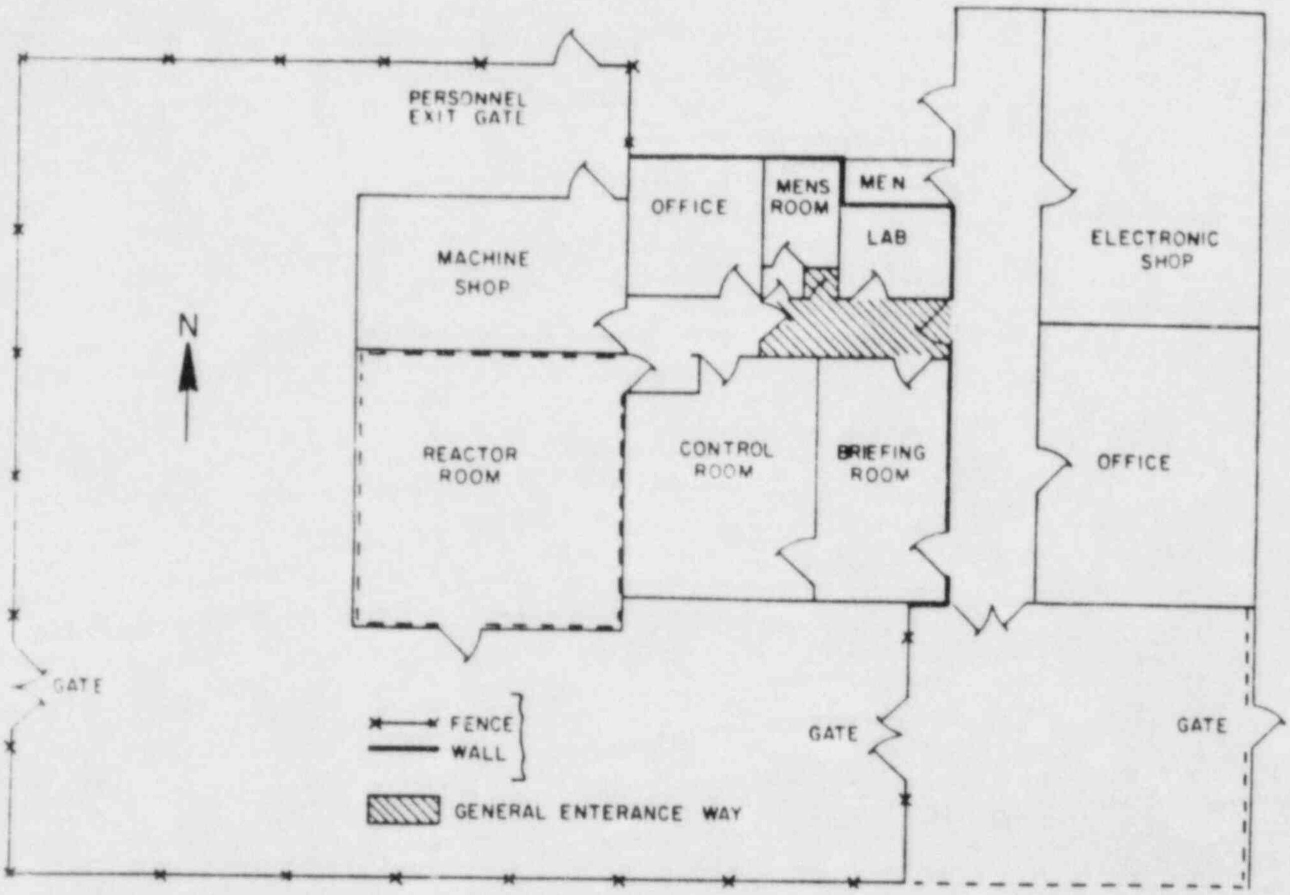


FIGURE 4.2.1: RESTRICTED AREA

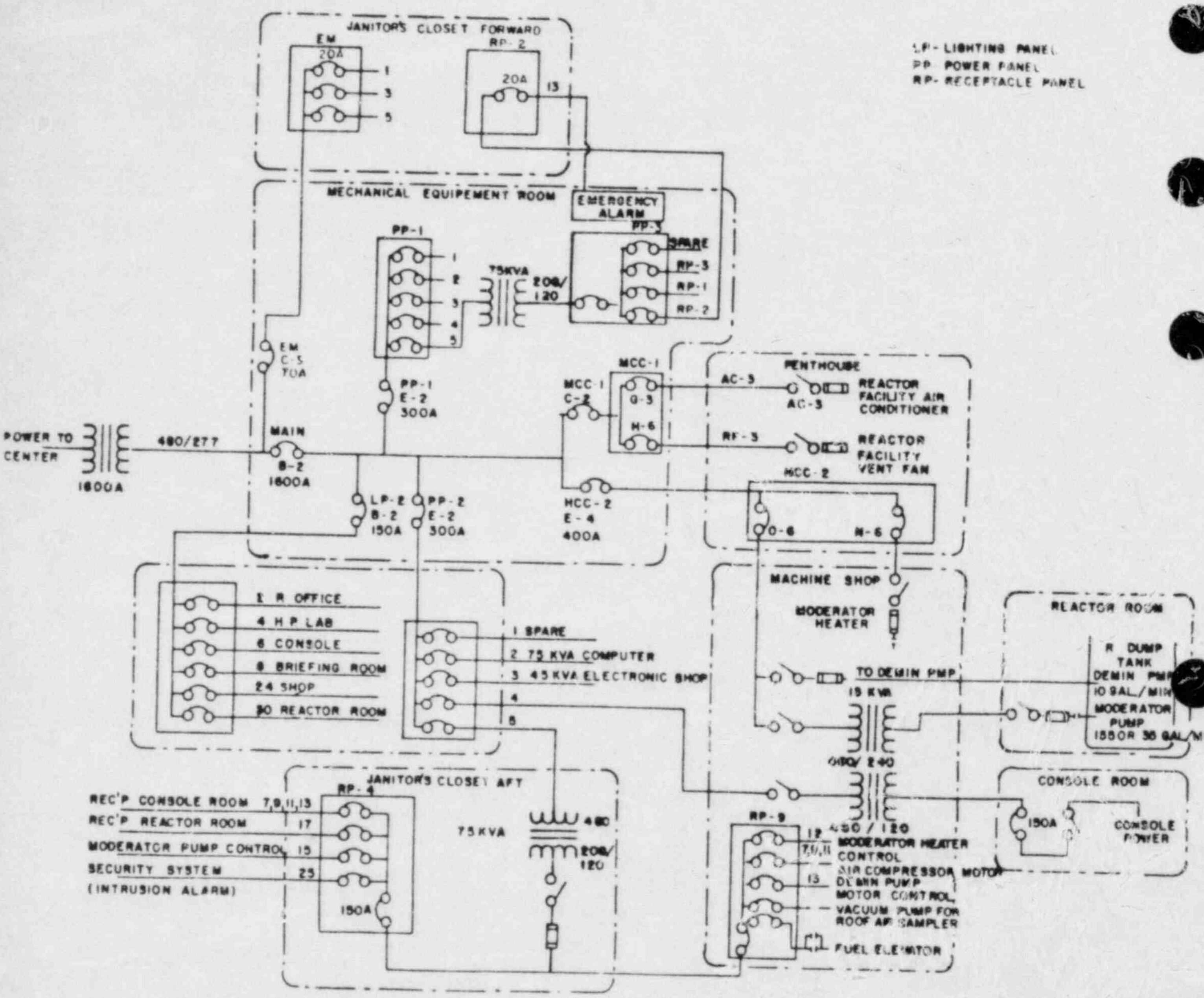


FIGURE 4.2.2: ELECTRICAL DISTRIBUTION

CHAPTER 5

ORGANIZATION AND ADMINISTRATION OF THE FACILITY

5.1 ADMINISTRATIVE ORGANIZATION

5.1.1 Corporate Organization

The administration and operation of the Westinghouse Nuclear Training Center is a function of Nuclear Training Services, Nuclear Services Integration Division, Water Reactor Division, Nuclear Energy Systems, Westinghouse Electric Corporation (Refer to Figure 5.1.1).

NUCLEAR ENERGY SYSTEMS

↓

WATER REACTOR DIVISIONS

↓

NUCLEAR SERVICES INTEGRATION DIVISION

↓

NUCLEAR TRAINING SERVICES

↓

W. N. T. C.
ZION, ILLINOIS

↓

NTR FACILITY

FIGURE 5.1.1: CORPORATE ORGANIZATION

5.1.2 Operating Staff Organization

The NTR facility operating staff organization is given Figure 5.1.2. All facility staff members are NRC licensed senior reactor operators who are required to maintain qualification in accordance with Appendix C. Under the direction of the Manager, the facility staff conducts scheduled training operations on the NTR as part of the Westinghouse Nuclear Training Programs and other operations including irradiations and maintenance. The participation of trainees in any reactor program is authorized as such by the facility Manager. However, trainees do not accrue any responsibility for operations and are always under the direct control of a licensed reactor operator.

5.1.2.1 NTR Facility Manager

The Manager is appointed by the higher management of Westinghouse Water Reactor Divisions. He is an individual who, by training and experience is capable of understanding the reactor, can exercise judgement as to the safety of its operation and administration, and can assume responsibility for changes and modifications in the reactor system. It is the responsibility of the Manager to assure the operation of the reactor facility within the limitations and constraints specified in the Facility License and Technical Specifications. The decisions of the Manager are subject to higher management review and approval, but this does not relieve the Manager from the final responsibility for the decisions. The Manager has an NRC Senior Operating License on the Facility.

NTR FACILITY
MANAGER

↓

TRAINING
REACTOR
COORDINATOR

↓

REACTOR
LEAD
ENGINEER

↓

SENIOR
REACTOR
OPERATORS

↓

REACTOR
OPERATORS

↓

COGNIZANT
PERSONS

FIGURE 5.1.2: OPERATING STAFF ORGANIZATION

5.1.2.2 Training Reactor Coordinator

The Training Reactor Coordinator is officially appointed by and reports directly to the Manager of the NTR. The Training Reactor Coordinator is responsible for the administration of the NTR Facility with respect to Operating Staff training, compliance with the Facility operating license and Technical Specifications, related documentation and records, and Facility reports. The Training Reactor Coordinator also offers technical support for the Facility, is required to be a licensed senior operator, and serves as the secretary of the Reactor Safeguards Committee.

5.1.2.3 Reactor Lead Engineer

The Reactor Lead Engineer is officially appointed by the Manager of the NTR. His responsibilities and duties are involved directly with the administration of the Facility and supervision of the operating staff. The Reactor Lead Engineer coordinates all the routine operational and maintenance activities on the reactor. He reports to the Manager through the Training Reactor Coordinator and must be intimately familiar with the reactor and its operations as well as the overall management of the Facility. The Reactor Lead Engineer is required to be a licensed senior operator.

5.1.2.4 Senior Reactor Operator

A Senior Reactor Operator is an individual who possesses a valid senior operator's license, issued by the Nuclear Regulatory Commission. His duties involve the normal operation and maintenance of the reactor, as well as the direction of Reactor Operators in the performance of their duties.

5.1.2.5 Reactor Operator

A Reactor Operator is an individual possessing a valid operator's license, issued by the Nuclear Regulatory Commission. His duties involve the normal operation and maintenance of the reactor.

5.1.2.6 Cognizant Person

In addition to the above staff personnel, there exists one other classification of personnel pertinent to the operation of the reactor. This individual is called a Cognizant Person, and by definition is a Westinghouse staff member deemed by the Manager to be capable of recognizing the need for taking emergency action. This person is likely to be a member of or to work closely with the operating staff and is well aware of the operating and emergency procedures of the Facility. His duties involving the reactor operation come into play only if the regular operating staff members become incapacitated under an emergency condition.

5.2 TECHNICAL SUPPORT

In addition to its own personnel in Nuclear Training Services the Westinghouse Nuclear Training Center can also seek expertise from the many varied disciplines of reactor technology encompassed throughout the Nuclear Power Systems faction of Westinghouse.

5.3 REACTOR SAFEGUARDS COMMITTEE

The NTR Reactor Safeguards Committee (RSC) consists of a group of persons of recognized capability in the nuclear and associated fields. The RSC basically functions as a review and advisory committee in matters pertaining to the safe operation of the reactor facility. The qualifications of the committee members must be consistent with the following:

(1) Each member must have a minimum of five (5) years industrial experience in nuclear and related fields and must have a minimum of three (3) years of active participation in his nuclear orientated discipline.

(2) The experience and knowledge of each member must be applicable to or pertain to the Committee's responsibility to properly review the facility and its operation from the standpoint of safety.

(3) Each member must be capable and willing to exercise his individual judgement in regard to all Committee reviews and decisions.

(4) The "nuclear orientated discipline" of a minimum of two Committee members must lie in the areas of reactor physics and reactor operations.

5.3.1 RSC Organization and General Practices

The reactor Safeguards Committee consists of five or more members, Westinghouse or non-Westinghouse employees, whose experience is in accord with the above qualifications. At least four members must not be in the line organization which is responsible for reactor operations (Nuclear Training Services Group, NTS). The NTR Facility Manager is automatically a member of the Committee and the NTR Training Reactor Coordinator serves as the RSC Secretary, a non-voting committee position. One of the non-line organization members is appointed Chairman by the NTS management and holds the position for not more than three years. The RSC meets at least once each six (6) months. A quorum of the Committee consists of at least four members and at least half of those present must be from non-line organizations.

5.3.2 RSC Responsibilities and Duties

The Reactor Safeguards Committee reviews NTR activities and advises the Manager and/or whatever echelon of Westinghouse management it feels appropriate on all matters pertaining to the safe operation of the facility. The matters which are to be specifically reviewed are listed below:

1. Proposed experiments, tests and operations not described in the Safety Analysis Report.
2. Proposed changes or modifications to the facility not described in the Safety Analysis Report.
3. Proposed changes to the Technical Specifications.
4. Proposed normal operating procedures and emergency procedures, and proposed changes thereto.
5. Facility operation for compliance with internal rules, procedures and regulations, and with license provisions.
6. Performance of facility apparatus and equipment.

Quarterly inspections of the facility and reactor operation are made by an individual committee member. A report of this inspection is made immediately if necessary, or at the next Committee meeting.

A semi-annual audit of the radiation safety practices and records is made by an experienced group of Committee members. A written report is made of this audit and distributed to the Committee Membership.

The official records of the RSC include the minutes of each meeting, special reports on proposed experiments reviewed including the Committee findings, and reports on facility inspection and radiation safety practice inspections. All Committee reports and meeting minutes are transmitted to the membership and through line management up to and including the Manager, NTS.

5.4 REACTOR OPERATIONS

5.4.1 General

The basic responsibility for all reactor operations including routine training, irradiation, maintenance and special operations rests with the facility Manager. All operations including fuel and radioactive material handling must be initially approved by the Manager. The execution of the approved operations is the responsibility of the operating staff. All reactor and related operations are documented in a facility operating manual and a manual of experiments and demonstrations. These documents are reviewed by the Reactor Safeguards Committee prior to their use. Changes to the operating manuals which do not affect the intent of the original procedures may be made by the facility Manager and then reported to the Safeguards Committee. Changes which can possibly change the intent of the original procedure must first be reviewed by the Reactor Safeguards Committee. This is also true for any new procedure, experiment or demonstration.

The reactor operations which are planned for the facility are covered in section 6.2 of this report.

5.4.2 Operating Procedures

All operating procedures must conform with the restrictions established in the Facility License and Technical Specifications. Approved written operating procedures are documented and followed for the following items in the Facility Operating Manual.

(1) Procedure for obtaining written approval for all operations from the facility Manager.

(2) Routine reactor operations.

(3) Radiation safety practices.

(4) Facility security operations and rules.

(5) Emergency procedures.

(6) Preventive and corrective maintenance procedures.

(7) Fuel handling, storage and changes.

(8) Radioactive material handling procedures.

(9) Radioactive waste handling procedures.

Approved written experimental and demonstrational procedures are documented and followed for the following items in the reactor's Manual of Experiments and Demonstrations.

(1) Routine training operational demonstrations.

(2) Routine training experiments.

(3) Routine irradiation experiments.

(4) Routine test experiments

5.5 RADIATION SAFETY PRACTICES

All radiation safety and protection practices of this Facility comply with the regulations set down in Title 10, Code of Federal Regulations, Part 20. The major responsibility for the proper control of radiation hazards at the NTR Facility rests with the Manager and the Operating Staff. In conjunction with the technical services supplied by Commonwealth Edison Health Physics Section at the Zion Site and off-site Westinghouse Health Physics groups the proper safety standards and procedures are to be established. These procedures are documented in the Facility Operating Manual and accurate records of the results of the practices employed are kept. The radiation safety and protection procedures which are followed include the following items:

(1) Personnel monitoring, dose limitations and exposure records.

(2) Environmental monitoring and records thereof.

(3) Periodic and routine radiation and contamination surveys and records thereof.

(4) Radioactive material handling, inventory, storage, receipt and shipment and records thereof.

(5) Radioactive waste storage and disposal and records thereof.

5.6 FACILITY SECURITY

Facility security is the responsibility of the Manager and Operating Staff. Security regulations and written procedures are implemented by the operating staff with the aid of the City of Zion Police Department. Certain persons, as designated by the Manager are permitted to open the facility at the start of each normal working day and after normal working hours. These individuals include the licensed operating staff members, and designated Training Center Management. Off-hour security inspections are made by the City of Zion Police patrol. Access to certain areas in the facility (e.g. reactor room) is limited to operating staff members or requires a staff member acting as escort.

Under emergency conditions access into the building and the areas other than the reactor room of the facility are provided by the City of Zion patrolman or Training Center Staff. Access into the reactor room can only be gained by an Authorized Individual. Any unscheduled facility entry must be reported to the NTR facility Manager.

The facility building (south wing of the Training Center) is fenced by a ten (10) foot woven wire fence which is buried in the ground a minimum of one (1) foot and is topped with three (3) strands of barbed wire. Two gates are located in the fence for the purpose of permitting truck access to the facility. These gates are padlocked

closed. The main entrance into the facility is located in the south end of the east hall of the Training Center. A key-locked and/or an electrical-locked control barrier door is provided. Figure 4.2.1 describes the facility and its restricted area.

Appendix D provides further discussion and reference to the Physical Security Plan.

5.7 EMERGENCY PLANS

Written emergency procedures are prepared to guide facility personnel in the event of unusual occurrences which endanger their lives. These procedures are included in the Facility Operating Manual and a shortened version is posted. The plans include all creditable emergencies and the proper use of the local emergency support groups, including:

1. Medical Aid and Hospitals
2. State and Local Police
3. Fire Department

The proposed emergency plan is included as Appendix B for reference. Periodic reviews and up-dates of the plan are conducted as described in the plan to assure that it remains a viable document.

CHAPTER 6

INITIAL REACTOR TESTING AND REACTOR OPERATIONS

6.1 INITIAL TESTING

6.1.1 Structural Strength Testing Program

The supporting structures of the reactor core, the core tank and the rod drive mechanisms have been designed with considerable margin of safety. However, the following tests have been undertaken to ensure that they are free from defects due to faulty material or poor workmanship.

6.1.1.1 Reactor Tank and Supporting Beams

The unfueled reactor tank and its support beams was tested first by gradually filling the reactor tank with water. The position of the bottom of the tank with respect to the dump tank and the levelness of the beams were monitored as the water level increased.

The water was then transferred to the dump tank. After the reactor tank was emptied, the dump tank was gradually filled with water. The position of the lower grid plate with respect to an independent upper reference and the levelness of the beams was monitored as the water level in the dump tank increased. The temperature of the water and the room during the test was recorded.

6.1.1.2 Core Structure

The lower and upper grid plates were each tested after installation by gradually loading with lead bricks and/or water bags. The lower plate and its supporting gussets was visually inspected under load for possible defects.

6.1.1.3 Supporting Structure for Rod Drive Mechanisms

The structure was gradually loaded by installing the drive mechansims and by adding lead bricks to simulate the weight of working personnel. The sag of the structure was monitored. The structure under maximum load was visually inspected.

6.1.2 Component and System Testing

Each component or each system of the reactor was individually tested after its installation, if such test were possible and useful. The results of testing was documented for future reference.

The testing included the following items in essentially sequential order.

6.1.2.1 Mechanical and Electrical Components and Systems

1. Electrical system in the reactor area
2. Water supply for the reactor

3. Compressed air system

4. Valves in the moderator system

5. Moderator pump

6. Level measuring system

7. Demineralizer system

8. Core structure

9. Fuel storage area

10. Moderator system

11. Control rod drive mechanisms and position indicating system

12. Control rod magnets and magnet power supply

13. Status indicating and alarm system

14. Interlock and reactor trip system

15. Source drive mechanism and position indicating system

6.1.2.2 Nuclear Components and Systems

1. Gamma sources

2. Neutron sources

3. Survey instruments

4. Area monitors

5. Counting equipment

6. BF_3 channels

7. Compensated ion chamber channels

8. Gamma sensitive detector channel

9. Control rods

10. Fuel elements

6.1.3 Nuclear Tests and Verification of Design and Operational Parameters

Since the reactor core is a reconstitution of a well-known core, the purpose of the nuclear tests were to confirm that all operational parameters were as they should be. The results were systematically documented.

6.1.3.1 Initial Criticality

Standard procedure for approach to critical by fuel loading using the inverse multiplication method was followed for the initial criticality test. Moderator water level lowering during fuel element insertion was not necessary since multiplication characteristics of the core are well known.

6.1.3.2 K-excess Measurement

The normal core loading will consist of 19 standard fuel elements plus five control rods with fuel followers forming a hexagon configuration in the center of the core structure. The fuel/control rod configuration is surrounded by 20 graphite reflector rods.

Standard procedure for loading to excess reactivity was followed to obtain the normal core.

The k_{excess} of the normal core was determined by poisoning the core with standard stainless steel rods in aluminum inserts and by measuring the reactivity worth of the poison rods and the portions of the control rods still in the core, or by measuring the differential rod worths in the normal core.

6.1.3.3 Control Reactivity

The differential and integral reactivity worths of the safety and shim rods were carefully determined under various conditions.

The reactivity effect of the moderator level was investigated and documented.

6.1.3.4 Coefficients of Reactivity

A reactivity computor was used to determine the moderator temperature coefficient, and the local and average void coefficients under various conditions.

6.1.3.5 Flux Mapping and Power Calibration

Detailed neutron flux and power density distributions were determined in the normal core utilizing activation foils. Power calibration of the neutron sensitive channels was obtained by

irradiating activation foils at the average power point in the core. The gamma sensitive channel was then intercalibrated with the linear neutron channel to obtain its power calibration.

6.1.3.6 Power Run and Radiation Survey

The reactor was operated at several increasingly higher power levels up to the full power. Neutron and gamma radiation levels at several key points were monitored. Neutron and gamma radiation levels around the reactor room were mapped with the reactor at full power.

The effect of the neutron source and power level on critical control rod position, and the changes of moderator temperature with power level were monitored.

6.2 OPERATIONS

The Manager of NTR Facility must be cognizant of each activity involving operation of the reactor, movement of the reactor fuel elements or irradiation of material in the reactor and is responsible for the proper execution of the activity.

All operations must comply with the operating procedures reviewed by the Reactor Safeguards Committee and approved by the NTR Facility Manager. The operations and procedures are documented in the

"Facility Operating Manual" and the "Manual of Experiments and Demonstrations", for the reactor.

6.2.1 Training Operations

All training operations must be scheduled in advance and approved by the NTR Facility Manager.

Training reactor operations will be executed only under the direction and direct supervision of a licensed senior reactor operator. It will be the instructor operator's responsibility to ensure that trainees execute all operations in a safe manner. The instructor operators or training engineers will endeavor to impart a correct attitude toward reactor safety and radiation safety to trainees and to make them aware of the responsibility of a reactor operator.

No training operation will be allowed on the reactor when unknown or undocumented cores or untried operating conditions are involved.

Training operations will consist of a formal program of demonstrations and operation practices using the following types of standard training experiments.

a. Reactor unloading
b. Fuel loading approach to critical by 1/M method

c. Simulated PWR core loading (poisoned core)

d. Rod bank 1/M approach to critical with poisoned core

e. Power run with poisoned core

f. Measurement of control rod worth in poisoned core

g. Fuel tube reactivity worth measurement

h. Neutron absorber reactivity worth measurement

i. Water hole reactivity worth measurement

j. Flux mapping by activation

k. Measurement of reactivity worth of the upper water reflector

l. Measurement of various reactor physics parameters

6.2.2 Experimental Operations

For the purpose of calibration and verification of operational parameters of the reactor, the following routine experiments may be periodically executed by the operating staff in accordance with procedures reviewed by the Reactor Safeguard Committee and approved by the facility manager.

a. Flux mapping and power calibration

b. Measurement of radiation levels in and around the reactor

c. Measurement of temperature coefficient

d. Measurement of k_{excess}

e. Measurement of void coefficient

f. Measurement of fuel reactivity worth

g. Measurement of poison reactivity worth

h. Measurement of moderator reactivity effect

6.2.3 Irradiation Operations

Irradiation of samples in the reactor, handling of samples before and after irradiation, acceptance of samples for irradiation and release of irradiated samples must be carried out in accordance with the operating procedures reviewed by the Reactor Safeguard Committee and approved by the NTR Facility Manager.

