NEDO-12727
80NED029
Class I
April 1981

GENERAL ELECTRIC
NUCLEAR TEST REACTOR
SAFETY ANALYSIS REPORT

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of the
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ABSTRACT

The General Electric Nuclear Test Reactor is described and a summary of the facility safety evaluation is presented. The description includes the General Electric Nuclear Test Reactor history; the Vallecitos Nuclear Center Site and area characteristics; a detailed facility description; descriptions of Irradiation Facilities, instrumentation and control systems; and facility administration, including the Quality Assurance programs and shielding around the facility. The safety evaluation contains a summary of the analyses performed and the consequences of normal and off-normal conditions, and postulated reactor accident conditions.

1. INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The General Electric Company (GE) designed, constructed, and is operating the Nuclear Test Reactor (NTR) as a part of the experimental facilities at its Vallecitos Nuclear Center (VNC) Site in Alameda County, California. The reactor was designed as an experimental tool to (1) advance the Company's progress in the nuclear energy program, (2) provide a source of neutrons for sample irradiations or exponential experiments, and (3) provide a sensitive device for measuring reactivity. Over 23 years of operation in the performance of a variety of experiments for General Electric and its customers has demonstrated the safety and effectiveness inherent in the reactor's design and the Company's operating methods.

1.2 PURPOSE

Descriptive information for the NTR facility was originally contained in GEAP-1005, Safeguards Report, Nuclear Test Reactor. This document was part of the application, dated June 5, 1957, for a construction permit and facility license; pursuant to this application, as amended, construction permit CPRR-19 and facility license R-33 (Docket 50-73) were issued. In 1958, General Electric amended the license application to incorporate changes in procedures and equipment. At that time, the information in document GEAP-3068, Summary Safeguards for the Nuclear Test Reactor (October 7, 1958), was substituted for the related information in the original application.

A new summary safeguards report for the nuclear test reactor (APED-4444) was submitted February 1, 1965, for updating and providing new information about design, operation and safety analysis. It accompanied a license application amendment requesting separate technical specifications and provided a documentation mechanism for simplifying amendatory actions.

A revision of APED-4444 (APED-4444A) was submitted November 21, 1968 and amended March 31, 1969 and May 28, 1969 as part of a license application amendment requesting an increase in authorized maximum steady-state power level from 30 to 100 kW.

As a revision of APED-4444A, NEDO-12727 has been prepared to:

1. Accompany an application for reactor license extension to permit continued 100-kW, steady-state operation of the NTR.

2. Describe in one document the changes that have been made to the Site, facility, and organization since APED-4444A was written.

3. Provide additional information and technical data obtained during a recently completed safety evaluation of the facility.

1.3 SUMMARY DESCRIPTION

The NTR is a heterogeneous, enriched-uranium, graphite-moderated and -reflected, light-water-cooled, thermal reactor, licensed to operate at steady powers up to 100 kW. The fuel consists of highly enriched uranium-aluminum alloy disks, clad with aluminum. The core is cooled either by natural or forced circulation of deionized lightwater circulated in a primary system constructed primarily of aluminum. Reactivity is controlled by six manually positioned cadmium sheets, four boron-carbide-filled safety rods (spring-actuated for reactor scram), and three electric-motor-driven boron-carbide-filled control rods. Conventional instrumentation is provided to indicate, record, and control important variables, and shut down the reactor automatically if assigned operating limits are exceeded. The reactor's irradiation facilities include a central sample tube, and penetrations through and into the reflector, the reflector faces, and the beams from any of these facilities.

When used as a neutron source, the reactor can provide unperturbed neutron fluxes (at 100 kW) of about 2×10^{12} thermal n/cm^2-sec and an epicadmium flux of about 1×10^{12} n/cm^2-sec. When used as a detector, reactivity effects can be measured with a precision of 10^{-6} $\Delta k/k$ without the use of a pile oscillator.

The reactor is located in a thick-walled concrete cell which, with the control room, north room, setup room and the south cell, houses the NTR facility. An over-all view of the facility, except the north room, set-up room, shop cage, and bunker, is shown in Figure 1-1. Principal equipment in the concrete reactor

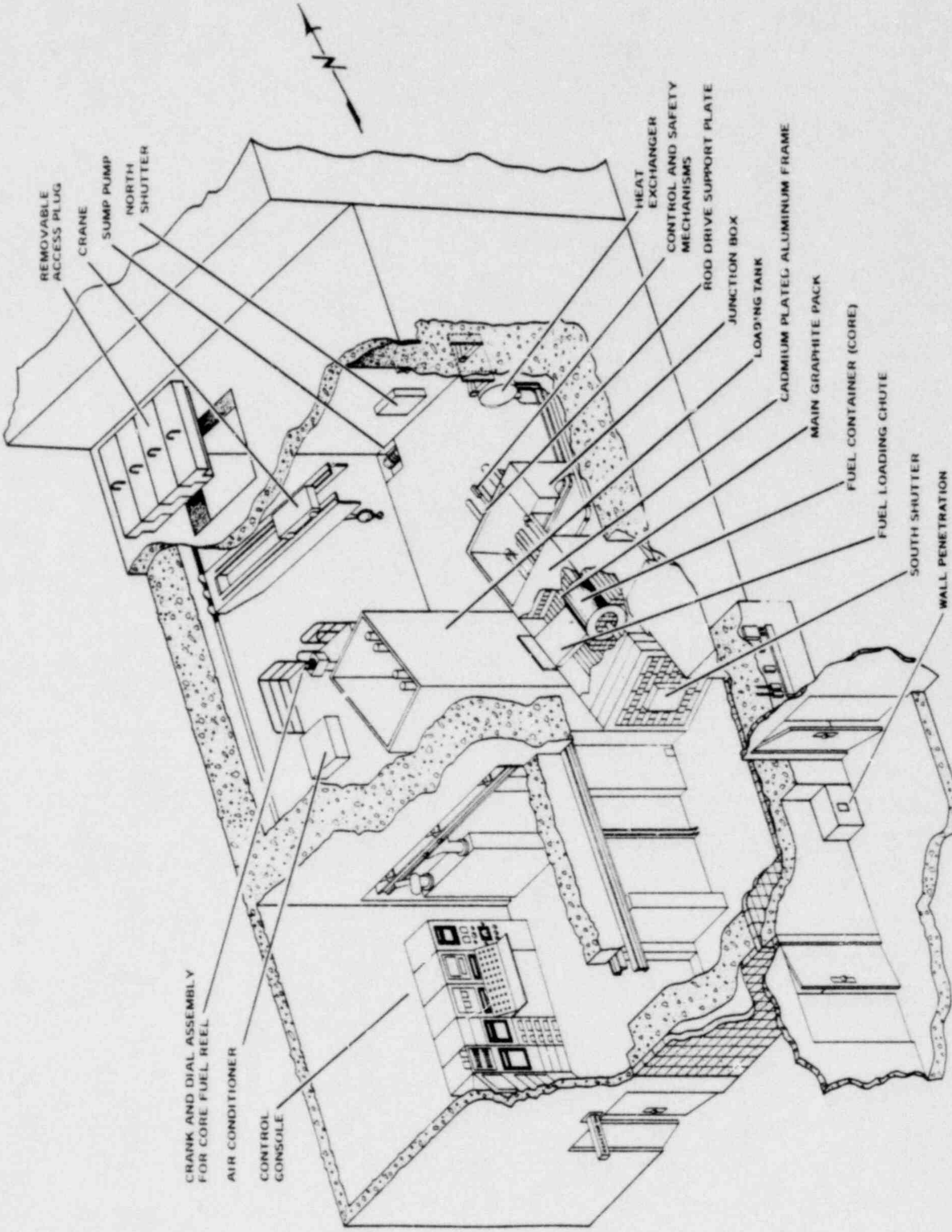


Figure 1-1. Nuclear Test Reactor Facility

cell includes the reactor, the reactor control mechanisms, the coolant system, a fuel loading tank which provides radiation shielding and the primary water system reservoir. The control room contains the control console and provides space for experiment equipment, preparation, and an operator work area. The south cell is a concrete-shielded room which provides access to the thermal column and the horizontal facility. The north room provides space for performing experiments utilizing the horizontal facility neutron beam and the Control and Instrumentation (C&I) test facility. The set-up room is used for storage and setup of experiments prior to irradiation.

The NTR facility is located at the VNC in Building 105. The adjacent two rooms on the west side contain laboratories. The adjacent room on the east side is a shop area which also contains the shop cage and supporting reactor instrumentation. Other areas in Building 105 are used for offices, laboratories, mechanical equipment, and storage. Services and facilities are available to the NTR as a component of Building 105, and include fire protection, security, radiation monitoring, emergency alarm system, water supply, electric load center, and service air supply.

The dimensions, measurements, and other numerical values given in this report may differ slightly from actual values, owing to normal construction and manufacturing tolerances, or normal accuracy of instrumentation.

1.4 DEFINITIONS

Following is a list of definitions, items of clarification etc., which shall be used in the interpretation of this report, Standard Operating Procedures, and other NTR items as applicable.

1.4.1 Reactor Safety

1. Safety Limits (SL) are limits upon important process variables which are found to be necessary to reasonably protect the reactor fuel from melting.

2. Process Variables, as generally used, are measurable operational variables of reactor power, inlet temperature, and primary flow. For the NTR, the only important process variable is the reactor power.

3. Limiting Safety System Settings (LSSS) are settings for automatic protective devices related to those variables having significant reactor safety functions.

4. Limiting Conditions of Operation (LCO) represents the lowest functional capability of performance levels of equipment required for safe operation of the facility.

5. Measuring Channel is the combination of sensor, lines, amplifiers, and output devices which are connected for the purpose of measuring the value of a process variable.

6. Measured Value of a parameter is the value as it appears at the output of a measuring channel.

7. True Value for a parameter is its exact value at any instant.

8. Indicated Value is the true value of a variable offset by the particular instrument uncertainty; e.g., calibration, circuit bias, and least count.

9. Inlet Temperature is the core coolant inlet temperature.

10. Quasi-Steady is a condition in which the time scales associated with a particular transient off-normal event are large enough to allow for neglect of the fuel plate and coolant rates of temperature change during that event.

11. Reactor Power is the total reactor thermal power, as determined by a primary coolant system heat balance.

12. Scram Systems are that combination of measuring channels and associated circuitry which forms the protective system of the reactor as listed in Table 8-1.

13. Nuclear Safety Systems consist of the linear and log N nuclear channels identified in Table 8-1 that are used to monitor reactor power.

14. Safety-Related Systems are that combination of measuring channels and associated circuitry which provides information requiring manual protective action to be initiated as listed in Table 8-2.

15. Reactor Safety System consists of the scram systems listed in Table 8-1 and the safety-related systems listed in Table 8-2.

1.4.2 Reactor Operations

1. Reactor Operation (the reactor is considered started up) is when all of the following conditions are satisfied:

   a. Console key is in the console keyswitch and the console is energized;

   b. More than one of the installed safety and control rods are withdrawn;

   c. Potential excess reactivity is greater than or equal to zero.

2. Reactor Shutdown is that subcritical condition of the reactor in which the negative reactivity of the xenon-free core would be equal to or greater than the shut-down margin with experiments removed, or when there is insufficient fuel to go critical with all manual poison sheets, safety rods, and control rods removed.

3. Reactor Secured is that overall condition in which all of the following conditions are satisfied:

   a. Reactor is shut down;

   b. Console is deenergized and the console keyswitch is in proper custody;

   c. No work is in progress involving in-core components, installed rod drives, or experiments.

4. Unscheduled Shutdown is any unplanned shutdown of the reactor caused by actuation of the scram channels, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, excluding shutdowns which occur during planned equipment testing or check-out operations.

5. Operable is when a system or component is capable of performing its intended function in a normal manner.

6. Abnormal Occurrence is any of the following:

   a. Any actual reactor safety system setting less conservative than specified in Subsection 2.2 of the Technical Specifications Limiting Safety System Settings.

   b. Operation in violation of a Limiting Condition for Operation in Section 3 of the Technical Specifications.

   c. Incidents or conditions which prevented or could have prevented the reactor safety system from performing its intended safety function.

   d. An uncontrolled or unplanned release of radioactivity in excess of the limits specified in the Nuclear Regulatory Commission requirements: Title 10, Code of Federal Regulations, Part 20 (10 CFR 20).

   e. An uncontrolled or unanticipated change in reactivity greater than 0.50$ while the reactor is critical.

   f. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy has caused the existence or development of an unsafe condition in connection with the operation of the reactor.

7. Potential Excess Reactivity is that excess reactivity which can be added by the remote control of poison rods plus the maximum credible

reactivity addition from primary coolant temperature change plus the potential reactivity worth of all installed experiments.

8. Shut-down Margin is the minimum shut-down reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating condition, and that the reactor will remain subcritical without further operator action.

1.4.3 Reactor Experiments

1. Experiment is any of the following:

   a. An activity utilizing the reactor's experimental facilities or its components, or the neutrons or radiation generated therein.

   b. An evaluation or test of a reactor system's operational surveillance, or maintenance technique.

   c. The material content of any of the preceding, including structural components, encapsulation or confining boundaries, and contained fluids or solids.

2. Experimental Facility is any location for experiments which is on or against the external surfaces of the reactor main graphite pack or thermal column, or within any penetration thereof.

3. Secured Experiment is any experiment, experimental facility, or component of an experiment which is deemed to be secured, or in a secured position, if it 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 natural phenomena or malfunctions.

4. Unsecured Experiment is any experiment, experimental facility, or component of an experiment which is deemed to be unsecured if it is not and when it is not secured as defined in the preceding paragraph. Moving parts of experiments are deemed to be unsecured when they are in motion.

5. Movable Experiment is one which may be inserted, removed, or manipulated in an experimental facility while the reactor is critical.

6. Static Reactivity Worth of an experiment is the absolute value of the reactivity change which may be measurable by calibrated control or regulating rod comparison methods between two defined terminal positions or configurations of the experiment.

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

   The evaluation must consider possible trajectories of the experiment in motion relative to the reactor, its orientation along each trajectory, and circumstances which can cause internal changes, such as creating or filling of void spaces or motion of mechanical components. For removable experiments, the potential reactivity worth is equal to or greater than the static reactivity worth.

8. Explosive Material is any solid or liquid which is categorized as a severe, dangerous, or very dangerous explosion hazard in "Dangerous Properties of Industrial Materials," Third Ed. (1968), by N. I. Sax, or is given an identification of reactivity (stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication, "Identification System for Fire Hazards of Materials," 704.M (1966), also enumerated in the "Handbook for Laboratory Safety," 2nd Ed. (1971), published by The Chemical Rubber Co.

9. Accumulation is any single item or group of fissile materials. One can of UO_2 powder, one can of UO_2 pellets, or one fuel rod are examples of an accumulation of fissile material. Examples of fissile materials are U-235, U-233, and Pu-239.

10. Assembly is accumulations of fissile material which are separated from each other by less than 12 inches.

11. Array is a collection of two or more accumulations and/or assemblies wherein each accumulation and/or assembly in the array is separated from the other by more than 12 inches, but is not nuclear-isolated from each other.

12. Nuclear Isolation involves two accumulations (assemblies) or arrays of accumulations (assemblies) which may be considered as being nuclearly isolated from each other only if an edge-to-edge separation exists which is not less than one of the following or its nuclear equivalent: 8 inches of water; the larger of 12 feet or the greatest distance across an orthographic projection of either accumulation (assembly) or array on a plain perpendicular to a line joining their centers.

13. Criticality Area is any physically identified area or location involving the handling of fissile materials under the direction of a single area manager. A criticality area may include more than one criticality limit area.

14. Criticality Limit Area (CLA) is a designated and physically identifiable locality within which a specific set of criticality control limits govern the use of fissile materials.

15. Criticality Control is the administrative and technical requirements established to minimize the possibility of achieving inadvertent criticality in the environment analyzed.

16. Safe Batch is an accumulation of fissile material which is 45% of the critical accumulation considering enrichment, full reflection, and optimum water moderation consistent with the form of the material.

1.4.4 Maintenance and Testing

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

2. Channel Test is the introduction or interruption of a signal into the channel to verify that it is operable (not to include the alarm or trip test).

3. Channel Calibration is a comparison and/or an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, if possible, including equipment actuation, alarm, or trip set-point test and shall include the channel test.

4. Instrument Check is a qualitative verification of acceptable performance by observation of the instrument's behavior. This verification, where possible, shall include a comparison of the instrument with other independent instruments measuring the same variable.

5. Instrument Test is the introduction or interruption of a signal (i.e., built-in internal signals) into an instrument to verify that it is operable (not to include verification of the alarm or trip test).

6. Instrument Calibration is a comparison and/or adjustment of an instrument such that its output corresponds with acceptable accuracy to known values of the parameter which the instrument measures. Calibration shall include instrument actuation, alarm, or trip test, and shall include the instrument test.

7. Component Check is a qualitative verification of acceptable performance by observation of the components' behavior. This verification, where possible, shall include a comparison of the component with other independent components or systems measuring the same variables.

8. Component Test is the introduction or interruption of a signal into the component to verify that it is operable (not to include the alarm or trip test).

9. Component Calibration is a comparison and/or an adjustment of the component such that its output corresponds with acceptable accuracy to known values of the parameter which the component measures. Calibration shall encompass the entire component, if possible, including equipment actuation, alarm, or trip test, and shall include the component test.

10. Alarm or Trip Test is the introduction or interruption of a signal into a channel, instrument, component, etc., as required to verify that the alarm or trip circuitry is functioning as intended.

11. Daily means prior to the first startup of each day of operation following a period when the reactor was secured.

12. Monthly means approximately at 1-mo. intervals (usually the first week of each month).

13. Quarterly means approximately at 3-mo. intervals (usually the first month of each fiscal quarter).

14. Yearly means approximately at 1-yr intervals.

1.4.5 General

1. Site is the area (~1600 acres) within the confines of the Vallecitos Nuclear Center (VNC) owned and operated by General Electric.

2. Facility is that portion of the building occupied by the reactor, reactor control room, and associated support areas.

3. Licensed Operator is a person who is licensed as a Reactor Operator (RO) or Senior Reactor Operator (SRO) pursuant to 10 CFR 55 to operate the controls of the Nuclear Test Reactor (NTR).

4. Operator Trainee is an individual who manipulates the controls of the reactor as part of a training program to qualify for an operator's license. This individual will be under the direction and in the presence of a licensed operator while manipulating the controls.

5. Senior Reactor Operator Readily Available On-Call is one who satisfies all of the following:

   a. Is within a reasonable driving time (1/2 hour) from the reactor facility;

   b. Can be promptly contacted by telephone;

   c. Has informed the reactor operator on duty of where he may be contacted.

1.5 SAFETY PRINCIPLES

The design of the NTR resulted from the evolution of a series of reactors designed by scientists at the Knolls Atomic Power Laboratory. The earlier reactors were known as thermal test reactors; three models were built and operated successfully. The logical evolution which led to the design of the NTR produced a versatile and safe reactor. Features which contribute to the safety of the reactor and which were incorporated into the design and construction of the reactor include:

1. Negative void coefficient of reactivity.

2. Negative or small positive coolant temperature coefficient of reactivity.

3. A control system extremely sensitive to changes in reactivity so that minute changes are detectable.

4. Safety and control functions that are completely separate, except all safety rods must be fully withdrawn to ensure negative reactivity is available if needed for scram before more than one control rod can be moved.

5. Manually positioned cadmium sheets that can be used to limit reactivity controllable from the console and to provide sufficient negative reactivity to preclude any possible danger or criticality during fuel loading.

6. An instrumentation system which includes fail-safe and redundant features as well as proven reliable components.

7. A system constructed from materials having properties compatible with their intended service.

Safety measures which have been incorporated into the operation of the facility include:

1. Very low heat flux, even at the maximum operating power.

2. Temperatures and pressures only a little above ambient.

3. Low operating power, resulting in a low fission-product inventory.

4. Rigid control by operations management of all experiments performed in the reactor facility.

5. Performance of all activities that can affect nuclear safety under the direction of an NRC-licensed reactor operator or NRC-licensed senior reactor operator, as required.

1.6 HISTORY OF THE FACILITY

The reactor was constructed under construction permit CPRR-19, issued October 24, 1957, as requested by General Electric's application, dated June 5, 1957. Operation of the reactor up to powers of 30 kW was authorized by facility license R-33, issued on October 31, 1957. Initial loading of the reactor began on November 7, 1957, and criticality was first achieved on November 15, 1957. Operation since that time was in accordance with license R-33, as amended. During that period of authorized operation, the reactor was operated at powers up to 30 kW for more than 5000 hours. On July 22, 1969, the license was amended, which revised the license in its entirety, authorizing operation of the reactor at power levels of up to a maximum of 100 kW steady-state power. Since then, the reactor has operated at power levels up to 100 kW for more than 17,000 hours while irradiating a wide variety of experiments. In 1976, the reactor core can sprung a leak in a weld area, necessitating replacement. The reactor fuel was removed and inspected and a major portion of the reactor was dismantled. The core can was replaced, as well as some of the graphite in the central area. Some modification of the irradiation facilities occurred at this time. The reactor was reassembled, utilizing the original fuel, and routine operation resumed. The safe and efficient operation of the NTR, now in its twenty-fourth year of operation, is evidenced by the 21 annual reports submitted to the NRC on operating experience pertinent to safety.

1.7 GENERAL CONCLUSIONS

Because of the many safety features provided and the strong administrative control applied to operation of the facility, the possibility of an accident involving high radiation exposure or the dispersion of substantial quantities of radioactivity is considered extremely remote. However, the protection of the health and safety of the public is ensured further by housing the reactor in a thick-walled concrete cell that provides radiation shielding and permits controlled release of airborne contamination. On the basis of the descriptive and analytical information provided in this report and the proven performance of the facility over an extended operating period, it is concluded that the design and operating methods of the NTR facility provide the reasonable assurance required by the regulations that the health and safety of the public will not be endangered by continued operation of the facility.