Each sample irradiation must be evaluated from the point of view of reactor, radiation and contamination hazards and the evaluation must be properly documented. A standard formula is established for routine type sample irradiations.

Reactor operations for sample irradiation must be scheduled in advance and approved by NTR Facility Manager.

6.2.4 Special Operations

Any reactor experiment of non-routine nature which is not covered by approved standard operating procedures must be thoroughly evaluated by the operating staff, reviewed by the Reactor Safeguard Committee, approved by the NTR Facility Manager, and be properly documented.

No training operation will be allowed on a reactor experiment of non-routine nature. Once a reactor experiment has been successfully

carried out and all pertinent parameters have been determined, it may be classified as a routine experiment by the NTR Facility Manager after review by the Reactor Safeguards Committee.

In accordance with 10 CFR 50.59, critical experiments involving a change in the Technical Specifications require prior approval by the Nuclear Regulatory Commission.

CHAPTER 7

SAFETY ANALYSIS

7.1 GENERAL

This chapter evaluates the safety aspects of the facility and demonstrates that the reactor will be operated safely and that credible accidents will not produce any hazardous conditions. The general safety of the facility in accordance with its intended function is deemed excellent as is evidenced by the reactor's use over the past twelve years at Zion. The chapter is divided into five sections. The next three sections deal with categories of conditions that can affect the safety of the facility; environmental, operational and radiation. As a general rule, the analysis of the environmental conditions from a safety standpoint involves evaluating the environment in terms of the increase in probability of unexpected occurrences causing accidents from which hazards can develop. Operational conditions are dependent on the operating system and components, the operational procedures and the operating personnel. The actual operations of the reactor do in fact present an accident potential. Against this potential, a rigorous redundant but flexible operational system has been developed with which experienced personnel can function in performing the reactors training and experimental program. the radiation conditions of the facility are analyzed for normal operations and credible incidents that could cause excess exposures

to personnel. Both environmental and operational occurrences could produce radiation hazards. The fifth section is an analysis of the maximum credible accident (MCA) which has been established for the reactor. The accident that is analyzed is not serious by comparison with the potential accidents of other reactor systems. A reactivity ramp insertion of 0.1$/second, simulating a startup accident is considered even though it is not possible that the means to produce such a ramp insertion to accident proportions is available. A reactivity step accident was not analyzed because it is not believed to be remotely credible. Evidence from reactor systems similar to the NTR, such as portions of the Borax and Spert Tests, indicates that a step accident, to destroy a reactor and cause major internal problems, is difficult to achieve even deliberately.

7.2 ENVIRONMENTAL CONDITIONS AND ANALYSIS

7.2.1 Flood

Section 3.6 describes the hydrology conditions surrounding the site. The maximum seiche level of 5.0 feet above lake level and the maximum wave run-up of 6.7 feet above lake level are seen to be of no consequence at the site since the elevation of the facility is 6.8 feet above the maximum recorded lake level. Flooding of the core would not be of immediate concern, could it occur, as during shutdown the core may be in a flooded condition, and no electrical equipment is required to maintain the core shutdown.

7.2.2 Earthquake

The key structural components of the reactor are designed to withstand loads from seismic accelerations greater than those which have been recorded or are expected at the NTR Site. The low fission product inventory contained in the reactor does not represent a significant hazard to the area surrounding the facility. Therefore, in the event of an earthquake of unexpected strength which causes damage to the reactor and facility, the only resulting radioactive hazards will be localized at the reator site. The possibility of the damaged reactor achieving inadvertent and dangerous criticality conditions, during and after the unexpected event whether initially in an operating or shutdown condition, is not credible.

7.2.3 Windstorm

The maximum expected wind (Sect. 3.7) of 70 mph is not severe enough to cause significant damage to the reactor building. Loss of power which may be caused by severe winds does not hinder the maintenance of the core in a safe shutdown condition.

7.2.4 Fire

The buildings at the site are of fireproof construction. Flammable materials inside the building are minimal. Fire fighting equipment

is located at several prime locations throughout the site and includes dry and wet chemicals as well as water. The core is well below grade and it is unlikely fire could ever reach it.

7.3 OPERATIONAL CONDITIONS AND ANALYSIS

7.3.1 Loss of Power

The control system is basically designed to fail in a safe manner. The loss of electrical power will cause a reactor trip by several means.

7.3.2 Uncontrolled Rod Withdrawal

If the period and power trips fail to function properly, the control interlock system fails and the operator is completey negligent, a so-called start-up runaway is possible. This type of accident, although highly improbable due to the compounded human errors and mechanical failures necessary to bring it about, may be visualized as occurring due to uncontrolled simultaneous withdrawal of all shim control rods at the maximum design rate up to, and beyond, criticality. Since the maximum rate of reactivity inserted due to control rod withdrawal is 0.1\$/sec, the minimum time interval between delayed critical and prompt critical is 10.0 seconds. Even for this rate, the operator has more than sufficient time to trip the reactor before the accident occurs. The safety system

circuitry, including that of the manual trip, is independent of the circuitry which normally moves the rods and, hence is not affected by failures in the rod programming system.

Because the ramp of 0.1\$/sec results in an accident of significant proportions, the control interlock system has been carefully designed to prevent it. As designed, only in the far-subcritical region is the "all shims" mode of rods used which could potentially produce this magnitude ramp if continued. The control rod withdrawal program involves an operation called safety rod cocking. The design limit for withdrawal of this rod to a position representing more than 10\$ shutdown, is experimentally determined for each core and conservatively set prior to such experimental evaluation. Beyond this position, the safety rod cannot be moved in the "safety rod" mode. If the cocked limit switch failed, the safety rod could continue up. However, the reactor could not acieve criticality even with the safety rod fully withdrawn. It is not possible to withdraw shim rods until the safety rod relay is actuated by the safety rod-cocked limit switch.

The all shim rod group withdrawal produces a peak ramp on the order of 0.1\$/sec. This rod group withdrawal is limited to reactor conditions where k_{eff} is less than 0.99. The limiting value of k_{eff} is set at the value determined experimentally or conservatively estimated prior to such experimental evaluation. Failure of the count rate cutout system could permit insertion of

the peak ramp. However, sufficient additional time is given the operator for corrective action that it is not likely this ramp would result in an accident.

The ramp insertion of 0.1$/sec is regarded as a credible accident, with a high degree of designed improbability through the interlock system. It is analyzed in section 7.5 as the maximum credible accident.

7.3.3 Uncontrolled Moderator Insertion

The only moderator ramp which can be achieved with the NTR critical is less than 0.012$/second. Since the moderator ramp of this magnitude can be achieved only when the core is being inundated by the moderator, less fuel will be involved in the criticality incident and the accident cannot approach the proportions of the uncontrolled control rod ramp.

A more rapid moderator ramp is possible when far subcritical. Failure of the interlock system to allow this to occur near critical is not credible. This would entail failure of two separate relays which, 1) deny power to fast moderator insertion unless shim rods are fully inserted, and 2) remove shim rod magnet power when power is applied to the fast moderator insertion mode. Further, the count rate cutout must fail to permit rapid moderator insertion to continue past k-effective equal to 0.99. If failure could occur,

the accident would be less than the 0.1$/second ramp analyzed as the maximum credible accident because of the reduced active core volume.

7.3.4 Failure of Experiments

7.3.4.1 Reactor Loadings

No means is envisioned by which an accident can occur through reactor loading unless a deliberate effort is made to circumvent the multiple guards against it.

It is not believed credible that the reactor will ever be loaded without control rods present. If control rods are present, they are interlocked to prevent withdrawal except with specific limitations.

The number of control rods required by the administrative limitations and the physical makeup of the core structure insures adequate shutdown in itself. Entry into the assembly room with control rods cocked cannot be made without satisfying interlocks which insure through redundancy circuits that the control system is in a safe configuration.

7.3.4.2 Temperature Coefficient Measurements

The nature of temperature coefficient studies in a low power reactor is an evaluation of small changes in reactivity as a function of system temperature. The bulk heat capacity of the

moderator-shielding water and metallic materials that make up the reactor limit the rate at which the system can be externally heated or cooled. Thus, no significant reactivity ramp rates will be involved in this experiment. If the reactor is heated by its own nuclear energy at full power no significant reactivity changes will take place since the maximum temperature rise in the core is less than 20°F.

7.3.4.3 Void Coefficient Measurements

The presence of static voids in an undermoderated water reactor represents a potential hazard because their sudden and unexpected removal from the reactor would cause positive reactivity insertions. Accidental void removal is possible because of their buoyancy characteristics and because of their possibility of flooding. Two independent hold down mechanisms are employed on any experimental apparatus employed which is lighter than water. No credibly collapsible experimental apparatus is employed. All equipment is tested and inspected in water prior to the performance of the experiment. The reactivity associated with voids that can be credibly removed is administratively limited to a value less than 0.8$.

Because of the physical makeup of the fuel elements and the core structure, the only readily voidable locations in the core are the inner water channels in each fuel element. In a normal core loading, the reactivity worth of this voided location is a maximum value in the central fuel element and is less than 0.50$.

7.3.4.4 Importance Function Measurements [Absorbers, Moderator (Void) and Fuel]

Reactivity studies of various types of materials are made as a function of location in the reactor and material composition. All materials placed in the reactor are contained properly and will not be inimical with a water system. The main hazard which exists for experiments of this type involves the consequences of unexpected removal (or insertion) of the material during operations. The unit reactivity worth of the material utilized is kept at a minimum, below a 0.80$. They are also physically placed in the reactor in such a manner so as to make a credible failure highly improbable.

7.3.4.5 Flux Distribution Measurements and Power Calibration

Spacial radiation level measurements are made using neutron and gamma detectors and activation techniques. The reactivity changes associated with the detectors is generally less than 0.20$ Proper holders, supports and containers are utilized. The loss of material in the reactor is highly improbable during the operation of the reactor.

7.3.4.6 Reactor Physics Parameter Measurements

This category of experiments involves the measurement of various reactor physics parameters using standard academic methods. Small quantities of materials are placed in the reactor for irradiation. The materials are then removed and analyzed to determine the required parameter. The credibility of causing an accident with substantial hazards is highly unlikely since the conditions are basically the same as those stated in 7.3.4.5.

7.3.4.7 Irradiation Experiments

Many of the above experiments can be included in this category of experiments, as well as any other experiment in which materials or components are placed in the reactor to study their behavior in a radiation environment or to produce activation by neutron absorption. By carefully preparing the reactor and experimental apparatus and by using proper installation techniques the accidental physical displacement of such experiments can be ruled out. Prior to the performance of an irradiation experiment, estimates of the activity that will be produced in the materials placed in the reactor must be made, as well as the thermal conditions that will exist during the irradiation. If accurate estimates are not possible, irradiations of small quantities of test samples are made at low power to correctly determine conditions for larger quantities of materials at high power. The reactor, at full power, produces a

maximum thermal neutron flux of less than $10^{11} n/cm^2$-s. This moderately high flux is capable of producing significant activity in high cross section material. The activity which is permitted to be produced is dependent on the handling conditions of the post-irradiated materials and their respective half lives. Since no special equipment is presumably available for shielding purposes at time of removal, activities must be kept well below 10 curies. The possibility of accidents occurring for this category of experiments is kept to a minimum by careful review of the proposed irradiation in regard to reactivity effects on the reactor, physical conditions which will prevail and resulting activity produced. The reactivity insertion limit per unit of material inserted and the resulting activity limit assures that the hazards from credible accidents are not excessive.

7.3.4.8 Operational Demonstrations and Experiments

This classification of experiments involves the use of the reactor with its normal core loading and the use of its controls to demonstrate fundamental reactor operations and neutron kinetics. Generally, no external or internal experimental equipment is utilized. Because of the design of the controls, interlock system and safety system, and the administrative restrictions and manual operation, no credible accident is possible that would produce hazardous conditions. The demonstrations and experiments performed are as follows:

1. Normal reactor startup
2. Reactor control and normal power level operations
3. Reactor kinetics studies (prompt neutron transients and delay neutron effects)
4. Reactor control studies and control rod calibration
5. Detector channel studies and behavior during operations

7.3.5 Mechanical Damage and Failure

Mechanical damage and failure of the reactor components and the reactor supporting structure which would jeopardize the safety of the facility and produce dangerous hazards affecting the public are not possible. The low fission product inventory normally in the core and the improbability of inadvertent criticality occurring in a damaged reactor justify the above statement.

The chance of any mechanical failure of the key support members and tanks has been minimized by conservative design in accordance with the applicable boiler code of the American Society of Mechanical Engineers (ASME), the American Institute of Steel Construction (AISC) Handbook and the Aluminum Construction Manaul. An initial testing and inspection program on all important reactor components and supports has been performed with the reactor installed (Chapter 6), and a periodic maintenance inspection program is required. Unintentional minor damage to the system components is a possibility during maintenance and experimental activities. The required pre-operational inspection of reactor components and the elaborate

protection safety and interlock system of the reactor will prevent actual operation with faulty components.

7.3.6 Reactor Maintenance

All Reactor maintenance is performed under the direction of a senior licensed operator. Procedures for periodic preventive maintenance are documented. All operational limitations, safety system requirements, "Key On" room entry conditions, the minimum shutdown reactivity requirement of 3$ for personnel access, fuel and radioactive material handling practices and radiation safety practices are to be followed during maintenance. The performance of special maintenance or modifications requires review and approval. Generally, the possibility of an accident occurring during maintenance operations is reduced by careful planning of all repair and work, and by the performance of the maintenance by an experienced operating staff.

7.4 RADIATION CONDITIONS AND ANALYSIS

7.4.1 Normal Operating Levels

The highest normal radiation levels that exist in the reactor and facility occur at full power operation. Gamma and neutron radiation constitutes the penetrating radiation that emanates from the reactor. The top of the core is located about 11.5 feet below the floor and ground level. In the cylindrical, 8 foot diameter reactor

tank, the normal core is covered by a water thickness of 5 feet on the top, there is about 4 feet of water below the bottom of the core and side water shield is more than 3 feet thick. The dump tank, its concrete pit, the reactor room concrete floor and the soil below the floor afford additional shielding at their respective locations. The annulus between the dump tank wall and the reactor tank allows scattered radiation to stream upward and escape absorption. Although the predominant direction of the radiation that leaves the reactor shielding is upward, the presence of the control rod drive platform and magnetic carriage structure over the reactor and above the floor level tends to scatter radiation in all directions and produce a beacon effect outside the reactor room. Actual measurements have been made at Zion with a five foot water height to determine the radiation levels at various locations. The levels obtained at 10 kilowatt operation were made at 4-1/2 feet above ground level. Directly above the core the radiation exposure fields were 1 R/hr gamma and 5 mrem/hr neutrons. At a location immediately outside the reactor room, the exposure fields were 4 mR/hr gamma and .5 mrem/hr neutrons. At the boundary of the controlled area, the exposure fields were 8 mR/hr gamma and 0 mrem/hr neutrons. The total annual energy output safety limit that has been established for the facility is 200 kilowatt hours. This indicates that the reactor can only be operated at 10 kW for 20 hours in any one year. Therefore, the provisions set down in 10 CFR 20 are satisfied for unrestricted area (assumed to be the restricted area boundary) and for exposures to operating staff personnel in the facility.

Over the past years at Zion, air and water samples have been analyzed to determine the radioactivity levels in them on a monthly basis. The records indicate that particulate contamination of the air has never been encountered to any significant degree and the activities (microcurie per milliliter) generally remain below 10^{-11} ($\beta - \gamma$) and 10^{-11} (α). The water contamination, (μCi/ml) has also remained at low levels, less than 10^{-7} ($\beta - \gamma$) and 10^{-10} (α), with a slight increase observed following a full power run. No long term increase in radioactive contamination in the water has been observed, indicating that the fuel integrity has been preserved over the years.

7.4.2 Loss of Shielding Water

A credible incident could occur regarding radiation conditions if an auxiliary reactor trip is initiated at full power and the gamma sensitive safety channel over the reactor malfunctions.

Normally upon actuation of the water dump at full power the gamma sensitive channel high level trip will automatically produce a reactor trip since water shielding is being removed between the detector and core. If it is assumed that no reactor trip takes place the reactor will begin to shutdown only when the water level reaches the reflector region of the core. This condition removes the upper reactor shield and would produce excessive levels of radiation in the facility. The reactor is finally shutdown as the water level drops into the core. The resulting radiation level

produced would rise to a maximum value over the minute it requires the water level to drop into the core. At this point, the neutrons escaping at the top of the reactor represent the major radiation hazard, the gamma contribution is much smaller. The radiation levels in the facility then turn downward as the reactor is made subcritical and the reactor power begins to drop, slowly at first then must faster. About 30 seconds after the maximum radiation levels were reached, the neutron level is very low and decreasing slowly, the decay gammas from the fission products represent the major contribution to the radiation level above the reactor. The reactor tank is finally emptied removing the water shielding completely from the reactor. Measurements and extrapolation thereof indicate that the maximum radiation levels which would be reached at the Zion facility would be high. However, the total integrated exposure dose over the two minutes that the exposure takes place would not be hazardous to personnel. At the same facility locations that were described in section 7.4.1, the following maximum radiation exposure levels and total integrated doses would be encountered: directly above the reactor 462 rem/hr and less than 4 rem, at the outside of the reactor room wall 4.5 rem/hr and less than 38 mrem and at the boundary of the facility there is no prompt radiation dose pertinent to 10 CFR 20. Since the reactor room walls represent the closest location that an individual can occupy, relative to the reactor centerline, the total exposure dose received during the incident will not be extreme. Following the incident the radiation levels, will continue to drop in accordance with fission product decay laws. For a typical full power run; one (1) minute

after the reactor is shutdown the level over the reactor is < 5 mr/hr.

In conclusion, the resulting radiation conditions produced by the incident are not extreme regarding personnel exposure and would not effect the general public. Also, the actual conditions produced during an auxiliary reactor trip should cause the gamma sensitive safety channel above the reactor to produce a reactor trip. The trip point for this channel is set at 120 percent of the value measured at 10 kW operation and with normal water height. The reactor would therefore be shutdown by a reactor trip shortly after the water dump was actuated. The resulting radiation condition would be caused by decay of fission products reaching a maximum as the reactor core is uncovered. The peak levels of radiation produced at various locations in the facility as compared with the above incident will be lower by more than a factor of ten.

7.4.3 Radioactive Material Storage Areas

7.4.3.1 Fuel Storage

Only the approved fuel storage locations are used to store fuel elements. The geometrical configuration of stored fuel in the main fuel storage area makes accidental criticality incidents impossible. A separate gamma sensitive detector, with readout and audible level trip capabilities is positioned to monitor the main fuel storage area. This detector acts as an area (reactor room)

monitor and as a de facto criticality detector satisfying the provisions set down in 10 CFR 70 regarding criticality monitors required by SNM licenses. The radiation conditions which exist when the fuel is stored in the main fuel storage are a function of the current irradiation history of the reactor. The storage area is constructed in such a manner that its shielding components do not permit a high radiation area to exist except inside the storage area or directly over it. The level of radiation at the facility boundaries is therefore the normal background level when fuel is in storage.

7.4.3.2 Radioactive Samples and Source Storage

All radioactive materials above exempt quantities are stored in the reactor room in appropriate containers and with adequate shielding. The radiation levels outside the shields are kept well below that which constitutes a high radiation area. Careful documented inventories are made periodically to assure that the location of all radioactive materials is known. Through proper management of these materials, the possibility of causing an incident that would produce hazardous conditions can be readily kept minimal.

7.4.4 Fuel Element Handling and Irradiated Material Handling

The practices utilized at the facility when handling radioactive material include: establishment of approved procedures, estimations of the activity to be handled and expected exposure to handlers,

adequate practice of procedures involving physical manipulations, careful monitoring of radiation levels when actually performing operations, and procedures to be following in case of credible accident. Since the quantities of radioactive materials to be handled are normally small (usually well below 10 Ci), manual handling of the materials is utilized using appropriate handling tools when necessary. The possibility of a physical accident occurring is credible; however, the resulting hazards produced by the accident will be readily contained within the reactor room and will be of no consequence to the public. The hazards produced are kept at a minimum to the operating personnel by initially assuring that the materials placed in the reactor are well contained and cannot be easily damaged. The clean-up problem which may exist will depend on the scope of the accident and the material in question but it is not expected to be significant.

7.5 MAXIMUM CREDIBLE ACCIDENT

7.5.1 Description and Summary of the Accident

The Maximum Credible Accident is an uncontrolled 0.1$/sec reactivity ramp during reactor startup. The accident is conceived as occurring in the following manner:

1. During the course of reactor startup with the safety rod cocked and the reactor shutdown by less than five percent, the operator withdraws the remaining control rods, using the "ALL SHIMS" rods withdrawal mode.
2. Because of operator inattention, failure of the count rate cutout, decading level trip, Log-N period and level trips, the reactor is brought to a supercritical condition on a 0.1$/sec ramp.
3. A power excursion occurs in which approximately 5 MW-sec energy is released to the peak of the power burst.
4. The excursion is self-terminated by the formation of voids due to moderator boiling. No damage is incurred by the reactor. The reactor chugs at a power less than 1 MW until the reactor is manually tripped.

7.5.2 Analysis of the Accident

7.5.2.1 Introduction

The event analyzed is an uncontrolled ramp reactivity insertion of 0.10$/sec that is initiated from a subcritical condition of approximately 1 milliwatt. The subsequent excursion and shutdown are based upon the well documented results of the SPERT-I Core A-17/28 experiments. Table 7.5.1 "Comparison of the NTR N-24-S and SPERT-I Core A-17/28" indicates the similarities between the two cores.

The quantity of fuel, amount of excess reactivity, and metal to water ratio in the A-17/28 core compare favorably with the NTR N-24-S core. It should be noted that the average void coefficient in the SPERT-I core is less negative than that for the NTR. Since the formation of voids is the mechanism by which the reactor is shutdown, the NTR would require less voiding, thus less heat to add the 1$ of negative reactivity required for shutdown. Therefore, the use of the SPERT-I experimental results is a conservative approach to this analysis.

7.5.2.2 Power Excursion

The power excursions of the SPERT-I core experiments were initiated by remote ejection of the transient center rod from the core producing essentially step-wise insertions of reactivity (Ref. 3). For the ramp insertion of reactivity the most important parameter is the rate of reactivity insertion; the reactivity coefficient which involves the average void coefficient, the prompt neutron lifetime, and the initial power is a secondary effect. The ramp and step excursions are approximately equivalent when the maximum reciprocal period in the ramp case equals the constant reciprocal period in the step case (Ref. 1, p. 503). A 0.10$/sec ramp in the NTR is equivalent to a 0.070 %Δk/sec ramp for the SPERT-I core. Figure 7.5.1, "Maximum Reciprocal Period Vs. Initial Power Level as a Function of Ramp Rate for SPERT-I Core A-17/28" (Ref. 1, p. 504), was used to determine the maximum reciprocal period in this analysis. A ramp rate of 0.09 %Δk/sec and an initial power of 10^{-3} watts were conservatively chosen. These values yield a maximum reciprocal period of approximately 21.5 sec^{-1}.

This value for reciprocal period can be used in conjunction with Figure 7.5.2, "Maximum Reactor Power Vs. Maximum Reciprocal Period During Ramp Tests on SPERT-I Core A-17/28" (Ref. 1, p. 504), to arrive at the maximum reactor power level of approximately 70 Mw. Accordingly using Figure 7.5.3, "Energy to Peak of Power Excursion Vs. Reciprocal Period for SPERT-I Core A-17/28" (Ref. 1, p. 501), the energy produced to the peak power was less than 5 Mw-sec. During this power excursion the maximum fuel plate surface temperature would be approximately 160°C as indicated by Figure 7.5.4, "Maximum Fuel Plate Surface Temperature Vs. Reciprocal Period for SPERT-I Core A-17/28" (Ref. 1, p. 503).

As half the thickness of the NTR metalic fuel rings is 0.16 cm, it is expected that the temperature gradient across the fuel plate will be small. After the power burst the reactor chugs at a power level less than 1 Mwt until the reactor is manually tripped.

7.5.2.3 Conclusion

As the estimated maximum fuel plate surface temperature (160°C) is substantially below its melting point (660°C), no damage to the core is expected and consequently no fission products are released. The prompt radiation dosage received outside the NTR reactor room, 4 1/2 feet above floor level, would be less than 60 mrem for neutron and 75 mR for gamma. This dosage is obtained with a five foot water height over the core. There is no off-site prompt radiation dose pertinent to restrictions of 10CFR20. As there is no release of fission products to the atmosphere, post excursion radiation doses are controlled by cleanup procedures.

7.5.2.4 References

1. Reactor Physics Contents, ANL-5800, 2nd Edition, Argonne National Laboratory, July 1963. Pp. 497 - 507.

2. "Safety Considerations for the 24 Element Graphite Reflected Core", December 3, 1980. (Submitted to NTC by letter dated March 19, 1981, A. J. Nardi to James R. Miller).

3. H. L. Witener, et. al., "Reactor Instrumentation and Test Procedures", Report on SPERT-I Destructive Test Results, Transactions of the American Nuclear Society 6, 1, p. 137 (June, 1963).

TABLE 7.5.1

COMPARISON OF THE NTR N-24-S CORE AND THE SPERT-I CORE A-17/28

	NTR[2]	SPERT[1]
Number of fuel assemblies	24	28
Fuel assemblies	3 concentric cylinders	MTR plate type
U-235 per fuel assembly, gm	200	168
Fuel length, cm	91.4	61.0
Plate spacing, cm	0.239, 6.35 center	0.297
Clad material	Al	Al
Clad thickness, cm	0.0927	0.0508
Critical mass, kg U-235	4.1	3.9
Total U-235 loaded, kg	4.8	4.7
H/U ratio	370	320
Metal to water ratio	0.76	0.79
Available excess reactivity, $	5.5	5.2
β	0.008	0.007
Temp. coefficient at 20°C, $/°C	-0.225×10^{-2}	-0.67×10^{-2}
Central void coefficient, $/cm^3	-8.08×10^{-4}	-9.3×10^{-4}
Average void coefficient, $/% void	-.40	-.34
Average void coefficient, (C_v), $/cm^3	-6.27×10^{-4}	-4.6×10^{-4}
Prompt neutron lifetime, sec.	$.5 \times 10^{-4}$	$.48 \times 10^{-4}$

1) Reference 1, page 498

2) Reference 2

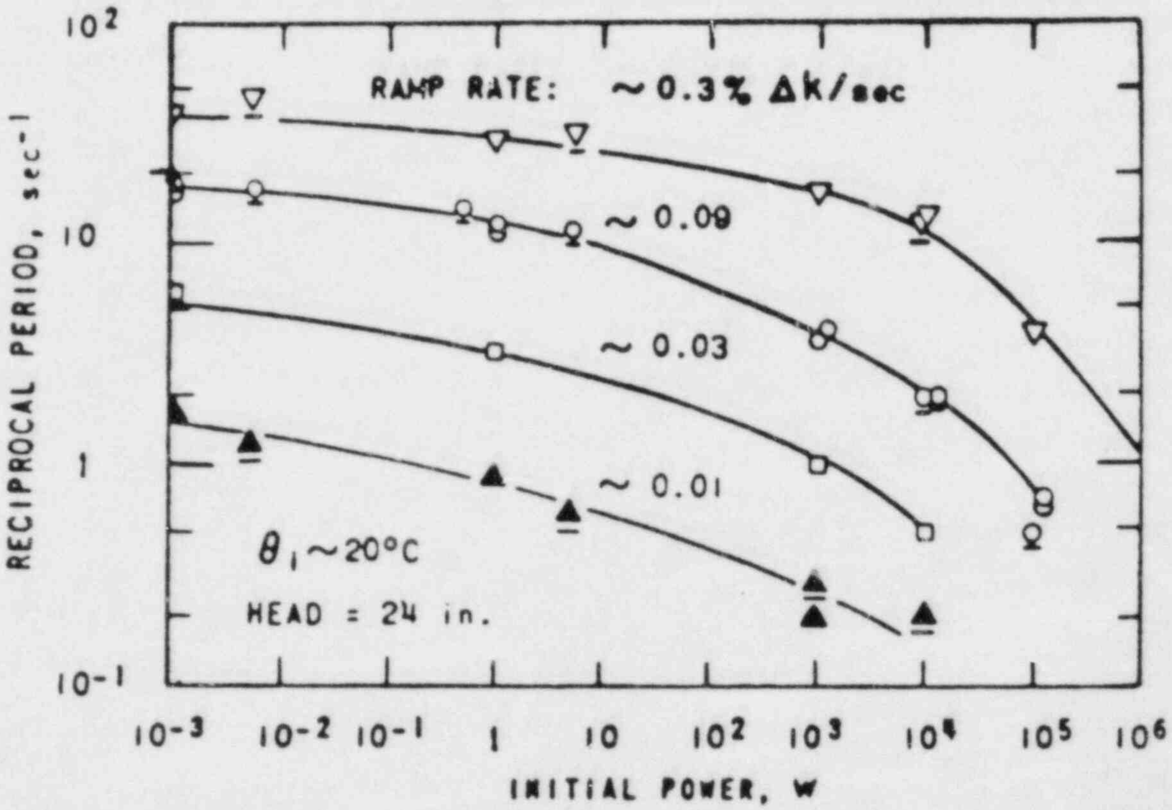


FIGURE 7.5.1: MAXIMUM RECIPROCAL PERIOD VS. INITIAL POWER LEVEL AS A FUNCTION OF RAMP RATE FOR SPERT I CORE A 17/28

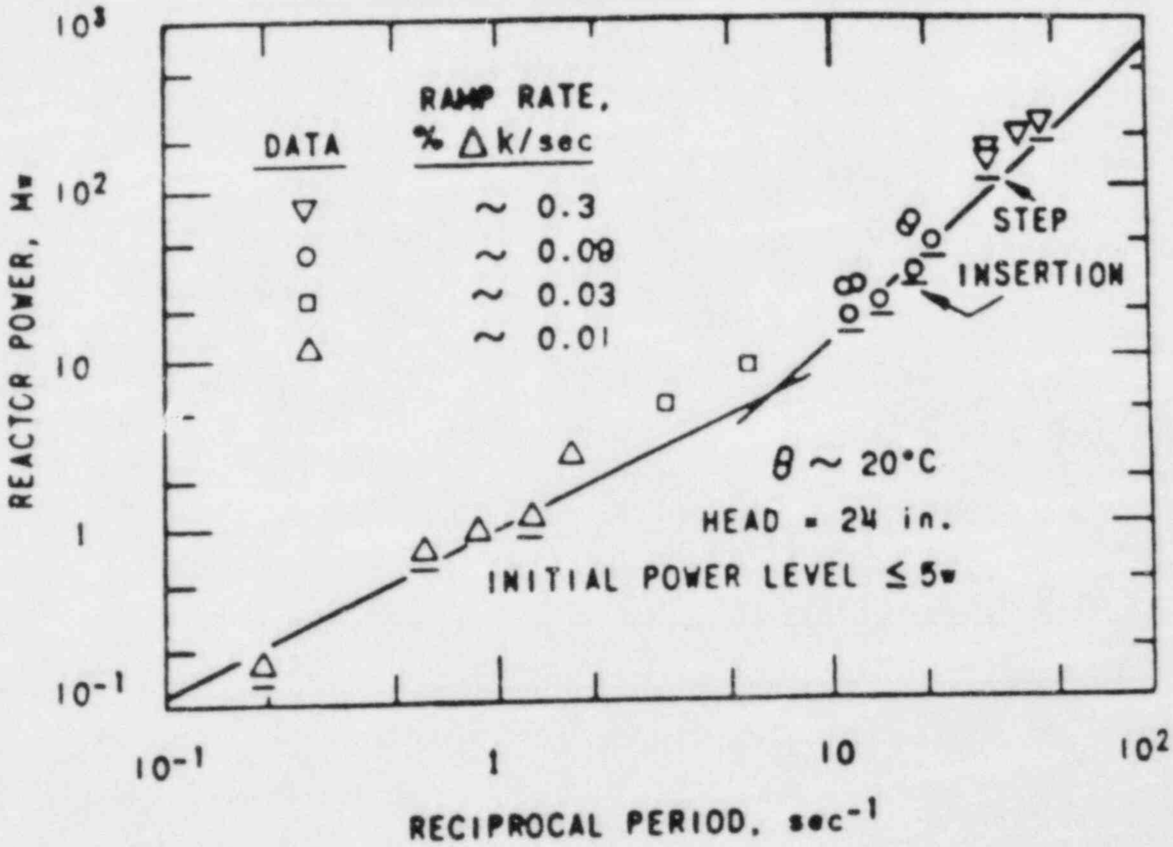


FIGURE 7.5.2: MAXIMUM REACTOR POWER VS. MAXIMUM RECIPROCAL PERIOD DURING RAMP RATE TESTS ON SPERT-I CORE A-17/28

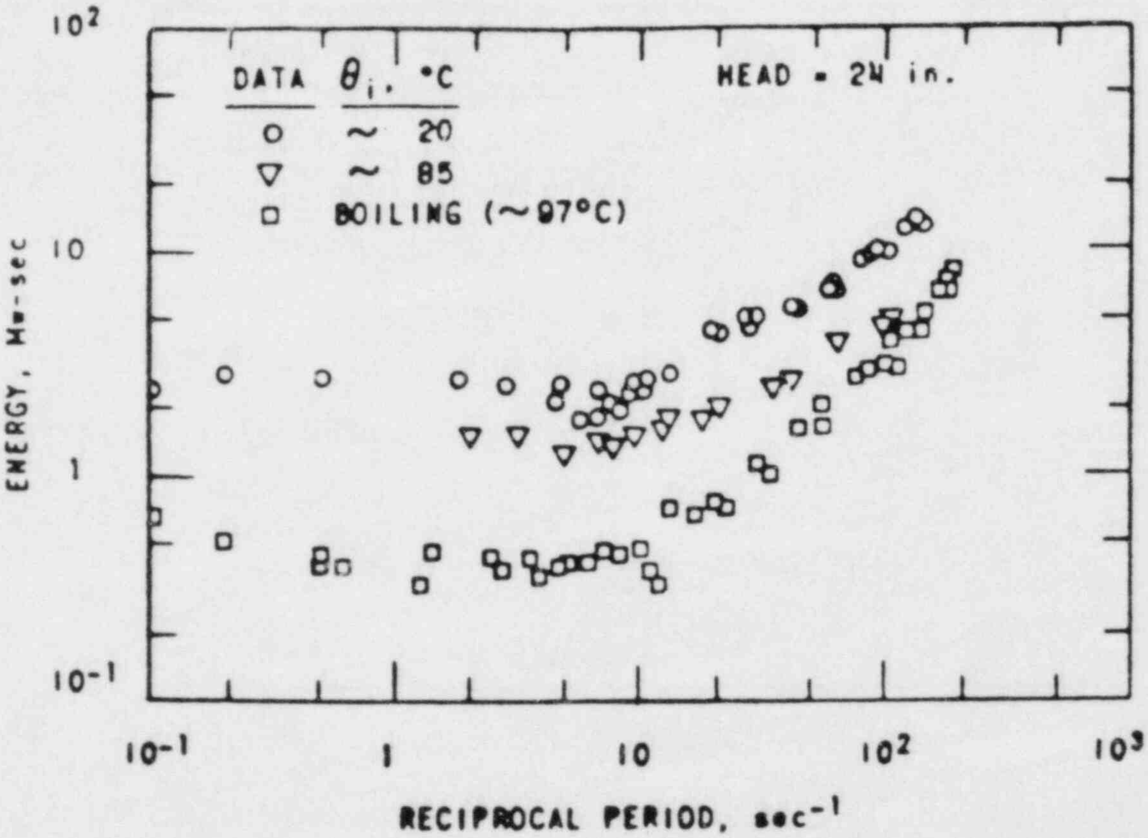


FIGURE 7.5.3: ENERGY TO PEAK OF POWER EXCURSION VS. RECIPROCAL PERIOD FOR SPERT I CORE A 17/28

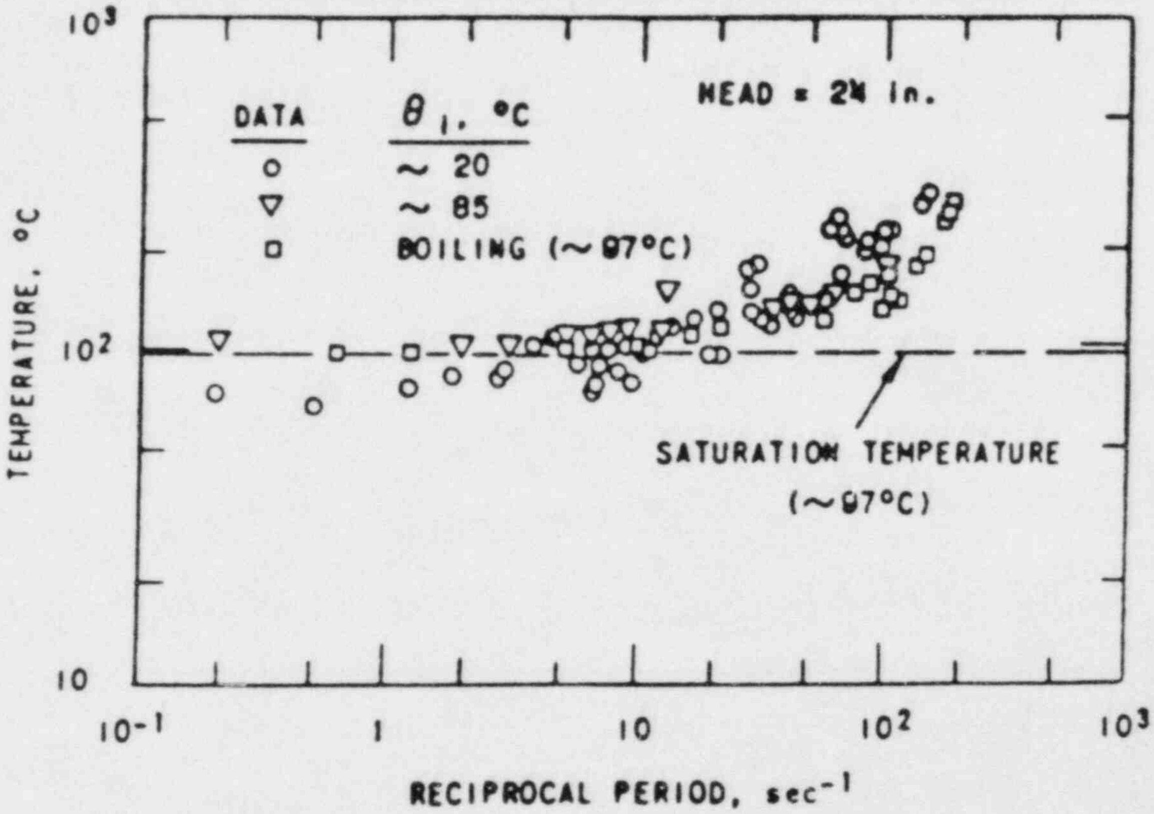


FIGURE 7.5.4: MAXIMUM FUEL PLATE SURFACE TEMPERATURE VS. RECIPROCAL PERIOD FOR SPERT I CORE A 17/28

SUMMARY OF TECHNICAL SPECIFICATIONS CHANGES

	Section	Pages	Content/Reason
1.0	Definitions	3, 4, 5, 6	Rearrange in alphabetical order and add three definitions
1.10	Movable Experiment	4	Definition added
1.14	Reactivity Worth of An Experiment	5	Definition added
1.20	Secured Experiment	6	Definition added
1.21	Unsecured Experiment	6	Definition added
2.1	Safety Limits	8	Correct maximum temperature increase
2.2	Limiting Safety System Settings	9	Clarify basis
4.0	Experiments	21	Redefine experimental limitations in terms of movable experiments and unsecured experiments. Limit positive reactivity worth of experiments.
		23	Reword bases to reflect these changes
5.0	Surveillance Requirements		Change wording to reflect accepted industrial format and provide operational flexibility [ANS 15.1-1982, page 6]
	5.1.1	24	
	5.1.2		
	5.1.3	25	
	5.1.5		
	5.1 Bases	26	Add paragraph outlining maximum intervals for surveillance
	5.2.1.A	27	
	5.2.1.B	28	
	5.2.1.C		
	5.3.1	29	Reflect increase interval from 3 months to 6 months well with industrial (ANS standards) and past operating history
	5.3.4	29	WORDING change

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Section		Pages	Content/Reason
5.3	Surveillance Requirements Bases	30	Reflect change in 5.3.1
6.2	Facility	32	Changes reflect amended controlled access area
6.4.3	Standard Fuel Element	33	Spelling
6.4.6	Graphite Reflector Rods	38	Correct dimensions and clarify language
6.5	Water Handling System	39	Correct flow rates
6.6	Fuel Storage	39	More closely reflect reality
7.3.4.2	Special RSC Reports	44	Correct spelling

FSAR

APPENDIX A

FACILITY LICENSE NO. R-119

PROPOSED

TECHNICAL SPECIFICATIONS

FOR THE

WESTINGHOUSE NUCLEAR TRAINING REACTOR

DOCKET NO. 50-87

0281C

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1.0 DEFINITIONS

The following terms are defined to aid in the uniform interpretation of the specifications.

1.1 Administrative Controls - the provisions related to organization and management, personnel requirements, procedures, record keeping, review and audit, and reporting that are considered necessary to assure operation of the facility in a safe manner.

1.2 Auxiliary Reactor Trip - consists of the dumping of the moderator-shield water through the ten-inch dump valve. The trip is initiated manually.

1.3 Channel Calibration - adjusting a channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip.

1.4 Channel Check - a check to verify by intercomparison of channel outputs whether or not a measuring channel is operable.

1.5 Channel Test - a test to verify, by introducing externally generated signals, that a measuring channel is operable.

1.6 Excess Reactivity - that reactivity above critical which may be added by manipulation of remote controls.

1.7 Experiment - any installed apparatus, device, or material within the confines of the reactor tank which is not a normal part of the assembly, or any core loading which is not the normal core loading.

1.8 Measuring Channel - an arrangement of components and modules as required to measure the value of a process variable. The output of the measuring channel is the measured value of the process variable and may be considered the true value within the accuracy of the measuring channel.

1.9 Moderator-Shield Water - the water that is placed in the reactor tank.

1.10 Movable Experiment - one where it is intended that the entire experiment may be moved into and out of the reactor while the reactor is operating.

1.11 Neutron Source - any neutron-emitting radioactive material, other than the reactor fuel, which is positioned in or near the reactor core to provide an external source of neutrons.

1.12 Operable - when a system or component is capable of performing its required function in a normal manner. (Operating means it is performing its function.)

1.13 Readily Available on Call - normally means within a 20-mile radius of the facility and that the operator-on-duty knows the location and telephone number of the senior operator on duty.

1.14 Reactivity Worth of an Experiment - the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter experiment position or configuration.

1.15 Reactor Operational - means the reactor is not secured.

1.16 Reactor Safety System - that combination of measuring channels that forms the automatic protection system for the reactor or that provides information which requires manual protective action to be initiated.

1.17 Reactor Secured - means that all control rods are in their down positions and the key is removed from the console lock, or when there is no fuel in the core.

1.18 Reactor Trip - consists of a gravity drop of control rods caused by the interruption of the electrical power to the magnet carriages. The trip can be initiated automatically by the safety system, manually by the manual reactor trip and manually by disconnecting the facility electrical power.

1.19 Safety Channel - an arrangement of components and modules as required to generate a single protective action signal when required by a facility condition.

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

1.21 Unsecured Experiment - any experiment that is not a secured experiment.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability

This specification applies to the reactor power level limitation and the annual integrated thermal power limitation.

Objective

The purpose of this specification is to establish the upper safety limit on power level for the reactor and the integrated thermal power produced annually.

Specifications

1. Maximum power level shall not exceed 20 kilowatts (thermal).
2. The integrated thermal power (thermal energy) for any calendar year shall not exceed 200 kilowatt hours.

Bases

A maximum power level of 20 kWt and an integrated thermal power of 200 kW-hr per year provide adequate flexibility for the performance of training and irradiation operations and at the same time appropriately limit the quantity of radioactive material available for release.

Calculations and measurements have shown that during steady-state operation of 20 kWt the average moderator temperature increase is less than 20°F, and therefore, the clad and fuel temperatures remain significantly below their failure point.

With the moderator-shield water level at a height of about 5 feet above the top of the core and the power level at 20 kWt, estimates based on shielding calculations and extrapolated experimental data show that the radiation level immediately outside the reactor room is below 30 mRem/hr and the radiation level at the controlled area boundary is less than 2 mRem/hr. Access into the reactor room during reactor operation is prohibited administratively and by a security interlock system.

The maximum operating power level of the reactor is 10 kWt. In general, the operating power level is kept as low as possible, consistent with the training operation requirements, and normally is less than one hundred watts.

2.2 Limiting Safety System Settings

Applicability

This specification applies to the settings for instruments monitoring parameters associated with reactor safety limits.

Objective

The purpose of this specification is to assure protective action before safety limits are exceeded.

Specifications

The limiting safety system settings shall be as follows:

Maximum Power Level	12 kWt
Minimum Flux Level	2.5 neutron/cm^2-s
Minimum Period	3 seconds
Maximum Gamma-ray Exposure Rate (above the water shield)	R/hr value experimentally determined for 12 kWt power level operation with normal moderator-shield water height

Bases

The maximum power level trip setting of 12 kWt is established by estimating an error of 20 percent in the absolute neutron flux measurement by activation methods and a maximum error of 25 percent in the nonlinearity of monitoring instruments. The safety margin prevailing between the safety limit and the limiting safety system setting is adequate to allow for these errors.

The minimum flux level in the core has been established to prevent a source-out startup. A minimum interlock setpoint of 2 cps on a source level detector shall be used to assure this minimum neutron flux level.

The minimum 3-second period is specified so that there is sufficient time for the automatic safety system to respond before the power level safety limit is exceeded. The transient would be terminated in less than 200 milliseconds after the reactor trip.

The maximum gamma-ray exposure rate setting is established to correspond with the exposure rate above the top of the reactor sheild that would occur during reactor operation at 12 kWt with the moderator-shield water level at the normal (5 foot) water height. This setting assures that increasing radiation

levels in the vicinity of the reactor room will be detected before they become excessive when the reactor is operated at moderator-shield water heights other than the normal level. An approximate value for this setting is estimated to be 500 mR/hr.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Control and Safety Systems

Applicability

These specifications apply to all methods of changing core reactivity available to the reactor operator.

Objective

The purpose of these specifications is to assure that an adequate shutdown method is available and that positive reactivity insertion rates are within those analyzed in the Safety Analysis Report.

Specifications

1. There shall be a minimum of five operable control rods. The maximum excess reactivity that can be loaded shall be such that the reactor shall be subcritical by a margin greater than 1$ with the control rod having the largest reactivity worth fully withdrawn.

2. The maximum control rod and moderator-shield water reactivity insertion rate shall be less than 0.10$/s when keff is less than 0.99 and less than 0.035$/s when keff is greater than 0.99.

3. The total control rod drop time for each control rod from its full-out position to its full-in position shall be less than or equal to 1.2 seconds. This time shall include a maximum magnet carriage release time of 0.125 second.

4. Negative reactivity shall be available in operable cocked control rods prior to adding the moderator-shield water to the reactor. At least 1$ of negative reactivity shall be available when core loadings, capable of becoming critical, are to be filled with the moderator-shield water.

5. The auxiliary reactor trip (moderator-shield water dump) shall add negative reactivity within one minute of its activation. Remote auxiliary reactor trip controls shall be available at the console and in the reactor room.

6. The normal moderator-shield water level shall be established at a minimum of 5 feet above the top of the core. Reactor operations at water levels below this normal level shall be permitted only when the operating power is lowered accordingly. (Refer to Specification 2.2, "Maximum Gamma-ray Exposure Rate".) Moderator level and adjustments near criticality shall be made only after first establishing criticality by control rod manipulation.

7. A manual reactor trip shall be included in the reactor safety system. Controls for the reactor trip shall be available at the console and in the reactor room.

8. A manual electric switch shall be provided in the facility for the purpose of disconnecting the electrical power of the facility and causing a reactor trip.

9. The minimum safety system channels that shall be operating during reactor operation are listed in Table 1.

10. The interlocks that shall be operable during reactor operations are listed in Table 2.

Bases

A minimum number of five control rods and a maximum excess reactivity are specified to assure that there is adequate shutdown capability even for the stuck control rod condition. The only authorized core configurations are centered in the core structure utilizing all five control rods. The minimum obtainable critical fuel loading in the reactor, in the best right cylindrical configuration centered in the core structure, consists of sixteen fuel elements and five control rods surrounded with a single ring of twenty graphite reflector rods. With control rods withdrawn, there are effectively twenty-one fuel elements (considering control rod fuel followers as fuel elements) in the minimum critical loading.