2. THE SITE

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 Reactor and Site Location

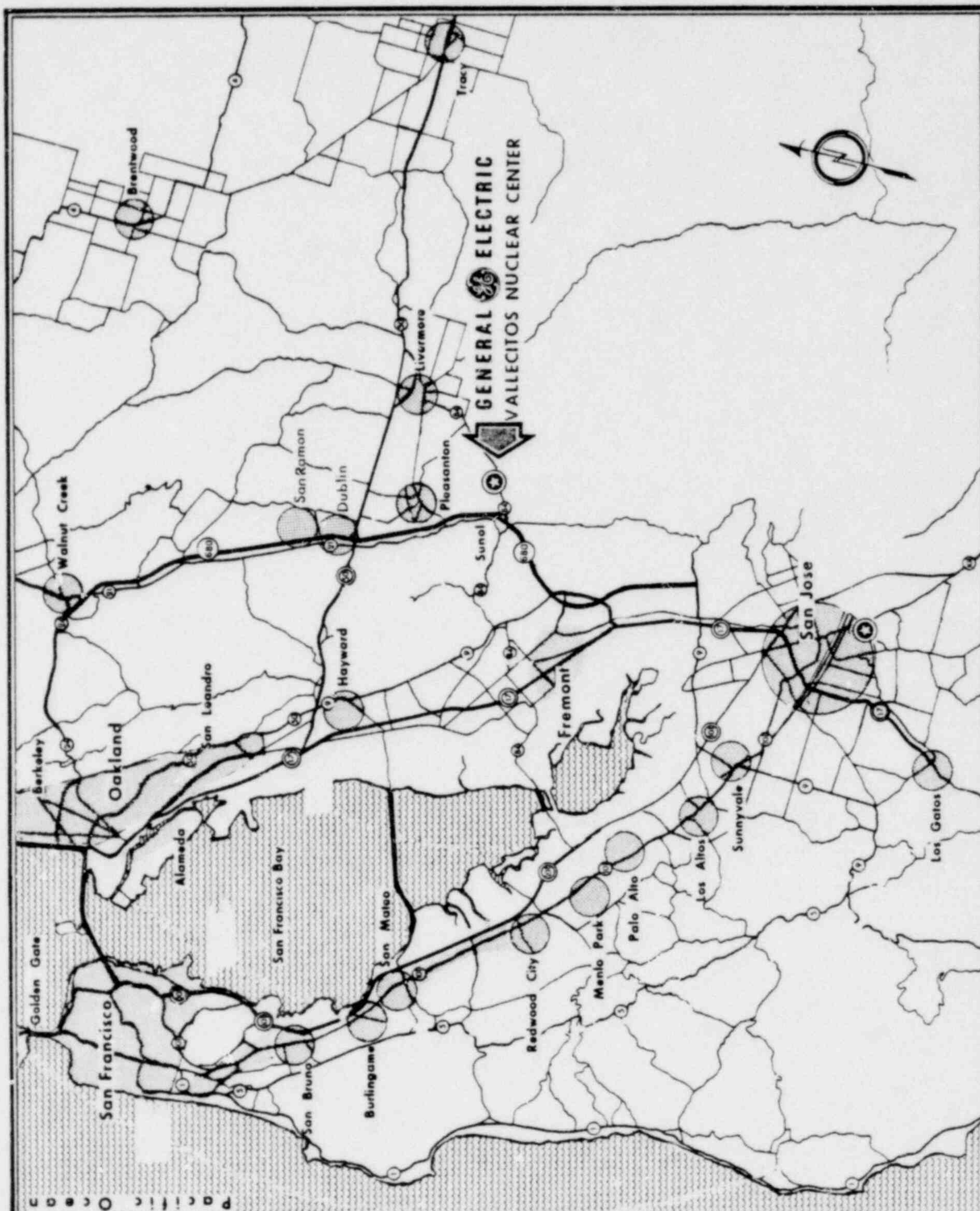
The NTR Site is located at the 1594-acre Vallecitos Nuclear Center (VNC). The Site is owned by the General Electric Company and is situated near the center of the Pleasanton Quadrangle of Alameda County. The nearest towns are Pleasanton and Sunol, within a 5-mi radius, and Livermore-Dublin and a part of Fremont, within a 10-mi radius of the site. The VNC is east of San Francisco Bay, approximately 35 air miles east-southeast of San Francisco and 20 air miles north of San Jose. Figures 2-1 through 2-4 show the general Bay Area, the Pleasanton Quadrangle, VNC and surrounding area, and the Site plot plan. respectively.

2.1.2 Site Description

The VNC is located on the north side of the Vallecitos Valley. The valley is approximately 2 miles long and 1 mile wide; its major axis is east-northeast and west-southwest. The valley is at an elevation of 400 to 500 feet above sea level and is surrounded by barren mountains and rolling hills.

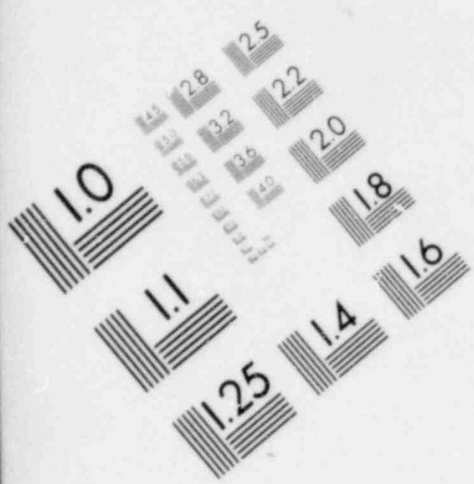
The Site consists of a quadrilateral, bounded on the west, north, and east by hilly terrain; in some places, the hills are about 700 feet above the general Site elevation. Vallecitos Road (State Highway 84) forms the southern boundary of the Site, from which an expanse of gently rolling grassland extends for about 3 miles. Beyond 3 miles, mountain ranges form a southern barrier which completes the encirclement of the Site.

Approximately one-third of the Site is gently sloping or rolling terrain. The remainder consists primarily of the southwestern slope of a ridge serrated by several small draws. The southern part of the Site, adjacent to the Vallecitos Road, is relatively flat and accommodates the NTR, laboratories, and administrative facilities. Site atmospheric-environmental monitoring stations are strategically located on the property.



BAY AREA MAP

Figure 2-1. Bay Area Map



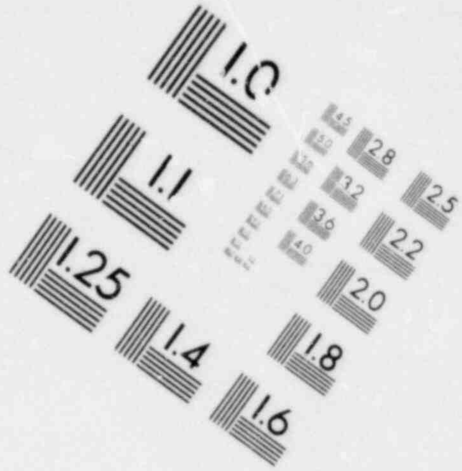
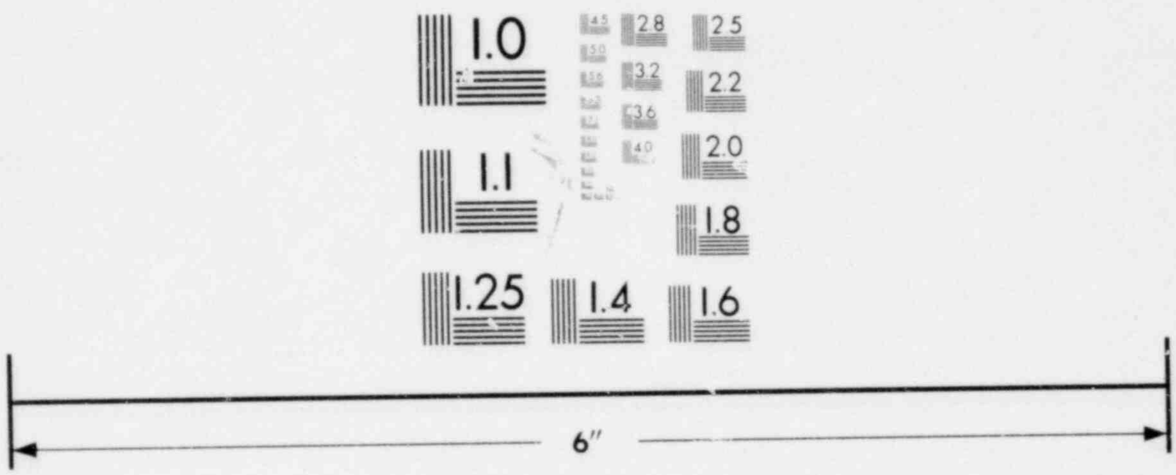
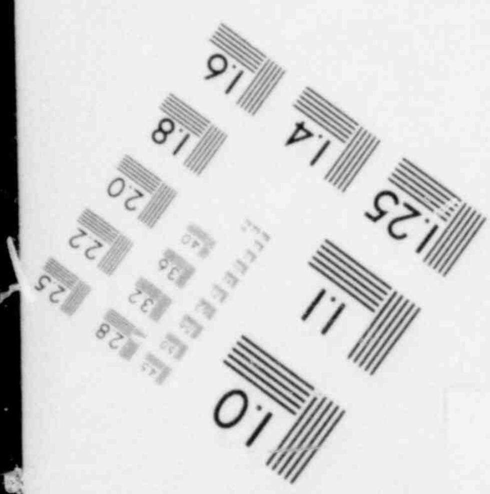


IMAGE EVALUATION
TEST TARGET (MT-3)





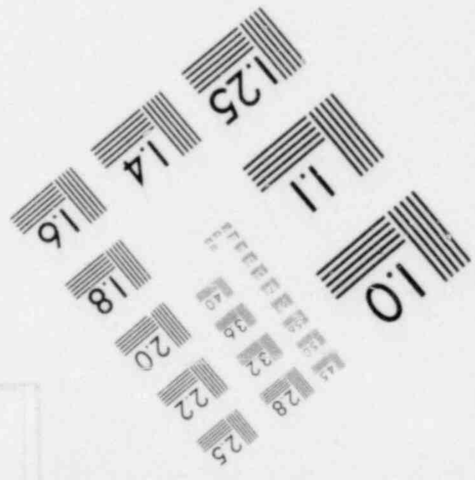


IMAGE EVALUATION
TEST TARGET (MT-3)

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Figure 2-2. Pleasanton Quadrangle

Figure 2-3. Vallecitos Nuclear Center and Surrounding Area

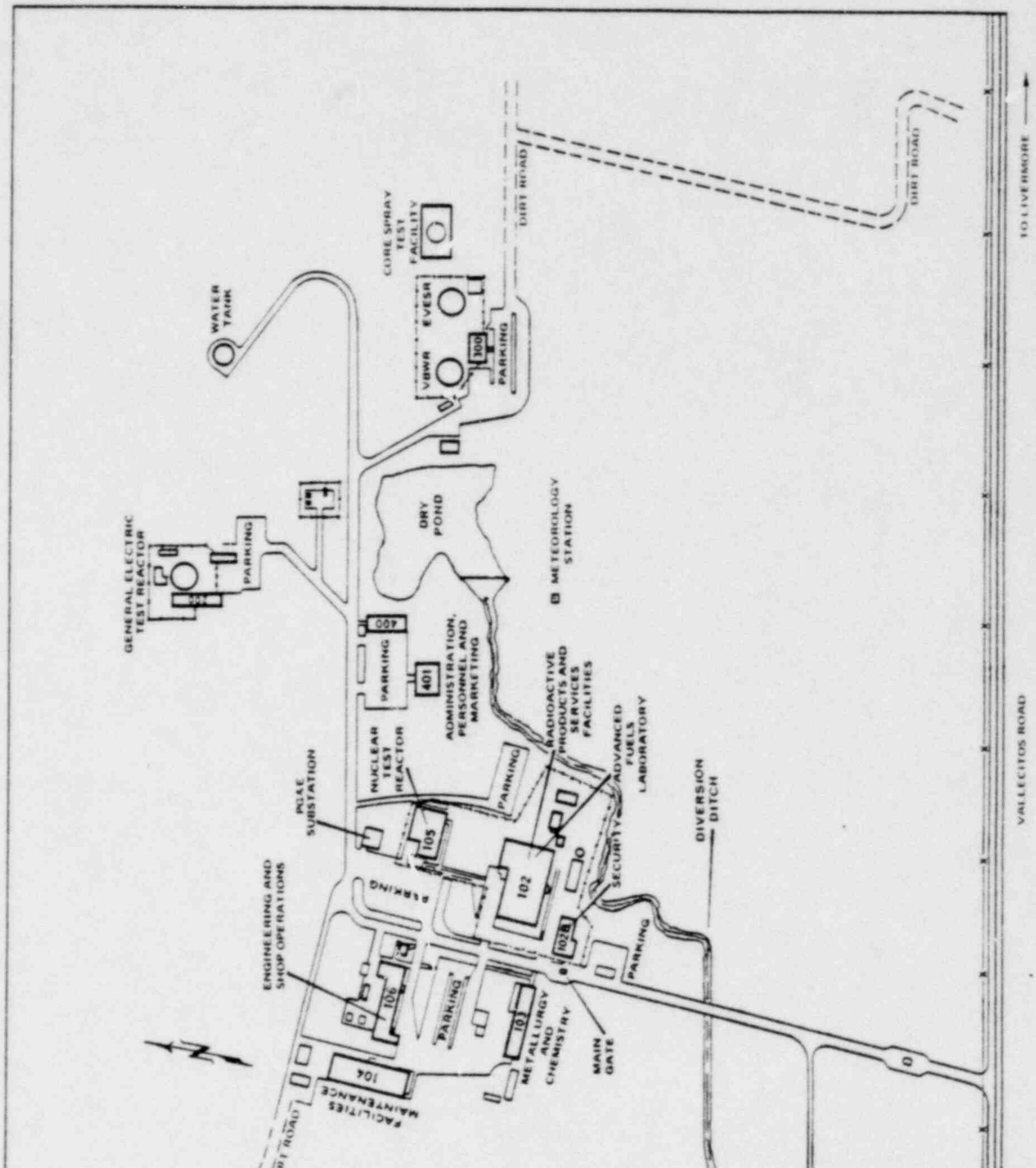


Figure 2-4. Vallecitos Site

Since the northeast portion of the Site is mountainous, the general security of the Site is increased. Approximately 1500 acres of the Site is normally leased for grazing and for cattle-feed crops. The land surrounding the Site is devoted to agriculture and cattle raising.

2.1.3 Population Distribution

The residential population density in the immediate vicinity of the VNC is very low. The farm land within a 3-mi radius is estimated to average less than 50 people per square mile. There are only four houses on the south side of Vallecitos Road and approximately 10 houses immediately west of the Site. Minor land development has occurred 2-1/2 to 3 miles to the west and northwest, associated with the expansion of the town of Pleasanton and the unincorporated areas of Happy Valley and Sunol. Population within a 10-mi radius of the Site is estimated to be 221,000 while that within a 20-mi radius is estimated to be 1,147,000.

Livermore is the largest population center within a 10-mi radius of the VNC, with a population of approximately 51,200. Located 7 miles northeast of the Site beyond a 1200-ft-high mountain range, Livermore is largely a bedroom community with some light industry and agricultural activities. A United States Veterans Administration hospital with a population of approximately 1000 is located between Livermore and the Site, approximately 5 air miles east of the Site. Approximately 8 miles northeast of the Site is the Lawrence Livermore Laboratory of the University of California.

The second largest town within a 10-mi radius of the Site is Pleasanton. A predominantly bedroom community with a population of approximately 32,000, Pleasanton is located beyond the hills 5 miles northwest of the Site. The towns of Livermore and Pleasanton contain a small amount of light industry, but in no way can be considered industrial centers.

Located to the southwest, at a distance of approximately 12 miles, is the city of Fremont. With a population of approximately 119,000, Fremont borders the eastern shore of San Francisco Bay and is separated from the Site by a mountain range which rises 1000 feet above Vallecitos Valley. Beyond Fremont, and generally to the southwest, beginning at a distance of 20 miles from the Site, is San Jose, which has a population of over 500,000. To the northwest, and beginning 15 miles from the Site, is the City of Hayward, with a population of 100,000. Oakland and San Francisco lie to the northwest approximately 30 and 35 miles, respectively. More detailed information on population distribution is given in a separate document.[1]

The rate of population growth in the 30-mi^2 area surrounding the VNC is expected to be slow in the coming decades. The growth rate within this 3-mi radius has been negligible over the last century. Excluding the General Electric property, a considerable portion of the land within the 3-mi radius is owned by the City of San Francisco; it is believed the city would not be inclined to sell the land. The major part of the remaining land is rugged terrain which does not attract industrial or residential buildings; substantial parcels of the privately owned land have been placed into the Alameda County Land Preserve Program under the California Land Conservation Act of 1965.

2.2 SITE CHARACTERISTICS

2.2.1 Security

Since its inception, VNC has operated under a controlled-access security plan. A perimeter fence maintains the Site as a restricted area to the general public. The entrance gate approached over VNC property from State Route 84 (Vallecitos Road) is guarded at all times to control the entrance and exit of personnel. Additional security and control within the Site and its facilities is extensive. The plan conforms to the NRC requirements delineated in 10 CFR, Section 73.40.

2.2.2 Safety

Violent storms are infrequent in this area. The main consequence of such storms would be the possible interruption of service from the Pacific Gas and Electric (PG&E) system (described in Subsection 2.2.5). Reactor shutdown is automatic in the event that normal electrical service is lost. There is a remote likelihood of major flooding at VNC. However, there is some possibility of substantial sheet flow caused by heavy rainfall and resultant runoff from the surrounding hills. All roadways and facilities are constructed with drainage to preclude damage caused by such an occurrence. Surface water drains away from the Site facilities to several natural ravines and man-made channels which empty into Vallecitos Creek.

2.2.3 Fire Protection

The design of the building containing the reactor and the reactor itself makes maximum use of noncombustible structural material. A 500,000-gal water storage tank is located on a hillside above the NTR; 100,000 gallons of this stored water is reserved for fire protection. The Site fire truck (pumper) is garaged at the Building 200 area and auxiliary firefighting equipment is also available at strategic locations. A fire alert system for personnel instruction is provided. Fire teams for potential fire emergencies at the Site have been organized and are maintained in a trained status.

2.2.4 Water Supply

Water is normally supplied to the Site from the Hetch-Hetchy Aqueduct, which provides water to the city of San Francisco. A 14-in., 15,000-ft-long pipe has been installed from the aqueduct to the Site. The pumps installed have a capacity of 1,000,000 gpd. The pipe line capacity is over 3,000,000 gpd. A 500,000-gal water storage tank (mentioned above) is located on Site and provides gravity flow. The Calaveras Reservoir, located about 8 miles south of VNC, provides backup for Hetch-Hetchy.

2.2.5 Electrical Power

Electrical service is supplied by PG&E to the Site substation, where it is distributed to Site facilities. The PG&E transmission line passing just south of the Site is fed from two directions in an electrical loop to ensure a most reliable and continuous parallel 60-kV supply to the VNC substation. This substation feeds electrical power to the NTR.

2.2.6 Reactor and Laboratory Facilities

Facilities located at the VNC are shown in Figure 2-4. The main laboratory buildings are located approximately 1700 feet north of Vallecitos Road. Building 102 contains the Radioactive Materials Laboratory and the Advanced Fuels Laboratory. In these two laboratories, post-irradiation studies and research and development activities are performed. Building 103 houses chemistry, metallurgy, and ceramics research and development activities, as well as extensive analytical chemistry laboratories. Extensions of some of these areas are located in special facilities in Building 300. Building 400 has been assigned to Chemical Engineering and Materials Development. Building 401 contains offices for Administration, Personnel, and Marketing.

Building 106 contains machine, sheet metal, and development shops, and Building 104 contains maintenance and Site warehouse facilities, and radiation protection offices, including the whole body counter. Facilities are available on the Site for handling, sorting, and processing liquid and solid radioactive wastes generated at all VNC nuclear facilities. Temporary storage of solid radioactive waste is accommodated at the hillside storage facility. A nonradioactive liquid waste chemical treatment plant and sewage treatment plant are located in the southwest corner of the Site. A liquid waste evaporator facility is located near the hillside storage facility.

Building 105 contains offices and laboratories and houses the NTR and another shielded cell that was formerly used as a critical experiment facility. The Vallecitos Boiling Water Reactor (VBWR) and the ESADA Vallecitos Experimental Superheat Reactor (EVESR), developmental reactors located on the Site, are now in a deactivated status.

2.3 METEOROLOGY

The VNC meteorology was studied in 1976 by Nuclear Services Corporation of Campbell, California. A summary of the results of their investigation is published in a separate document.[1]

2.4 HYDROLOGY

The hydrology of VNC was studied in 1976 by Nuclear Services of Campbell, California. A summary of the results of their investigations is published in a separate document.[1] Further studies were performed by the U.S. Geological Survey. Results of these studies are published in a separate document.[1a]

2.5 TOPOGRAPHY

Topography of the Pleasanton Quadrangle and the VNC Site is shown in Figures 2-2 and 2-5. Vallecitos Valley is about 2 miles long and 1 mile wide, with the major axis running east-northeast and west-southwest. The Valley is at an elevation of 400 to 500 feet above sea level and is surrounded by barren mountains and rolling hills ranging from 100 to 700 feet above the Site. The flat section of the Site is in the southern part, adjacent to Vallecitos Road, and accommodates all the present facilities.

2.6 GEOLOGY AND SEISMOLOGY

Comprehensive geological and seismological studies have been conducted at and near VNC in the years 1977-1980. The results of these studies (References 2 through 6) have been reported and submitted to the Nuclear Regulatory Commission in relation to the General Electric Test Reactor, also located at VNC. The Nuclear Test Reactor design and the excess reactivity limit under which it operates makes off-normal condition evaluations (see Section 11) insensitive to geological conditions and seismological parameter values.

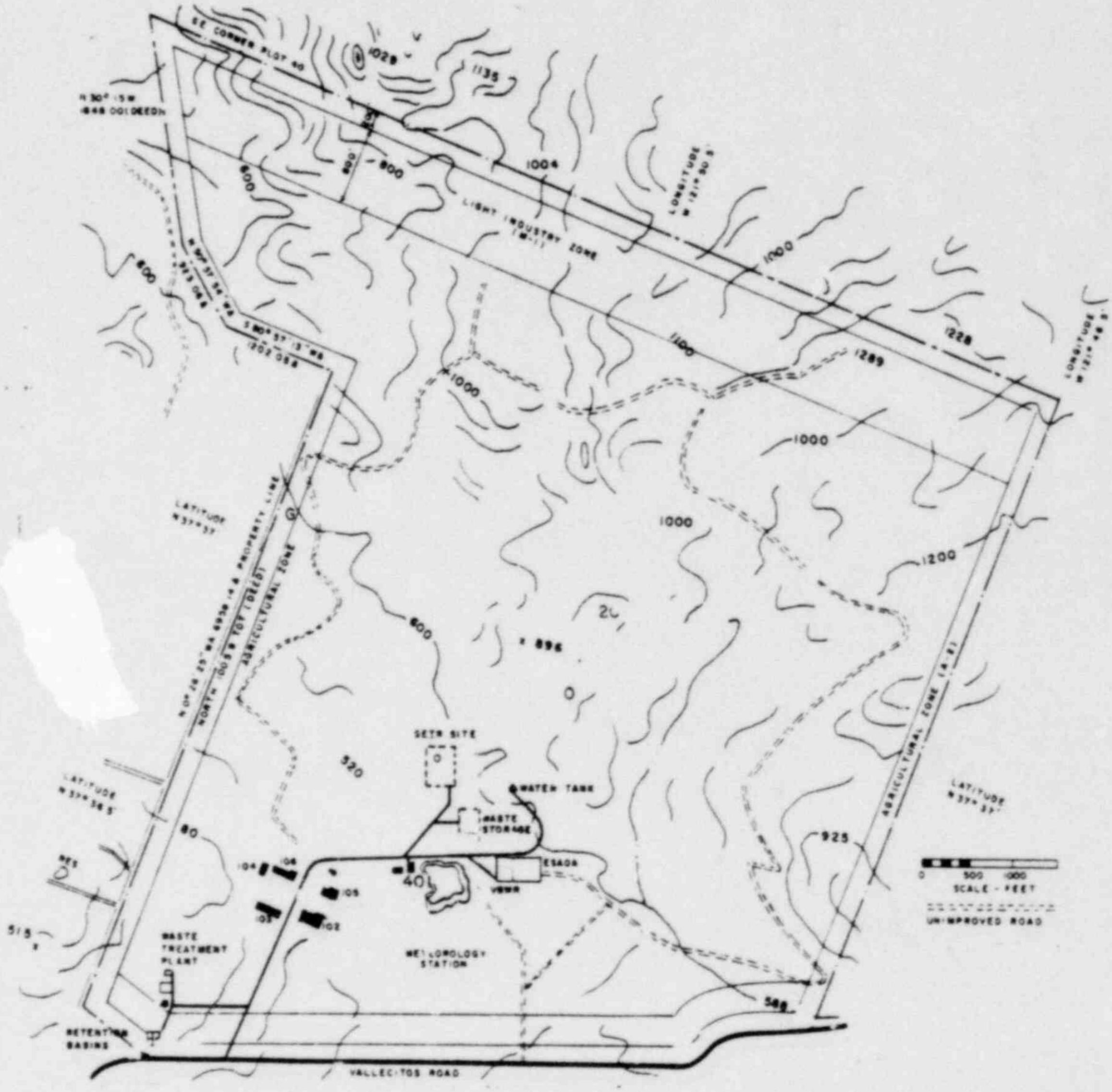


Figure 2-5. Topography Contour of Vallecitos Nuclear Center

POOR ORIGINAL

3. REACTOR CELL

3.1 DESCRIPTION

The reactor is installed in a concrete-shielded room (the reactor cell) located in the northeast corner of Building 105. The radiation shielding which this cell provides for the reactor and its cooling system is discussed in Section 10. Although the cell is not designed to provide gas tight containment, controlled release of airborne reactivity is possible through the operation of the cell ventilation system. Figure 3-1 shows the floor plan of Building 105, and Figure 3-2 shows plan and elevation views of part of the area that contains the NTR facility, including the NTR and reactor cell, and the north room with its shielded Neutron Radiography Facility. These illustrations are not necessarily current with respect to the arrangement of the office and laboratory areas. Other details that are not pertinent to safety considerations for the NTR facility may not be as shown.