The core loading consisting of nineteen fuel elements and five control rods in a hexagonal configuration, surrounded by a single ring of twenty graphite reflector rods, centered in the core structure represents the normal loading. The normal core loading is used to establish a maximum excess reactivity of <6$ for the reactor.

The maximum reactivity insertion rates, far from and near criticality, are specified to assure that the reactivity addition rate is less than that analyzed in the maximum credible accident (MCA). The maximum control rod withdrawal rate and the moderator-shield water addition rate are controlled by these limitations.

The insertion time of less than 1.2 seconds for each control rod from its fully withdrawn position is specified to assure that the insertion time does not exceed that assumed when establishing the minimum period specified in specification 2.2 as a limiting safety system setting.

The required control rod withdrawal prior to adding the moderator-shield water is specified to assure that reactor trip will have the capability of adding negative reactivity during reactor startup.

The auxiliary reactor trip is specified to assure that there is a secondary mode of shutdown available during reactor operations. The requirement that negative reactivity be introduced in less than one minute following activation of the trip is established to limit the consequences of a potential power transient. By including a remote auxiliary reactor trip control in the reactor room, the trip may be activated readily by individuals in the room under emergency conditions.

The normal moderator-shield water level of the reactor is established at a minimum of five feet above the top of the core to assure that an adequate shield is provided at the maximum power level of the reactor. When reactor operations require a lower moderator-shield water height (down to a one foot level), the operating power must be lowered accordingly so that the gamma-ray exposure rate limit setting is not exceeded over the core. When moderator water level reactivity control is to be utilized (water level below one foot from the top of the core), the gamma-ray exposure rate limit setting is lowered accordingly - to approximately 1/10 of its maximum permissible value - to further reduce the possibility of operating with a high neutron and gamma radiation field in the vicinity of the reactor room. Controlling the reactor by moderator level near criticality is permitted only when the reactor is first made critical by control rod movements. This assures that the control rod is the primary mode of reactivity control in a critical reactor.

A manual reactor trip assures that a reactor trip can be readily initiated by an operator at his demand. Including a manual reactor trip control in the reactor room assures that the trip may be activated readily by individuals in the reactor room under emergency conditions.

By providing a method of disconnecting the facility electrical power, an additional mode is established to manually cause a reactor trip.

The safety system channels listed in Table 1 provide a high degree of redundancy to assure that human or mechanical failures will not endanger the reactor facility of the general public.

The interlock system listed in Table 2 assures that only authorized personnel can operate the reactor, that the proper sequence of operations is performed, that no one can accidentally enter the reactor room when the reactor is operating, and that the reactor room is entered with proper conditions prevailing when the master console key is on.

3.2 Reactor Parameters

Applicability

These specifications apply to core nuclear parameters and moderator-shield water physical parameters.

Objective

The purpose of the specifications on reactivity coefficients is to assure that the inherent reactivity feedback mechanisms of the water moderator are safe. The purpose of the specification on purity of the water moderator is to assure adequate corrosion control in a room temperature, open air aluminum-water system.

Specifications

1. At temperatures greater than 80°F, the temperature coefficient of reactivity shall be negative and shall have a minimum absolute value of 1×10^{-3} \$/°F.

2. The void coefficient of reactivity shall be negative and shall have a minimim absolute value of 1×10^{-1} \$/percent void fraction.

3. The moderator-shield water shall have a pH within the range of 4.5-8, inclusive, and a resistivity greater than 200,000 ohm-cm when averaged over a two-month period.

Bases

The minimum absolute value of the temperature coefficient of reactivity is specified to assure that an adequate inherent negative reactivity effect takes place when the reactor temperature increases above the value where the coefficient becomes negative. At lower temperatures, where the coefficient together with the slow rate at which controlled temperature increases may be effected provide assurance that positive reactivity insertion due to controlled temperature increases will be small enough and slow enough as to be safely controllable. Uncontrolled heatups will quickly raise the moderator temperature to the range where the coefficient is negative while providing only a negligible positive reactivity effect.

The minimum absolute value of the void coefficient of reactivity is specified to assure that the negative reactivity insertion due to void formation is greater than that which was calculated to occur in the SAR.

The moderator-shield water quality is specified to assure adequate corrosion control in a room temperature, open air, aluminum-water system. This corrosion is a long-term reaction. Based on years of experience, a two-month period for averaging has proven adequate to avoid quality degradation which would result in appreciable corrosion.

3.3 Radiation Monitoring

Applicability

These specifications apply to the minimum radiation monitoring requirements for reactor operations.

Objective

The purpose of these specifications is to assure that adequate monitoring is available to preclude undetected radiation hazards or uncontrolled releases of radioactive material.

Specifications

1. The minimum radiation monitoring systems for reactor operation shall include:

   A. A critical detector system which monitors the main fuel storage area and also functions as an area monitor. This system shall have an audible alarm in the reactor room.

   B. An area gamma monitor in the console room with an audible alarm.

2. Instruments to permit the periodic sampling and measuring of radioactivity in the air and the moderator-shield water shall be provided.

3. Portable detection and survey instruments shall be provided.

Bases

The continuous monitoring of radiation levels in the reactor room and the console room assures the warning of the existence of any abnormally high radiation levels. The availability of instruments to measure the amount of radioactivity in the air and moderator-shield water will assist in monitoring fuel clad integrity and assures continued compliance with the requirements of 10 CFR Part 20. The availability of the required portable monitors provides assurance that personnel will be able to monitor potential radiation fields before an area is entered.

4.0 EXPERIMENTS

Availability

These specifications apply to all experiments placed in the reactor tank.

Objective

The objective of these specifications is to define a set of criteria for experiments to assure the safety of the reactor and personnel.

Specifications

1. No experiment shall be performed until a written program, which has been developed in sufficient detail to permit good understanding of the safety aspects, is reviewed by the Reactor Safeguards Committee (RSC) and approved by the Facility Manager.

2. No experiments shall be conducted if the associated experimental equipment could interfere with the control rod functions or could adversely affect the nuclear instrumentation.

3. The maximum reactivity change for withdrawal and insertion of movable experiments shall be 0.25$.

4. The maximum reactivity worth of any individual unsecured experiments shall be limited such that the failure of any experiment or associated equipment will not result in a positive reactivity addition greater than 0.80$. In

addition the reactivity worth of all unsecured experiments is limited such that a common mode failure of all such experiments and their associated equipment will not result in a positive reactivity addition greater than 0.80$.

5. The maximum positive reactivity worth of all experiments in the core shall be limited to 1$.

6. Experiments shall not contain explosives or other material which may produce a violent chemical reaction and/or airborne radioactivity.

Bases

The experiments to be performed in the reactor programs are discussed in the Safety Analysis Report (SAR). The present programs are oriented almost exclusively toward fundamental reactor technology training. Other special programs may involve the use of the reactor as an irradiation facility. To assure that experiments are well planned and evaluated prior to being performed, detailed written procedures for all new experiments must be prepared, reviewed by the RSC and approved by the Facility Manager.

Since the control rods enter the core by gravity and are required by other Technical Specifications to be operable, no

experiment should be allowed to interfere with their functions. To assure that specified power limits are not exceeded, the nuclear instrumentation must be capable of accurately monitoring core parameters.

All reactor experiments are reviewed and approved prior to their performance to assure that the experimental techniques and procedures are safe and proper, and the hazards from possible accidents are minimal. A maximum reactivity change is established for movable experiments to assure that the reactor controls are readily capable of controlling the reactor.

A positive reactivity addition of 0.80$ due to failure of an unsecured experiment or associated equipment would cause the reactor power to rise on a stable period greater than one second. Thus, the reactor safety systems would be able to trip the reactor before an excessive power level is reached.

The limit on positive reactivity worth of experiments assures an ample shutdown margin in the reactor.

Restrictions on irradiations of explosives and highly flammable materials are imposed to minimize the possibility of explosions or fires in the vicinity of the reactor.

To minimize the possibility of exposing facility personnel or the public to radioactive materials, no experiments will be performed with materials that could result in a violent chemical reaction and/or produce airborne radioactivity.

5.0 SURVEILLANCE REQUIREMENTS

5.1 Reactor Control and Safety

Applicability

These specifications apply to the surveillance of the safety and control apparatus and instrumentation of the facility.

Objective

The purpose of these specifications is to assure that the safety and control equipment is operable and meets the criteria established in the design bases.

Specifications

1. The total control rod drop time and magnet release time shall be measured semiannually to verify that the requirement of specification 3.1, item 3, is met.

2. The moderator-shield water dump time shall be measured semiannually to verfiy that the requirement of specification 3.1, item 5, is met.

3. The maximum control rod and moderator-shield water reactivity insertion rates shall be validated annually to verify that the requirements of specification 3.1, item 2, are met.

4. The following shall be performed each day prior to initial reactor operation, except when continuous reactor operations are scheduled, then they shall be performed once each day.

   A. A visual inspection of reactor components.

   B. An operability check or test of safety system channels.

   C. An operability check of the interlock system.

   D. An operability check of the radiation monitors and alarm setpoints.

5. The safety system channels shall be calibrated semiannually.

Bases

Past performance of control rods and control rod drives, and the moderator-shield water fill-and-dump valve system have demonstrated that testing at intervals of six months is adequate to assure compliance with specification 3.1, items 3 and 5, and validation at intervals of twelve months assures compliance with item 2 of specfication 3.1.

Visual inspection of the reactor components, including the control rods, prior to operation is to assure that the components have not been damaged and that the core is in the proper condition. Since redundancy of all safety channels is provided, random failures should not jeopardize the ability of the overall system to perform its required functions. The interlock system for the reactor is designed so that its failure places the system in a safe or non-operating condition. However, to assure that failures in the safety channels and interlock system are detected as soon as possible, frequent surveillance is desirable and thus specified. The frequent checks of the area radiation monitors and their alarm points assures the availability of the system to perform its required functions.

Past experience has indicated that, in conjunction with the daily check, calibration of the safety channels at intervals of six months assures that proper accuracy is maintained.

Surveillance testing intervals shall also contain maximum intervals as set out below to provide operational flexibility and not to reduce frequency. Established frequencies shall be maintained over the long term.

a. Five years (intervals not to exceed six years)
b. Two years (intervals not to exceed two and one-half years)
c. Annual (intervals not to exceed 15 months)
d. Semiannual (intervals not to exceed seven and one-half months)
e. Quarterly (intervals not to exceed four months)
f. Monthly (intervals not to exceed six weeks)
g. Weekly (intervals not to exceed ten days)
h. Daily (must be done during the calendar day

5.2 Reactor Parameters

Applicability

These specifications apply to the verification of control rod reactivity worths, temperature and void coefficients of reactivity, and reactor power level, which are pertinent to the reactor control and transient analysis and to water quality.

Objective

The purpose of these specifications is to assure that the analytical bases are and remain valid and that the reactor is safely operated.

Specifications

1. The following parameters shall be determined during the initial physics measurements of each new reactor core configuration or composition and validated periodically with the stated frequency:

   A. Individual control rod reactivity worths (semiannually).

   B. Temperature and void coefficients of reactivity (annually).

   C. Reactor power calibration (semiannually).

2. The reactor moderator-shield water quality shall be determined monthly.

Bases

Measurements of the above core parameters are made when a new reactor configuration or composition is assembled. Whenever the core configuration or composition is altered, the core parameters are evaluated to assure that they are within the limits of these specifications and the values analyzed in the SAR. During the initial startup test period of the reactor, measurements and determinations of the core parameters will be made for all standard assemblies which are to be utilized in the reactor's operational programs. Verification of these parameters are made periodically to assure that changes have not taken place.

Past experience indicates that monthly measurements of the moderator-shield water quality are adequate to comply with the requirement of specification 3.2, item 3.

5.3 Radiation Monitoring

Applicability

These specifications apply to the surveillance of the radiation monitoring equipment and activities of the facility.

Objective

The purpose of these specifications is to assure the continued validity of radiation protection standards in the facility.

Specifications

1. The area radiation monitors and the portable radiation survey instruments shall be calibrated semiannually.

2. The air in the reactor room shall be sampled and measured for particulate activity monthly.

3. The water in the reactor tank and dump tank shall be sampled and measured for radioactive contaminants monthly.

4. The reactor facility shall be surveyed for radioactive contamination semiannually.

Bases

Experience has demonstrated that calibration of the area radiation monitors and the portable survey instruments semiannually is adequate to assure that significant deterioration in accuracy does not occur.

The specified frequencies for monitoring radioactive contamination in the air and water in the reactor room as well as in the overall reactor facility is based on previous experience.

6.0 DESIGN FEATURES

6.1 Site

The NTR site shall be located on Commonwealth Edison property adjacent to the exclusion area for the Zion Station. An exclusion radius of 450 feet shall define the exclusion area for the site.

6.2 Facility

The reactor facility shall be housed in the Nuclear Training Center building. The main security entrance into the facility shall be in an interior access through the Training Center.

The controlled area of the facility shall be bounded by walls and appropriate doors. The distance from the centerline of the NTR to the edge of the controlled area shall not be less than 13 feet.

6.3 Reactor Room

The reactor room shall be an eight inch concrete block enclosure with approximate floor dimensions of 27 x 27 feet. The height from the ground floor to the ceiling shall be about 27 feet. One exterior side wall shall have an equipment-emergency door and one interior wall shall have a main access door. A 12-foot diameter, aluminum dump tank shall be centered in the reactor room with its top below the floor level. The dump tank shall be surrounded by 18 inches of reinforced concrete and shall rest on a 30-inch thick reinforced concrete pad. A steel structure approximately 10 feet above the floor level and supported on beams at the side walls shall be used to support the control rod drive platform.

6.4 Reactor

6.4.1 Reactor Tank

The aluminum reactor tank shall have a capacity of approximately 7000 gallons of water with no co e components in place. The tank nominal dimensions shall be 8 feet in diameter and 19 feet high. The reactor tank shall fit inside the dump tank in an eccentric fashion and shall hang on a floor level

steel structure which lies over the dump tank. A horizontal support shall be provided near the bottom of the tank. The tank has openings for the inlet water line of the water fill system, the drain line at the bottom of the tank, and the water dump line. An annular support plate shall be provided to support the core structure.

6.4.2 Reactor Core

The aluminum reactor core structure shall be comprised of upper and lower grid plates, core shroud tubes for fuel elements, graphite reflector rods and control rod locations and tie-down rods. The two horizontal grid plates are separated by approximately 44 inches. The bottom plate is approximately 48 inches in diameter and varies in thickness from 2 inches at its edge to a nominal 6 inches in the center portion. The upper grid plate is an elongated hexagon, 1 inch thick. The distance between the flats along the two short axes is approximately 31 inches and along the long axis 34 inches. The center-to-center spacing of the position holes is 3.125 inches. There is a total of ninety-three position holes, five of which contain control rod shroud tubes, seven are experimental hole

positions normally containing fuel element insert adapters and the remainder contain fuel element shroud tubes. The core structure is radially centered in the reactor tank and positioned vertically so that the top of the reactor core is approximately 11.5 feet from the floor level.

6.4.3 Standard Fuel Elements

A standard fuel element shall be composed of 3 concentric tubes of an aluminum-clad fuel alloy that are held together and positioned at each end by aluminum brackets. The fuel alloy (13 w/o U-Al, 93.5 percent U-235 enriched) has a thickness of 0.053 inch and the clad thickness is a nominal 0.036 inch. the aluminum grade used is no. 1100 (2S) or its equivalent.

Each standard element nominally contains 200 gms of U-235 (outer tube - 82.4 gms, middle tube - 66.6 gms and inner tube - 50.9 gms). The length of the fuel alloy is 36 inches and is centered lengthwise in the 42-inch fuel tube. The total fuel element length, including the top and bottom aluminum brackets, is 47-5/16 inches. The outside diameter of the outer, middle and inner tubes is 2.50 inches, 2.06 inches and

1.62 inches, respectively. The nominal water gap between the fuel tubes is 0.094 inch.

When the fuel elements are disassembled and reassembled in any combination of one or two fuel tubes, the resulting element shall be considered a special, non-standard partial fuel element.

6.4.4 Control Rods

The control rods shall consist of a cadmium neutron absorbing section, clad with aluminum and an attached standard fuel element follower. The length of the poison section of the rod is 36 inches. The maximum diameter of the rod is 2.5 inches. The aluminum-clad cadmium tube and the fuel element follower are linked by an axial stainless steel support rod. This linkage extends to and includes a shock absorber piston at its lower end, a stop cap above the piston section, and a magnet carriage armature at its upper end.

The control rods are guided through their respective core position holes by aluminum shroud or guide tubes. The shroud tubes extend above and below the core grid plates. The enclosed lower section with

appropriate holes serves as a water dash pot and the upper section provides a physical stop to prevent further downward motion of the control rod. There are five control rod locatons in the core structure. The control rod in the location lying closest to the center of the structure is the safety rod and those in the four outer locations are shim rods.

6.4.5 Control Rod Drive Assembly

The drive assembly for the control rods shall consist of a magnet carriage, a vertical drive, a position indicator and suitable drive motors. Indications of magnet carriage "up", magnet carriage "down" and control rod "down" are provided. The position indicator gives an indication of the control rod or its magnet carriage position throughout its distance of travel (maximum 41 inches).

The drive assemblies shall be mounted on an overhead platform located approximately ten feet above the floor level and centered over the reactor tank.

6.4.6 Graphite Reflector Rods

A graphite reflector rod shall consist of a type G83 graphite rod, 2.625 inches in diameter, forty-eight inches long with a 0.5 inch hole bored axially the full length of the rod. A 0.5 inch aluminum support rod shall be inserted in the axially bored hole and have support spacers bolted on each end. A ball joint handling adapter shall be screwed and pinned to the top of the aluminum support rod.

The normal core loading shall consist of twenty (20) graphite reflector rods loaded in core positions immediately adjacent to and completely surrounding the core positions occupied by fuel elements and control rods arranged in a normal core configuration.

6.5 Water Handling System

The water handling system shall allow remote filling and emptying of the reactor tank. It shall provide for a water dump in the event that an auxiliary reactor trip is necessary. The filling system shall be controlled by the operator who must satisfy the sequential interlock system

before adding water to the reactor tank. A sump pump shall be provided to add the moderator-shield water from the storage-dump tank into the reactor tank. Slow and fast fill rates of about 35 gpm and 120 gpm shall be possible. A nominal 10-inch valve shall be installed in the dump line and have the capability of emptying the reactor tank on demand of the operator. A valve shall be installed in the bottom drain line of the reactor tank to provide for completely emptying the reactor tank.

6.6 Fuel Storage

When not in use, the fuel elements shall be stored in the reactor room in facilities which will provide adequate isolation and radiation safety. No more than six fuel elements shall be out of the core or the approved storage areas at one time. The following facilities shall be considered approved storage areas:

1. Reactor tank storage tubes, a maximum number of six, shall be located inside the reactor tank and each shall be capable of holding not more than one fuel element. These tubes shall be located approximately nine feet below the reactor tank top and next to the reactor tank wall in a single slab array.

2. The main fuel storage area shall consist of a single slab array, each of which is capable of holding not more than one fuel element. The main fuel storage area shall be located along the west interior wall of the reactor room and provide adequate shielding from stored fuel.

3. When the control rods are unloaded from the core and the fuel element followers are not removed, the rods shall be hung in the control rod storage rack in the reactor room. This rack shall have storage locations with spacing equal to or greater than 5-3/4" in a single line and shall have no more than 9 positions.

7.0 ADMINISTRATIVE CONTROLS

7.1 Organization

The Manager of the Westinghouse Nuclear Training Reactor Facility shall be responsible for the safe operation of the reactor. The Manager reports through the normal line of management as indicated in Table 3 to higher levels of management. The facility organization shall consist of the Manager, a Training Reactor Coordinator, a Reactor Lead Engineer and a staff of reactor operators. All members of the facility operating staff shall be licensed operators or operators in training.

7.2 Personnel Requirements

7.2.1 When the reactor is not secured, the reactor console shall be under the surveillance of a licensed operator.

7.2.2 A licensed senior operator shall be "readily available on call" at all times during reactor operations. A licensed senior operator shall be present at the reactor facility during initial loading and approach to critical and power following each configuration change, recovery from an unplanned shutdown or significant reduction in power, and fuel handling and refueling. However, a licensed senior operator's presence will not be required during recovery from an unplanned shutdown or significant reduction in power when the cause has been clearly established and corrected. The identity of an a method for rapidly contacting the senior operator on duty shall be known to the operator on duty.

7.3 Review and Audit

7.3.1 The Reactor Safeguards Committee (RSC) shall include at least four scientists or engineers who are not in the line organization responsible for reactor

operations (i.e., Nuclear Training Services Group - NTS) and shall represent at least one half of the Committee membership. the minimum qualifications of the RSC members with regard to nuclear experience shall be:

1. Each member must have a minimum of five years industrial experience in nuclear and related fields and must have a minimum of three years of active participation in his nuclear oriented discipline.

2. The experience and knowledge of each member must be applicable to or pertain to the Committee's responsibility to properly review the facility and its operation from the standpoint of safety.

3. Each member must be capable and willing to exercise his individual judgement in regard to all Committee reviews and decisions.

4. The "nuclear oriented discipline" of a minimum of two Committee members must lie in the areas of reactor physics and reactor operations.

7.3.2 The RSC shall meet at least once each six months. A quorum of the RSC shall consist of at least four members and at least half of those present shall be from organizations outside the line organization responsible for reactor operations.

7.3.3 The RSC shall review activities and advise the Manager of NTR and/or whatever echelon of management it feels appropriate on all matters pertaining to the safe operation of the NTR. The reviews shall cover:

1. Proposed experiments, tests and operations not described in the Safety Analysis Report.

2. Proposed changes or modificatioans to the facility not described in the Safety Analysis Report.

3. Proposed changes to the Technical Specifications.

4. Proposed normal operating procedures and emergency procedures, and proposed changes thereto.

5. Facility operation for compliance with internal rules, procedures and regulations, and with license provisions.

6. Performance of facility apparatus and equipment.

7.3.4 Recording and reporting requirements for the RSC shall include:

1. Minutes of each meeting.

2. Special reports on experiments reviewed and facility inspections, including the Committee's findings.

3. Special reports on facility radiation safety practices and records made semiannually.

4. All Committee reports and meeting minutes shall be transmitted through the line management up to and including the Manager, NTS.

7.4 Operating Procedure

7.4.1 Personnel entry into the reactor room shall be subject to the following procedures and conditions:

1. No person shall be allowed in the reactor room unless the reactor is subcritical by at least 3$.

2. No person shall be allowed in the reactor room if remote changes are being made to the reactor which may produce positive reactivity effects.

3. When personnel enter the reactor room to perform activities which may affect the reactor's condition in any way, the following requirements shall be met:

   A. An inter-communication system shall be operational providing voice communications between the reactor room and the control room.

   B. An audible signal of the reactor source multiplication shall be heard in the reactor room.

7.4.2 Approved written operating procedures shall be followed for the following items.

1. Facility security alarms.

2. Routine reactor operations.

3. Radiation safety procedures.

4. Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.

5. Non-routine operations and emergency situations.

6. Fuel handling, storage and changes in the core.

7.4.3 New procedures and changes in the operating procedures shall require review by the RSC and the approval of the Facility Manager.

7.4.4 Temporary changes in the operating procedures which do not change the intent of the original procedures may be made by the Facility Manager. Such changes shall be recorded in the operating records and reported to the RSC.

7.5 <u>Actions to be Taken in the Event of a Reportable Occurrence</u>

7.5.1 Reportable occurrences shall include but not necessarily be limited to the following:

1. A violation of a limiting safety system setting.

2. A violation of a limiting condition for operation.

3. An engineered safety system component malfunction or other component or system malfunction which could render the reactor safety system incapable of performing its intended safety function.

4. An uncontrolled and unanticipated change in reactivity.

5. A personnel action that may cause an unsafe condition in connection with the operation of the reactor.

7.5.2 In the event of a reportable occurrence, reactor operation shall not be resumed until the cause is known and appropriate corrective measures are taken.

7.5.3 The occurrence shall be reported to the NRC in accordance with Section 7.7.1 of the specifications.

7.5.4 A report shall be prepared which shall include an analysis of the causes of the occurrence and recommendations for action to prevent or reduce the probability of recurrence. The report shall be submitted to the RSC for review and shall be maintained as part of the facility records.

7.6 Action to be Taken in the Event a Safety Limit is Exceeded

7.6.1 If a safety limit is exceeded, the reactor shall be secured or otherwise placed in a safe condition and reactor operation shall not be resumed until authorized by the NRC.

7.6.2 An immediate report of the occurrence shall be made to the NRC in accordance with Section 7.7.1 of the specifications.

7.6.3 A complete analysis of the incident together with recommendations for preventing or reducing the probability of recurrence shall be prepared and submitted to the RSC and to the NRC when authorization to resume operation is sought.

7.7 Reporting Requirements

In addition to reports otherwise required by applicable regulations:

7.7.1 The licensee shall inform the Commission of any reportable occurrence or violation of a safety limit. For each occurrence, the licensee shall notify, within 24 hours by telephone or telegraph, the Director of

the appropriate Nuclear Regulatory Commission Regional Compliance Office listed in Appendix D of 10 CFR 20 and shall submit within ten days a report in writing to the Director, Division of Reactor Licensing (hereinafter, "the Director, DRL").

7.7.2 The licensee shall report to the Director, DRL, in writing within 30 days of its observed occurrence any significant change in the transient or accident analyses as described in the Safety Analysis Report, as ammended, or any changes in the facility organization structure.

7.7.3 The licensee shall submit a report within 60 days after criticality of the reactor, in writing to the Director, DRL, upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level or the installation of a new core, describing the measured values of the operating conditions or characteristics of the reactor under the new conditions, including:

1. Total control rod reactivity worth.

2. Reactivity worth of the single control rod of highest reactivity worth.

3. Total and individual reactivity worths of any experiments inserted in the reactor.

4. Minimum shutdown margin both at room and operating temperatures.

7.7.4 The licensee shall submit in writing, to the Director, DRL, an annual operating report within 60 days after the end of each calendar year, providing the following information:

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

2. The energy generated by the reactor and the number of hours the reactor was operational.

3. The number of inadvertent reactor trips, including the reasons therefor.

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

5. A summary description of changes in the facility or procedures, and tests and experiments carried out under the conditions of Section 50.59 of 10 CFR 50..

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

7.8 Records

Records maintained by the licensee shall include, but not necessarily be limited to, the following:

7.8.1 Reactor operating records, including power levels and periods or operation at each power level.

7.8.2 Records of inadvertent reactor trips, including reasons therefor.

7.8.3 Records of experiments, including any unusual events involved in their performance and in their handling.

7.8.4 Records of reportable occurrences.

7.8.5 Records of tests and measurements performed pursuant to the Technical Specifications.

7.8.6 Records of maintenance operations involving substitution or replacement of reactor equipment or components.

7.8.7 Records of fuel inventories and transfers.

7.8.8 Records showing radioactivity released or discharged into the air or water beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

7.8.9 Records of facility contamination and radiation survey results.

7.8.10 Records of radiation exposures for all facility personnel and visitors.

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

Items 1 through 6 shall be retained for at least five years; items 7, 8, 9 and 11 shall be retained for the life of the facility, and item 10 records shall be retained indefinitely or until the Commission authorizes their disposal.

TABLE 1

MINIMUM SAFETY SYSTEM CHANNELS

Reactor Conditions and Ranges	Channels	Minimum Number	Functions
Source Range	Linear or Log Neutron Level	1	High Neutron Level Reactor Trip
(keff < 99)	Linear or Log Neutron Level	1	
Startup and Power	Linear or Log Neutron Level	2	High Neutron Level Reactor Trip
Range	Period	1	Period Trip
(keff $\geq$ 0.99)	Linear or Log Gamma Level	1	High Gamma Level Reactor Trip

TABLE 2

MINIMUM INTERLOCKS

INTERLOCKS	ACTION IF INTERLOCK NOT SATISFIED
Console Master Key "On"	Reactor Trip
Reactor Room Door Closed (a)	Reactor Trip and prevents control rod withdrawal and moderator insertion
Neutron Flux Up (a)	Prevents control rod withdrawal (b) and moderator insertion
Safety Rod Cocked (a)	Prevents control rod withdrawal (b) and moderator insertion
Water Level Up (a)	Prevents control rod withdrawal (b)
Reactor Room Access Key "On"	Prevents control rod withdrawal (b)
Count rate cutout (high and low)	Prevents bank control rod withdrawal (b) and "fast fill" mode of moderator insertion

(a) During maintenance checks, special operations and "console master key on" reactor room entry, these interlocks may be temporarily bypassed using special individual key switches.

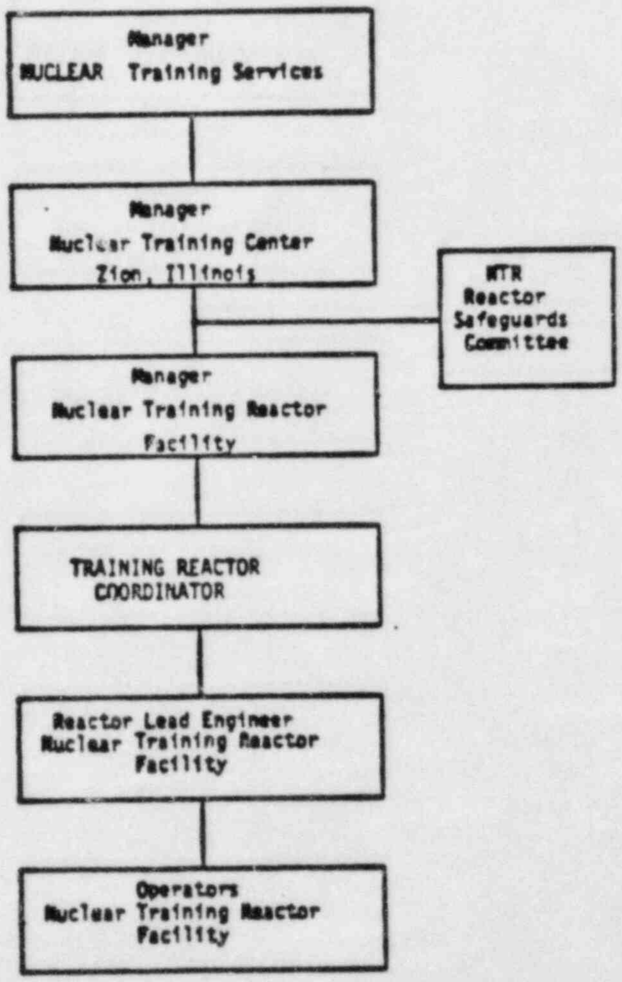
(b) With following exception: safety rod is moved to cocked position prior to any other positive insertion operation.

TABLE 3

ORGANIZATIONAL CHART

(W) Nuclear Training Reactor Facility

Nuclear Training Center, Zion, Illinois



EMERGENCY PLAN

FOR THE

WESTINGHOUSE NUCLEAR TRAINING REACTOR

DOCKET NO. 50-87

WESTINGHOUSE NUCLEAR TRAINING CENTER

NUCLEAR TRAINING REACTOR

<u>EMERGENCY PLAN</u>

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WESTINGHOUSE NUCLEAR TRAINING CENTER

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PREFACE

Title 10, Chapter 1, "Code of Federal Regulations", Part 50.34 (b)(6)(v) states that plans for coping with emergencies are required and the plans shall include the items specified in Appendix E of that part. These plans are to be included in the final safety analysis report as part of the facility license application.

Chapter 5, section 5.7 of the Nuclear Training Reactor (NTR) Final Safety Analysis Report contains Emergency Plan information. That section refers to the NTR Operating Manual. Chapter 6 of the NTR Operating Manual is devoted to Emergency Procedures.

The "NTR Emergency Plan" is to provide reasonable assurance that appropriate measures can and will be taken in the unlikely event of an NTR Facility emergency to protect public health and safety and prevent damage to property. This unlikely event is supported by Chapter 7, "Safety Analysis", of the revised NTR Final Safety Analysis Report (FSAR). Chapter 7, section 7.6 of the FSAR deals with the Maximum Credible Accident. The conclusion, Chapter 7 section 7.5.2.5, points out that no fission products are released and that the total prompt radiation dosage received outside the reactor room would be less than 135 millirem. No off-site exposure will result.

Chapter 7, section 7.4.2 of the FSAR, Loss of Shielding Water, considers radiation conditions if an auxiliary reactor trip is

initiated at full power and both of the gamma sensitive channels malfunction. The analysis states, "In conclusion, the resulting radiation conditions produced by the incident are not extreme regarding personnel exposure and would not effect the general public".

Furthermore, the NTR Facility is considered to be a "Negligible-Risk Research Reactor", as defined in Section 3 of ANS Standard 15.15, "Criteria for the Reactor Safety Systems of Research Reactors". (See Appendix 1 for the definition and emergency planning basis.)

It is therefore assumed in the writing of this "NTR Emergency Plan" that no off-site exposure will exist and evacuation of the Westinghouse Nuclear Training Center (WNTC) and adjacent areas will not be required.

The "NTR Emergency Plan" is referenced by Chapter 6 of the NTR Operating Manual. This plan is an addendum to the Operating Manual; however, to enhance the review and implementation of the plan, it is published under separate cover.

1.0 DESCRIPTION AND USE

The reactor facility is housed in a 3200 square foot enclosure which makes up the south wing of the Westinghouse Nuclear Training Center. The boundaries of the facility are established to provide the proper radiation safety control and facility security. The reactor is installed in a separate reactor room and the reactor core is situated approximately eleven (11) feet below ground level. Adjacent rooms house the support facilities for the reactor including the console room. The reactor is a light water-moderated, graphite reflected and light water shielded, highly enriched uranium-aluminum, low power system. The reactor core, core support structure, moderator-shield water, graphite reflector rods and instrumentation are contained in an open eight (8) foot diameter, aluminum tank. Access into the core is gained only from the top of the reactor tank at ground level.

The Nuclear Training Reactor (NTR) is owned and operated by the Westinghouse Electric Corporation. The Nuclear Training Services Group, Nuclear Services Integration Division, Water Reactor Divisions will be responsible for the management of the reactor facility. The reactor is located in the City of Zion, Illinois, and is part of the Westinghouse Nuclear Training Center. (See Appendix 2 for building layout and area maps.) The Nuclear Training Center (NTC) is a joint venture between Westinghouse and the Commonwealth Edison Company of Illinois (Commonwealth Edison). The NTC building and land is owned by Commonwealth Edison and the internal training and support equipment is owned by Westinghouse. The training programs performed at the NTC and their execution is the responsibility of Westinghouse and the NTC is operated solely by Westinghouse employees.

The primary use of the reactor facility will be in support of nuclear training programs conducted by the Westinghouse Nuclear Services Integration Division. The reactor is utilized to conduct demonstrations and to provide operating experience for (W) customer personnel in the areas of fundamental reactor physics and reactor operations. Other minor activities of the reactor facility include irradiation experiments and reactor instrumentation studies.

1.1 DEFINITIONS

Emergency

A Condition calling for immediate action beyond the scope of normal operating procedures, to avoid an accident or mitigate the consequences of one.

Negligible-Risk Research Reactor

A Research Reactor for which, in the postulated event of the complete failure of the Reactor Safety System coincident with the occurrence of the most adverse Design Basis Event, the radiological consequences would be negligible.

Negligible Radiological Consequences

An exposure/release of radioactivity, in one day due to an accident, in a quantity which would not exceed the limit permitted to be released over a year due to routine operations. Specifically, the consequences could not exceed:

1. The exposure of the whole body* of an individual in an unrestricted area to 0.5 rems of radiation or the exposure of "any other organ" of such an individual to 1.5 rems of radiation; or

2. The exposure of the whole body* of an individual located at an allowed position in a restricted area of the reactor facility to 5 rems of radiation or the exposure of "any other organ" of such an individual to 15 rems of radiation; or

3. The release of radioactive materials in concentrations at a point where a number of the public could be located which, if averaged over a period of 24 hours, would exceed 365 times the limits specified for such materials in 10CFR20, Appendix B, Table II.

* The "whole body" value shall also apply to the active blood-forming organs, gonads, and lenses of eyes.

Unusual Event

Events which indicate a potential degradation of the level of safety at the facility. The situation may or may not have caused damage to the facility, but if there is damage, it does not necessarily require an immediate change to the operating status.

The actual or projected radiological effluents at the site boundary in an unusual event are 10 times the MPC when averaged over 24 hours or 15 mrem whole body accumulated dose in 24 hours.

Alert

Events which invoke actual or potential substantial degradation of the level of safety of the facility. The actual or projected radiological effluents at the site boundary in an alert are 50 times the MPC when averaged over 24 hours or 75 mrem whole body accumulated in 24 hours. Actual or projected radiation levels at the site boundary are 20 mrem/hr for 1 hour whole body or 100 mrem thyroid dose.

NTR Emergency Support Groups

a. The City of Zion Police Department, Zion, Illinois.
b. The City of Zion Fire and Rescue Department, Zion, Illinois.
c. Victory Memorial Hospital, Waukegan, Illinois.
d. EDRA Radiological Emergency Assistance.
e. Illinois Radiological Assistance Team, Springfield, Illinois.
f. Westinghouse Water Reactor Division Medical Director, Monroeville, PA.
g. Manager, Licensing, Safeguards and Safety Administration, Monroeville, PA.
h. Nuclear Energy Systems, Emergency Committee, Monroeville, PA.
i. Westinghouse Gateway Center, Pittsburgh, PA.

Emergency Action Levels

Specific instrument readings, or observations; radiological dose or dose rates; or specific contamination levels of airborne, waterborne, or surface-deposited radioactive

materials that may be used as thresholds for establishing emergency classes and initiating appropriate emergency measures.

Emergency Classes

Emergency classes are classes of accidents grouped by severity level for which predetermined emergency measures should be taken or considered.

Emergency Plan

An emergency plan is a document that provides the basis for actions to cope with an emergency. It outlines the objectives to be met by the emergency procedures and defines the authority and responsibilities to achieve such objectives.

Emergency Planning Zone (EPZ)

Area for which offsite emergency planning is performed to assure that prompt and effective actions can be taken to protect the public in the event of an accident. The EPZ size depends on the distance beyond the site boundary at which the Protective Action Guide (PAG) could be exceeded.

Emergency Procedues

Emergency procedures are documented instructions that detail the implementation actions and methods required to achieve the objectives of the emergency plan.

Offsite

The geographical area that is beyond the site boundary.

Onsite

The geographical area that is within the site boundary.

Operations Boundary

The area within the site boundary such as the reactor building (or the nearest physical personnel barrier in cases where the reactor building is not a principal physical personnel barrier) where the reactor chief administrator has direct authority over all activities. The area within this boundary shall have prearranged evacuation procedures known to personnel frequenting the area.

Protective Action Guides (PAG)

Projected radiological dose or dose commitment values to individuals that warrant protective action following a release of radioactive material. Protective actions would be warranted provided the reduction in individual dose expected to be achieved by carrying out the protective action is not offset by excessive risks to individual safety in taking the protective action. The projected dose does not include the dose that has unavoidably occurred prior to the assessment.

Research Reactor

A device designed to support a self-sustaining neutron chain reaction for research, developmental, educational, training, or experimental purposes, and which may have provisions for production of nonfissile radioisotopes.

Site Boundary

The site boundary is that boundary, not necessarily having restrictive barriers, surrounding the operations boundary wherein the reactor administrator may directly initiate emergency activities. The area within the site boundary may be frequented by people unacquainted with the reactor operations.

WNTC

The Westinghouse Nuclear Training Center located at 505 Shiloh Blvd., Zion, Illinois.

NTR

The Westinghouse Nuclear Training Reactor Facility located as an integral part of WNTC.

Designated Alternate

A person designated to act in the behalf of a principal assigned specific duties in support of this plan. Principals and designated alternates are as follows:

Principal	Designated Alternate
Manager, WNTC	Manager, NTR
Manager, NTR	Reactor Coordinator
Reactor Coordinator	Reactor Lead Engineer
Reactor Lead Engineer	Duty Senior Reactor Operator
Emergency Coordinator	Radiation Safety Coordinator
Radiation Safety Coord.	Asst. Radiation Safety Coord.

1.2 EMERGENCY PRIORITIES

When emergency conditions exist at the NTR Facility, the immediate actions taken and their priorities should be based on the following: First, in all emergencies the saving of human life is paramount. Second, the prevention of the spread of any hazards associated with the accident conditions should be initiated. And third, the destruction of the Facility and its equipment should be prevented.

1.3 NON-STAFF PERSONNEL

It is realized that a special problem exists at the NTR Facility regarding emergencies. At any one time, there can be a large number of non-NTR personnel such as trainees and visitors, present or participating in various programs. Under all circumstances these individuals are under the direct control of a NTR Facility Staff member.

1.4 EMERGENCY PLAN AND PROCEDURES

The Emergency Plan provides the basis for actions to cope with an emergency. It outlines the objectives to be met by the emergency procedures and defines the authority and responsibilities to achieve such objectives.

Emergency Procedures set forth in Chapter 6 of the NTR Operating Manual detail the implementation actions and methods required to achieve the objectives of the Emergency Plan.

The Facility emergency procedures utilize the above emergency action plan principles. It is important that these procedures be complete but uncomplicated in order to minimize confusion.

1.5 WNTC EMERGENCY PLAN

Emergencies which arise may affect the property and occupants in the NTR Facility or possibly in the overall Training Center. The NTR Facility emergency procedures are established to be compatible with "WNTC Emergency Plan". The "WNTC Emergency Plan" coordinates the actions of personnel for the overall Training Center.

1.6 ZION EMERGENCY OFFSITE FACILITY

In case of an actual emergency at the Zion Generating Station requiring activation of the Emergency Offsite Facility, administrative actions will be initiated to limit operations in order to avoid any possible compounding of an emergency.

2.0 EMERGENCY PROCEDURES - GENERAL

2.1 EVACUATION AND ASSEMBLY POINTS

Unless conditions or instructions dictate otherwise, all personnel in the NTR Facility will promptly assembly in the Control Room or Entryway. The NTR Facility Staff personnel aware of the emergency condition will be responsible for the redirection of assembly to Critique Room No. 3 if no contamination is involved. (Refer to Diagram 1 in Appendix 2.) If contamination is involved and conditions are such that no radiation hazard exists, decontamination of personnel will be accomplished prior to redirection to Critique Room No. 3.

2.2 APPROPRIATE ACTION

The senior NTR Facility Staff member present in the Facility will ascertain the nature and extent of the problem and assume responsibility for appropriate action listed in Section 6.0, "Emergency Operations".

2.3 EMERGENCY PROCEDURES

Chapter 6 of the NTR Operating Manual details Emergency Procedures. Appendix 5 of this plan lists those procedures.

2.4 RECOVERY PROCEDURES

Accidents which involve the hazards of radioactive materials may require the evacuation of the Reactor Room or the NTR Facility. The recovery from these accidents will require

personnel to eventually re-enter these areas. The responsibilities and coordination of the recovery follows the procedure set forth in the NTR Operating Manual, Responsibilities.

3.0 ORGANIZATION AND RESPONSIBILITIES

3.1 EMERGENCY ORGANIZATION

The purpose of the emergency organization is to (1) respond to emergency situations and take appropriate action to mitigate the consequences of the emergency, (2) recover from the emergency and (3) maintain emergency preparedness.

A block diagram of the emergency organization is depicted in Diagram 1.

3.2 RESPONSIBILITIES

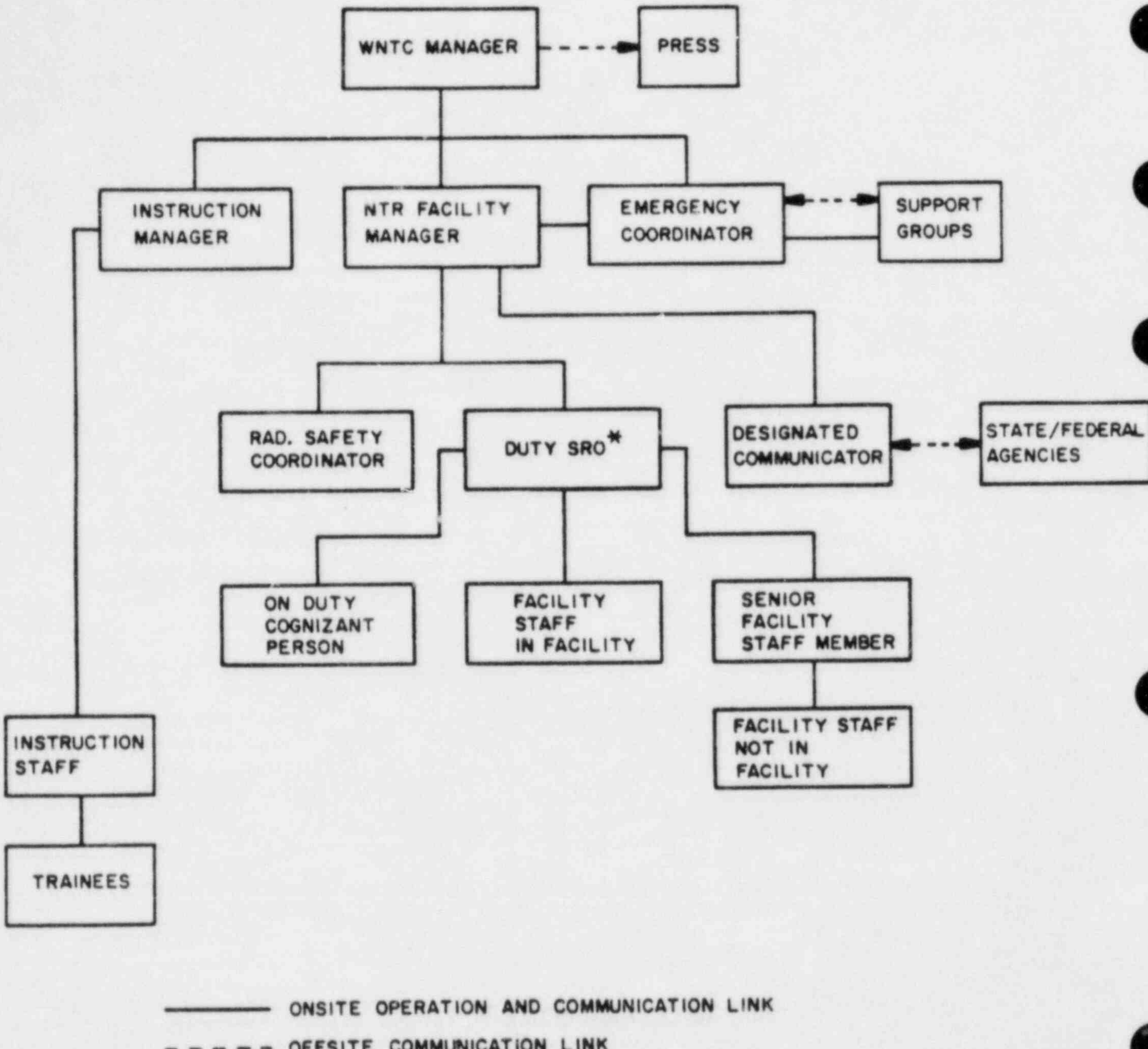
WNTC Manager

The safe operation of the NTR is the direct responsibility of the Manager, WNTC, or a designated alternate. With respect to emergencies, responsibilities include:

a. Provide necessary equipment and personnel to effectively meet the foreseeable emergencies involving the NTR.

b. Provide direction, coordinate, support the emergency organization, and designate Assistance Teams as necessary based on availability of qualified personnel and emergency needs.

c. Provide press releases as required.



*WHEN NO OPERATIONS ARE SCHEDULED, THE NTR LEAD ENGINEER ASSUMES THIS RESPONSIBILITY.

DIAGRAM 1

EMERGENCY ORGANIZATION DIAGRAM

NTR Manager

The NTR Manager or designated alternate is responsible for the NTR staff, contractor, or consultant personnel in the NTR Facility. With respect to emergencies, responsibilities include:

a. Insure that the Emergency Plan and procedures are written and implemented for emergency conditions which might exist within the NTR Facility.

b. Insure that proper emergency equipment and support personnel are available.

c. Advise and supervise the efforts of the Duty SRO, Emergency Coordinator, and Radiation Safety Coordinator.

d. Insure prompt and proper notification of support and regulatory groups.

e. Authorize exposure limits to combat emergencies or conduct recovery effects.