The reactor cell is a rectangular-shaped room with approximate internal dimensions of 22 feet wide by 23 feet long by 24 feet high. A heavy-concrete-shielded alcove surrounds the reactor in the southeast corner and a mezzanine is located above the cell door in the southwest corner. Equipment of appreciable size located within the cell includes the reactor, reactor cooling system, fuel loading tank, holdup tank, 5-ton bridge crane, air-conditioning unit, and storage shelves. Approximate gross volume of the cell is 11,300 ft^3 and, with the above-mentioned equipment installed, the net air volume is approximately 10,500 ft^3.

Normal access to the cell is through the large doorway in the south wall. During reactor operation, the doorway is normally closed by a 1-ft-thick, motor-driven, sliding concrete door lined with 1.25 inches of steel. Power for moving the cell door in either direction is interlockd with a key-operated switch so that it is possible for the reactor operator to control all cell entries. A manually operated, 1-ft-thick paraffin door covered with aluminum and located just inside the reactor cell is normally closed to reduce further the radiation dose rates outside the reactor cell door.

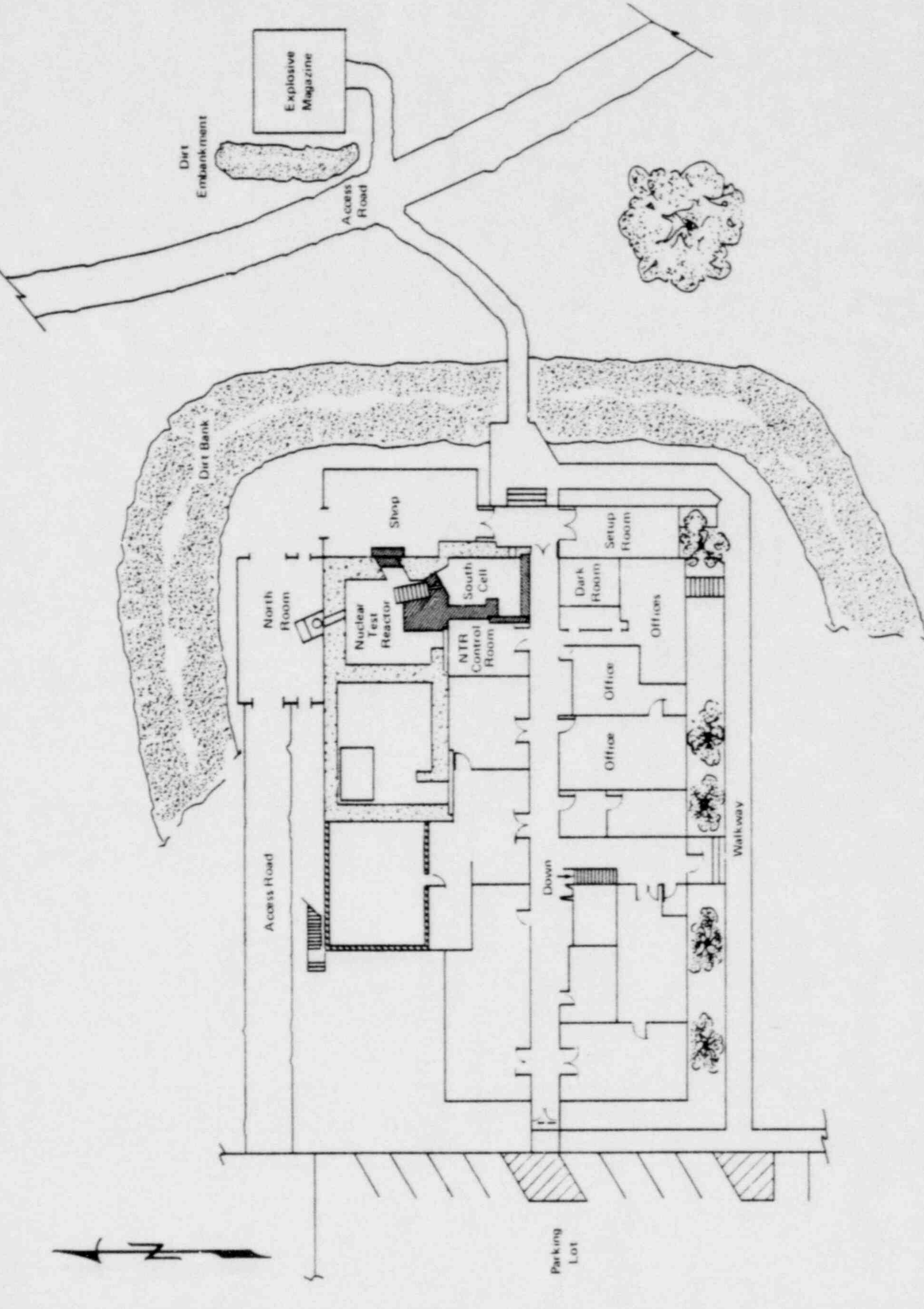


Figure 3-1. Building 105 Floor Plan

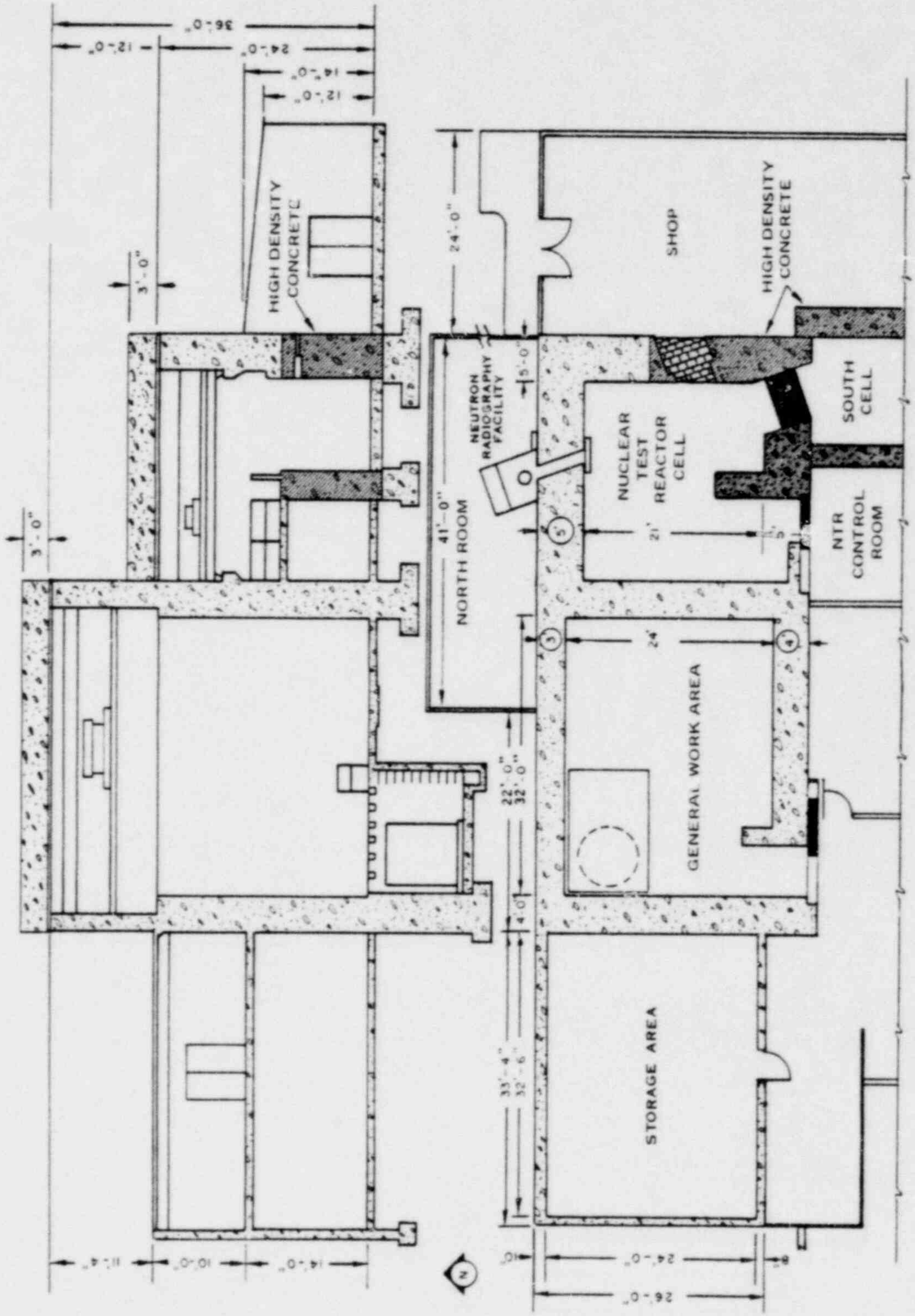


Figure 3-2. Part Floor Plan and Elevation

A large removable equipment hatch is provided in the cell roof. Use of this hatch is limited to very special occasions when use of the cell door is impossible or impractical.

The refueling of the reactor and the maintenance of equipment in the cell are performed only with the reactor shut down (i.e., sufficient manual poison sheets and safety rods inserted to satisfy minimum shut-down margin requirements). Normally, the radiation and contamination levels are quite low; therefore, these activities can be performed with the cell door open. If expected radiation levels, results of radiation monitoring, or some other nonroutine nature of these activities makes closing the door desirable, either maintenance or refueling may be done with the door closed.

3.2 PENETRATIONS

For discussion, the penetrations into the reactor cell have been divided into two types--reactor and experiment services. Reactor service penetrations normally are used for passing water, electric power, and air into and out of the cell; their use generally does not change from day to day. Experiment service penetrations normally are used to move materials and equipment for experiment programs into the reactor cell. The list of penetrations below is the present arrangement. Future changes in the number, location, or type designation of any of the penetrations will be made in a manner to retain the effectiveness of the cell to control radiation, contamination and security.

In addition to the cell door and roof hatch, the existing reactor service penetrations include:

1. One 1.5-in. secondary cooling water supply line;
2. Two 1.5-in. water lines for the air conditioner and supply return;
3. One 1.5-in. deionized water supply line;
4. One 2-in. drain line to the Site retention basins;

5. Twelve 3-in. electrical conduits for wiring between the cell and control room;

6. Eight small-diameter ($\leq$1.5 inches) electrical power and lighting conduits, including spares;

7. One 0.75-in. compressed air supply line;

8. Two 8-in. by 8-in. ventilation exhaust ducts through the cell roof;

9. Four 1.5-in. pipes to the adjoining laboratory; and

10. Two 4-in. holes approximately 10 feet above the cell floor, one of which penetrates into the control room and the other of which penetrates over the control room roof

Existing experiment service penetrations are listed below. Use of these penetrations is controlled carefully to ensure that the effectiveness of the cell to contain radiation and radioactive materials is not significantly reduced.

1. South thermal column into south cell;

2. Horizontal facility tube into south cell;

3. A 24-in.-diam hole through the north wall at approximately core centerline height (blocked on the south end by an aluminum plate and a motor-driven shutter which is controlled from the control room; blocked on the north end by the Modular Stone Monument, which is described in Subsection 7.12).

4. Stepped hole (6-8 inches in diameter) through the east wall approximately 10 feet above the cell floor;

5. Hole for future thermal column through the east wall (presently filled with unmortared concrete bricks); and

6. Two holes (3 and 4 inches, respectively) in the north wall, penetrating into the north room.

3.3 VENTILATION AND AIR CONDITIONING

The NTR exhaust system includes a 3000-ft^3/min fan located on the reactor cell roof. The fan draws air from the reactor cell, south cell, north room Neutron Radiography Facility, and the SNM cage in the northwest corner of the Building 105 shop. The air goes through a prefilter and a bank of absolute filters and is then discharged through a stack of sufficient height to disperse the exhaust upward.

An air-sampling system samples the fan discharge to provide continuous monitoring for radioactive particulate material and nonfilterable radiogases released from the facility. The air-monitoring system is discussed in Subsection 6.7.

An air-conditioning unit is located on the mezzanine inside the cell. Both air inlet and exhaust for this unit are within the cell. The only function of this unit is to control the temperature and humidity inside the cell to levels which are comfortable for operating personnel. The air-conditioning unit has no appreciable effect on the operation of the reactor.

3.4 INSTRUMENTATION

Cell instrumentation consists of a remote radiation-monitoring unit, a differential pressure gauge and the intercommunication system. Each of these systems is discussed in Section 8.

4. REACTOR CORE

4.1 GENERAL DESCRIPTION

The NTR is a light-water-cooled, high-enriched-uranium, graphite-moderated and -reflected, thermal reactor with a nominal power rating of 100 kW. The reactor is 14 feet long (including thermal column and control rod drives) by 6 feet wide by 6 feet high. The reactor consists of a core in the form of an annular cylinder centered in a 5-ft cube of ACOT-grade graphite. The core container is a horizontal aluminum cylinder 20 inches long and 18 inches in diameter, with an inner annulus diameter of 11.5 inches. The 11.5-in.-diam cylindrical space is filled with graphite and traversed by the horizontal facility. The annulus thus formed is fitted with 16 fuel assemblies, each consisting of 40 aluminum-clad 2.75-in.-outside diameter (o.d.) uranium-aluminum fuel disks. The disks are spaced on aluminum rods to give an active length of about 15.25 inches. Up to six manual poison sheets, four safety rods, and three control rods are arrayed around the outside of the fuel container (Figure 4-1).

4.2 FUEL CONTAINER

Figure 4-2 is an assembly drawing of the present reactor fuel container. This container was put into service in 1976 after the previous container, which had been in service for approximately 18 years, sprung a leak in a weld area. The annular ends of the container, 0.5-in. aluminum plates, are welded to the inner and outer cylindrical skins, which are rolled aluminum sheets 0.25 and 0.0625 inch thick, respectively. The outer cylinder is made from two pieces welded together opposite the loading chute. Attached to the inside surface of each end plate is an aluminum circular raceway which supports and guides the core reel assembly. The reel assembly, in turn, supports the fuel assemblies. The core reel assembly is described in more detail in Subsection 4.4.

Openings are provided in the north end plate for the 1.5-in. primary coolant inlet and outlet lines. The inlet pipe is connected to a flow-distributor tube located inside the container below the core. A row of 25 0.25-in. holes is drilled into the lower side of the 1.375-in. flow-distributor tube, with the holes near the core midplane closer together than those at the ends to distribute water flow to correspond to power distribution along the core. The center-to-center distance between the 10 holes nearest the midplane is 0.4375 inch; the next three holes, toward each end, are on 0.5-in. centers; the next

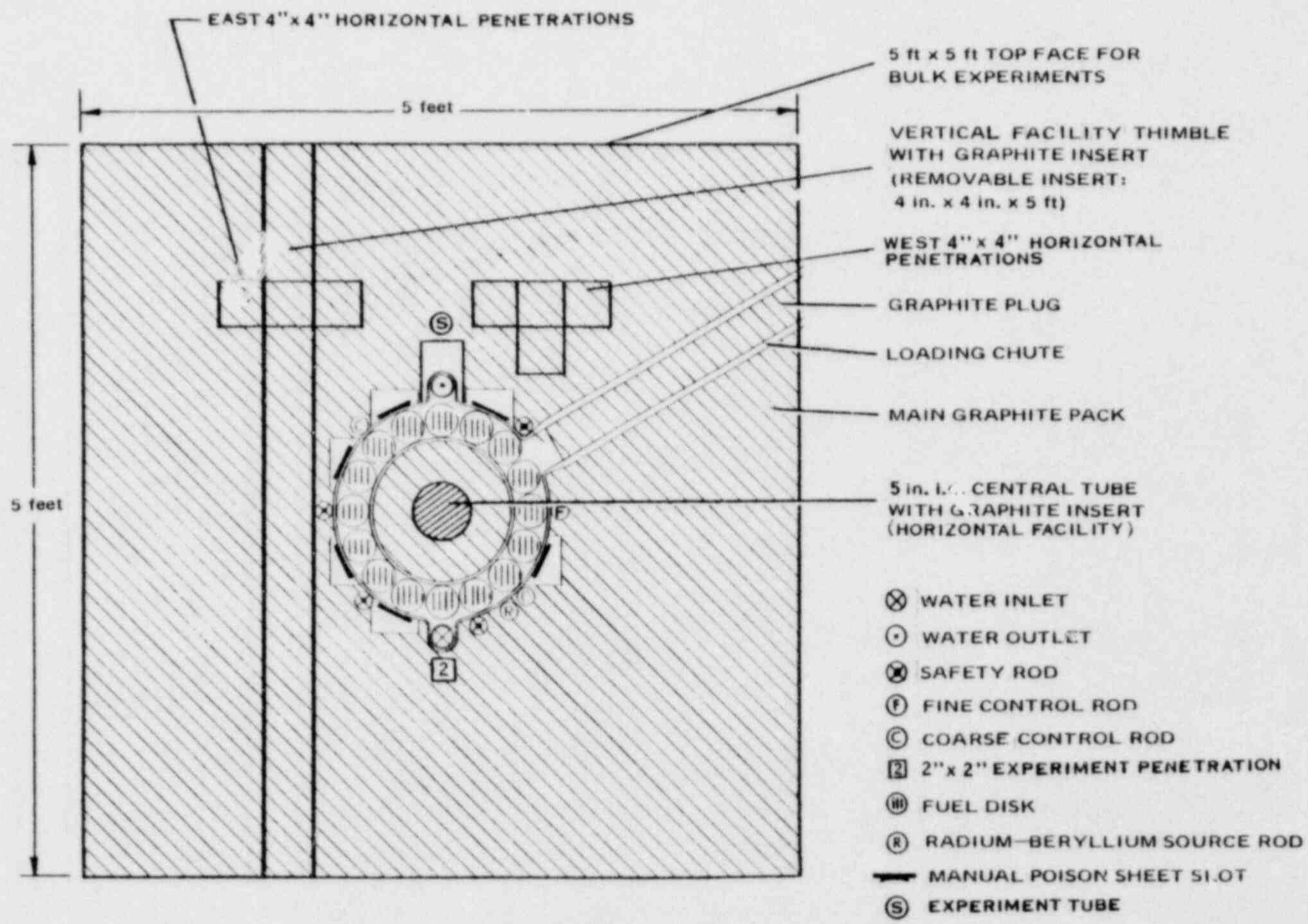


Figure 4-1. Vertical Section Through the NTR

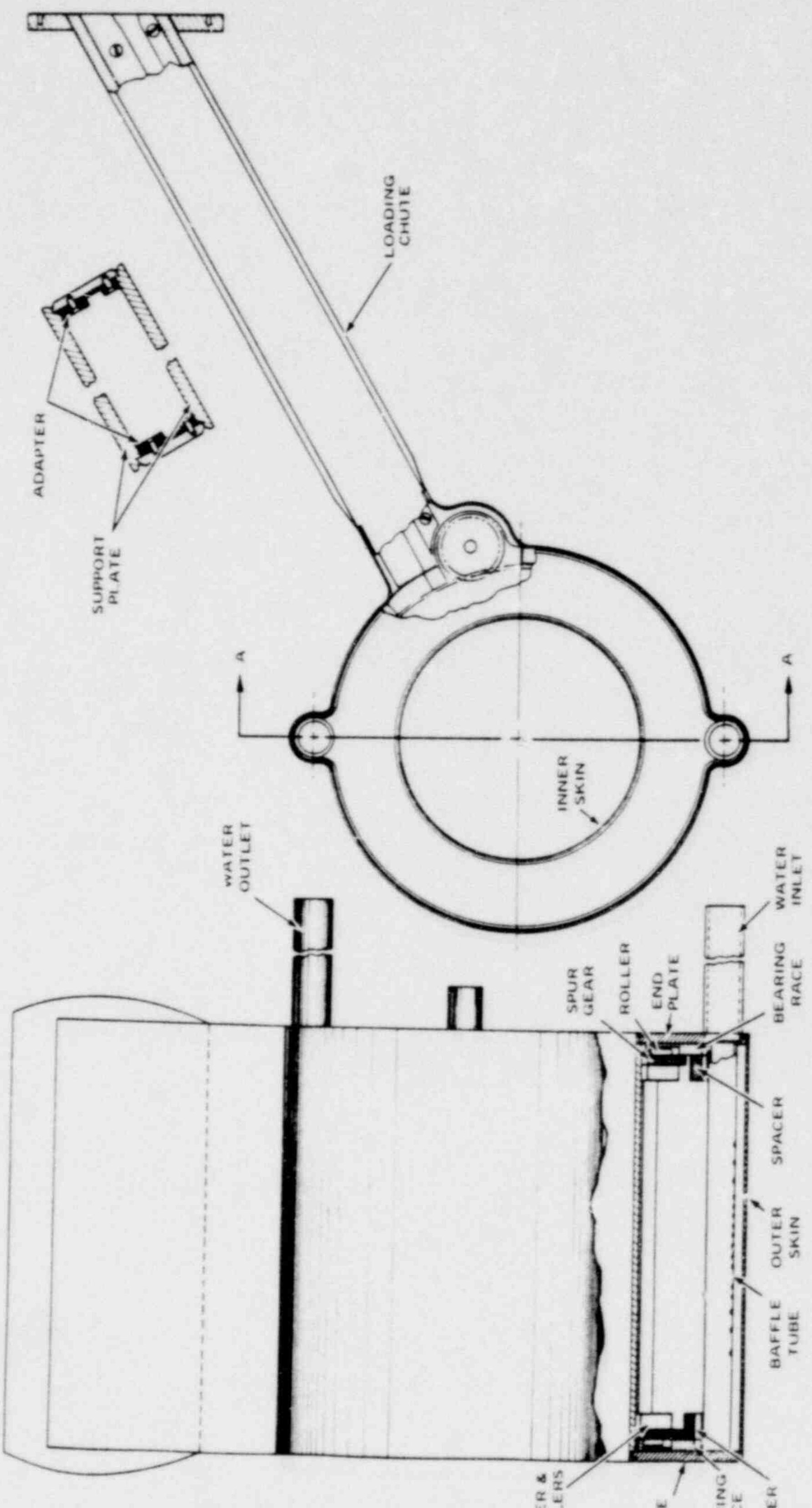


Figure 4-2. Fuel Container Assembly

two holes on 0.75-in. centers; and the last two holes on 1-in. centers. An identical baffle tube located above the core (with holes on the top side of the tube) is connected to the outlet line.

A 3.25-in. opening in each end plate accommodates the drive mechanism for the reel assembly. These openings are in the area just below the junction between the loading chute and the fuel container. This drive mechanism is discussed in Subsection 4.4.

Attached to the outer wall of the can, inclined upward at an angle of about 30 degrees with the horizontal, is a rectangular aluminum loading chute about 30 inches long, 20 inches wide, and 4 inches high. The chute is connected to the fuel loading tank. Slotted adapters fastened inside the chute provide a guide for the chute plug. The slots in the adapters line up with radial slots in the two circular raceways to guide the fuel loading tool to the core reel during refueling operations. When not in use, the loading chute is filled with an aluminum-clad graphite plug, and the chute opening is at least partially covered with an aluminum gate located in the storage tank.

Eight 0.75-in. aluminum tubes supported from brackets attached to the end plates run horizontally along the outside surface of the fuel container to the north face of the reactor. These tubes are guides for the control, safety, and neutron source rods. Six slotted graphite ways attached to the north end plate, oriented parallel to these guide tubes, serve as guides for the manually positioned poison sheets. The only other attachment to the container is a bracket fastened to the south end plate to help support the 5-in. horizontal facility.

The fuel container rests on the sections of the 5-ft graphite cube pack beneath it. The graphite was machined to close tolerance, then fitted around the container and loading chute to provide maximum support. An aluminum box constructed of 0.375-in. plate and 2-in. angle aluminum contains the 5-ft graphite cube on all faces except the bottom and the south face. The south face is joined to the 4-ft graphite cube thermal column. A 0.031-in. cadmium liner is provided for the north and east sides of the box. The box containing the graphite cube rests on a base consisting of a 0.625-in.-thick aluminum plate fastened to a framework of 5-in. aluminum I-beams. The I-beam base is clamped to steel support plates anchored to the reactor cell floor. At the time the reactor was installed, the base was shimmed level and grouted.

4.3 FUEL ASSEMBLIES

There are 16 fuel assemblies in the NTR core. Each fuel assembly consists of 40 fuel disks and spacers skewered on a shaft to form a shishkabob-type assembly. Lateral motion of the disks and spacers on the shaft is prevented by lock nuts placed on both ends of the shaft. All available spaces in the core support reel are filled by the 16 assemblies. There are 640 fuel disks in the core. These 640 fuel disks contain approximately 3.6 kg U-235. All the fuel disks in the core are from the original fuel load fabricated in 1957. When the fuel container was replaced in 1976, the fuel was removed, inspected, and leak checked. No cleaning, replacement, or repair was necessary.

Each space between the disks contains a 0.180-in.-thick aluminum spacer, and an additional 0.031-in.-thick aluminum washer is located in every other space. This arrangement produces an assembly with an active length of approximately 15.25 inches with the face-to-face distance between disks alternating between 0.24 and 0.27 inch.

Each fuel disk (Figure 4-3) is composed of a fuel bearing, flat, doughnut-shaped sandwich and an inner and outer edge ring. The three pieces were brazed together to clad the uranium-aluminum alloy meat. The sandwich consists of the uranium-aluminum alloy meat, which contains, on the average, 6.06 g of high-enriched uranium (approximately 93%) after 2.76% depletion as a 23.5 wt % alloy, plus 0.027 inch of 1100-series aluminum cladding on each face. The finished, flat, doughnut-shaped sandwich is 0.142 inch thick and has a 2.68-in. o.d., with a 0.58-in.-diam center hole. The inner edge ring (0.516-in. inside diameter (i.d.), 0.033 inch thick, and 0.20 inch wide) fits into the center hole of the disk and is brazed to the faces of the sandwich cladding. A 2.75-in. o.d. outer edge ring, with the same width and thickness as the inner ring, fits around the circumference of the disk and is also brazed to the faces of the sandwich.

A 0.75-in. length of each end of the 0.5-in. aluminum support shaft is machined to provide a tip suitable for supporting and positioning the fuel assembly accurately in the core reel. Tolerances on the shafts, reel, and fuel container

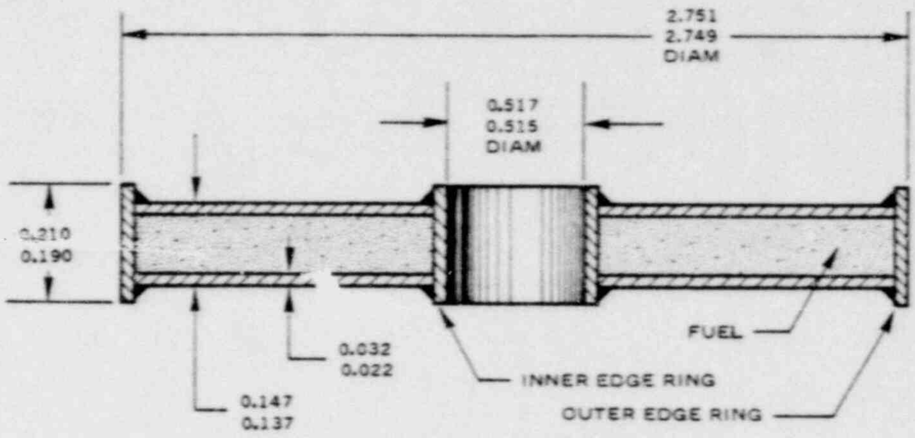


Figure 4-3. Fuel Disk

were set so that the maximum radial and circumferential movements of the shaft, and hence a fuel assembly, are less than 0.125 and 0.016 inch, respectively. The support tip extends past the ends of the core reel about 0.375 inch into the raceway; it is this section of the fuel assembly that is engaged by a tool during fuel handling.

4.4 CORE SUPPORT STRUCTURE AND ROTATING MECHANISM

Located within the fuel container can is the core reel assembly, which consists of a pair of spur gears tied together with eight separator bars. Radial slots in these spur gears receive the machined tips of the fuel assembly shafts to support and position each fuel assembly. Stainless steel rollers attached to the outer face of each gear guide and support the reel in the radial raceways attached to the fuel container end plates (Figure 4-2). A reel drive mechanism is provided to rotate the entire reel assembly to any desired position with respect to the loading chute.

The two large spur gears are almost identical. These gears, made of 0.34-in.-thick aluminum, have a 16.3-in. o.d. and a 12.9-in. i.d. Eight stainless steel rollers are bolted to the outer face, and eight triangular-shaped separator bars are bolted to the inner face (through a 0.75-in.-thick spacer ring) of each gear. The roller and raceway on the north end are V-shaped to prevent lateral motion of the entire reel. Sixteen equally spaced slots, 0.189 inch wide, are cut into each gear to receive the machined tips of the fuel assembly shafts. These radial slots terminate at an inner radius that places the center of fully inserted fuel elements on a 7.48-in. radius.

The reel drive assembly consists of two pinion gears (3-in. o.d. by 0.34 inch thick) keyed to a single 0.625-in. shaft. The shaft seal is a double O-ring seal with a tattle-tale petcock. Outside the reactor shield the shaft extends through a right-angle gear box, to the top of the fuel loading tank, to a hand-operated drive wheel with a dial indicator. The dial indicates the orientation of the reel and the position of any fuel assembly. The reel may be rotated to any desired position for core work; once the work is complete, the reel is no longer moved. Movement of the reel assembly is permitted only when the reactor is shut down. Since the reel can be rotated only from within the reactor cell and is locked in position, unauthorized or unintentional movement during reactor operation is not considered credible.

4.5 GRAPHITE REFLECTOR-MODERATOR

The graphite reflector-moderator is a 5-ft cube of reactor-grade (AGOT) graphite which not only serves as a neutron reflector and moderator, but also physically supports the fuel container. The fuel container is centered in the reflector with the core cylindrical axis horizontal. Many small pieces (primarily 4 inches by 4 inches by varying lengths) were machined carefully and stacked together to form the 5-ft cube. The reflector is contained and supported by the aluminum box and base discussed in Subsection 4.2.

Among the numerous items penetrating the reflector are (1) the fuel loading chute through the west face, (2) the control rod, safety rod, and neutron source guide tubes, (3) the manually positioned poison sheet slots, and (4) the core reel drive shaft through the north face. The horizontal facility tube and the experiment tube traverse the reflector from the north to the south face (the horizontal facility tube continues through the thermal column); the vertical facility extends from the top to the bottom of the reflector. Section 7 contains a discussion of the vertical and horizontal facilities and the thermal column. Several modifications were made to the main graphite pack during the major outage of 1976 and are also described in Section 7. The modifications added new irradiation facilities capabilities.

Two special sections of the reflector were designed to be removable: the set of blocks situated between the fuel container and the north face; and, the group of blocks that fill the 11.5-in.-diam hole formed by the inner skin of the fuel container. These sections make it possible to inspect the fuel container without disturbing the rest of the reflector.

4.6 SAFETY RODS, CONTROL RODS, AND MANUAL POISON SHEETS

Movable neutron absorbers located about the periphery of the fuel container include:

1. Two boron-carbide-filled motor-driven coarse control rods;

2. One boron-carbide-filled motor-driven fine control rod;

3. Four boron-carbide-filled safety rods, motor-driven carriage with an electromagnet that attaches to the poison section, and scram force by cocked springs; and

4. Up to six poison sheets (aluminum sheets with cadmium inserts) manually positioned from within the cell.

Orientation of these rods about the core is shown in Figure 4-1. Detailed discussions of these poisons as well as the methods for positioning them are given in Subsection 8.5.

4.7 NUCLEAR CHARACTERISTICS

Normal operation of the NTR is at powers no greater than 100 kW, with maximum temperature and pressure in the core at about 150°F and 20 psia, respectively.

As a result of these very conservative operating conditions, none of the nuclear characteristics (except the water moderator temperature coefficient of reactivity) varies significantly with normal temperatures.

Several important features of the NTR that affect the nuclear characteristics result from an effort to enhance the performance of the reactor as a sensitive detector of reactivity changes. Among these features are the low critical mass, the fuel-to-sample geometry, and sensitive control system. The reactor is constructed so that samples placed in the horizontal facility are in a neutron flux that is higher than the flux in the fuel lattice. The sensitivity of the reactor as a detector is proportional to the ratio of the thermal flux at the sample to that in the fuel lattice. A number of nuclear parameters are listed in Table 4-1.

Figures 4-4 and 4-5 illustrate the thermal neutron flux profiles in the horizontal facility and in three of the manual poison sheet slots. The profiles in the manual poison sheet slots are the profiles of the three upper slots on the east side of the core and are expected to correspond very closely to the axial neutron flux and thermal power distribution in the adjacent section of the core.

Discussions of temperature coefficients of reactivity usually separate the total coefficient into a nuclear cross section effect and an effect caused by density and volume changes in the system. These two major effects are subdivided further according to the location of material that is affected (i.e., fuel, moderator, or coolant) and the speed with which the effect occurs. For an NTR-type reactor, such a complete breakdown is not necessary. By far, the dominant effect for accident analysis is that of density changes, including displacement of cooling water by expansion of fuel within the fuel annulus. Although the results of earlier studies indicate that a positive effect may result from heating the reflector graphite, this temperature change would be too slow (on the order of minutes) to affect a nuclear excursion significantly. The effect from a temperature change in the fuel annulus is observed in fractions of seconds. The over-all temperature coefficient of the fuel annulus was measured and found to be positive up to 124°F. As temperature is increased above 124°F (the turnover point), the coefficient becomes negative. The coefficient was measured between 65 and 156°F, and, for analyzing accidents, it

Table 4-1

NUCLEAR PARAMETERS

Fuel Loading	
Critical mass (cold, 0.28 inch between disks)	3.0 kg U-235 (512 disks)
Actual initial loading	3.7 kg U-235 (640 disks)
Actual loading after 80 MWd of operation (2.76% depletion)	3.6 kg U-235
Reactivity Worth of Movable Nuclear Poisons (Subsection 8.5)	
All three control rods (typical[a] operational core)	0.016 Δk/k (2.3$)
All four safety rods (minimum)	0.014 Δk/k (2.0$)
All six Manual Poison Sheets (MPS)	0.021 Δk/k (3.0$)
Reactivity (Console Excess with Typical[a] Operational Core)	
All four safety rods and three control rods withdrawn	+0.002 Δk/k (+0.3$)
All four safety rods withdrawn and all three control rods inserted	-0.014 Δk/k (-2.0$)
All four safety rods inserted and all three control rods withdrawn (minimum)	-0.012 Δk/k (-1.7$)
All four safety rods inserted and all three control rods inserted (minimum)	-0.028 Δk/k (-4.0$)
All four safety rods inserted and all three control rods inserted and all six manual poison sheets inserted (minimum)	-0.043 Δk/k (-6.1$)
Reactivity (Console Excess)	
All four safety rods and all three control rods inserted and all manual poison sheets withdrawn	-0.022 Δk/k (-3.1$)
Reactivity Addition from Primary Coolant Temperature change (from 75 to 124°F)	+0.00048 Δk/k (+0.07$)

[a] 1/2 MPS in slot #1, 3/8 MPS in slot #5, neutrography source log in horizontal cavity, graphite in vertical and other experiment facilities (or similar arrangement); excludes temperature and experiment transient worth.

Table 4-1

NUCLEAR PARAMETERS (Continued)

Miscellaneous Reactivity Effects	
Removing all graphite from central sample tube (3-in. cavity)	-0.004 $\Delta k/k$ (0.6\$)
Filling central sample tube with water (3-in. cavity)	-0.02 $\Delta k/k$ (-3\$)
Removing all graphite from vert cal facility	-0.008 $\Delta k/k$ (-1.1\$)
Removing the fuel loading chute plug	-0.009 $\Delta k/k$ (-1.25\$)
Equilibrium xenon at 100 kW	-2.3×10^{-3} $\Delta k/k$ (-0.3\$)
Yearly fuel burnup (typical use)	-4.6×10^{-4} $\Delta k/k$ (-0.06\$)
1 g U-235 in small (25 g) low-enriched fuel samples in central sample tube	$\sim+2 \times 10^{-4}$ $\Delta k/k$ (-0.03\$)
1 cm^2 of absorber in small (50 g) nonfuel sample in central sample tube	$\sim-1.8 \times 10^{-4}$ $\Delta k/k$ (-0.025\$)
Coefficients of Reactivity	
Temperature coefficient in	
Water coolant (measured)	-5.7×10^{-3}(T-124)¢/°F
Inner graphite (calculated)	$+0.17 \times 10^{-3}$¢/°F
Outer graphite (calculated)	$+4.1 \times 10^{-3}$¢/°F
Average void coefficient	-5.7 ¢/% void
Doppler coefficient	Negligible
Mean Lifetime of Prompt Neutrons	2×10^{-4} sec
Neutron Fluxes at 100 kW	
Average thermal flux in fuel	7×10^{11}nv
Peak thermal flux in central sample tube	2.5×10^{12}nv
Thermal flux at face of thermal column	7×10^{8}nv
Thermal flux at face of 5-ft graphite cube	5×10^{10}nv

Table 4-1

NUCLEAR PARAMETERS (Continued)

Miscellaneous Parameters After 80 MWd of Operation	
Reactivity lost due to fuel burnup	0.0035 $\Delta k/k$ (0.6\$)
Reactivity lost due to aggregate fission product poisoning	0.0028 $\Delta k/k$ (0.4\$)
Reactivity lost due to samarium-149 poisoning	0.007 $\Delta k/k$ (1.0\$)

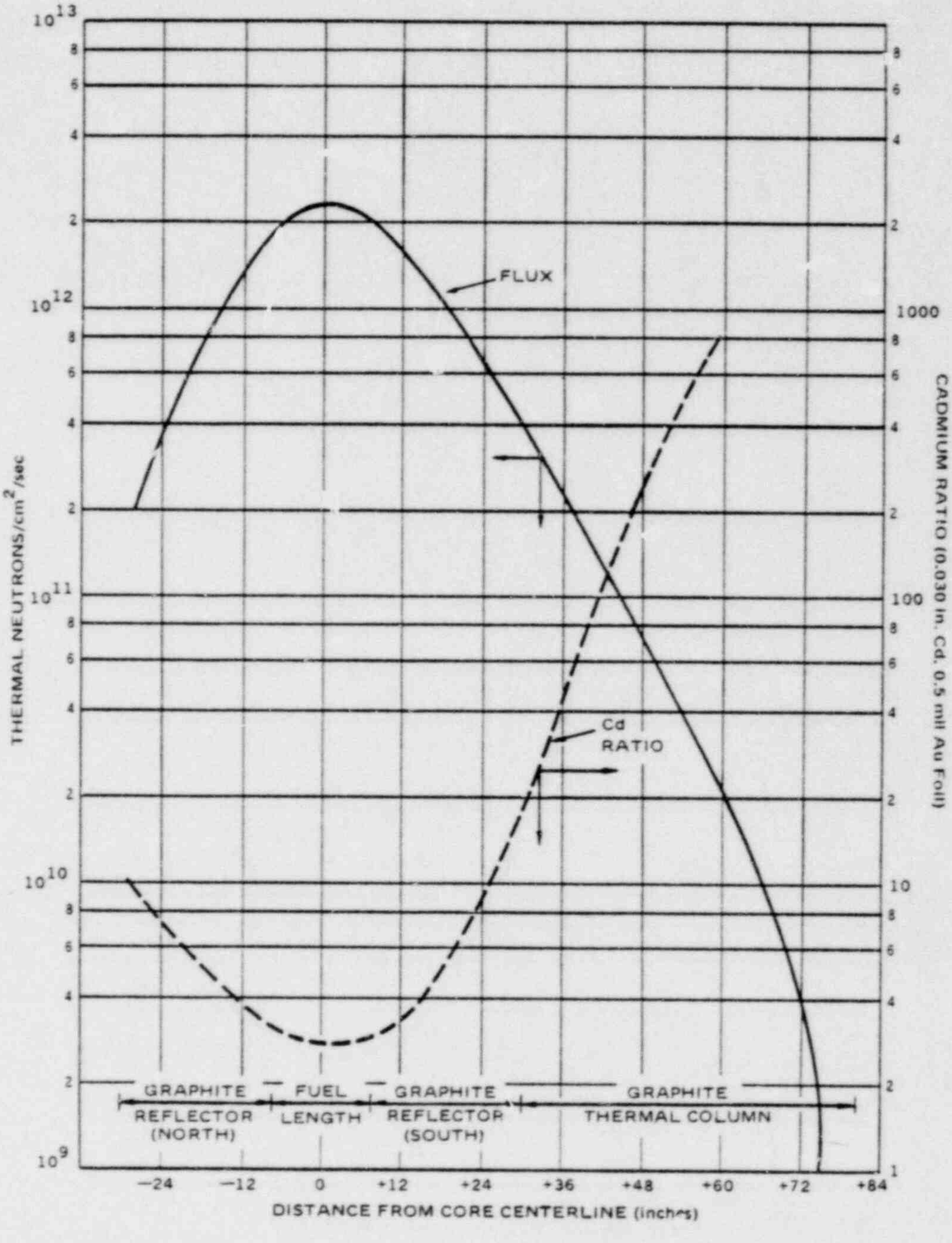


Figure 4-4. Thermal Neutron Flux and Cadmium Ratio Traverse of Horizontal Facility Reactor Power (100(kW)

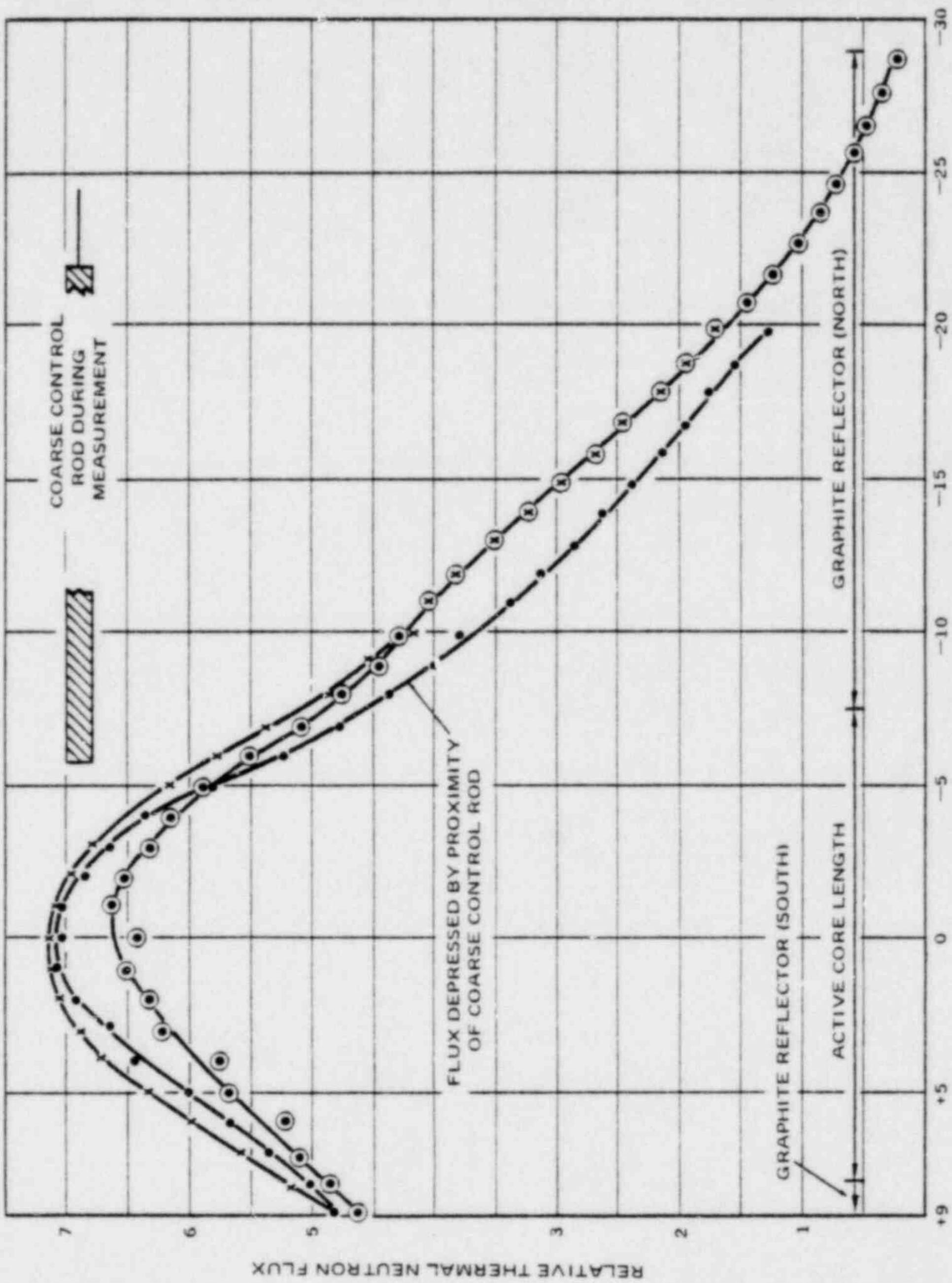


Figure 4-5. Thermal Neutron Flux Traverses in Three Manual Poison Sheet Slots

is assumed the data can be extrapolated to boiling. The measured coefficient is given by:

$$d\rho/dt = -5.7 \times 10^{-3}(T-124)\ ¢/°F \quad (1)$$

where T is the primary coolant temperature in (°F) and ρ is the reactivity of the system (¢). This coefficient is not affected by fuel burnup and is not expected to vary significantly with core life.