Emergency Coordinator

The Emergency Coordinator is responsible to, and acts for the Manager, WNTC, in implementing and coordinating the Emergency Plan at WNTC. With respect to emergencies, responsibilities include:

a. Conceive and develop plans and procedures for anticipated emergencies and incorporate them into an overall plan for WNTC.

b. Coordinate services of outside groups for emergencies.

c. Review and update the WNTC Emergency Plan as necessary.

d. Conduct training and drills to insure understanding and compliance

Radiation Safety Coordinator

a. The Radiation Safety Coordinator will perform or supervise the performance of the following:

- Radiation surveys
- Ascertain exposure doses of personnel
- Supply portable radiation instruments
- Supply decontamination supplies and equipment
- Escort any radiation injury to the hospital or assign an escort
- Advise hospital personnel of any radiation hazard associated with treatment of personnel injured in the NTR Facility

Instruction Manager

Instruction Managers are responsible for instructors and trainees not designated as NTR Staff members or involved in NTR Training. With respect to emergencies, responsibilities include:

a. Insure that the instructors and trainees are accounted for during an emergency.

b. Provide assistance to the emergency teams as requested by NTR Manager.

Duty SRO

The assigned Duty SRO will take action to combat the emergency, if safe to do so. The Duty SRO will also perform the additional actions listed in Section 6.0 of this Plan, "Emergency Operations", and will assume the position of the Senior Staff member until a more Senior Staff member is present and fully briefed.

On-Duty Cognizant Person

The on-duty cognizant person will assist the SRO in whatever way is necessary to handle the emergency. He or she may be called upon to perform any of the following:

a. Take charge of trainees and visitors to insure their safety.

b. Make radiation surveys.

c. Assist an injured person.

d. Act as a communicator.

e. Interrupt electrical power to the NTR.

f. Obtain needed equipment or supplies.

Senior Facility Staff Member

Upon declaration of an emergency, the Senior Facility Staff Member will proceed directly to the Zion Simulator Room (See

Appendix 2), establish communications with the Duty SRO and direct the actions of Facility staff members not present in the Facility at the time of the emergency.

Facility Staff

a. Facility staff members present in the Facility at the time an emergency is declared will take direction from the Duty SRO.

b. Facility staff members not present in the Facility at the time an emergency is declared will assembly in the Zion Simulator Room and take direction from the Senior Facility staff member present.

3.3 SUPPORT GROUPS

Local Groups

Letters of agreement are maintained with local support groups (see Appendix 3) for the NTR. These support groups provide services of:

1. Law enforcement
2. Fire protection
3. Ambulance service
4. Medical and hospital support
5. Radiological assistance

State/Federal Agencies

The State of Illinois has the statutory responsibility and authority for protecting the health and safety of the public of Illinois.

The Illinois Plan for Radiological Accidents (IPRA) is based upon implementation of five basic functions:

1. Command and coordination
2. Notification and warning
3. Accident assessment
4. Protective actions
5. Parallel operations

The role of the Nuclear Regulatory Commission during an emergency consists of verifying that Emergency Plans and procedures have been implemented, assuring the public health and safety are protected, and conducting investigative activities associated with the incident. The NRC will assist in coordination of federal response resources and provide the NTR, state and local agencies advisory assistance associated with assessing and mitigating hazards to the public.

4.0 COMMUNICATIONS AND ALARMS

4.1 COMMUNICATIONS

Direct verbal communications with other locations in the Training Center and off-site can be made by commercial telephone. Separate telephone lines are available in the NTR Facility. One of these lines passes through the Training Center switchboard and is equipped with an emergency power

supply. A second and independent telephone line is connected directly to the local telephone system.

A Training Center public address system is available through the building. The public address system unit in the NTR Facility has the capability of paging "All Stations" within the Training Center. Take-a-phones and battery-operated bullhorns may also be available for communications.

The emergency and fire alarms are also considered as means of communications from the Facility.

4.2 ALARMS

Area Monitors

The radiation area monitors in the NTR Facility entryway and control room give an audible alarm when tripped This alarm will be heard only in the NTR Facility or immediately outside the main entrance door. The area radiation monitor/criticality monitor in the Reactor Room alarms by sounding a bell. This alarm will generally be heard only in the NTR Facility.

Emergency Alarm

The emergency alarm for the NTR Facility is a horn, similar to an automobile horn. This alarm is sounded by manually actuating one of two switches in the NTR Facility. The horn will be heard throughout the Training Center.

Fire Alarm

The fire alarm is initiated automatically by smoke and heat detectors. The alarm can also be initiated manually from pull stations. The pull station in the NTR Facility is located in the entryway. The alarm sound is a continuous gonging and is heard throughout the Training Center. It also sounds at the local law enforcement agency. The local law enforcement agency, in turn, immediately relays the alarm to the local fire department.

5.0 EMERGENCY FACILITIES AND EQUIPMENT

5.1 EMERGENCY SUPPORT CENTER

The Zion Simulator Room is designated as the primary emergency support center. The simulator room is large enough to stage specific groups of emergency action teams and telephone/public address communications are installed. The simulator room is also located to provide ready access to the number 1 emergency equipment station in the north-south hallway. (See Appendix 2 for location.)

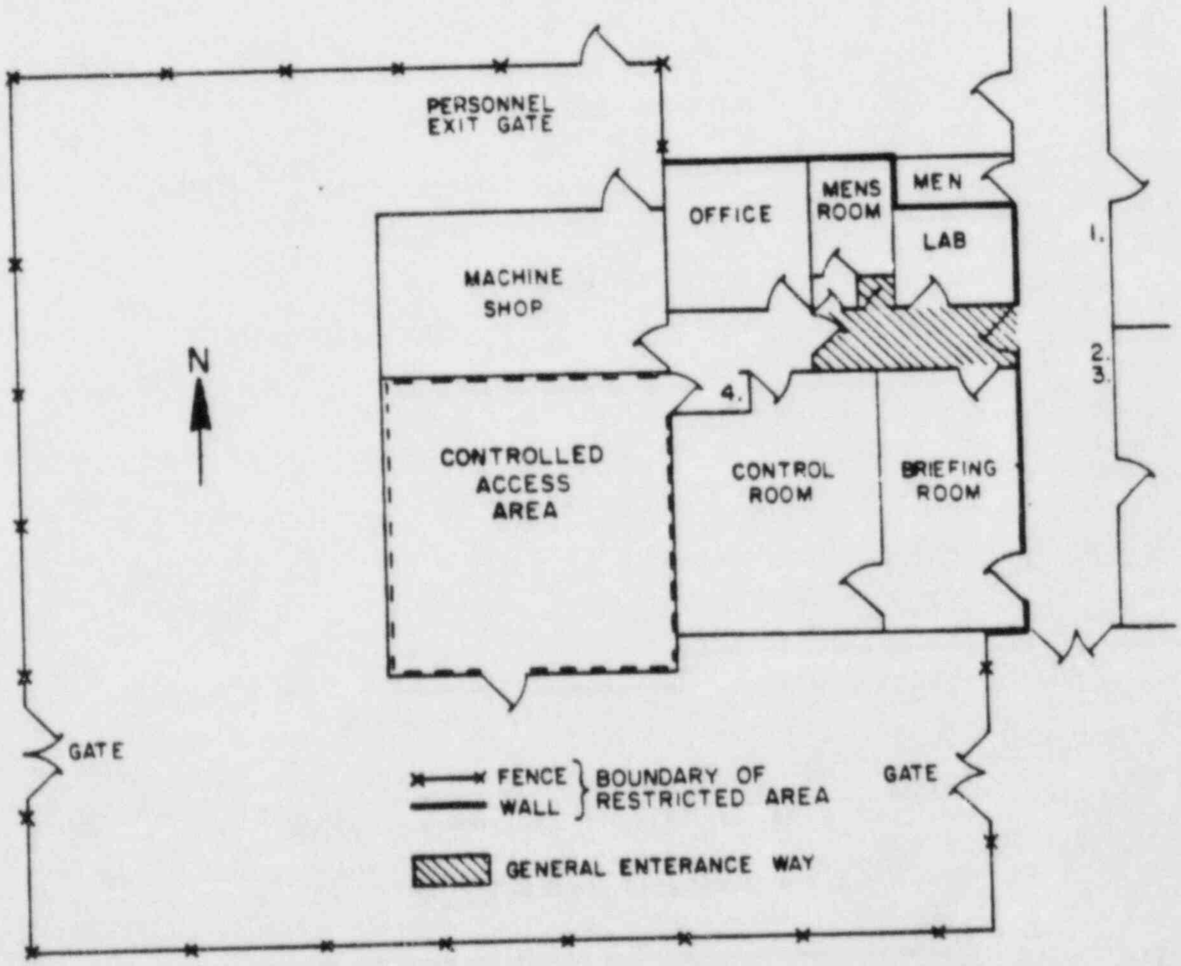
5.2 EMERGENCY EQUIPMENT

This plan specifies equipment and supplies that must be maintained readily available to the emergency response personnel. The operational readiness of equipment and supplies is ensured by quarterly inspection and inventory and use during required drills and exercises. Appendix 2 depicts the location of emergency equipment/supplies location and Appendix 4 lists those equipment/supplies by location. Portable radiation survey monitoring equipment is available inside the NTR Facility and at the Emergency Equipment

Station located in the north-south hallway outside the Fa ility. Diagram 5 shows the specific location of the equipment.

In addition to equipment/supplies listed in Appendix 4, portable radiation survey monitoring equipment is available inside the NTR Facility and at the Emergency Equipment Station located in the north-south hallway outside the Facility. Diagram 5 shows the specific location of the equipment.

DIAGRAM 5



1. Emergency Instruments & TLD's
2. Emergency Equipment
3. Emergency Supplies Locker
4. Emergency Instruments inside the NTR

Table I lists the types, ranges, and uses of portable survey instruments found inside the NTR Facility. Table II lists the types, ranges, and uses of portable survey instruments found at the Emergency Equipment Station in the north-south hallway.

The survey instruments listed are calibrated quarterly and may be replaced by instruments of similar type for maintenance and calibration purposes.

5.3 PERSONNEL MONITORING EQUIPMENT

Radiation doses to personnel on-site during an emergency will be determined by TLD badges and pocket dosimeters. TLD's can be read and reported back by the contractor within 24 hours. Pocket dosimeters can be read and recorded as often as needed. Emergency TLD's are kept at the Emergency Equipment Station located in the north-south hallway, as shown in Diagram 5.

Every reasonable effort shall be made to limit personnel external occupational exposures to those limits specified in the NTR Operating Manual, NTR Permissible Work Limits. If these limits must be exceeded, then personnel exposures shall be controlled so as not to exceed the limits and requirements specified in 10CFR20 limits. Higher adult exposure limits as specified below may be considered only in extreme emergency situations. Prior approval of the NTR Manager must be obtained before exposing personnel to doses in this range.

- Exposure of the whole body - total dose, measured in air: Up to 25 rem.

TABLE I

PORTABLE RADIATION SURVEY MONITORING EQUIPMENT
FOUND INSIDE THE NTR FACILITY

Instrument	Range	Use
Two (2) Eberline E-530 Geiger Counters	0-200 mr/hr	Low level beta-gamma
One (1) Eberline RO-1 Ion Chamber	0-500 r/hr 0-500 mr integrated	Mid-range beta-gamma
One (1) Eberline RO-7 Ion Chamber	0-199.9 R/hr	Mid-range beta-gamma
One (1) Victoreen Radector III Ion Chamber	0.1-100 mr/hr 0.1-1.0 kr/hr	Mid to high range beta-gamma
One (1) Eberline PAC 1SA Scintillation Detector	0-2.0x10^6 cpm	Mid-range alpha
One (1) Eberline PNR-4 BF_3 Proportional Counter	0-5000 mr/hr	High-range neutron

TABLE II

PORTABLE RADIATION SURVEY MONITORING EQUIPMENT FOUND AT THE EMERGENCY EQUIPMENT STATION

Instrument	Range	Use
One (1) Victoreen Radector III Ion Chamber	0-1.0 kr/hr	High level beta-gamma
One (1) Eberline PAC-1SA Scintillation Detector	0-2x10^6 cpm	Alpha
One (1) Eberline E-120 Geiger Counter	0-200 mr/hr	Low level beta-gamma

- Local exposure - any adult. Dose measured in air and additional to whole body dose:
  - Hands and forearms, up to 100 rem
  - Feet and ankles, up to 100 rem

5.4 DECONTAMINATION FACILITIES

Facility/equipment decontamination will be accomplished as directed by the Senior Facility Staff member present utilizing the emergency decontamination kit located in the Emergency Equipment Locker (See Diagram 5).

Personnel decontamination will be accomplished at Victory Memorial Hospital, Waukegan, Illinois. The hospital has full decontamination facilities and personnel trained to perform decontamination. Care and transport of contaminated personnel will be guided by the Emergency Plan for Radiation Casualties (see Appendix 3).

5.5 FIRST AID AND MEDICAL FACILITIES

WNTC maintains a written agreement with Victory Memorial Hospital, Waukegan, Illinois (see Appendix 3).

This service is supported by the hospital's "Nuclear Disaster Plan" and their "Decontamination of Radioactively Contaminated Patient from the WNTC at the Victory Memorial Hospital Plan".

WNTC maintains a written agreement with Zion Emergency Rescue Department to supply ambulance and first aid services as needed for any emergency at WNTC (see Appendix 3).

The written agreements with the above support groups will be renewed in the event that a change in the WNTR Emergency Plan is made that affects the service provided.

5.6 COMMUNICATION EQUIPMENT

See Section 4.1 of this plan.

6.0 EMERGENCY OPERATIONS

6.1 INITIATION OF PLAN

The NTR Emergency Plan may be initiated by the senior NTR Facility staff member present in the Facility whenever any of the following occurs:

a. A radiological or other hazardous materials accident which present a significant and continuing hazard to personnel and requires assistance to bring under control.

b. A fire, explosion, civil disturbance, or similar accident which requires assistance to protect the health and safety of personnel or to limit the damage to facilities or property.

c. Possible damage of nuclear fuel or neutron source and the likelihood of a related release of radioactive material.

d. Any other accident of equivalent magnitude as determined by the senior NTR Facility staff member.

6.2 EMERGENCY ACTION LEVELS

Emergency situations are classified into four major categories: Unusual Event, Alert, Site Area Emergency, and General Emergency.

Section 7.5 of the FSAR presents the description and analysis of the Maximum Credible Accident. Under this condition no off-site release or exposure is expected or postulated. Therefore, of the four emergency classifications only the Unusual Event and Alert are of interest and significance. A definition of these emergency categories is given in Section 1.1 of the Emergency Plan.

6.3 CLASSIFICATIONS OF EMERGENCY CONDITIONS

After all available information pertinent to an emergency condition has been assembled and evaluated, classification of the event shall be made. Table III is a compilation of the emergency action levels for the WNTR. Table III provides descriptions of events, references to the NTR Operating Manual emergency procedures, and conditions which define an unusual event or alert.

TABLE III

TABLE OF WNTR EMERGENCY ACTION LEVELS

EMERGENCY CONDITION/ NTR OPERATIONS MANUAL PARAGRAPH NUMBER	UNUSUAL EVENT	ALERT
1. Fire or smoke - para. 6.4.3	If outside assistance is required	If SNM is threatened by fire or CAA integrity is threatened
2. Bomb threat to the Facility - para. 6.4.4	If it is suspected that a bomb is in the Facility or if a bomb threat is received	If a bomb is discovered in or near the Facility
3. Acts of civil disorder at Facility - para. 6.4.5	If the NTR Facility is threatened	If the SNM is threatened or the integrity of the CAA is broken
4. Radioactive material spill - para. 6.4.6	Actual or projected radiological effluents at the site boundary exceeding 10 MPC when averaged over 24 hrs or 15 mrem whole body accumulated in 24 hrs	Actual or projected radiological effluents at the site boundary exceeding 50 MPC when averaged over 24 hrs or 75 mrem whole body accumulated in 24 hrs or 20 mrem/hr at site boundary or 100 mrem thyroid dose
5. Reactor excursion - para. 6.4.7	If an L.S.S. or L.C.O is violated or if conditions of 4A above are met	If the integrity of the fuel cladding is breached, or if conditions of 4B above are met
6. Accidental Criticality - para. 6.4.8	If conditions of 4A above are met	If conditions of 4B above are met

TABLE 2 (Continued)

EMERGENCY CONDITION/ NTR OPERATIONS MANUAL PARAGRAPH NUMBER	UNUSUAL EVENT	ALERT
7. Human Casualty - para. 6.4.9	Transportation of radioactivity contaminated injured person to hospital or personal exposures greater than limits specified in 10CFR20	
8. Abnormal Occurrence - para. 6.4.10	Any occurrence that threatens the safety of the Facility or of Facility staff members	Any occurrence that threatens the safety of the general public

6.3.a Unusual Event Implementing Procedures

The following implementing procedures will be used for an emergency classified as an unusual event. These procedures are found in Chapter 6 of the NTR Operating Manual, a copy of which is in the Emergency Cabinet in the north-south hallway.

Fire or Smoke	Para. 6.4.3
Bomb Threat to the Facility	Para. 6.4.4
Acts of Civil Disorder - NTR Facility	Para. 6.4.5
Radioactive Material Spill	Para. 6.4.6
Reactor Excursion	Para. 6.4.7
Accidental Criticality	Para. 6.4.8
Human Casualty	Para. 6.4.9
Abnormal Occurrence	Para. 6.4.10

6.3.b Alert Implementing Procedures

The following implementing procedures will be used for an emergency classified as an alert. The classification of an event will depend upon the severity of the situation. These procedures are found in Chapter 6 of the NTR Operating Manual, a copy of which is in the Emergency Cabinet in the north-south hallway.

Fire or Smoke (If SNM is threatened by fire)	Para. 6.4.3
Bomb Threat to the Facility (If bomb is suspected to be in the CAA)	Para. 6.4.4
Radioactive Material Spill (If limits specified in 1.1 of this Plan are likely to occur)	Para. 6.4.6

Reactor Excursion (If limits specified in 1.1 of this Plan are likely to occur)	Para. 6.4.7
Accidental Criticality (If limits specified in 2.5 above are likely to occur)	Para. 6.4.8
Abnormal Occurrence (If limits specified in 1.1 of this Plan are likely to occur)	Para. 6.4.10

6.4 IMMEDIATE ACTIONS

6.4.1 The Duty SRO or senior NTR Facility staff member present will insure the reactor is shutdown and secured.

6.4.2 Unless conditions or instructions dictate otherwise, all personnel in the NTR Facility will promptly assembly in the Control Room or entryway. The NTR Facility staff personnel aware of the the emergency condition will be responsible for the redirection of trainees and visitors to Critique Room No. 3.

The senior NTR Facility staff member present will insure that all personnel in the Facility at the time of the emergency are accounted for by sight count and comparison to the sign-in sheet. Persons unaccounted for will be located and removed from the Facility using methods and equipment as dictated by existing conditions.

In the event of a potential contamination situation, the senior NTR Facility staff present will insure that all personnel evacuating the Facility are surveyed for contamination prior to exiting the Facility. These personnel

found to be contaminated will be directed to critique room no. 3 for further disposition. Those personnel found not to be contaminated will be directed to Classroom No. 3. (See Appendix 2 for locations of classroom no. 3 and critique room no. 3.)

6.4.3 The Duty SRO shall relay information regarding the emergency to the cognizant person and NTR Manager.

6.4.4 The DUTY SRO shall insure appropriate NTR emergency support groups are notified. Notification lists of NTR emergency support groups and respective telephone numbers are posted in the Facility entryway and on the bulletin board outside of NTR door no. 1 in the north-south hallway.

6.4.5 The Duty SRO shall follow the action required by appropriate emergency procedure listed in Appendix 5 and detailed in Chapter 6 of the NTR Operating Manual.

6.4.6 General steps and amplification of specific steps are summarized below:

1. One person will be directed to take any necessary preliminary radiation/contamination surveys. Based on results of surveys, appropriate isolation barrier location and access control of the Facility will be directed by the Duty SRO.

2. One person will be directed to communicate the existing conditions to appropriate persons and agencies using the most effective means of communications available. If the initiation of the emergency plan was the result of a reportable occurrence as described in the NTR Technical Specifications or an incident requiring notification as

described in 10CFR20.403, the NRC shall be immediately notified by telephone. The notification shall be to the Director of Region III Office by the telephone number posted on the NTR bulletin board located in the north-south hallway.

The person designated to communicate with the NRC shall use the Event Notification Worksheet as published in NUREG-0835. Verification of receipt of communication shall be obtained by use of a call back number. A copy of the Event Notification Worksheet is located in the emergency equipment locker in the north-south hallway.

The following occurrences require 24 hour notification as per Section 7 of the NTR Technical Specifications:

- A violation of a limiting safety system setting
- A violation of a limiting condition of operation
- An engineered safety system component malfunction or other component or system malfunction which could render the reactor safety system incapable of performing its intended safety function
- An uncontrolled and unanticipated change in reactivity
- A personnel action that may cause an unsafe condition in connection with the operation of the reactor

Section 7 of Technical Specifications also require immediate (24 hours) notification if a safety limit is exceeded.

The following occurrences require immediate notification as per 10CFR20.403:

0006N

- Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual of 150 rems or more of radiation; or exposure of the feet, ankles, hands, or forearms of any individual to 375 rems or more of radiation
- The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 5,000 times the limits specified for such materials in Appendix B, Table II
- A loss of one working week or more of the operation of any facilities affected
- Damage to property in excess of $200,000.

3. Personnel not necessary to assist in bringing the emergency condition under control will be removed from the NTR Facility when it is safe or expedient to do so.

4. The senior NTR Facility staff member will coordinate recovery efforts with the emergency support groups.

If a radiation casualty is involved, the steps outlined in Appendix 3, "Emergency Plan for Radiation Casualties" shall be performed.

7.0 RECOVERY

Accidents which involve the hazards of radioactive materials may require the evacuation of the Reactor Room or the NTR Facility. The recovery from such accidents will require personnel to eventually re-enter these areas. The responsibilities assigned and coordination of the re-entry/recovery effort will be governed by existing conditions but will follow general principles presented below.

Protective clothing, equipment and supplies (including a copy of this Plan) are stored in the Emergency Equipment Locker, which is located in the north-south hallway. Additional personnel emergency equipment and a decontamination kit are also stored in this area. A second location for emergency equipment is near the Training Center Staff Office area, across from the ladies' restroom.

A radiation field may be entered for investigative purposes or to rescue or safeguard personnel provided appropriate protective clothing, devices and equipment are utilized and the following requirements are followed and limits are not exceeded.

7.1 EXPOSURE LIMITS

a. Every reasonable effort shall be made to limit personnel external occupational exposures to those limits specified in the NTR Operating Manual, NTR Permissible Work Limits.

b. If the limits specified above must be exceeded, then personnel exposures shall be controlled so as not to exceed the limits and requirements specified in 10 CFR 20.

c. Higher adult exposure limits specified below may be considered only in extreme emergency situations. Prior approval of the NTR Site Radiation Officer, the NTR Manager and the Radiation Safety Coordinator must be obtained before exposing personnel to doses in this range.

1) Exposure of the whole body - total dose, measured in air: Up to 25 rems.

2) Local exposure - any adult. Dose measured in air and additional to whole body dose:

a) Hands and forearms, up to 100 rems
b) Feet and ankles, up to 100 rems

7.2 REQUIREMENTS

a. The emergency exposure limits shall not be used except in case of a bonafide emergency involving risk of life or limb, or the destruction of valuable property.

b. Under no circumstances shall an individual be deliberately exposed to an occupational dose in this range twice in a lifetime. "Once in a lifetime" should be interpreted to mean one episode in a lifetime.

c. When possible, respiratory protection and protective clothing should be worn to reduce the hazards of internal and external contamination to levels comparable to those encountered under normal operating conditions.

d. This emergency assignment will be restricted to those NTR Facility Staff members whose NRC-4 Forms are current.

e. Personnel exposures resulting from an emergency situation shall be estimated and recorded.

f. The W WRD Medical Director shall be promptly notified of any exposures exceeding the quarterly limits and of any evidence of significant internal exposure such as contaminated injuries or highly contaminated nasal smears.

g. Personnel exposures in excess of the quarterly limits shall be investigated by management and reported to the NRC and to the individual as required by 10 CFR 20.

h. Individuals who might be suspected of significant internal disposition as determined by nasal smears or other means shall submit a urine sample to the Radiation Safety Coordinator for analysis within twenty-four hours following the incident.

7.3 CONTAMINATION LIMITS

Contamination limits are discussed in the NTR Operating Manual. During a re-entry, it may be necessary to determine the presence of contamination with portable survey instruments. Materials are to be considered contaminated when smears or surface dose rates indicate any of the following conditions.

a. More than 100 c/m beta-gamma, above a background of less than 150 c/m, as measured directly with a portable GM survey instrument.

b. More than 100 c/m beta-gamma smearable (total area or 100 square centimeters, whichever is smaller), as measured

above a background of less than 150 c/m with a portable GM survey instrument.

c. More than 60 c/m alpha as measured directly on a portable survey instrument.

d. More than 60 c/m alpha smearable (total area or per 100 square centimeters, whichever is smaller), as measured with a portable survey instrument.