An experiment was performed to check the sign of the void coefficient of reactivity. In this experiment, the reactivity effect of moving pieces of aluminum from the core was positive; therefore, the void coefficient was negative, as required. The magnitude of the void coefficient was not measured directly, but was determined from the results of the temperature coefficient experiment. In this determination, the source of reactivity change in the temperature coefficient is considered to be caused by the density changes only and is interpreted as an effect from void buildup. Extrapolation of the temperature coefficient data yields a void coefficient of -5.7 ¢/% void above the temperature coefficient turning point of 124°F.

Low power generation of the NTR makes reactivity changes from fuel burnup and fission product poisoning small. During the past 23 years, the reactor has accumulated approximately 80 MWd of operation. Based on this history, the total reactivity losses are estimated to be 0.6$ from the fuel burnup, 0.4$ from aggregate fission product poisoning, and 1$ from samarium-149 poisoning.

Changes in reactivity caused by inserting materials during experiments are largest for experiments in the horizontal facility. Several measured reactivity effects in the horizontal facility and the vertical facility are given in Table 4-1. As indicated by the fact that the thermal column increases the flux at the south face of the reflector, experiments at the face of the 5-ft graphite cube, which contain large quantities of reflector materials, could have a small reactivity effect. However, during experiments performed to date, such an effect has never been observed.

4.8 THERMAL AND HYDRAULIC CHARACTERISTICS

Maximum authorized steady-state power for the NTR is 100 kW. High-power trips are routinely set at powers no higher than 138 kW and a core outlet high-temperature scram is set to ensure that the core outlet temperature is less than 222°F. For powers above 0.1 kW, forced circulation of deionized water is used to transfer the heat from the core to a heat exchanger, as described in Section 5. When forced circulation is required, the reactor will scram if flow is less than 15 gpm. At powers less than 0.1 kW, operation is permitted without forced circulation (i.e., the primary recirculation pump need not be operating and the low-flow scram is bypassed). The 0.1-kW limitation for natural circulation operation is extremely conservative (established in the past) but will continue to be used even though the analysis for the loss-of-flow accident described in Section 11 shows the core can be adequately cooled by natural circulation at much higher powers. Under both operating conditions, natural or forced circulation, the performance of the core is good with regard to the avoidance of natural thermal limits. These thermal limits include melting of the fuel and cladding, and burnout of the fuel cladding.

The maximum authorized steady-state operating power, 100 kW thermal with a rated recirculation flow of 20 gpm, has been used for the reactor to establish steady-state values for the thermal and hydraulic characteristics of the reactor core. A summary of these characteristics given in Table 4-2 shows that the thermal loading on the core is quite modest. The core inlet coolant temperature is typically 90°F; the core average exit temperature is 120°F, and, in the hottest channel, the exit temperature is only 150°F. The saturation temperature of the coolant corresponding to the average reactor pressure, 20 psia, is 228°F. Thus, the state of the coolant is far removed from boiling at the design operating condition.

The cladding surface temperatures were established on the basis of known coolant temperatures and the heat flux distribution in the core. The flow through the core is laminar, and the surface film heat-transfer coefficients were calculated from a known laminar correlation. Fuel-plate temperatures increase with power up to a certain point; however, when the surface temperature is elevated to a value that will support local boiling of the coolant, the heat-transfer mechanism undergoes a marked change. There is substantial

Table 4-2

TYPICAL NTR CORE THERMAL AND HYDRAULIC CHARACTERISTICS

Maximum thermal power level (scram)	138 kW
Maximum steady-state thermal power level	100 kW
Average fuel disk surface heat flux	6600 Btu/h-ft^2
Maximum fuel disk surface heat flux	10600 Btu/h-ft^2
Total fuel to coolant heat transfer area	52.7 ft^2
Total core power peaking factor	1.58
Core average pressure level	20 psia
Coolant flow characteristic	
Total core flow area	0.39 ft^2
Channel flow area	0.70 $in.^2$
Channel hydraulic diameter	0.51 in.
Total recirculation flow rate	20 gpm (9800 lb/h)
Inlet velocity, average channel	0.14 ft/sec
Inlet velocity, hottest channel	0.07 ft/sec
Mass flow rate, average channel	122 lb/h
Mass flow rate, hottest channel	64 lb/h
Coolant inlet temperature	90°F
Coolant exit temperature, average channel	120°F
Coolant exit temperature, hottest channel	150°F
Coolant saturation temperature	228°F
Fuel disk cladding temperature	
Average channel	170°F
Hottest channel	195°F
Maximum temperature difference, fuel-to-cladding surface	1°F

increase in the heat-transfer coefficient, and, consequently, the plate surface temperature is practically "held" at a maximum value, corresponding to the value needed to establish local boiling. The Jens-Lottes correlation[7] was used to predict the local value of wall superheat necessary to establish local boiling. This phenomenon is important because metal temperatures are limited to values well below melting, which is particularly evident during certain accidental transients discussed in Section 11.

The core flow distribution out of the inlet header (described in Subsection 4.2) is such that adequate cooling of all portions of the core is achieved. The pipe is orificed to give higher-than-average flow rates in the horizontally central region of increased power generation, and lower-than-average flow to the end regions.

The peaking factors used in this evaluation were maximum expected values that result from operation of the reactor with neutron flux peaked on one side of the core. The circumferential power distribution used resulted in a circumferential power peaking factor of 1.25. The longitudinal shape is symmetrical, with a total axial peaking of 1.15. The total over-all power peaking in the core is 1.58, which includes a local peaking factor of 1.1.

Of considerable importance is the ability of the recirculation systen to maintain a mode of natural circulation flow when the primary pump is not operating and core power is up. In the absence of pump head, the driving pressure difference around the recirculation loop is the net elevation head of the coolant. This is directly proportional to the density differences between the water in the core and riser section and the water leaving the heat exchanger. Again, this density difference is a function of core power. The length of piping over which this density difference exists is slightly more than 5 feet. System response to loss or recirculation pumping is discussed in Section 11.

Fuel plate steam blanketing is a condition that may occur even in a pressurized-water system and can be of considerable concern. This condition is caused by going from local surface boiling into film boiling upon reaching very high surface heat fluxes. This could be of concern because the steam film degrades the heat transfer, and the fuel plate temperature increases greatly as a result. However, during steady-state operation, this is of no real concern in the NTR for these reasons: Heat fluxes at maximum power in the reactor are quite small because of the low power rating, and the burnout heat fluxes, or the heat flux necessary to cause steam blanketing, are very high for the coolant conditions existing in the reactor, as evidenced by experimental data. For instance, in the hottest channel in the core, the data indicates a burnout heat flux of 227,000 Btu/h-ft^2 for the hydraulic conditions at which the channel is operating. The actual maximum heat flux in this channel, for 100-kW operation, is 10,300 Btu/h-ft^2. Thus, the burnout ratio, or the ratio of burnout heat flux to maximum operating heat flux, is 22. This is a considerable margin and represents a highly safe condition.

5. COOLANT SYSTEMS

5.1 GENERAL

Forced circulation of deionized light water cools the reactor core at powers above 0.1 kW; below 0.1 kW, no coolant circulation is required for typical operating periods, although it may be utilized if desired. The primary water system removes the heat from the core and transfers it through a heat exchanger to the secondary water system. Figure 5-1 shows the NTR coolant systems. Typical conditions for reactor power of 100 kW are: 35 gpm secondary water, 20 gpm primary water, 90°F core inlet temperature, and 120°F core outlet temperature.

5.2 PRIMARY SYSTEM

The entire primary system (Figure 5-2) is located in the reactor cell. Practically all piping in the system is 1.5-in., Schedule 40 aluminum. The internal parts of equipment, such as the pump and heat exchangers, are stainless steel. Water flows from the pump, through a check valve and throttle valve, to the core; from the core to the heat exchanger, it flows through the air trap (originally an in-line heater tank) and suction valve back to the pump. Other piping considered to be part of the primary system is installed to provide for makeup, cleanup, over-flow, draining, venting, and calibrating. Instrumentation for the system is discussed in Section 8. The primary system contains about 28.5 gallons, of which 9 gallons are in the fuel container. The primary source of system leakage is the pump seal, which averages less than 1 gallon per week.

The primary pump is a centrifugal pump driven by a 1-hp electric motor and rated at 25 gpm with discharge head of 30 feet. The pump is mounted on the reactor base north of the control rod drive support plate. Electrical control stations for the motor are provided locally at the pump and remotely at the reactor console.

Water enters the fuel container beneath the core, flows up through fuel assemblies in each one-half of the fuel container, and exits from above the

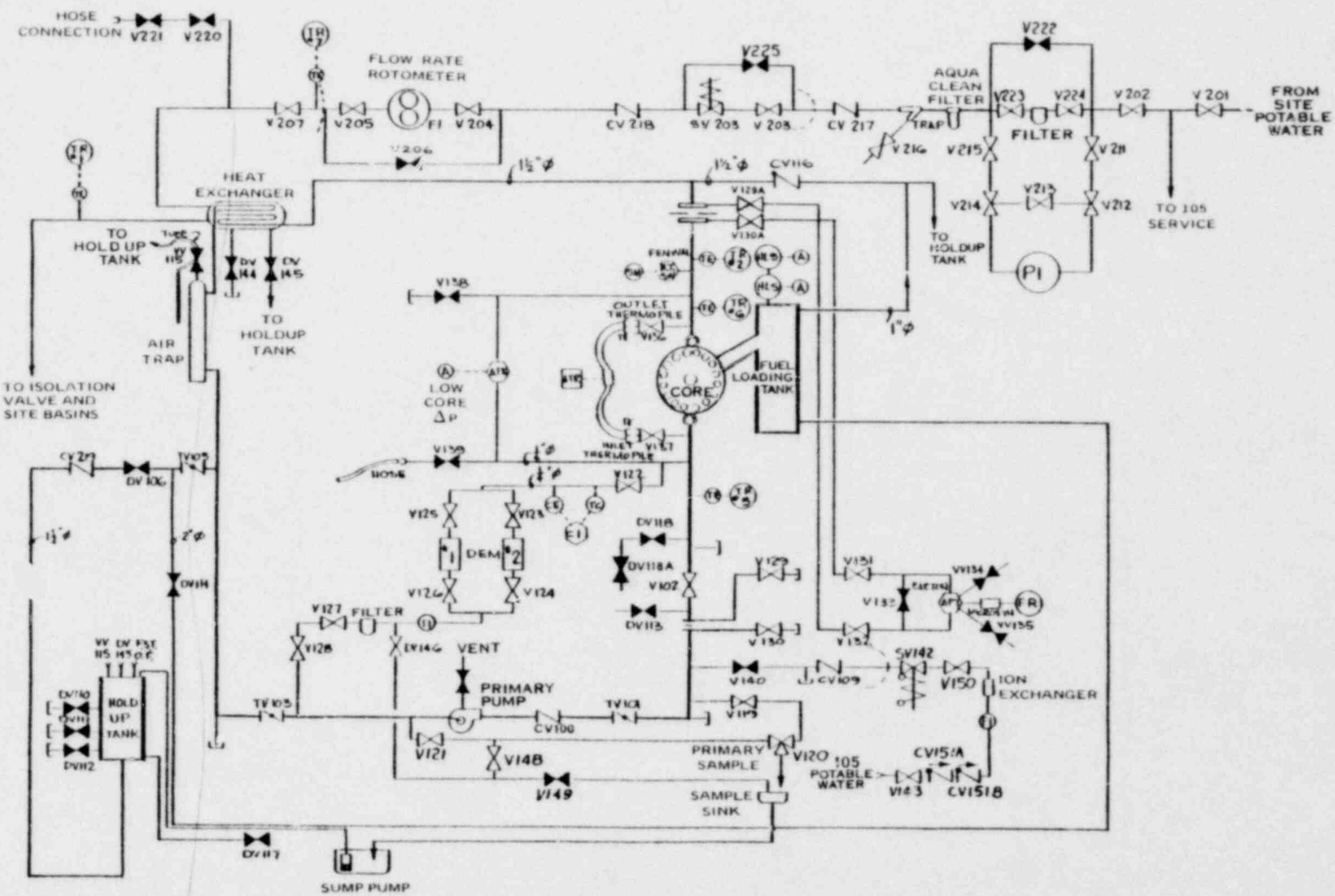


Figure 5-1. Nuclear Test Reactor Coolant Systems

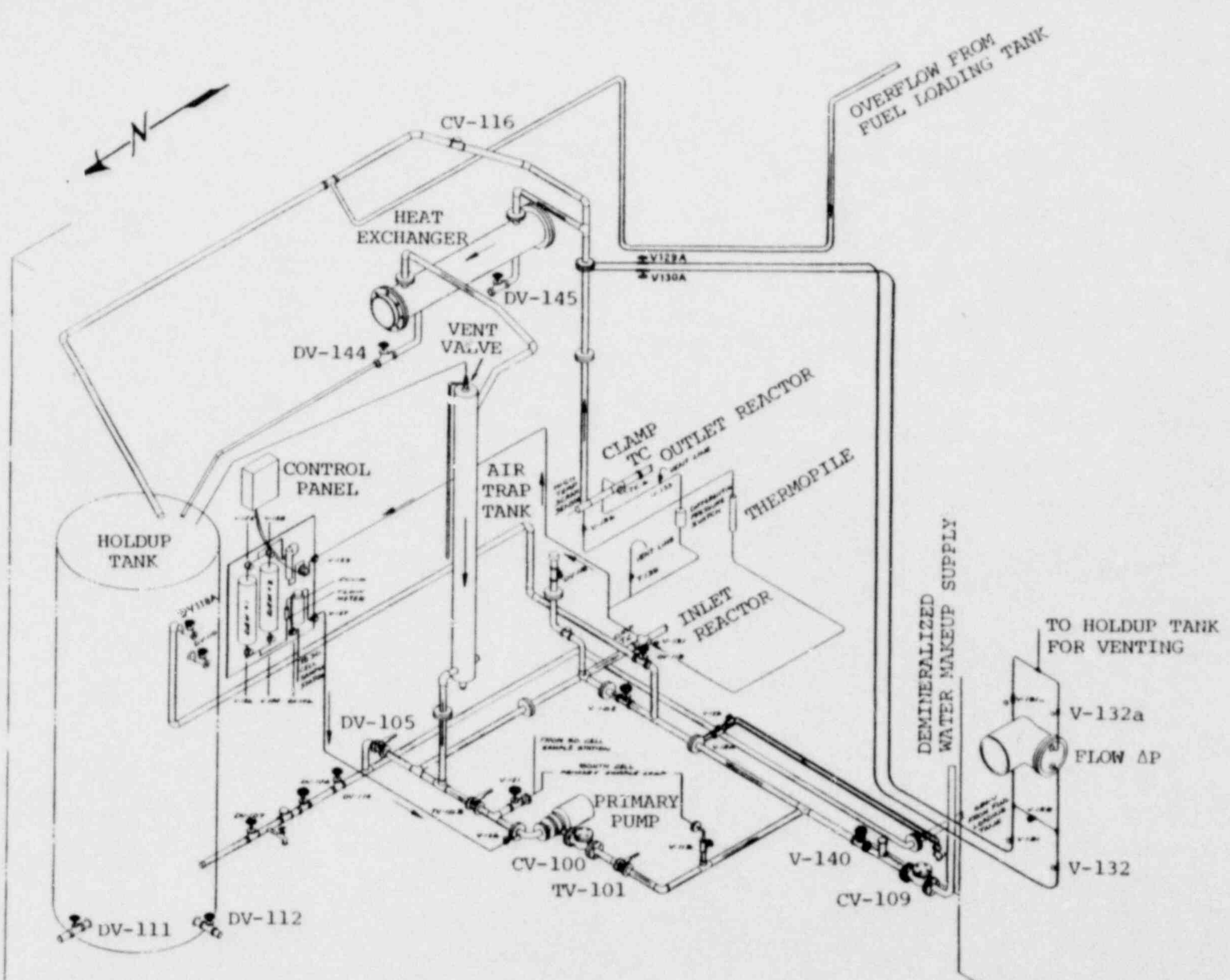


Figure 5-2. Primary Coolant System

core; both inlet and outlet are provided with flow distributors described in Subsection 4.2. Normal pressure drop across the core is about 1.5 psig.

Adequate heat-transfer capability is provided by a single two-phase heat exchanger. The unit is designed to eliminate expansion strains from temperature differential to a greater extent than is possible in the NTR system. Primary water flows through the shell side of the two-phase fixed tube and shell heat exchanger. The heat exchanger is capable of transferring $3.4 \times 10^{[illegible]}$ Btu/h. The tube bundle consists of 0.25-in.-diam, 36-in.-long stainless steel tubes, and provides a heat-transfer surface of approximately 36 ft^2. The heat exchanger is located on the east wall of the reactor cell about 6 feet above the reactor core. The design specifications pertinent to maintaining system integrity are given in Table 5-1.

The air trap tank is a 5-ft length of 4-in. aluminum pipe that originally contained Cal-Rod-type heaters rated at 5 kW. The heaters have been removed from the system but the tank remains and is utilized as a system air trap. A

Table 5-1

HEAT EXCHANGER SPECIFICATIONS

	Shell Side	Tube Side
Fluid (coolant)	Primary	Secondary
Fluid flow rate (gpm)	20	35
Fluid velocity (ft/sec)	1.48	3.43
Temperature in (°F)	124	60
Temperature out (°F)	90	80
Pressure drop (psig)	1.6	4.0
Design Pressure (psig)	300	150
Test pressure (psig)	500	300
Design temperature (°F)	400	400
Material (stainless steel)	316	316

bypass cleanup system consists of two mixed bed deionizers and a flowmeter. If both units are used, the system will service 32 gph. Resins are contained in replaceable cartridges. Primary water conductivity is indicated in the reactor cell. It is recorded at appropriate intervals to ensure that any abnormal contamination or corrosion is detected. If these conductivity readings indicate a significant decrease in the purity of the water, grab samples will be taken and analyzed if appropriate to determine the cause, and appropriate corrective action will be taken. Any significant decrease in water purity probably would be accompanied by an increase in the radiation level from the deionizer. An increase in this level would be detected by routine surveys.

Makeup water to the primary system is from the Building 105 deionizer unit which contains two check valves. The makeup line, which contains a solenoid-actuated valve outside the reactor cell, enters the reactor cell through the floor and contains an additional check valve and a manual shutoff valve inside the cell. This line is connected to the primary system between the primary pump throttle valve and the core inlet. The makeup system also supplies the 1800-gal fuel loading tank, which serves as a reservoir for the primary system. The fuel loading tank is discussed in Subsection 6.1.

Primary water may be drained to the 500-gal holdup tank in the northeast corner of the cell where it can be retained or transferred to other tanks for transfer from the facility. The holdup tank also receives the discharge from the primary system atmospheric vent line, which is connected to the inlet of the heat exchanger. This overflow line is the highest point in the system and provides a continuous vent to atmosphere for air and other gases and prevents over-pressurizing the primary system. An overflow line from the fuel storage tank connects into the primary system atmospheric vent line. A sump pump is located in a sump which is in the northwest corner of the cell. Any water collected in this sump is automatically pumped into the 500-gal holdup tank.

Primary water samples can be taken at the sample station located in the northwest corner of the south cell. Water may be sampled either before or after the deionizers.

5.3 SECONDARY SYSTEM

Secondary cooling water (Figure 5-3) for the NTR comes from the Building 105 potable water supply, which is fed from the Site raw water main supplied from the Site's 500,000-gal storage tank. The 1.5-in. supply line to the NTR facility passes through a filter and then enters through the ceiling of the control room, where it passes through a shut-off valve, a check valve, and a flow indicator before entering the reactor cell. Inside the cell, the line goes directly to the tube side of the heat exchanger and then through a manual valve to the facility drain, which discharges to the Site retention basins. Pressure at the inlet of the heat exchanger is normally about 50 psig.

The probability and consequences of a leak between the two systems in the common heat exchanger have been evaluated. The evaluation showed that the probability of leaking contaminated water from the primary to secondary system is extremely low; furthermore, should such leakage occur, the contaminated water would drain to the Site retention basins. The basin water is analyzed for radioactive material content before it is released.

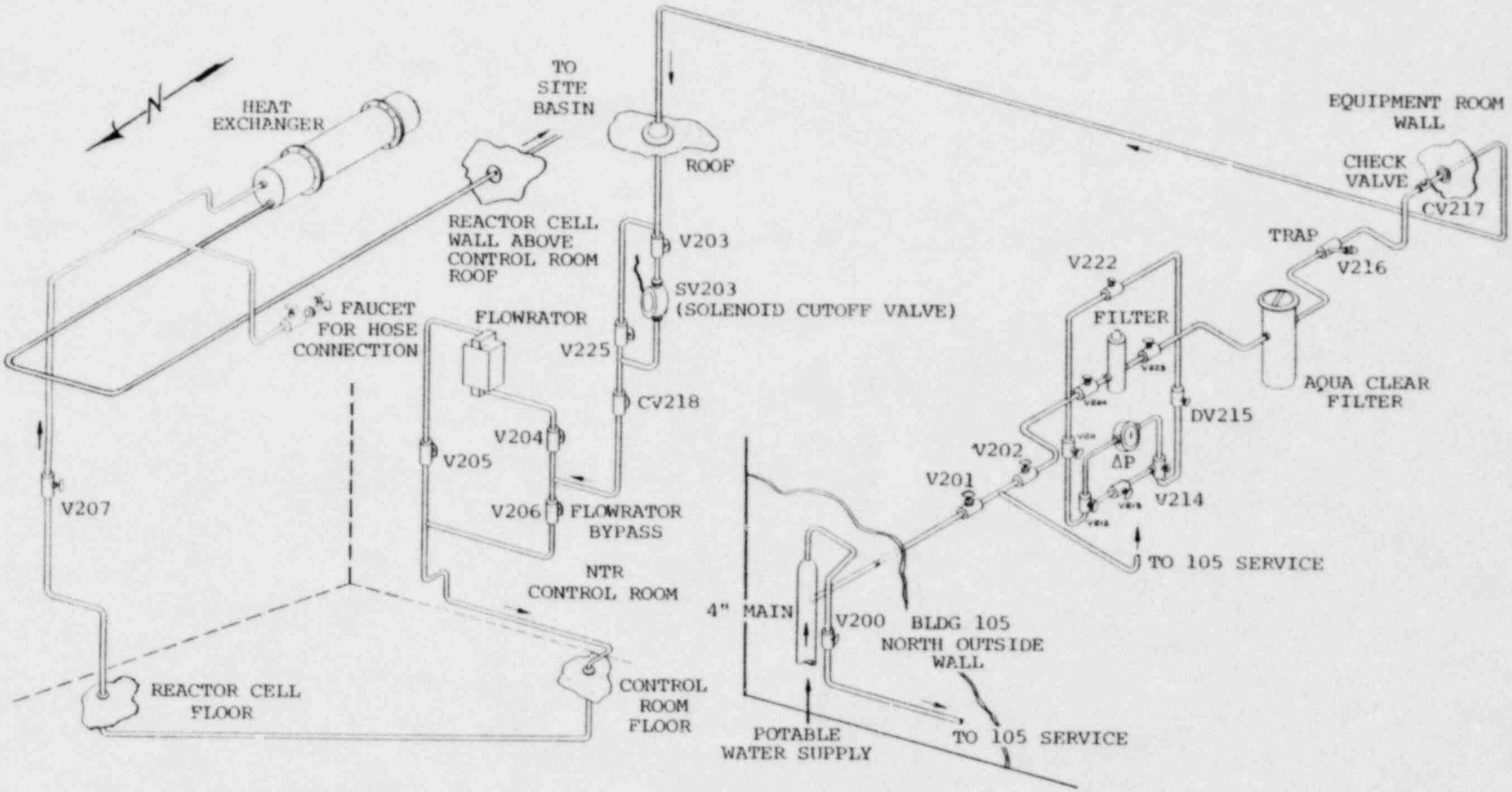


Figure 5-3. Secondary Cooling System

6. AUXILIARY SYSTEMS

6.1 FUEL HANDLING SYSTEMS

It is anticipated that only a trivial amount of fuel handling, confined to inspection of the fuel presently on hand, will be done at the NTR. However, facilities are available to perform any fuel handling operations that might become necessary in conjunction with operation of the reactor. A handling tool is provided for remote underwater transfer of the assemblies between the core and the fuel loading tank. Special arrangements may be made to use transfer equipment and storage facilities elsewhere on the Site if it becomes necesary to remove more than one element from the reactor. Arrangements can be made to transfer irradiated fuel away from the NTR facility. Proper authorization would be obtained before such transfers were made, and procedures would be developed to ensure safe handling with adequate consideration for radiation protection and criticality control.

The fuel loading tank, approximately 12 feet high by 5 feet long by 4 feet wide, is located adjacent to the west face of the graphite reflector. An expansion joint connects the west end of the reactor fuel loading chute to the east side of the tank. An aluminum gate for the loading chute is attached to the inside of the east wall of the tank and is normally in a partially closed position. A loading platform which essentially extends the loading chute into the tank may be attached to the inside of the east wall of the tank, as required. Affixed to the opposite tank wall is a storage rack for the fuel loading chute's aluminum-covered graphite plug. A pulley for the plug cable-lift is attached to the storage rack. Two 4-in.-diam and one 2-in.-diam aluminum thimbles installed in the tank are used to hold detectors for reactor nuclear instrumentation or samples for irradiation in a low-flux region. Level switches indicate high and low water level in the tank by energizing annunciator lights at the console. Access to the loading tank is from the reactor cell mezzanine. With a normal level in the tank, there is about 5 feet of water for shielding above the loading platform.

6.2 ELECTRICAL POWER SYSTEMS

A 480-V load center in Building 105 is fed from the Site's 12-kV bus, and, in turn, feeds power and lighting distribution panels for the NTR facility. The arrangement described below is the one now in existence, but future changes may be made if they are not detrimental to reactor safety. Two 220/115-Vac circuit breakers in the control room feed individual breakers for the primary coolant pump, service outlets, facility lights, and for the reactor console. Power supplied to the console is used for the reactor instrumentation and the control and safety rod drive motors. A 24-Vdc power supply is provided for operating relay circuits in the reactor safety and control systems.

Semiportable emergency lighting units are installed at several locations in the facility. Each unit contains a battery that maintains its charge from a 15-Vac system. On loss of ac power, the units are energized automatically to provide light for emergency action, personnel exit, etc.

6.3 FACILITY AIR SYSTEM

Compressed air for the facility is supplied from the Building 105 service air compressor located in the second-floor mechanical equipment room. The compressor will deliver 50 scfm of free air and is capable of a discharge pressure of 100 psig. A relief valve at the air compressor maintains system pressure at less than 120 psig. Air is supplied to a breathing-air purifying system mounted in the NTR control room. High temperature and low pressure sensors give both visual and audible alarms in the control room. Regulated output of the purifier will supply a manifold inside the reactor cell entranceway. The manifold contains four regulators that supply four individual hose reels, also mounted in the reactor cell entrance. In addition, compressed air is supplied to the air piston operator for the south cell door and to an air-operated shutter used for radiation shielding for the south radiation beam, which can be emitted from the horizontal facility. Conveniently located outlets are provided to supply experiment equipment or for service air.

6.4 FIRE PROTECTION SYSTEM

Fire protection equipment and procedures for the NTR are similar to conventional industrial plant fire protection equipment and procedures established for the existing Site. Equipment, buildings, procedures, etc., are in accordance with company-wide standards, state and local regulations, and the recommendations of insurance agencies.

Six-in. fire mains, which are legs of a loop surrounding Buildings 102 and 105, are located on the east and west sides of Building 105. These mains supply outdoor fire hydrants located at the northeast, southeast, and southwest corners of Building 105, and an extensive sprinkler system located within the building. Fire hoses and nozzles are kept permanently located in the Building 105 hallway and the southeast corner of the building. The 500,000-gal Site raw water storage tank, located approximately 130 feet higher than Building 105, is the source of water for the fire mains; 100,000 gallons of this water is reserved for fire protection. In addition to the water system, conventional portable fire extinguishers are located throughout the NTR facility and Building 105.

A Site-wide fire brigade, under the direction of the VNC Industrial Safety Specialist, provides fire protection for the entire Site. The brigade members are trained in the use of the Site fire truck and other fire fighting equipment. All members are familiar with work under radiation conditions and requirements of the Site instructions for radiation protection.

6.5 RADIOACTIVE SOLID WASTE SYSTEM

The NTR produces the typical variety of radioactive solid waste resulting from the operation and maintenance of a research reactor. Based on experience to date, most of the waste consists of slightly contaminated rags, paper, and other clean-up-type materials. Lesser contributors include the clean-up resins and small pieces of equipment or tools that cannot be decontaminated economically. Occasionally, there may be a large piece of equipment or volume of construction material that requires disposal.

Solid wastes generated at the NTR are handled by one or more of the several methods, procedures, and controls established for the entire Site. Waste is normally packaged promptly, labeled, and transferred to the Site waste facilities or shipped to approved disposal areas; however, waste may be temporarily stored at the NTR until transferred to a waste disposal contractor or to the Site waste facilities. Shipping containers for waste are licensed, inspected, and surveyed to ensure they meet all applicable specifications and regulations governing the packaging, labeling, and shipment of radioactive waste. Off-site disposal is by an authorized agency.

6.6 RADIOACTIVE LIQUID WASTE SYSTEM

Normally, radioactive liquid from the NTR is transferred to the reactor cell hold-up tank, which is vented to the filtered ventilation exhaust system. If sufficient quantities of liquid accumulate in the reactor cell hold-up tank, they may be transferred to portable holding tanks. The liquid in these tanks would be subsequently disposed of using established site practices and procedures. Subsequent handling of this liquid may include chemical treatment, storage at other facilities, concentration or dilution, transfer for disposal to approved disposal areas, or evaporation.

6.7 REACTOR CELL VENTILATION

Potentially contaminated air and other gases are collected in the reactor cell ventilation system and passed through absolute filters before being discharged from the NTR stack.

An air-monitoring system provides continuous indication of the concentration of radioactive material in the ventilation effluent and energizes an alarm at the reactor console if the concentration reaches a set point which has been selected to ensure that the airborne release does not exceed established limits (Table 6-1). Separate detection channels and alarms are used for particulate material and for nonfilterable radioactive gases. A continuous sample is drawn from the discharge of the NTR ventilation stack and passes through the particulate detector, a charcoal cartridge, the nonfilterable

Table 6-1

STACK RELEASE RATE LIMITS

Isotope Group	Annual Average (μCi/sec)	Short Term (μCi/sec)
Halogen, >8d, $T_{1/2}$	$(2.0x10^7)x(MPC_a)$	$(1.4x10^9)x(MPC_a)$
Particulate, >8d, $T_{1/2}$		
Beta-gamma	$(2.0x10^7)x(MPC_a)$	$(1.4x10^9)x(MPC_a)$
Alpha	$(2.0x10^7)x(MPC_a)$	$(1.4x10^9)x(MPC_a)$
All other (including noble gas)	$(1.4x10^{10})x(MPC_a)$	$(1.0x10^{12})x(MPC_a)$

Note: (MPC_a is the Maximum Permissible Concentration in air allowed in an unrestricted area. These values are listed in Appendix B, Table II, of 10 CFR 20.

radioactive gas detector, flow control valve, and a central vacuum pump, and is released through the Building 105 NTR stack at an elevation of about 45 feet above the ground. Particulate materials are collected on a high-efficiency filter paper and their emissions measured with a shielded Geiger-Müller detector. Nonfilterable radioactive gases are detected by an internal gas flow ionization chamber with a relatively high sensitivity for beta emitters. Current from the chamber is measured by a picoammeter. Each channel is recorded on a strip chart recorder. The charcoal cartridge and particulate filter are changed periodically (normally weekly) and counted by the VNC Counting Lab for I-131 and gross β-γ and α, respectively. The stack sampling/monitoring system is discussed in detail in Appendix A.

The annual average dilution factor from the NTR stack to the Site boundary based on 1976 and 1977 meteorological conditions and a stack flow rate of 3000 cfm equals approximately 20,000; that is, the concentration at the Site boundary of any release from the NTR stack will be no greater than 1/20,000 of the concentration at the stack when averaged over one year.

The beta-gamma particulate monitor and the noble gas monitor are equipped with alarm capabilities. The purpose of these alarms is to alert operations personnel to abnormal releases of radioactive effluents from the stack. The normal operation releases are well below the annual average release limits; therefore, the alarm set points are conservatively selected at, or slightly below, the annual average release limits for the most restrictive isotopes (Sr-90 and Kr-87). These release levels are orders of magnitude below the short-term release limits. The alarm points are routinely set less than or equal to the values in Table 6-2 based on present operation releases.

The NTR stack release limits in Table 6-1 and the NTR stack monitor alarm points in Table 6-2 were derived as shown in Appendix A.

6.8 WORK AREA PARTICULATE SAMPLING SYSTEM

Fixed air sample stations are installed in the control room, shop cage, south cell, and the north room (three each). These filters are normally changed approximately every week, and are counted by the VNC Counting Lab. A fixed filter Continuous Air Monitor (CAM) is located in the control room to remotely monitor the reactor cell airborne beta-gamma particulate contamination levels, as required. The reactor cell has two fixed air sample stations that are not used continuously, but may be used, as required, as a backup, or in addition to the CAM. Flow through the fixed air sample stations and the CAM is provided by the Building 105 central vacuum system.

Table 6-2
STACK MONITOR ALARM POINTS

Monitor	Set Point
Beta-gamma particulate	1×10^{4} cpm
Noble gas	2×10^{-11} amps

7. IRRADIATION FACILITIES

7.1 GENERAL

The NTR was designed and constructed for use in the research, development, analytical, and commercial programs of General Electric and its customers. Special design features of the reactor make it especially useful as a detector of reactivity effects, as well as a versatile source of neutrons. Generally, the reactor is operated at powers up to 100 kW when used as a neutron source for beam experiments or sample irradiations, and at lower powers for use as a detector for reactivity effect experiments. As an indication of the wide versatility of the experiment facilities, some of the potential uses for the NTR include the following:

1. A variable level neutron source;
2. Reactor materials quality control (fuel and structural materials);
3. Reactor mockup experiments;
4. Cross section investigations;
5. Neutron detector foil calibrations;
6. Biological investigations;
7. Neutron beam experiments;
8. Food preservation studies;
9. Limited isotope production;
10. Radiation damage studies;
11. Personnel training;

12. Neutrography (neutron radiography) Service* - radioactive and nonradioactive materials; and

13. Neutron detector quality control.

During the past 23 years, the NTR has been used for most of the purposes listed above. Presently it is primarily used for neutron radiography of materials, neutron detector quality control, reactor materials quality control and as a variable level neutron source.

7.2 SAFETY CRITERIA

Safety criteria applicable to all irradiations in NTR experimental facilities include the following:

1. No object involved in the irradiation may be moved during reactor operation unless its potential reactivity worth is less than 0.5$ and the operation is performed with the knowledge of the licensed operator at the console. All remotely controlled mechanisms for moving an object into the reactor core must be energized through the reactor console; however, movement of the object may be initiated from another location.

2. No irradiation will be performed which could credibly interfere with the scram action of the safety rods at any time during reactor operation.

3. Experimental objects shall not be allowed inside the core tank when the reactor is at a power greater than 0.1 kW.

4. Experimental objects located in the fuel loading chute shall be secured during reactor shutdown to prevent their entry into the core region during reactor operation.

5. A maximum of 10 curies of radioactive material and up to 50 grams of uranium may be in storage in a neutron radiography area where

*Service Mark of the General Electric Company explosive devices are present (i.e., in the south cell or north room). The storage locations must be at least 5 feet from any explosive device. Radioactive materials, other than those produced by the neutron radiography of the explosive devices and imaging systems, are not permitted in the set-up room if explosive devices are present.

6. Unshielded high-frequency generating equipment shall not be operated within 50 feet of any explosive devices.

7. The cumulative radiation exposure for any explosive device shall not exceed 3×10^{12} n/cm^2 from thermal neutrons and 1×10^4 roentgens from gamma.

8. Experimental capsules to be utilized in the experimental facilities shall be designed or tested to ensure that any pressure transients produced by chemical reaction of its contents and/or leakage of corrosive or flammable materials will not damage the reactor.

9. Experimental fuel elements containing plutonium to be utilized in the experimental facilities shall be clad and other experimental devices containing plutonium shall be encapsulated.

10. The maximum possible chemical energy release from the combustion of flammable substances contained in any experimental facility shall not exceed 1000 kW/sec. The total possible energy release from chemical combination or decomposition of substances contained in any experimental capsule shall be limited to 5 kW/sec, if the rate of the reaction in the capsule could exceed 1 watt. Experimental facilities containing flammable materials shall be vented external to the reactor graphite pack.

11. The sum of the reactivity worths of all devices inserted into all irradiation facilities at any one time must not exceed 0.76$ when

combined with the reactivity effect of the primary coolant temperature coefficient (measured from core inlet) and potential excess reactivity due to the withdrawal of the control rods.

12. The radioactive material content, including fission products, of any singly encapsulated experiment to be utilized in the experimental facilities, shall be limited such that the complete release of all gaseous, particulate, or volatile components from the encapsulation could not result in doses in excess of 10% of the equivalent annual doses stated in 10 CFR 20. This dose limit applies to persons occupying (1) unrestricted areas continuously for 2 hours, starting at time of release or (2) restricted areas during the length of time required to evacuate the restricted area.

13. The radioactive material content, including fission products, of any doubly encapsulated or vented experiment to be utilized in the experimental facilities, shall be limited such that the complete release of all gaseous, particulate, or volatile components from the encapsulation or confining boundary of the experiment could not result in (1) a dose to any person occupying an unrestricted area continuously for a period of two hours starting at the time of release in excess of 0.5 rem to the whole body or 1.5 rem to the thyroid or (2) a dose to any person occupying a restricted area during the length of time required to evacuate the restricted area in excess of 5 rem to the whole body or 30 rem to the thyroid.

The above safety criteria are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure and serve as a guide for the review and approval of new and untried experiments by the facility personnel. A written description and analysis of the possible hazards involved for each type of experiment is evaluated and approved by the Facility Manager or his designated alternate before the experiment can be conducted. Records of such evaluation and approval are maintained.

7.3 EXPERIMENT TYPES

A few typical types of irradiations are described briefly in the following paragraphs. It is anticipated the reactor will continue to be utilized for similar work, as well as for other types of irradiations. The facility used for a particular irradiation depends on the type and size of the irradiation device, the desired neutron flux, its potential reactivity worth and other considerations. The location of most of the irradiation facilities can be seen in Figures 1-1, 4-1, and 7-1 through 7-3.

1. Type I - Reactivity Worth Measurement

Reactivity worth measurements are made to detect variations in enrichment or impurity content of small samples of fuel, structural and control rod materials, and other nonfuel materials. The technique used is determination of sample worth by control rod position change. Sample sizes range from a few to hundreds of grams. Most measurements are made with reactor power in the vicinity of 10-15 watts.

2. Type II - Radiation Damage Studies

Semiconductor devices have been the object of study in most of the radiation-damage work done in the past with the NTR. Typically, the samples are installed in the vertical or the horizontal facility, and frequently lead wires are brought from the devices to monitoring instrumentation located in the control room. For some irradiations, power is varied stepwise as data are collected; for others, the irradiation is performed at a single power level.

3. Type III - Small-Sample Activations

Usually, either the horizontal or vertical facility is used for small-sample activations, which are performed as one step in an activation analysis carried out to determine the composition of a material. These irradiations may be performed on both fuel and nonfuel materials in a wide range of physical and chemical forms with a typical sample size of a few grams. For studies on short-lived isotopes, in the past, a rapid-transfer tube or shuttle has been installed into the horizontal or vertical facility to permit rapid removal and

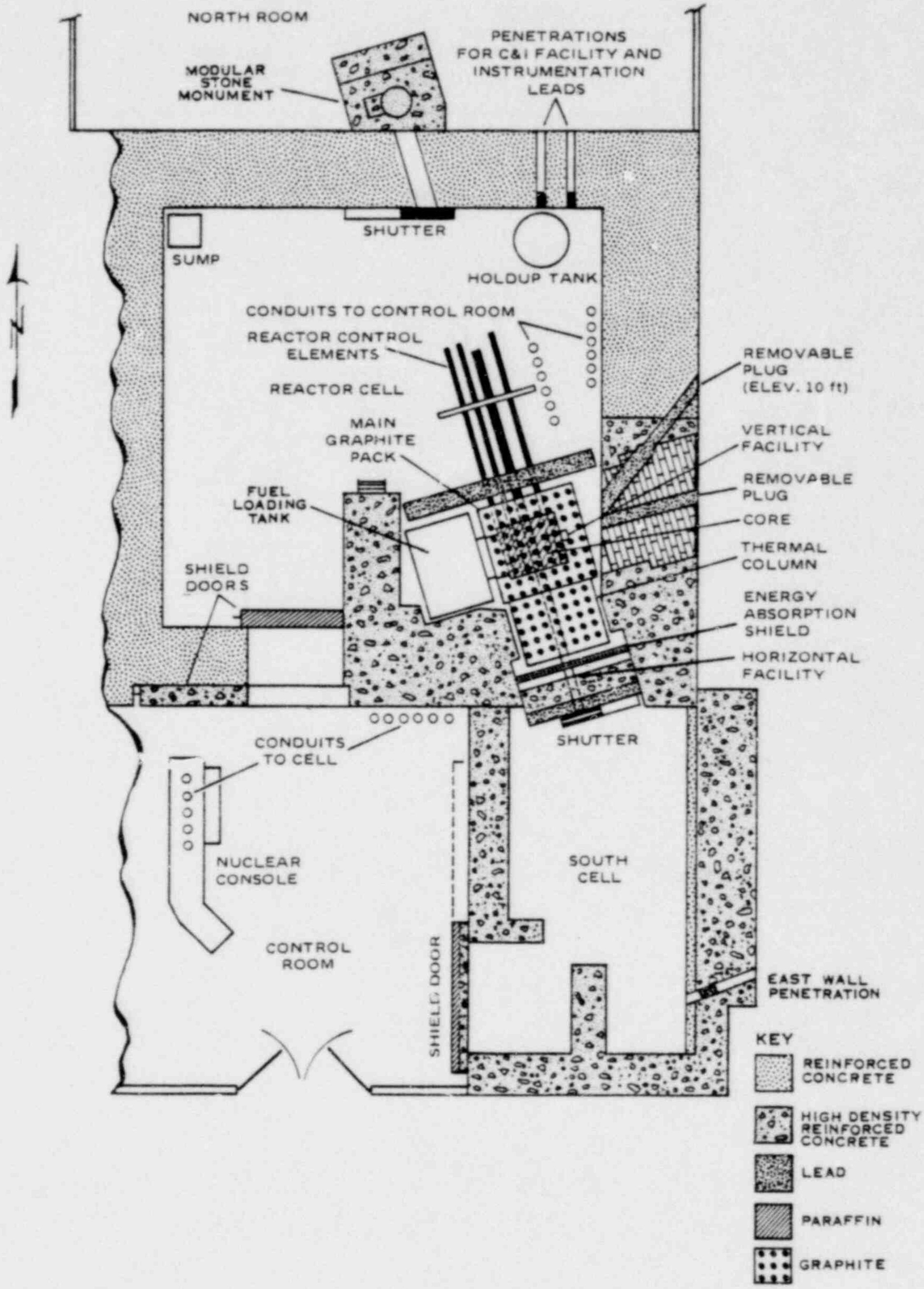


Figure 7-1. Nuclear Test Reactor Facilities (Top View)

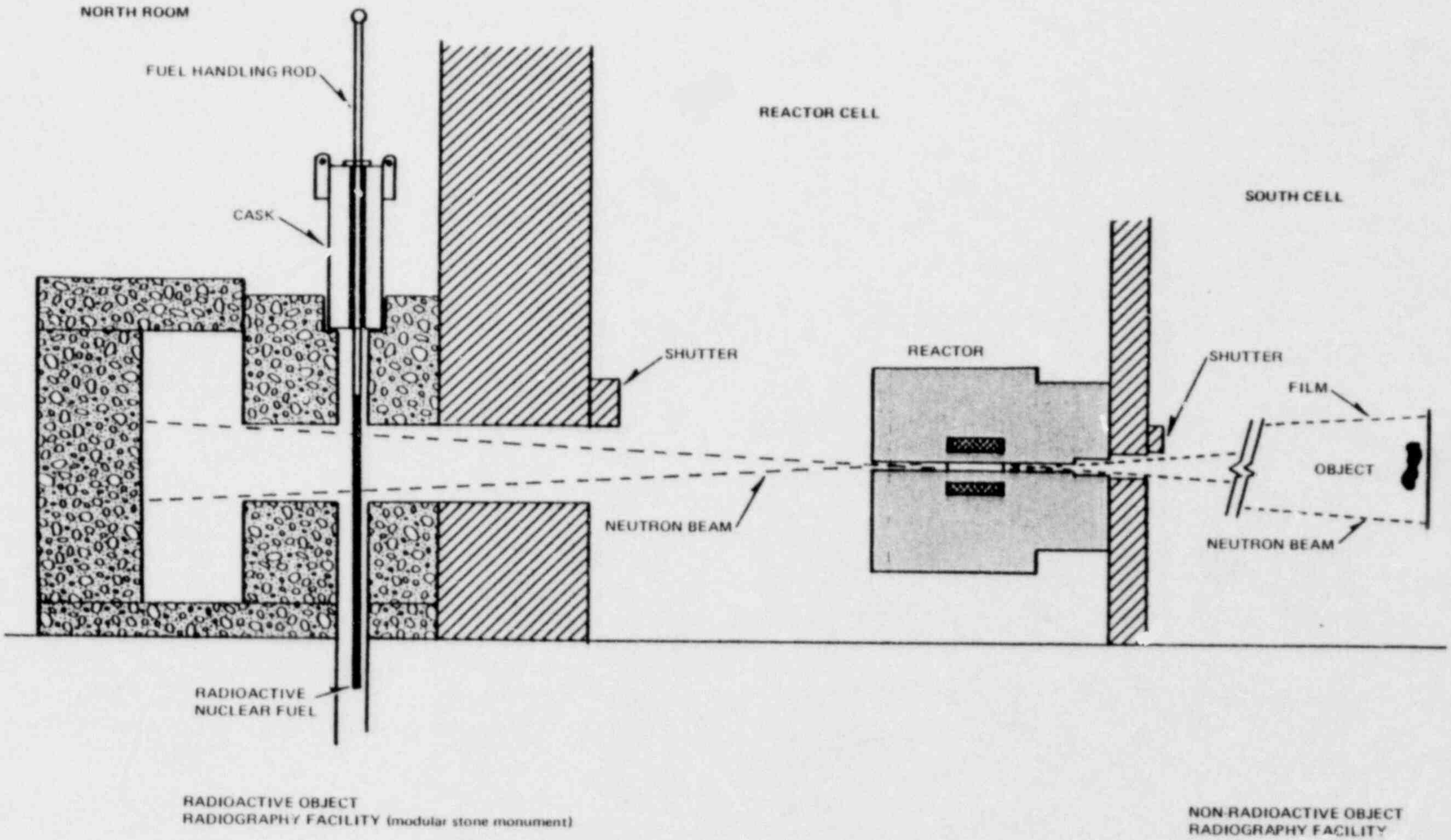


Figure 7-2. Nuclear Test Reactor Neutron Radiographic Facilities (Side View)