In addition to the above limits, all materials likely to contain entrapped radioactive materials will require positive pressure, supplied air respiratory equipment to be utilized before welding, flame cutting, grinding, or heating operations will be permitted.

7.4 RESUMPTION OF NORMAL ACTIVITY

Resumption of normal activity in the NTR facility shall occur only when radiation levels are such that the criteria of 10 CFR 20 are met and approval of NTR Manager is granted.

8.0 FACILITY STAFF AND SUPPORT GROUP TRAINING

The proficiency of Facility staff personnel responsible for responding to NTR Facility emergencies is insured by the following means:

1. Assigning personnel to emergency response duties which are similar to those performed as part of their regular work duties

2. Initial and periodic requalification training on contents of the Emergency Plan, implementing procedures, applicable techniques, and use of emergency equipment

3. Participation in exercises and drills designed to sharpen those skills which they are expected to use during an emergency

The training program for emergency response personnel allows each member to meet the following objectives:

- Know the objectives of the Emergency Plan;
- Understand the Emergency Action Level Classification System;
- Display an adequate knowledge of responsibilities and duties as set forth in the Emergency Plan;
- Know the persons with whom they may interface while performing Emergency plan functions; and
- Display a functional knowledge of the documents (e.g., procedures) necessary to fulfill their role in the Emergency Plan.

Emergency Plan support organizations personnel receive training provided by individual organization management. Each support organization maintains proficiency in emergency response tasks by day to day involvement in specific areas of responsibilities.

9.0 DRILLS

The adequacy and implementation of the Plan will be verified annually by conducting either a full-scale or small-scale exercise. The full-scale exercise shall be conducted every

three years. A small-scale exercise shall be conducted during each intervening year.

A full-scale exercise will consist of the following general steps:

a. Contact the members of the Emergency Support Group listed in Addendum 6-1 to Chapter 6 of the NTR Operating Manual. Advise them of the emergency exercise in progress and request confirmation of their ability to respond.

b. Personnel in the WNTC are to respond according to the WNTC Emergency Plan.

A small-scale exercise will consist of the following general steps:

a. Contact the Zion Emergency Rescue Department, Hospital, and the office of the W WRD Vice-President. Advise them of the drill and request confirmation of their ability to respond at that time.

b. Personnel in the WNTC are to respond according to the WNTC Emergency Plan.

Following the conduct of a large or small scale exercise, the Emergency Coordinators shall conduct a critique of the drill and prepare a written report which will document the adequacy, errors, and needed improvements in the Plan.

The report will be presented for the NTR Facility Manager. A copy of the report will then be forwarded to the WNTC Manager and the RSC.

Action planned and taken to correct or improve the Plan and its implementation will also be recorded by the Emergency Coordinator.

The reports will be maintained for a period of at least six years.

10.0 REVIEW OF THE NTR EMERGENCY PLAN

The Reactor Safeguards Committee is responsible for the review of the NTR Emergency Plan. This review will be documented and placed on file.

The review of the NTR Emergency Plan shall be completed at two year intervals.

WESTINGHOUSE NUCLEAR TRAINING CENTER

NUCLEAR TRAINING REACTOR

EMERGENCY PLAN

APPENDIX 1

EMERGENCY PLAN PLANNING BASIS

Negligible-Risk Research Reactor. A Research Reactor for which, in the postulated event of the complete failure of the Reactor Safety System coincident with the occurrence of the most adverse Design Basis Event, the radiological consequences would be negligible. Negligible radiological consequences are taken to be an exposure/release of radioactivity, in one day due to an accident, in a quantity which would not exceed the limit permitted to be released over a year due to routine operations. Specifically, the consequences could not exceed:

1. The exposure of the whole body* of an individual in an unrestricted area to 0.5 rems of radiation or the exposure of "any other organ" of such an individual to 1.5 rems of radiation; or

2. The exposure of the whole body* of an individual located at an allowed position in a restricted area of the reactor facility to 5 rems of radiation or the exposure of "any other organ" of such an individual to 15 rems of radiation; or

---

*The "whole body" value shall also apply to the active blood-forming organs, gonads, fetuses and lens of eyes.

3. The release of radioactive materials in concentrations at a point where a number of the public could be located which, if averaged over a period of 24 hours, would exceed 365 times the limits specified for such materials in 10CFR20, Appendix B, "Concentrations in Air and Water Above Natural Background", Table II.

WESTINGHOUSE NUCLEAR TRAINING CENTER

NUCLEAR TRAINING REACTOR

EMERGENCY PLAN

APPENDIX 2

GENERAL BUILDING LAYOUT PLANS AND AREA MAPS

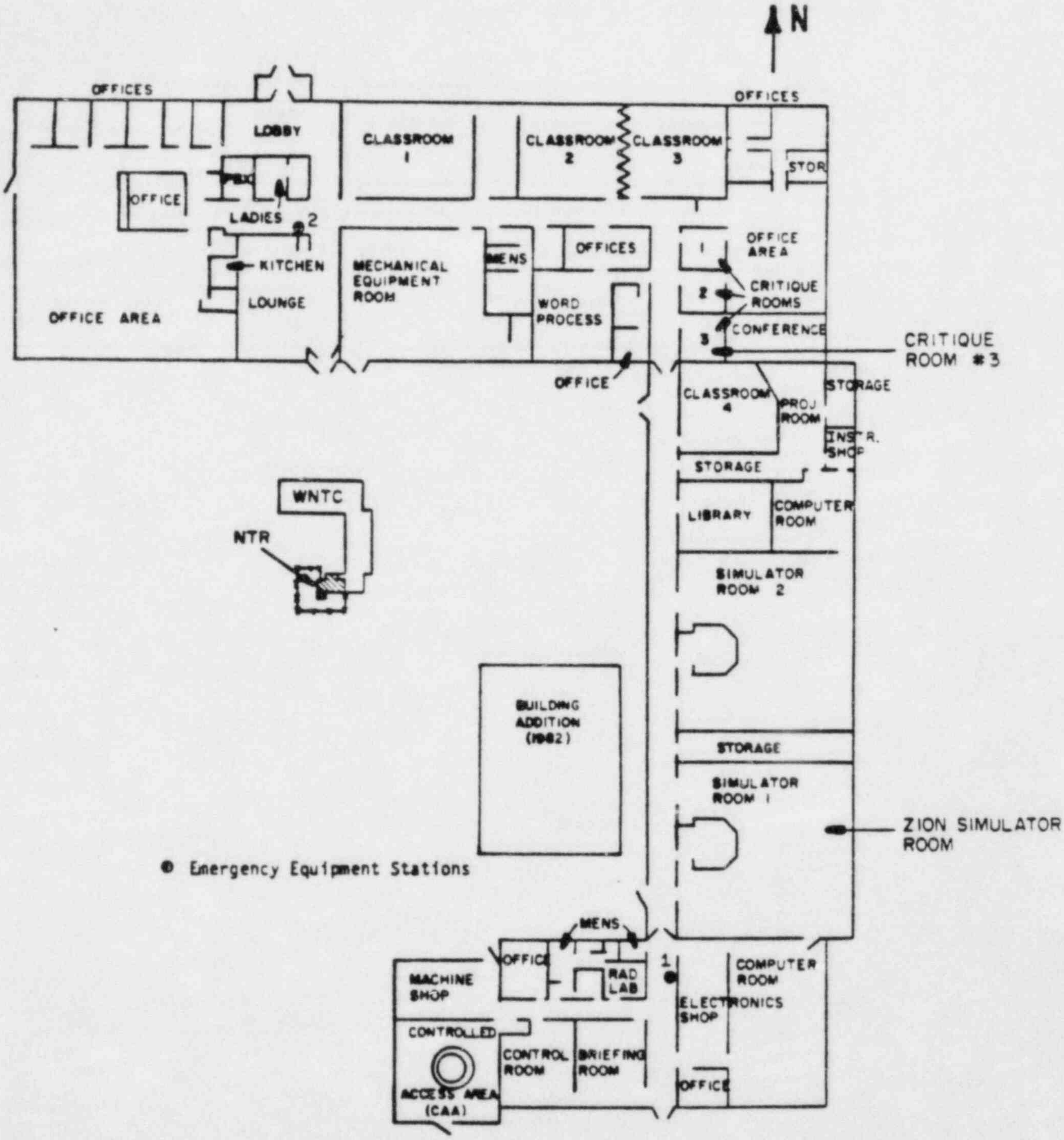


DIAGRAM NO. 2

WNTC AND NTR FACILITY BUILDING

GENERAL LAYOUT

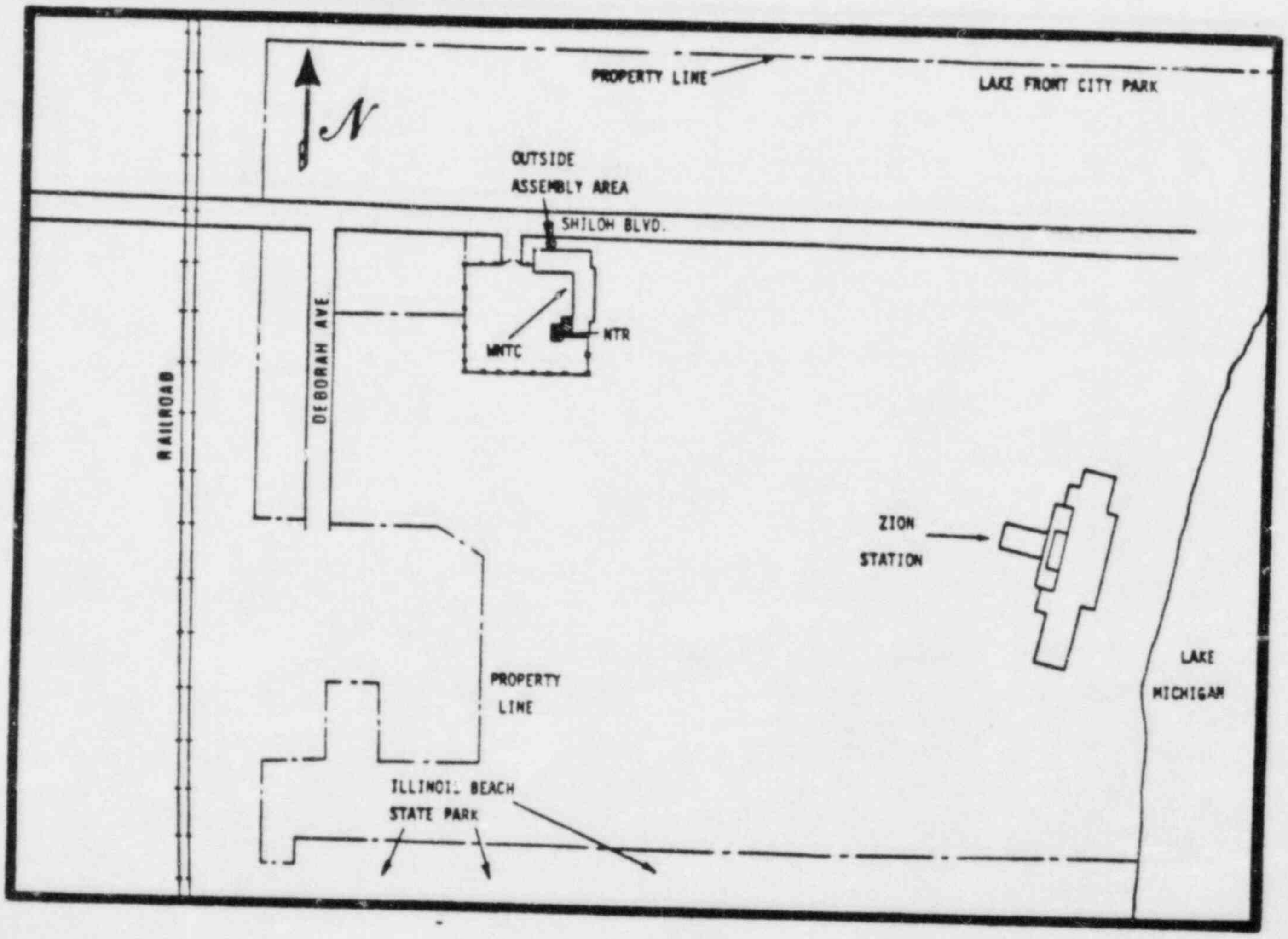


DIAGRAM NO.3

ZION SITE MAP

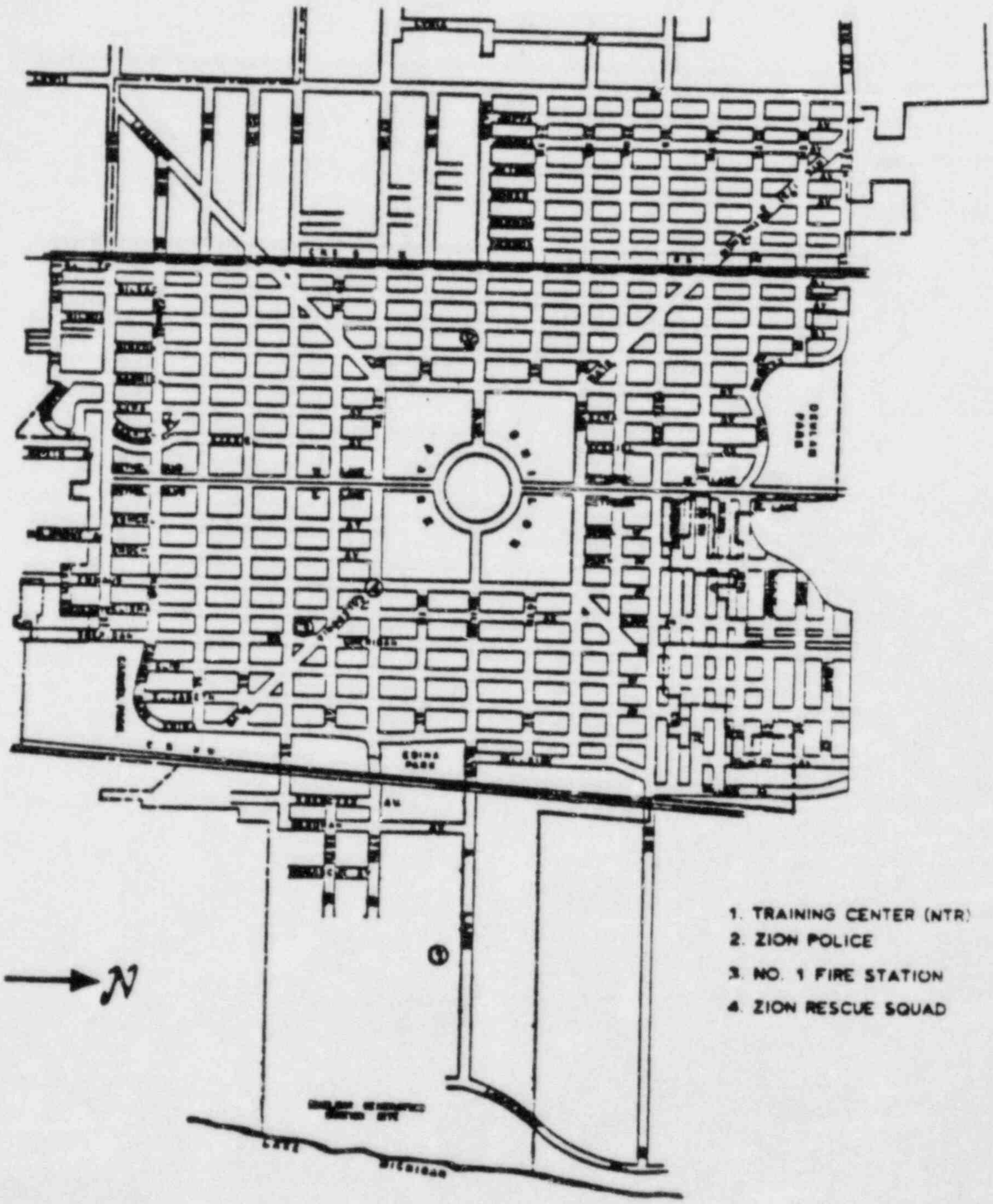

N
1. TRAINING CENTER (NTR)
2. ZION POLICE
3. NO. 1 FIRE STATION
4. ZION RESCUE SQUAD

DIAGRAM 3A

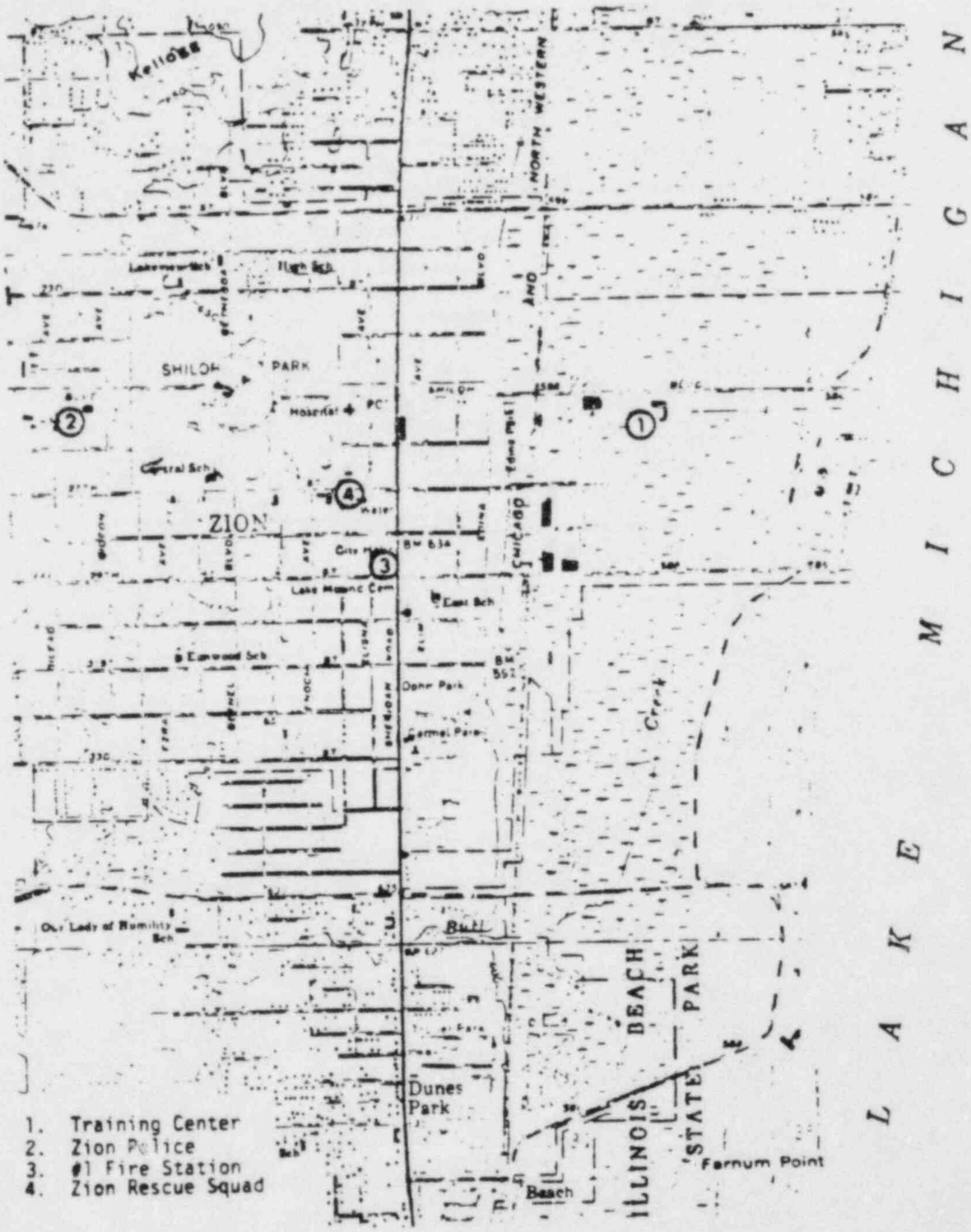


DIAGRAM NO. 4

MAP OF ZION

WESTINGHOUSE NUCLEAR TRAINING CENTER

NUCLEAR TRAINING REACTOR

EMERGENCY PLAN

APPENDIX 3

AGREEMENT LETTERS

AND

EMERGENCY PLAN FOR RADIATION CASUALTIES

victory memorial hospital

one people. one purpose. your health 1324 north sheridan road waukegan, illinois 60085 312-688-3000

June 30, 1983

Mr. F. E. Ellis
NTR Emergency Coordinator
Nuclear Training Center
505 Shiloh Blvd.
Zion, IL 60099

Dear Mr. Ellis:

Please let this letter serve as notice that the hospital has received a copy of the "Westinghouse Nuclear Training Center Reactor Emergency Plan" dated November 2, 1982, with changes and telephone numbers as indicated. The hospital will continue to provide support as stated in the plan. We do not anticipate any changes in the near future.

Enclosed please find a copy of the RMC Decontamination Procedure Manual and the organization chart you requested. Please be advised that RMC is presently revising our plan to include the latest remodeling of the decontamination chamber. As soon as we receive a final copy from RMC I will be sure you too will have a copy for your files.

If you have any further questions or comments concerning this matter please feel free to contact me.

Thank you very much.

Yours truly,

Timothy D. Enright

Timothy D. Enright
Asst. Vice-President/Facilities Management

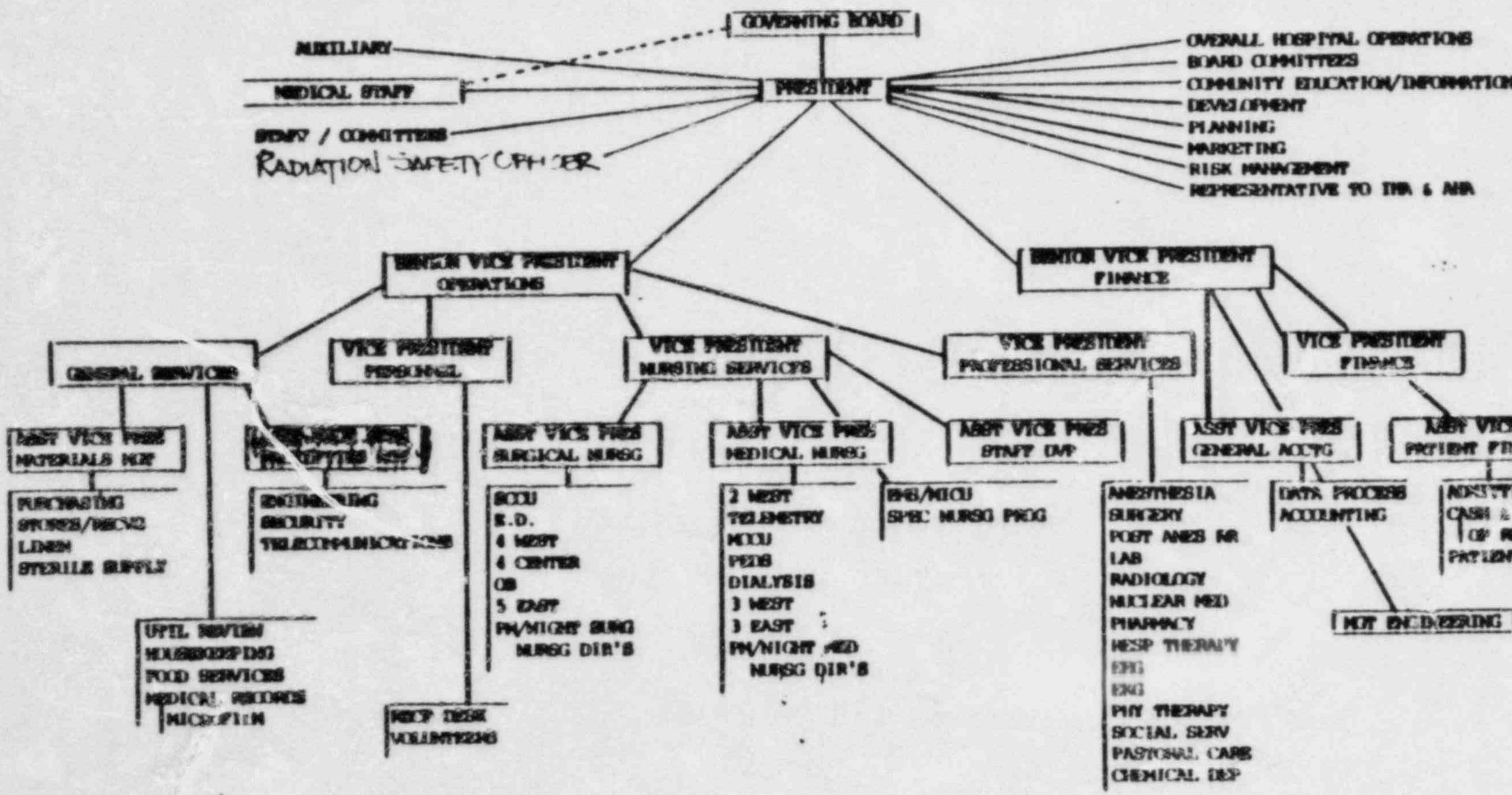
TDE:ta

enclosures

cc: D. Wasson

0006N

VICTORY MEMORIAL HOSPITAL



Donald Wasson
PRESIDENT

July, 1982
DATE

City of Zion Illinois
FIRE & RESCUE DEPARTMENT
2828 SHERIDAN ROAD
ZION, ILLINOIS 60099
(312) 746-3366

ANDREW W. NEARGARDER
CHIEF

June 28, 1983

Mr. F.E. Ellis
Safety and Security Coordinator
Westinghouse Nuclear Training Center
505 Shiloh Boulevard
Zion, Illinois 60099

Dear Mr. Ellis:

I, as Chief of the City of Zion Fire and Rescue Department, commit our full support to your facility in the event of any emergency. We have reviewed the "Westinghouse Nuclear Training Center Emergency Plan for Nuclear Training Reactor" dated November 2, 1982. We will provide support and assistance as outlined in the above stated plan.

I will also provide personnel for preplanning, training, and public education with your personnel relative to fire and rescue service, if requested. I am available for any cooperative effort which may benefit your facility and our organization.

Sincerely,

Andrew W. Neargarder

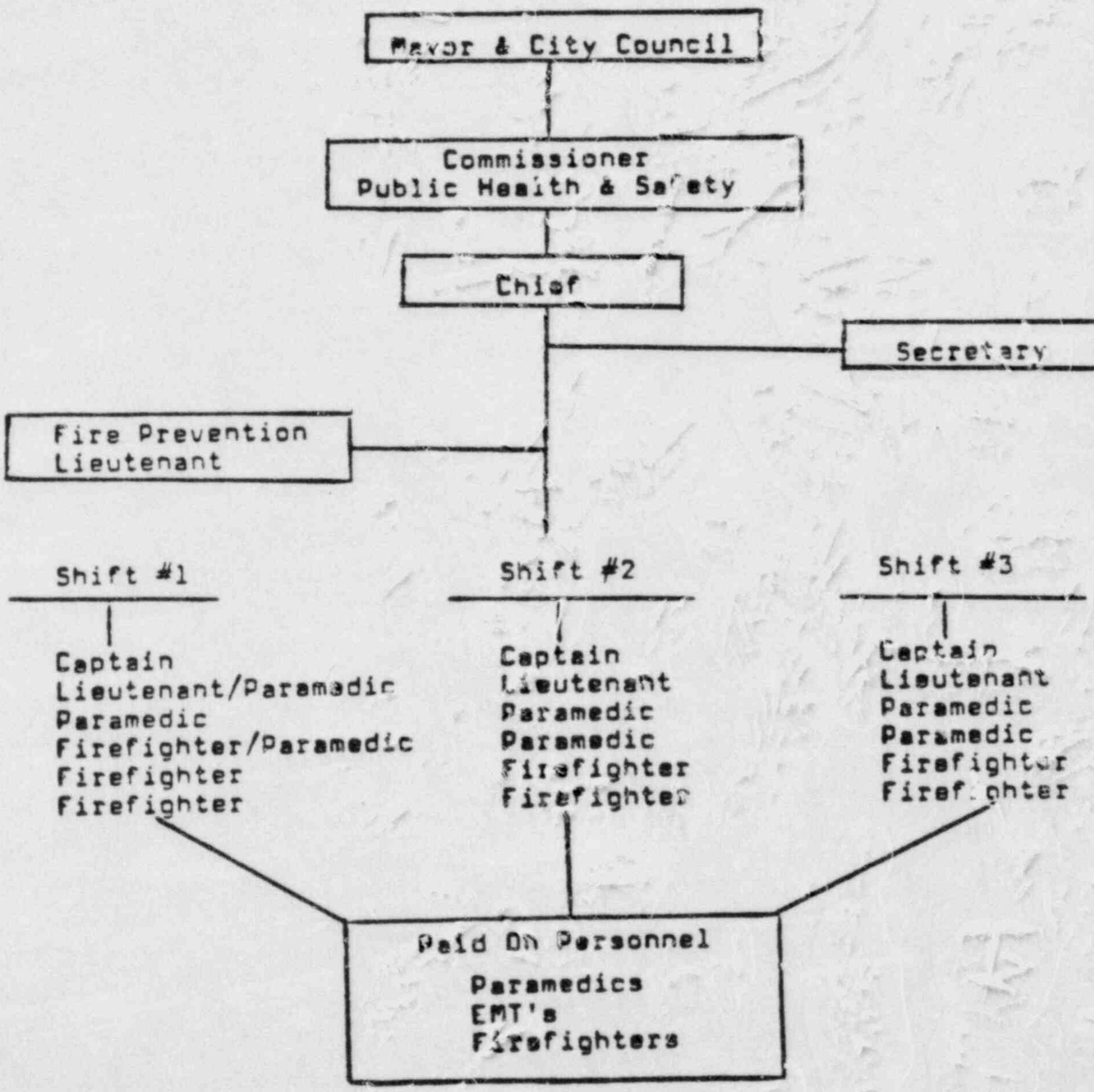
Andrew W. Neargarder
Chief

AWN:ar

cc: Commissioner Booth, Public Health & Safety
Mayor Howard P. Everline

City of Zion
Fire and Rescue Department

Organizational Chart



<u>EMERGENCY PLAN FOR RADIATION CASUALTIES</u>

1. Remove the individual from the radiation source.

2. Minimize the spread of possible contamination.

3. Extend aid to the victim(s).

4. Call ambulance (872-4505) and state that the accident involves radio-activity.

5. Call Victory Memorial Hospital (688-3000) and provide the following information:
   a. Number of injuries
   b. Number of contaminated injuries
   c. Special precaution for the hospital staff

6. Direct the ambulance to Victory Memorial Hospital's Radiation Emergency Area.

7. A WNTC Staff member will accompany the injured to the hospital with the radiation casualty packet.

8. Call Victory Memorial Hospital when the ambulance leaves WNTC (688-3000).

WESTINGHOUSE NUCLEAR TRAINING CENTER

NUCLEAR TRAINING REACTOR

EMERGENCY PLAN

APPENDIX 4

EMERGENCY EQUIPMENT LISTING AND LOCATION

EMERGENCY EQUIPMENT AVAILABLE AT

LOCATION 1

(SEE BUILDING GENERAL LAYOUT APPENDIX 2)

EMERGENCY CABINETS CONTENTS

Blankets
Air Pack W/2 spare tanks
Plastic bags
Step-off pads
Rubber gloves
Poly gloves
Cotton gloves
Plastic Bottles
Masking Tape
Talc Power
Up-to-date copy of the NTR Operating Manual
Radiation Signs and labels, assorted
Smear discs and envelopes
Respirators full face and half face w/spare filters
Disposable protective clothing
Cloth protective clothing
Plastic protective clothing
Alconox detergent
Radiation tape
Radiation rope
Asbestos Gloves
Pens and Pencils

Radiation Casualty Package (Contains)

Plastic gloves
Plastic sheet
Plastic bags
6 tags - Notice Save All Body Waste
6 tags - blank
6 tags - Danger Radiation Hazard
6 tags - Caution Contamination
4 signs - Caution Radiation Area
4 Signs - Caution Contamination Area
4 Stickers - Contaminated Waste
4 Stickers - Contaminated Material
2 Stickers - Radiation Tape
Pencil - pen - tablet

Additional Equipment at Location

Resuscitator kit
Emergency oxygen kit
Scott Air Pac
Stretcher
Bull horn
Fire extinguisher
Fire blanket

Emergency lantern
Low range beta/gamma detector
High range beta/gamma detector
Emergency TLD's
Emergency decontamination kit

EMERGENCY EQUIPMENT AVAILABLE AT LOCATION 2

(SEE BUILDING GENERAL LAYOUT APPENDIX 2)

Resuscitator kit
Emergency oxygen kit
Stretcher
Bull horn
Fire extinguisher
Fire blanket
Emergency lantern
Up-to-date copy of NTR operating manual

WESTINGHOUSE NUCLEAR TRAINING CENTER
NUCLEAR TRAINING REACTOR
EMERGENCY PLAN

APPENDIX 5

LIST OF WRITTEN PROCEDURES THAT IMPLEMENT THE PLAN

PROCEDURES	NTR OPERATING MANUAL REFERENCES
1. Emergency alarm response procedure	Para. 6.4.2
2. Fire or smoke procedure	Para. 6.4.3
3. Bomb threat procedure	Para. 6.4.4
4. Act of civil disorder procedure	Para. 6.4.5
5. Radioactive material spill procedure	Para. 6.4.6
6. Reactor excursion procedure	Para. 6.4.7
7. Accidental criticality procedure	Para. 6.4.8
8. Human casualty procedure	Para. 6.4.9
9. Abnormal occurrance procedure	Para. 6.4.10

FSAR

APPENDIX C

FACILITY LICENSE NO. R-119

OPERATOR REQUALIFICATION PROGRAM

FOR THE

WESTINGHOUSE NUCLEAR TRAINING REACTOR

DOCKET NO, 50-87

OPERATOR REQUALIFICATION PROGRAM

Revision Record

Revision	Date	Page	Comments
Original	12/30/77		NRC approval per P. F. Collins letter dated 12/2/77
Revision 1	12/18/81	-	Cover page added
		3	References
		10	Addition of 6 different reactivity manipulations
		10	Revised to reflect 5 control rod N-24S core
		11	3.4 title changed
		14-15	Incorporated appendix into text
		19-22	Added justifications for significant deviation from 10CFR55, appendix A

WESTINGHOUSE NUCLEAR TRAINING REACTOR

OPERATOR REQUALIFICATION PROGRAM

Table of Contents

1.0 Administration
  1.1 Responsibility
  1.2 Definitions

2.0 Schedule

3.0 Content
  3.1 Written Examination
  3.2 Reactor Operations
  3.3. Reactivity Manipulations
  3.4 Document Index

4.0 Absence from Authorized Activities
  4.1 Requirement
  4.2 Applicability

5.0 Exemptions

6.0 Records

7.0 Justification for Significant Deviations from 10CFR55, Appendix A

List of References

Reference A - NTR Training Manual (Customer Training Lessons), June 1981
Reference B - NTR Operating Manual, June 1981
Reference C - 10CFR55, "Operators' Licenses", August 1, 1980
Reference D - Standard ANSI/ANS 15.4 - 1977 "Standards for Selection and Training of Personnel for Research Reactors.

1.0 ADMINISTRATION

1.1 Responsibility

The Nuclear Training Reactor (NTR) requalification program is administered by the NTR Facility Manager, who may designate a licensed staff member to supervise the details of administration.

1.2 Definitions

1.2.1 Individuals participating in the program are persons licensed by the U. S. Nuclear Regulatory Commission to supervise the manipulation of the controls of the Nuclear Training Reactor. These persons are called Senior Reactor Operators (SRO's).

1.2.2 The Duty SRO is responsible for the NTR Facility during the designated shift assignment. The Duty SRO is a licensed Senior Reactor Operator.

2.0 SCHEDULE

The first program shall be scheduled to run from January 1, 1982, through December 31, 1983. Successive programs shall begin on January 1 of each even-numbered year and shall run through December 31 of the next odd-numbered year.

3.0 CONTENT

3.1 Written Examination

3.1.1 Schedule

Early in each two-year requalification program, a written examination will be administered to each licensed SRO except as noted in Section 5.0.

3.1.2 Content

The written examination shall consist of seven parts, as follows. All parts should normally have approximately the same number of questions or points. The examination should normally be designed so as to require an average of about four hours to complete.

A. Nuclear Theory and Principles of Reactor Operation

B. Design and Operating Characteristics of the NTR

C. Instrumentation and Control Systems

D. Reactor Trip, Interlock and Alarm Systems

E. Normal and Emergency Procedures

F. Radiation Control and Safety

G. Facility License, Technical Specifications and Bases

3.1.3 Evaluation and Additional Training

3.1.3.1 An individual who achieves an overall score of 80 percent or more on the written examination successfully completes the requalification written examination requirement for that two-year requalification program.

3.1.3.2 An individual who achieves an overall score of less than 80 percent but greater than or equal to 70 percent on the written examination requires additional training. The nature and content of such additional training shall be specified by the NTR Facility Manager of a designee, and shall be completed by the end of the two-year requalification program. The additional training may take any one or more of the following forms.

A. Lesson presentation or attendance
B. Self-study
C. On-the-job training
D. Tutoring

3.1.3.3 An individual who achieves an overall score of less than 70 percent on the written examination requires additional training. The nature and content of the additional training shall be specified by the NTR Facility Manager or a designee, and shall be completed within four months of the grading of the written examination. Within one month of the grading of the written examination, the NTR Facility Manager shall conduct an evaluation to determine if the deficiencies uncovered are such that the individual should not be designated a Duty SRO until after completion of the additional training. The evaluation shall take into account the individual's past performance record, past evaluations, past test scores, and current deficiencies. A special oral examination may also be given to aid in the evaluation. The additional training may take any one or more of the forms listed in Section 3.1.3.2

3.2.3.4 Regardless of the score, if an individual's test record indicates a deficiency in a critical area that affects safety, a training program shall be administered to promptly correct the critical deficiency.

3.2 REACTOR OPERATIONS

3.2.1 Operational Examination

3.2.1.1 Twice during each two-year requalification program, at approximately annual intervals, an operational examination will be administered to each licensed SRO, except as noted in Section 5.0. The duration of the operational examination should be about one hour if the individual manipulates the controls or about two hours if it is an evaluation of the conduction of a training lesson. The operational examination shall be conducted by the NTR Facility Manager or a designee and may take either or both of the following forms.

A. Evaluation of the individual conducting a training lesson utilizing the NTR.

B. Manipulation of the reactor controls by the individual in performing one or more training lessons or surveillance tests.

3.2.2 Evaluation and Additional Training

3.2.2.1 The operational examination shall evaluate and document that the individual has the knowledge, competence and dexterity to safely operate the reactor and to take proper action in response to situations that may arise.

3.2.2.2 Additional training in any one or more of the forms prescribed in section 3.2.3.2 shall be provided to correct any weakness or deficiency uncovered. The nature and content of such additional training shall be specified by the NTR Facility Manager or a designee, and shall be completed prior to the next operational exam.

3.2.2.3 If any weakness or deficiency is uncovered in a critical area that affects safety, then the NTR Facility Manager shall promptly conduct an evaluation to determine if the weakness or deficiency is such that the individual should not be assigned to licensed functions until after completion of additional training. The evaluation shall take into account the individual's past performance record, past

evaluations, past test scores and current deficiencies. A special oral, written or operational exam may also be given to aid in the evaluation.

3.3 Reactivity Manipulations

During each two-year requalification program, each licensed SRO shall perform or supervise the performance of at least ten reactivity manipulations which include six different reactivity manipulations from the following list. The performance or supervision should be spread out over the course of the two years.

3.3.1 Reactor fuel loading or unloading where 13 or more elements are handled.

3.3.2 Reactor startup from keff $\leq$ .99 to supercritical and then critical.

3.3.3 Reactor shutdown from critical or supercritical to keff $\leq$ .99.

3.3.4 Individual rod or shim rod bank calibration at two or more data points by positive period or reactivity computer measurements.

3.3.5 Control rod trading while maintaining the reactor critical or near critical where one rod moves at least 2.5 turns from its initial to its final position.

3.3.6 A subcritical or supercritical run from an initially critical condition in order to demonstrate delayed neutron effects.

3.3.7 Differential moderator worth measurement at two or more data points.

3.3.8 Intercalibration and linearity check of the nuclear instruments at two or more power levels.

3.3.9 A programmed power level change from one critical condition to another, where the difference between initial and final power levels is at least 50 percent of the lower power level.

3.4 DOCUMENT INDEX

3.4.1 Emergency Procedures

Each licensed SRO shall review the emergency procedures and the emergency plan twice during each two-year requalification program, at approximately

annual intervals. Any one of the following shall constitute satisfactory completion of this requirement.

3.4.1.1 Reading the documents.

3.4.1.2 Presenting or attending a lesson on emergency procedures for Cognizant Persons or unlicensed WNTC Managers, or an equivalent lesson.

3.4.1.3 Participating in rewriting the emergency procedures or emergency plan.

3.4.2 Change Notices

Each licensed SRO shall perform the following, which may occur at irregular times during the course of any two-year requalification program.

3.4.2.1 Read each new or revised Standing Order in a timely manner.

3.4.2.2 Become aware of changes to the Technical Specifications, to the design of the NTR Facility, or to other significant items in a timely manner by reading routed documents or attending or presenting briefing sessions.

4.0 ABSENCE FROM AUTHORIZED ACTIVITIES

This section is intended to meet the requirements of Section 55.31.e of reference C, and section 6.7 of reference D.

4.1 REQUIREMENT

4.1.1 Any licensed SRO not actively performing licensed functions for a period in excess of four months shall demonstrate to the NTR Facility Manager or a designee that his/her knowledge and understanding of the operation and administration of the NTR Facility are satisfactory before returning to licensed duties. This shall be accomplished through any one or more of the following means.

4.1.1.1 Interview and evaluation

4.1.1.2 Written examination

4.1.1.3 Operations examination

4.1.2 Any deficiencies uncovered shall be corrected by additional training in any one or more of the forms prescribed in section 3.1.3.2. The nature and content of such additional training shall be specified by the

NTR Facility Manager or a designee, and shall be completed before the individual returns to licensed duties.

4.2 APPLICABILITY

4.2.1 Due to the nature of their positions, the NTR Facility Manager, the Training Reactor Coordinator and the Reactor Lead Engineer continually perform licensed functions unless absent from the Nuclear Training Center in excess of four months due to such circumstances as illness, accident or prolonged off-site assignment.

4.2.2 Other licensed SRO's must perform or supervise the performance of licensed operator functions every four months in order to be actively engaged in licensed functions. Listed below are functions that are deemed to actively engage the licensed operators. The operator may actually perform the tasks or supervise the performance if assigned as the Duty SRO or Instructor.

1. Open the NTR Facility.
2. Close the NTR Facility.

3. Perform the tasks and complete the Reactor Daily Checklist.

4. Perform the tasks and complete the Reactor Daily Operation Checklist.

5. Instructing or performing any authorized NTR experiments (lessons).

6. Instructing or performing any of the reactivity manipulations listed in section 3.3 of the Requalification Program.

7. Performing or assisting with surveillance testing or maintenance as listed in section 5.5.2 through 5.5.7 of the NTR Operating Manual.

8. Writing or evaluating NTR Facility procedures.

9. Writing or evaluating NTR Facility lessons.

5.0 EXEMPTIONS

5.1 The person who prepares the written examination or the grading key for the written examination for a training program, normally a designee of the NTR Facility Manager, is not required to take a written examination for that two-year requalification program. This person is presumed to have achieved an overall score of 80 percent or greater on the examination.

5.2 An SRO who is initially licensed during, or within four months prior to the beginning of, a two-year requalification program may receive applicable credit towards the requirements of that requalification program for the training received during the Initial Training Program.

5.3 In individual whose license is to be renewed during the course of a two-year requalification program may not have completed the program at the time of submission of the renewal application. In such a situation, compliance with all applicable provisions of this program shall be deemed equivalent to satisfactory completion of the program for purposes of meeting the requirement of Section 55.33.a.4 of 10CFR55.

5.4 A person who administers an operational examination under the provisions of Section 3.2.1 of this program is considered to have satisfied the requirement for taking an operational examination at that time with no weaknesses or deficiencies uncovered.

5.5 The NTR Facility Manager approves all changes to the NTR Facility and therefore becomes aware of changes without necessarily meeting the requirements of Section 3.4.2.

6.0 RECORDS

Each of the records listed below shall be retained for at least two years from the date of the recorded event in a format and location amenable to audit by Nuclear Regulatory Commission inspectors.

6.1 Written examinations given, their answer keys, and the answers and grades of each SRO who takes the exam. The time required for each SRO to write the answers shall be recorded.

6.2 Records of additional training and evaluations under the requirements of Sections 3.1.3, 3.2.2 and 4.1.

6.3 Records of the removal from and restorat'on to licensed duties of individual SRO's under the requirements of Sections 3.1.3.3, 3.2.2.3 and 4.1.1.

6.4 Descriptive summaries of each operational examination administered, including any deficiencies or weaknesses discovered, the duration of the exam, and the identity of the examiner.

6.5 Records of the reactivity manipulations performed by each SRO. Not every manipulation need be recorded, but there must be a record that the individual has satisfied the requirements

of Section 3.3. It is sufficient to retain copies of personnel assignment schedules without noting specific reactivity manipulations, since references A and B specify what manipulations are to be performed. Such schedules will also serve to identify the experience obtained by an SRO during the effective license period for inclusion in the license renewal application.

6.6 Records of licensed operator functions performed to meet the requirement of Section 4.2.2.

6.7 Documentation of the reviews completed by each SRO to meet the requirement of Section 3.4.1.

6.8 Documentation of any reading, presentation or attendance completed by each SRO to meet the requirements of Section 3.4.2.

6.9 Documentation of Nuclear Regulatory Commission approval of this requalification program, revisions to it, and notifications made to the Commission of changes made to the program without prior Commission approval.

7.0 Justification for Significant Deviations From 10CRF55 Appendix A

Paragraph 7 of Appendix A to 10CFR55, "Operators' Licenses", makes provision for requalification programs for operators licensed on research and test reactors to deviate from the requirements of Appendix A, provided that any significant deviations are supported by written justification and approved by the Nuclear Regulatory Commission. Significant deviations between the Requalification Program for Licensed Operators for the Westinghouse Nuclear Training Reactor (NTR) Program and the requirements of Appendix A are identified and justified below.

The following is a listing of significant deviations with justification for these deviations.

1. Contrary to paragraph 2 of Appendix A, the NTR Program does not require a lecture series. This is in keeping with the "specialized mode of operation" of the NTR -- i.e., its almost exclusive use for training. Training operations keep the licensed senior operators continually aware of the theory and principles of operation, the general and specific operating characteristics, the instrumentation, control and safety systems, etc., by virtue of the need to communicate this information to trainees.

2. Contrary to paragraph 4.a of Appendix A, the NTR program calls for a written examination every two years instead of annually. Justification is the same as for item 1. Additionally, the training operations carried out on the NTR call for extensive and frequent power maneuvers, thus keeping the licensed operator continually aware of the items covered in the written examination. These justifications are in keeping with ANSI/ANS 5.4-1977.

3. Contrary to paragraphs 4.d and 4.c of Appendix A, the NTR Program does not require simulation of emergency or abnormal conditions, nor evaluations of operator actions to be taken during such simulation. Due to the simplicity of the design and operation of the NTR, the immediate action called for in nearly all emergencies is to shut down and secure the reactor. There is little training value in performing or simulating this simple action.

4. Contrary to Appendix A and to ANSI/ANS 15.4-1977, neither of which makes provision for exempting any operator actively and extensively engaged as an operator from any portion of the requalification program, the NTR Program permits the exemptions listed in Section 5. Justification for specific exemptions is listed below. Numbers in parentheses refer to sections in part 5.0 of the NTR Program.

A. The person who prepares a grading key for a written examination (5.1) will derive the same or more benefit from the preparation process as from actually taking the exam. The person who prepares an exam must take into account the answer in order to phrase the question, and must evaluate the answer in assigning points to the question. In this way, the person would also derive at least the same benefit as from actually taking the exam.

B. An operator who receives an initial license during, or within four months of the start of, a two-year qualification program (5.2) would have completed the initial operator qualification training program. The operator may be presumed to be entering the two-year period with knowledge and skills sharpened beyond those of other operators by virtue of the fact that the initial training program is more extensive and intensive than the requalification program.

C. An individual whose license is to be renewed during the course of a two-year requalification program (5.3) may still be in the process of requalification training at the time of license renewal. The regulations in Section 55.33.a.4 of 10CFR55 require a statement that the requalification program has been "satisfactorily completed" at the time of renewal. Satisfactory completion may not be accomplished until the end of the two-year program, after the effective term of the

individual's license. It appears to be within the intent of the regulation to require that the individual complete all applicable portions of the program up to the time of renewal. For example, an individual may have taken only one operational exam, or reviewed the emergency procedures only once, or supervised the performance of only five reactivity manipulations at the time of license renewal. These should be adequate to ensure continued competence so long as all other applicable requirements of the NIR program are met.

D. A person who administers an operational examination (5.4) must be considered competent to judge the knowledge, competence and dexterity of another person.

E. Justification for Section 5.5 is contained within that section itself.