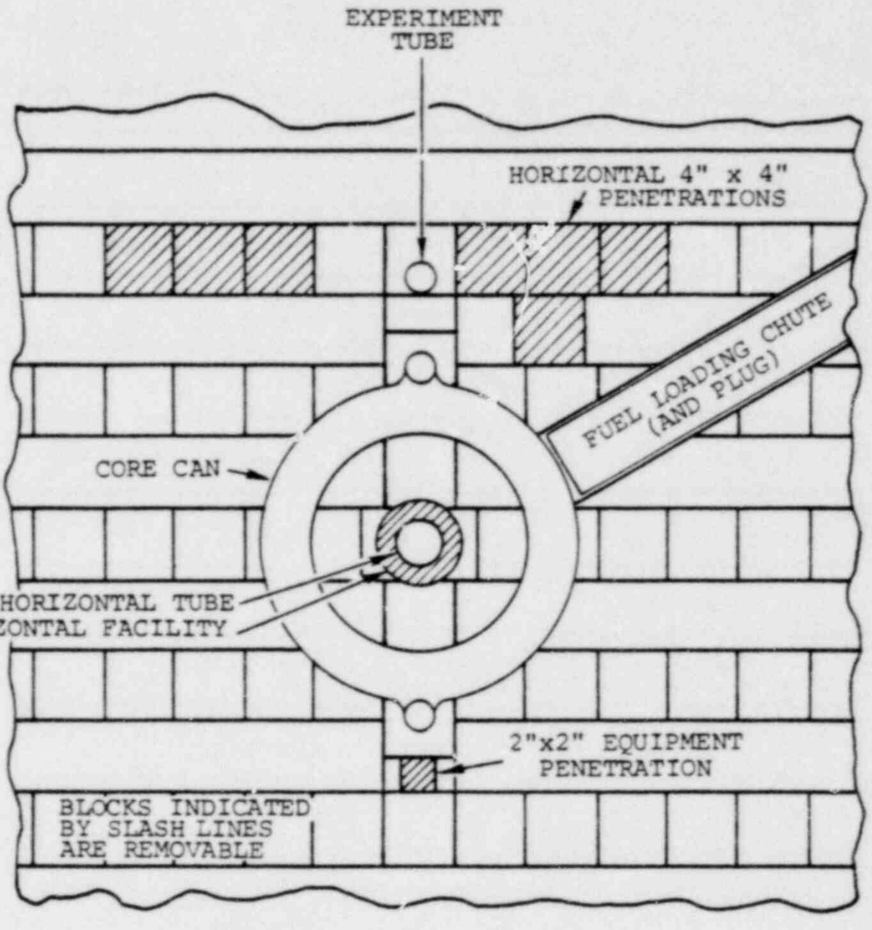


Figure 7-3. Horizontal Penetrations

counting of the sample after it has been activated. Shuttle tubes, typically about 1 inch in diameter, utilized pneumatic force to move a small sample container into or out of the reactor rapidly. Usually, plastic or aluminum tubing has been used. Driving gas is normally carbon dioxide, helium, or air at pressure less than 150 psig. A valve system may be provided to pressurize or exhaust either end of the tube to permit injection or ejection of the sample. Reactivity effect of inserting and removing a sample is usually no more than a few cents.

4. Type IV - Large-Sample Irradiations

It may be desirable to irradiate objects which are too large to be installed in the horizontal or vertical facility, or would have a high reactivity effect. These objects, such as fission foils and electronic equipment, are irradiated at one of the available faces of the main graphite pack; the top face or the vertical east face is most frequently used for these irradiations. (Note: Small samples also may be irradiated on these faces when a lower flux than that normally present in the horizontal or vertical facilities is desired.)

5. Type VI - Neutron Radiography

Neutrography Service* involves the nondestructive examination of irradiated and unirradiated nuclear fuel and test capsules. In addition, examinations of explosive devices (primarily used in aerospace programs) and other items are performed. The process involves placing an object in a neutron beam emitted from the reactor. Neutrons pass through the object, and with the aid of a converter screen, normally create an image on a sheet of radiographic film. The south cell and Modular Stone Monument (MSM) in the north room are the most frequently used target locations for this process. Figure 7-2 shows these neutron radiography facilities. The MSM is described in detail in Subsection 7.12.

*Service mark of the General Electric Company

6. Type VII - Nuclepore* Filter

Nuclepore* filter, a unique porous material, is a transparent plastic film penetrated with uniform cylindrical holes. The special holes result from a nuclear track registration and chemical etching process. The filters are made by exposing very thin plastic tape to fission products produced by a fission plate device irradiated by the NTR. These fission products make damage tracks in the tape. After irradiation, the tape is transferred to a chemical processing facility to batch-etch the tracks into uniform pores. The Nuclepore* filter irradiation machine is operated on the top face of the reactor, as required.

7.4 REVIEW AND APPROVAL

A written description and analysis of the possible hazards involved for each type of irradiation or experiment must be evaluated and approved by the facility manager or designated alternate before the irradiation or experiment may be conducted. Routine experiments — those that are repetitive and involve well-characterized materials and configurations as defined in the Experiment Type Approval (ETA) — may be reviewed and approved by a Licensed Senior Reactor Operator. Those experiments that are not routine, but still fall within the jurisdiction of the ETA, must be approved by the Reactor Supervisor or designated alternate. Review of new irradiations or experiments shall specifically cover the following items, as applicable:

1. Satisfactory application of materials involved;

2. Credible chemical reactions;

3. Satisfactory physical integrity for handling, irradiation, and containment of radioactive materials;

4. Effect on core reactivity;

*Trade Name of the Nuclepore Corporation

5. Effect on major core parameters;

6. Adequate cooling;

7. Operating instructions, including specification of any operation limits or instrument set points required to ensure safety of the irradiation device, the experiment, the reactor, and personnel; and

8. Applicable regulatory requirements.

7.5 HORIZONTAL FACILITY

The horizontal facility (sometimes called the central tube) is a cavity traversing the horizontal axis of the reactor (Figure 7-3). The facility arrangement for normal use is as follows: Starting in the south cell at the south face of the reactor shielding, an 8-in. opening passes through the south radiation shield and into the thermal column 10 inches, where the cavity decreases to a 5-in. diameter. The 5-in. cavity then continues through the core reflector (south side), core annulus, core reflector (north side), and the north radiation shield wall. Within the 5-in. cavity is a 3-in.-o.d. aluminum tube fitted into a 40-in.-long, 5-in.-o.d. graphite sleeve. The 40-in. graphite is centered in the main graphite pack. To provide broader capabilities for irradiations and improved neutron radiography beams, this sleeve can be removed, enlarging the horizontal facility to 5 inches. The east-west centerline of the reactor is about 113 inches from the south shield face and 47 inches from the north shield face. A penetration in the north wall of the reactor cell is provided with an electrically operated radiation shield shutter. The horizontal facility is accessible from either the south cell or reactor cell, and objects may be irradiated in any position within the sample tube or in the radiation beams streaming from the tube ends.

Material or equipment to be irradiated in the horizontal facility may be fastened to an extension rod and positioned manually from the south cell. Specially machined graphite logs suitable for use as sample holders for specific irradiations are available. Electrical leads, cooling lines, and associated equipment for instrumented devices can be brought out either end of the horizontal facility for connection to equipment in the reactor cell, south cell, or through available penetrations to or above the control room, shop area, or north room.

During irradiation, reactor instrumentation, as well as any instrumentation associated with the irradiation device, is observed carefully. In the event of an unexpected change in neutron multiplication, critical rod position, radiation levels, or reactor or irradiation device behavior, the operator will take whatever immediate action is required to ensure the safety of personnel, the reactor, and the irradiation device. He will then notify a Licensed Senior Reactor Operator, as required, who will evaluate the situation and initiate whatever further action is necessary.

Unloading of the horizontal facility is usually done with the reactor shut down; however, if the reactivity effect of the sample and the radiation level permits, the sample may be removed with the reactor operating. If the radiation level from the sample is such that conventional tweezers or pliers that are normally used to handle samples are inadequate to control exposure properly, temporary shielding and remote handling tools will be utilized. To minimize radiation exposure to the operators, irradiated samples are normally allowed to decay as long as is practical before handling them.

7.6 CONTROL AND INSTRUMENTATION TEST FACILITY

The Control and Instrumentation (C&I) Test Facility is used for testing various types of neutron detectors, principally Boiling Water Reactor (BWR) Traversing Incore Probes (TIPs) and Local Power Range Monitors (LPRMs). The facility simulates a BWR temperature environment and provides a thermal neutron flux environment compatible with C&I requirements.

The C&I test section is positioned in the experiment tube (Figure 7-3), which runs horizontally across the top of the reactor core can. Access tubes leave the north room (Figure 7-1), go through a lead pig, and terminate in a funnel assembly, which allows each detector to be sequenced into the test section. The pig is radiation shielding for the storage of irradiated detectors that have been withdrawn from the reactor but are too radioactive to be pulled into the north room.

7.7 MISCELLANEOUS HORIZONTAL PENETRATIONS

The main graphite pack has the capability of providing up to seven 4-in. by 4-in. penetrations in the horizontal plane. Two of the graphite layers have been modified to accommodate the penetrations (Figure 7-3). Dimensions of the graphite blocks in the seven penetrations have been reduced to permit removal of the blocks. The north end of each block is drilled and tapped to facilitate removal.

Additionally, there is the capability of removing a section of graphite below the core can 2 inches by 2 inches by 36 inches, which is suitable for small-sample irradiations or a rabbit tube facility, should one be required (Figure 7-3).

Each of the horizontal penetrations is accessible from inside the reactor cell through holes in the north shield wall. These holes are normally filled with lead or borated lead polyethylene plugs.

7.8 VERTICAL FACILITY

The vertical facility is defined by a 5-in.^{2} by 5-ft-long aluminum can, which extends vertically through the graphite reflector approximately tangent to the east side of the fuel container. A piece of reflector graphite normally fills the facility when it is not in use. The facility is accessible only from the top of the reactor inside the reactor cell. Irradiations may be performed at any position within the aluminum can or in the beam streaming from the top.

Devices to be irradiated in the vertical facility are usually attached to a wire or extension rod supported from the top shield of the reactor. Since entry into the reactor cell during critical operation is normally forbidden, samples must be positioned manually before startup or provided with a means for remote positioning. Space not occupied by the device may be filled with graphite blocks. Leads from the device can be brought out the top of the facility for connection to equipment in the reactor cell or through cell penetrations to or above the control room, shop area, or north room.

The precautions and procedures during irradiation are the same as those discussed in Subsection 7.5 for the horizontal facility. The only significant difference is that the shut-down radiation level of the reactor core may contribute appreciably to the radiation exposure received during the unloading. However, radiation monitoring is required for sample unloading at all times, which ensures that the operator is aware of radiation from all sources. In addition, the cell remote area radiation monitor (with readout in the control room) would indicate unexpected high radiation levels.

7.9 LOADING CHUTE FACILITY

The aluminum-covered graphite plug for the fuel loading chute may be removed to provide access to the inside of the fuel container for irradiations. Use of this facility is necessary for performing experiments, such as some of those required to determine the nuclear characteristics of the reactor. No experimental objects are permitted inside the core tank when reactor power is greater than 100 watts. Experimental objects in the fuel loading chute will be secured to prevent their entry into the core region during all normal operating conditions.

7.10 REACTOR FACE FACILITIES

Radiation escaping through any one of the faces of the 5-ft^3 graphite reflector may be utilized for experimentation. The aluminum box (partially cadmium-lined) that contains the reflector is provided with 4-ft by 4-ft removable sections on the top and east faces. The removable section for the east face contains a 20-in. by 18-in. hinged section, which can be opened to eliminate the cadmium from this area. Limited space between the reflector and the top shield slab can be used without removing the 48-in.-o.d. concrete plug in the shield. However, the plug would be removed to accommodate experiments such as the Nuclepore* filter irradiation machine discussed in Subsection 7.3.

*Trade name of the Nuclepore Corporation

Entry into the reactor cell is necessary for access to the face facilities. Therefore, objects to be irradiated are generally positioned before reactor startup or are provided with remote positioning devices. Irradiation devices utilizing the face facilities have a negligible effect on core reactivity.

7.11 EXTERNAL THERMAL COLUMN

The external thermal column is a 4-ft^3 of high-purity graphite located against the south side of the reactor's 5-ft^3 graphite reflector. The thermal column is traversed by the horizontal facility. A 20-in. wide by 20-in.-high by 4-ft-long centrally located section, made of 4-in. by 4-in. by 4-ft logs was designed to be removed partially or entirely to accommodate experiments inside the thermal column or to provide external radiation beams. The south face of the thermal column is accessible from the south cell. Radiation shielding on this face consists of a 0.375-in. boral plate, a 26-in.-thick wall of concrete bricks, and a 4-in. wall of lead bricks. Sections of the biological shield may be removed to provide access to the graphite face of the thermal column or to permit use of radiation beams. An air-piston-operated shutter is installed at the face of the lead brick wall to provide shielding from the horizontal cavity.

Irradiations utilizing the thermal column normally have a negligible effect on core reactivity and may be loaded or unloaded with the reactor operating or shut down. Radiation exposure to the operator is usually of more concern during such activities than the effects of the irradiation device on the reactor.

7.12 MODULAR STONE MONUMENT

The Modular Stone Monument (MSM) is a dual neutron radiography facility, located in the north room, allowing the capability of performing neutron radiography on unirradiated or irradiated objects (Figures 7-2 and 7-4). The design involves six concrete blocks that make up the shield and structural unit. A 12-in.-i.d. stainless steel pipe capped off the bottom penetrates into the ground beneath the MSM for 20 feet. This penetration allows neutron radiography of long objects to be performed by lowering them into the pipe.

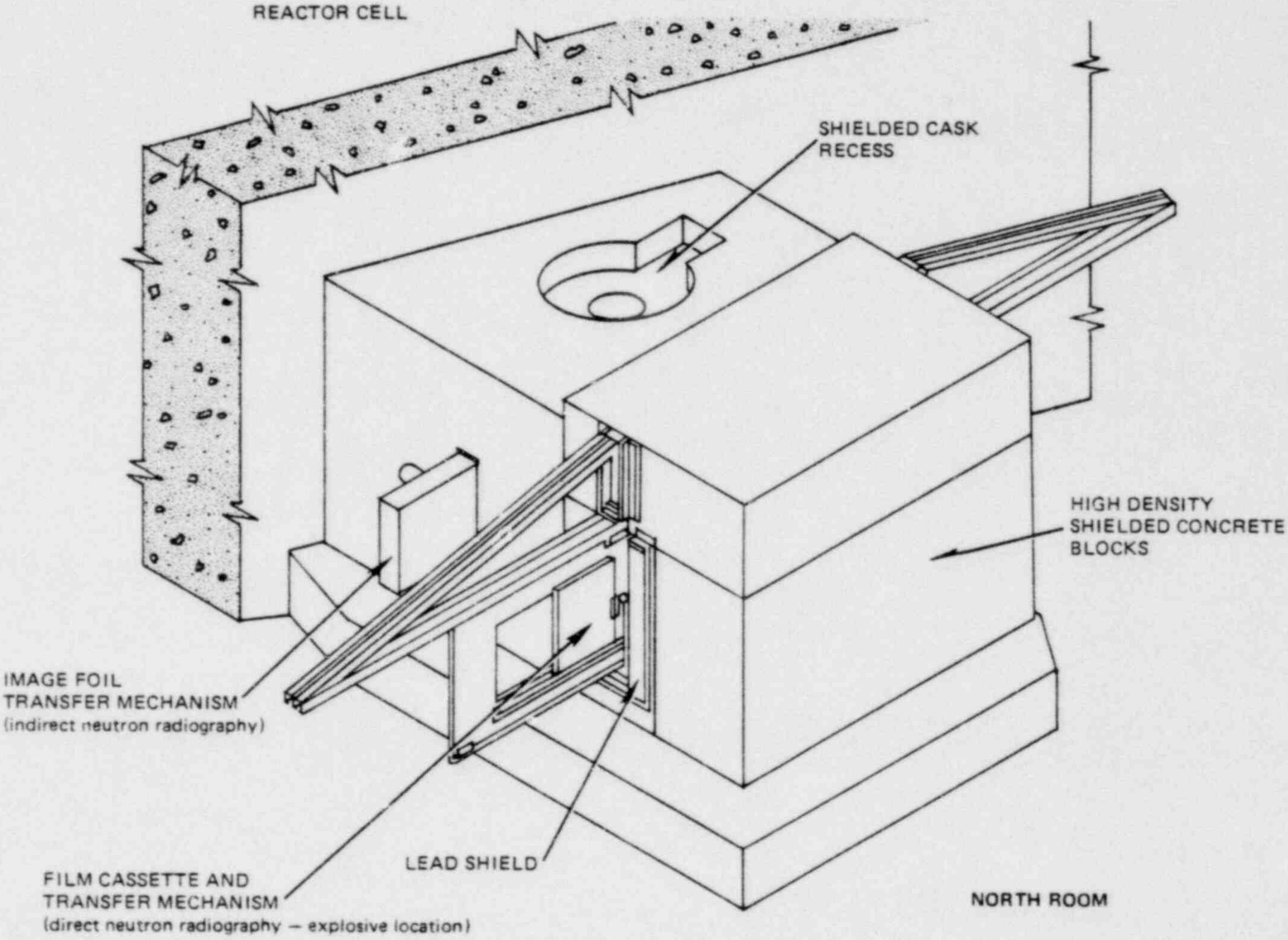


Figure 7-4. Modular Stone Monumen. Neutron Radiography Facility

Mechanisms for changing neutron radiography imaging foils without returning irradiated objects to their casks have been incorporated into the design.

Irradiated objects normally arrive at the NTR in large casks which are placed on top of the MSM, using the overhead crane. The objects can then be lowered down into the MSM in front of an imaging foil and the neutron radiography is then performed.

Unirradiated objects are moved into a facility on the north end of the MSM, usually by a trolley arrangement. The imaging system is placed behind the object and neutron radiography or irradiation is performed.

7.13 IRRADIATION DEVICES CONTAINING HAZARDOUS MATERIALS

Sometimes small quantities of corrosive, flammable, or explosive materials are inserted into the irradiation facilities. To ensure that the nuclear safety of the facility is not compromised, the following restrictions are placed on these irradiations:

1. Irradiations must be performed in accordance with written procedures approved by the Facility Manager or his designated alternate.

2. The negative reactivity worth of any components that could be ejected from the reactor by a chemical reaction must be less than 50c.

3. Hazardous materials inserted into the reactor must be in capsules that have been (if applicable) (a) designed or tested to ensure the pressure transients produced by any possible chemical reaction of their contents and leakage of corrosive or flammable materials will not damage the reactor; (b) designed to provide adequate protection against accidental exposure of the material to shock, heat, undesired chemical reaction, or electrical spark; and (c) designed to contain or direct into an area that is vented through an absolute filter (such as the reactor cell) the radioactive material contained in the capsule, thus preventing an uncontrolled release of this material, should a fire or explosion occur.

4. The maximum chemical energy release from the combustion of flammable substances inserted into the reactor shall be evaluated and shall not exceed 1000 kW/sec. The total possible energy release from the chemical combination or decomposition of hazardous substances contained in any experimental capsule shall be evaluated and shall be limited to 5 kW/sec, if the rate of reaction in the capsule could exceed 1 watt.

7.14 USE OF EXPLOSIVES

Because the use of explosives within a reactor facility is recognized as a safety concern, the following restriction shall apply based on the analysis in Appendix B.

The maximum amounts of explosives permitted in the NTR facilities are:

1. South cell: $W = (D/2)^2$ with $W \leq 9$ pounds and $D \geq 3$ feet;

2. North room (without MSM): $W = D^2$ with $W \leq 16$ pounds and $D \geq 1$ foot;

3. North room (with MSM): $W = 2$ pounds:

4. Set-up room: $W = 25$ pounds;

where W is the total weight of explosives in pounds of equivalent TNT, and D is the distance in feet from the south cell blast shield or the north room wall.

A maximum of 10 curies of radioactive material and up to 50 grams of uranium may be in storage in a neutron radiography area where explosive devices are present (i.e., in the south cell or the north room). The storage locations must be at least 5 feet away from any explosive device. Radioactive materials other than those produced by neutron radiography of the explosive devices and imaging systems are not permitted in the set-up room if explosive devices are present.

Unshielded, high-frequency generating equipment shall not be operated within 50 feet of any explosive device.

The cumulative radiation exposure for any explosive device shall not exceed 3×10^{12} n/cm^2 from thermal neutrons and 1×10^4 roentgens from gamma rays.

7.15 CRITICALITY

7.15.1 Introduction

The criticality analysis for the Nuclear Test Reactor (NTR) permits the storage and/or handling of fissile materials in four Criticality Limit Areas (CLAs) (Figure 7-5). The criticality analysis specifies fissile material limits for U-233, U-235 enriched to $\leq$93.5 w/o U-235 in uranium, UO_2 with enrichments $\leq$5 w/o in uranium, and plutonium which contained at least 5 w/o Pu-240. Mixtures of these fissile isotopes are also permitted in each CLA according to a pro formula which is specified in the criticality analysis. Each unit of fuel in a CLA satisfies the criterion: k-effective $\leq$ 0.9 under conditions of optimum water moderation and full water reflection. Further, the array that could occur by the simultaneous handling or storage of fissile material in each CLA satisfies the solid angle criterion: $\Omega \leq$ 9-10 k-effective.

7.15.2 Criticality Safety Control

The Manager, NTR, is fully responsible for the safety of operations and for the control of activities in NTR criticality areas. Criticality control limits governing the handling, use, and storage of fissile materials in the criticality areas external to the NTR core are established considering all pertinent process conditions and failure possibilities for activities or changes in activities. Written operating procedures incorporating the criticality control limits are established.

In the current criticality analysis applicable to the NTR, four CLAs are selected for control of fissile material external to the NTR core (Figure 7-5). The specification of maximum fissile mass limits for each CLA is the method chosen for criticality control because this method provides for the greatest flexibility for NTR activities. Administrative controls (written procedures) are used to monitor the fissile material in each CLA to insure that it is always less than the maximum permitted fissile limit. The criterion that the k-effective $\leq$ 0.9 under conditions of optimum water moderation and full

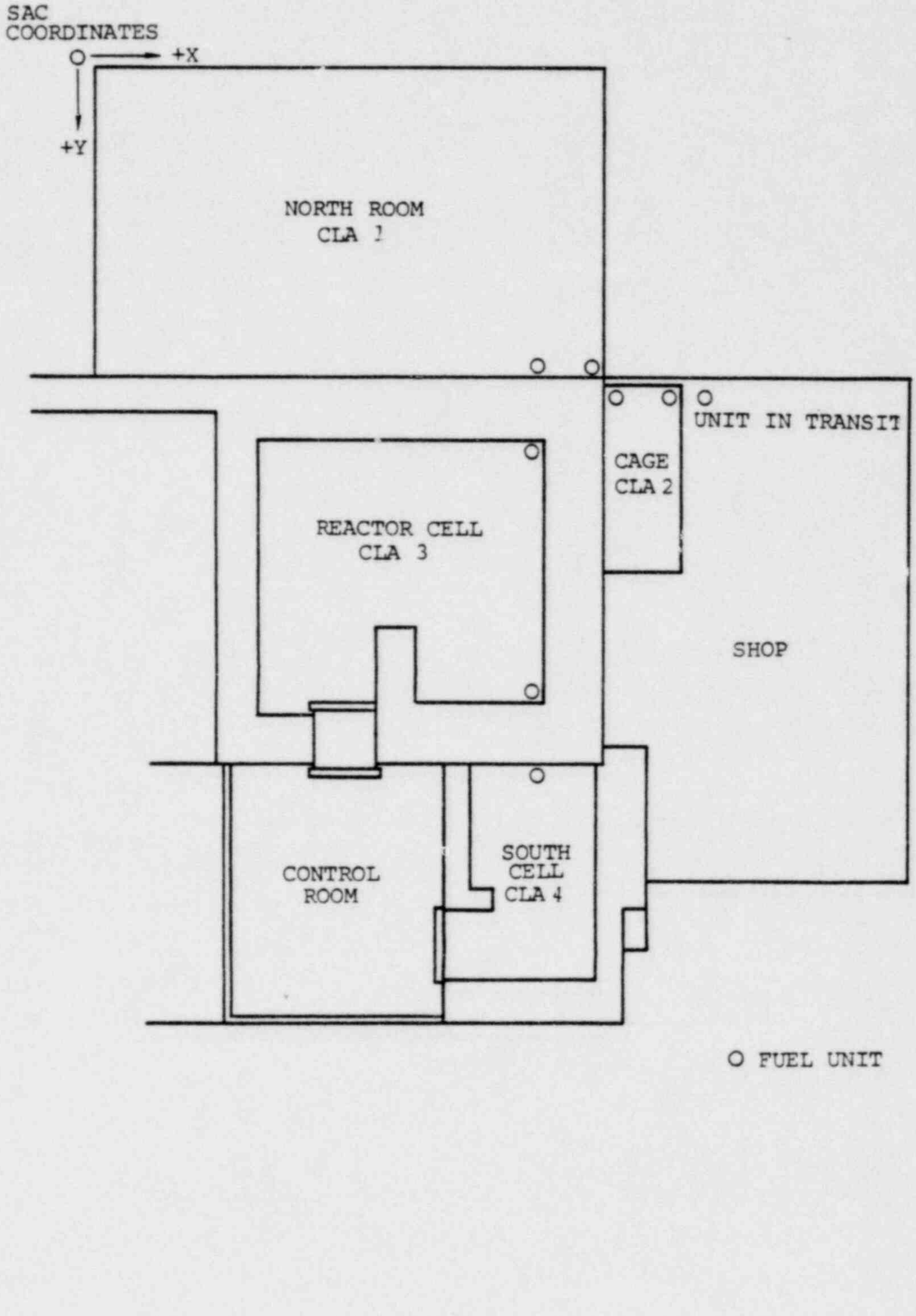


Figure 7-5. Criticality Limit Areas

water reflection is used [illegible] the maximum fissile mass that is permitted in each CLA. Fissile limits are specified for U-233, uranium enriched to ≤93.5 w/o U-235 in uranium, uranium enriched to ≤5 w/o U-235 in uranium, plutonium with ≥5 w/o Pu-240 and a prorated mixture of these fissile materials.

In evaluating the subcriticality of individual accumulations and assemblies or arrays of fission material in criticality analyses, the two-contingency criteria is applied; that is, two independent and unrelated accidents must occur before criticality is credible. Failure of one administrative control is not considered as one of the contingencies.

7.15.2.1 Computational Tools Used in NTR Criticality Analysis

In these analyses, three distinct computational tools were used. They are hand calculations, the KENO-IV Monte Carlo Criticality Code[8], and the solid angle calculation. These tools are described below.

7.15.2.1.1 Hand Calculations

Hand calculations are used to estimate the minimum critical mass for fissile accumulations. These calculations are performed with the one group approximation formula:

$$k_{eff} = k_\infty/(1+M^2B^2)$$

where

- k_{eff} is the k-effective multiplication factor;
- k_∞ is the infinite multiplication factor for the fuel mixture under consideration
- M^2 is the migration area, cm^2; and
- B^2 is the geometric buckling, cm^{-2}.

Values for the nuclear parameters K_∞ and M^2 and the extrapolation lengths necessary to calculate the geometric buckling were obtained from the Atlantic Richfield Hanford Company Criticality Handbook[9] and from the Dupont Savannah River Report.[10]

7.15.2.1.2 KENO-IV Monte Carlo Criticality Code

The KENO-IV Monte Carlo Criticality Code is used in conjunction with the 16-group Hansen and Roach Cross Section Set[11] as modified by Knight[12] to perform the criticality calculations. KENO-IV is used to calculate k-effective for critical accumulations of fissile materials in this document, and to calculate k-effective for prorated mixtures of fissile materials. The KENO-IV Code is also used to calculate k-effective values for masses of fissile material determined from hand calculations to give a k-effective value of 0.9.

The KENO IV code is a three-dimensional, multi-group Monte Carlo Criticality Code which was developed by the Oak Ridge National Laboratory. The principal result from the KENO-IV Code is an estimate of the k-effective for some accumulation (assembly) or array of fissile materials modeled as input to the code.

7.15.2.1.3 Solid Angle Method

The Solid Angle Method is used to specify safe parameters for an array of accumulation (assemblies) of fissile material without first determining the critical values for the array. This method was developed as an empirical means of evaluating interaction between small numbers of moderated fissile units.[13] The Solid Angle Method has been studied further in References 14 through 16 for arrays of fissile materials, and the results have been compared with results from the KENO-IV Code.

Solid Angle Method accumulations in an array are spaced so that each accumulation of fissile material is allowed to see only a limited number of other accumulations. The maximum allowable solid angles subtended by surrounding accumulations is inversely related to the effective multiplication factor (k-effective) for the accumulation under study. The value of k-effective for each accumulation is determined by using an independent calculation, such as the hand calculation or the KENO-IV Code described previously. The solid angles are determined by using point-to-plain formulas[14-17] or by exact formulations.[14-18] The point-to-plain formulas give conservative results when compared to those obtained from more exact integral formulas.

If the array being analyzed consists of identical accumulations, only the most centrally located accumulation need be considered as a point for the solid angle calculation. In other cases, it is necessary to choose more than one center point accumulation to assure that the k-effective solid angle relation is satisfied for all accumulations.

The relation between the k-effective value of an accumulation and the maximum allowable solid angle that may be subtended due to other accumulations is shown in Figure 7-6. The key to the symbols is presented in Table 7-1. This figure is discussed in detail in Subsection 7.15.3. The curve in this figure is as specified in Reference 17, which was originally derived in Reference 19. The Solid Angle Method specifies a maximum allowable solid angle subtended at any unit, with a neutron multiplication factor k-effective by all other units in the array. A given array is then judged to be subcritical if the actual solid angle is $\leq$ the allowed solid angle given by:

Ω allowed $\leq$ 9-10 k-effective

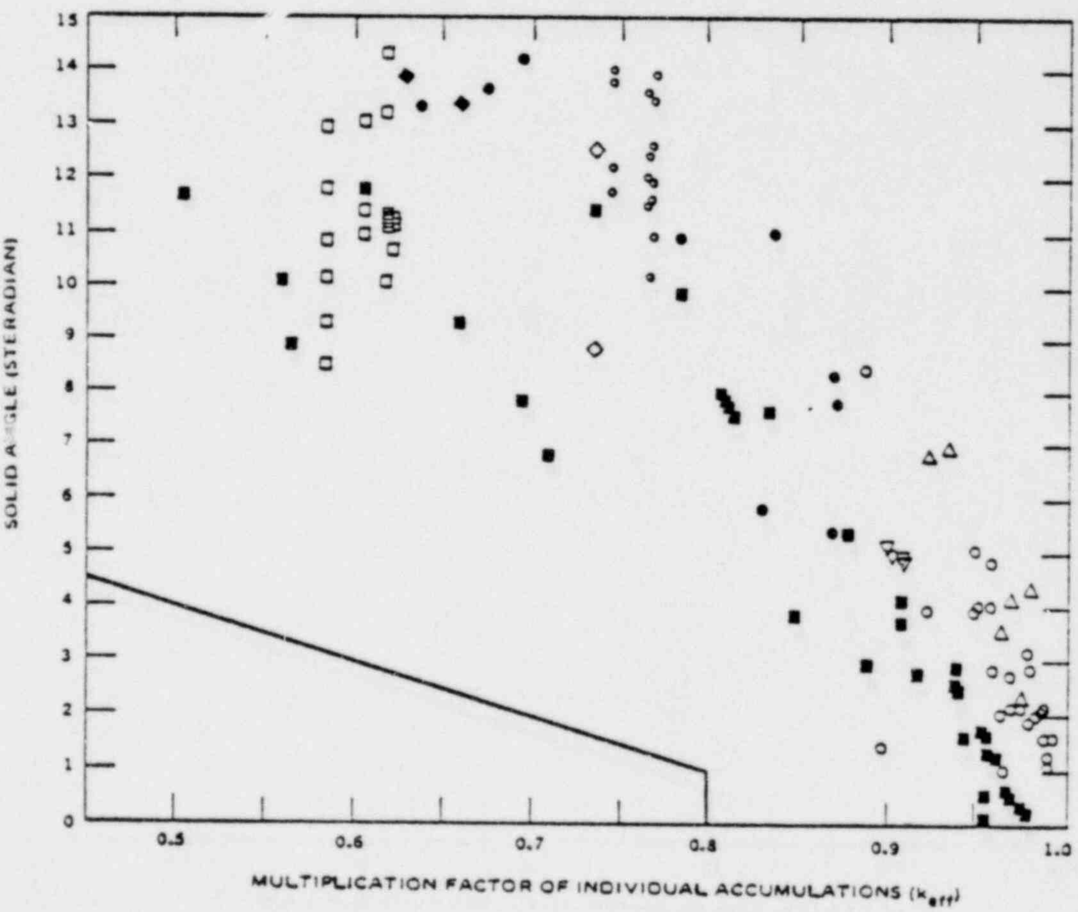


Figure 7-6. k_{eff} - Solid Angle Relation

Table 7-1

KEY TO SYMBOLS

Symbol	Number of Points	Type of Accumulation	References
∘	15	H/X 493 4.9% Enrichment Cylinders	(20) ORNL-3714, Vol. 1, Figure 2.2.1
o	24	H/X 297 93.2% Enrichment Cylinders	(21) Y-1272, Figure V 3.3
•	9	HX/ 309 93.2% Enrichment Cylinders	(21) Y-1272, Figure V 3.4
◆	2	H/X 50.1 93.2% Enrichment Cylinders	(21) Y-1272, Figure V 3.5
△	6	H/X 14.3 93.2% Enrichment Cylinders	(21) Y-1272, Figure V 3.6
□	19	H/X 59 92.6% Enrichment Cylinders	(22) TID-7028, Figure 76
▽	4	H/X 50 93.2% Enrichment Slabs	(21) Y-1272, Figure V 4.1
□		H/X 44 93.2% Enrichment Cylinders	
□	24	H/X 169 93.2% Enrichment Cylinders	(21) Y-1272, Figure V 1.1
□		H/X 330 93.2% Enrichment Slabs	
◇	2	H/X 59 92.6% Enrichment Cylinders	(22) TID-7028, Table 21

where

Ω allowed equals the allowed solid angle in steradians subtended at the center of any accumulation by the remainder of accumulations in the array, and

k-effective equals the unreflected neutron multiplication factor for the accumulation under study.

The following conditions shall also be satisfied in order to apply the Solid Angle Method:

1. The unreflected k-effective of any unit shall not exceed 0.8.

2. Each accumulation shall have a k-effective $\leq$0.9 when completely reflected by water.

3. Each accumulation shall be evaluated as optimumly moderated (e.g., aqueous solutions) at the point of maximum k-effective.

4. The minimum separation between each accumulation shall be at least 12 inches.

5. The allowed solid angle shall not exceed 6 steradians.

6. The effectiveness of the reflector around the array shall be no greater than that of a thick (12 inches) water reflector spaced at distances from the accumulations comparable to the spacing between the accumulations.

7. For full reflection of an array by concrete thicker than 4.8 inches, the allowable solid angle shall be reduced by 40%.

Conditions 1, 2, and 4-7 are as specified in Reference 17. Condition 3 is specified because of the results of the calculations in References 14 and 15 and to avoid the difficulties found in Reference 16.

7.15.3 Computational Models

Individual accumulations of fissile material are modeled as homogeneous mixtures of fissile material-water in a spherical shape. The effect of clumping for uranium fuel having uranium enrichments $\leq$5 w/o U-235 in uranium is also considered.

In Reference 10, results from critical experiments in calculations for critical masses of homogeneous and heterogeneous UO_2-H_2O mixtures are given for U-235 enrichments ranging between 0.9 w/o and 5 w/o U-235 in uranium. These results show the minimum critical mass to be smaller for heterogeneous mixtures than it is for homogeneous mixtures. For example, at a 5 w/o enrichment, the minimum heterogeneous critical mass is given as 1.56 kgs U-235 for a 0.1-in.-o.d. rod of UO_2; the corresponding homogeneous minimum critical mass is given as 1.85 kg U-235. Therefore, both homogeneous and heterogeneous spheres are modeled for 5 w/o U-235 in uranium.

Models for the various arrays are shown in Figure 7-6. One hundred fifteen (115) data points were plotted. Table 7-1 shows the number of data points used for each keyed symbol, the H/X ratio, enrichment, and geometry of the individual accumulation used in the array for that symbol, and the reference for the information presented. The information in this table indicates that most of the individual accumulations of fissile materials are solutions that are contained in cylinders of varying heights and diameters. Many of the cylinders are arranged in plainer arrays that have square pitches and triangular pitches of which a few are three-dimensional. Some of the individual accumulations are represented as slabs. The arrays are represented by slabs separated at various distances.

The models in Table 7-1 indicate that most of the individual accumulations of fissile material in arrays are uranium solutions in which the uranium is enriched to $\geq$92 w/o U-235 in uranium. Only 15 of the data points represent arrays containing accumulations of solutions of fissile material in cylindrical form enriched to 4.9 w/o U-235 in uranium. Detailed information about the accumulation of fissile material in each array and the shape of the array is given in the references indicated in Table 7-1.

7.15.4 Calculation Results

In Table 7-2, calculated results are given for homogeneous mixtures of U-233, U-235, and Pu-239 - H_2O spheres that are fully reflected by water. The hydrogen-to-fissile ratios used in the calculations are also stated. The source for the minimum critical mass values and the nuclear properties needed with the buckling equations to perform the hand calculations are also referenced. The hand calculation results give k-effectives slightly larger than 1 for the three homogeneous metal-water mixtures. Although the mean value of k-effective for the KENO-IV results are slightly lower than 1 for each of the three cases, the values for k-effective at the 2-sigma upper limit are all slightly larger than 1 for each of the three cases.

Calculated results are also given in Table 7-2 for homogeneous and heterogeneous mixtures of 5 w/o enriched uranium dioxide-water spheres that are fully water reflected. The hydrogen to fissile ratio used in these hand calculations is also given. Two minimum critical mass values for this mixture were obtained from two different references. Reference 9 gives a minimum critical mass of 1890 grams, which was read from a plotted curve. Reference 10 gives a minimum critical mass of 1850 grams, which was read from a table. The calculated results all give k-effective equal to 1 at least two decimal places.

It is noted that the minimum critical mass for a heterogeneous mixture of $U(5)O_2$-H_2O is about 300 grams smaller than for a homegeneous mixture of the same material, as shown in Table 7-2.

In Table 7-3, a k-effective value of 0.9 is assumed, and hand calculations are used to calculate the fissile mass and corresponding sphere size as a function of the hydrogen fissile ratio for homogeneous metal-water mixtures that are fully reflected by water. The fissile mixtures include U-233, U-235, and Pu-239-H_2O mixtures. From these calculations, the minimum fissile mass giving a k-effective value of 0.9 is found at some particular hydrogen-to-fissile ratio, and this information is used to prepare a model as input to the KENO-IV Code to calculate the k-effective. These k-effective results are compared in Table 7-3.

Table 7-2

CALCULATED K-EFFECTIVE RESULTS FOR CRITICAL FUEL ASSEMBLIES

Missile Material System	Minimum Critical Mass GMS Fissile	Reference	K_{eff}	Hand Calculations Nuclear Properties Reference	H/ Fissile	KENO-IV Calculations 8100 Neutrons Histories: K_{eff}	±	1 σ	2 σ K_{eff}
Homogeneous Sphere U233-H_2O	570	(ARH-600) Vol. III Fig. III.C.2 & Fig. III.C.6(100)-3	1.0115	Reference 9, Vol. III Fig. III.C.10(100)-3,4	427.57	0.99250	±	0.00769	1.00828
Homogeneous Sphere U235-H_2O, FWR	820	(ARH-600) Vol. II Fig. III.B.2 & Fig. III.B.6(100)-2	1.0094	Reference 9, Vol. II Fig. III.B.10(100)-1,2	47.07	0.98737	±	0.00918	1.00572
Homogeneous Sphere Pu239-H_2O, FWR	531	(ARH-600) Vol. II Fig. III.A.6 & Fig. III.A.6(100)-3	1.0066	Reference 9, Vol. II Fig. III.A.10(100)-3,4	880.1	0.99345	±		1.00525
Homogeneous Sphere U(5)O_2-H_2Om, FWR	1980	(ARH-600) Vol. II Fig. III.B.6-6	0.999	Reference 9, Vol. II Fig. III.B.10(5)-1,2	524.7	-		-	
Homogeneous Sphere U(5)O_2-H_2O, FWR	1850	(DP-1014) P. 39	1.0001	Reference 10 P. 58	551.46	-		-	
Heterogeneous Sphere U(5)O_2-H_2O, FWR, 0.1" O.D. UO_2 Rods	1560	(DP-1014)	1.0002	Reference 10	496.33	-		-	

NOTE: FWR means Full Water Reflection

Table 7-3

CALCULATED K-EFFECTIVE RESULTS

Fissile Material System	Fissile Mass, GMS	Hand Calculations		H/ Fissile	KENO-IV Calculations 14,100 Neutron Histories			
		K_{eff}	Nuclear Properties Reference		K_{eff}	±	1 σ	2 σ K_{eff}
Homogeneous Sphere U-233-H_2O, FWR	339	0.900	Reference 9, Vol. III. Fig. III.C.10(100)-3,4	427.57	0.85568	±	0.00738	0.87044
Homogeneous Sphere U-235-H_2O, FWR	476	0.900	Reference 9, Vol. II Fig. III.B.10(100)-1,2	470.7	0.87166	±	0.00609	0.88384
Homogeneous Sphere Pu-239-H_2O, FWR	299	0.900	Reference 9, Vol. II. Fig. III.A.19(100)-3,4	880.1	-0.87541	±	0.00633	0.88807
Homogeneous Sphere $U(5)O_2$-H_2O, FWR	923	0.900	Reference 10 Pg. 58	551.46	-		-	
Heterogeneous Sphere $U(5)O_2$-H_2O, FWR 0.1" O.D. UO_2 Rods	817	0.900	Reference 10 Pg. 59	551.46				

NOTE: FWR means Full Water Reflection

The k-effective results for the KENO-IV calculation in Table 7-3 show the upper limit k-effective at 2 sigma to be smaller than 0.9 by 0.011 to 0.03 for the three mixtures involved. This means that the hand calculations are more conservative than the KENO-IV Code results. This is to be expected since the KENO-IV Code is more accurate than hand calculations.

The minimum fissile mass giving a k-effective value of 0.9 for $U(5)O_2H_2O$ mixtures that are both homogeneous and heterogeneous in a spherical shape that are full water reflected are also given in Table 7-3.

A comparison of the minimum critical mass and the minimum fissile mass giving a k-effective value of 0.9 may also be made from the results of Table 7-2 and Table 7-3. For example, the ratio of minimum critical mass to minimum fissile mass where k-effective value equals 0.9 is 1.68 for the U-233 mixture, 1.72 for the U-235 mixture and 1.78 for the Pu-239 mixture. Similarly, for the 5 w/o enriched UO_2 mixtures, this ratio is 1.91 for the heterogeneous mixture and 2.00 for the homogeneous mixture.

Results for comparison of the solid angle criteria with data for critical arrays is given in Figure 7-6. The location of a solid angle value for a data point is obtained for the accumulation of fissile material having the highest solid angle in the array being represented. This is a function of the geometry of the accumulations in the array, the number of accumulations in the array, and the spacing between the accumulations in the array. The k-effective value for a particular array represents the bare k-effective value for an accumulation of fuel in that array. Some of the arrays are fully reflected by water and other materials, and some of the arrays are unreflected.

The results in Figure 7-6 show the solid angle criteria to be conservative in determining the criticality safety of an array of fissile material when compared to the data presented.

7.13.5 Conclusions

To assure that a criticality accident at the NTR due to handling and/or storage of fissile material external to the NTR core is not credible, controls must be specified that can be demonstrated to meet criteria that are safe. Safe criteria for activities at the NTR under normal operating conditions are as follows:

1. For individual accumulations and/or assemblies of fissile material, the k-effective shall be ≤ 0.9 under conditions of optimum water moderation with full water reflection.

2. The k-effective shall be ≤ 0.9 under conditions of optimum water moderation for individual accumulations of fissile material in an array with full water reflection around the array, or for arrays of fissile material, the solid angle criterion omega $\leq 9-10$ k-effective shall be satisfied where k-effective represents unreflected accumulation of fissile material in the array under conditions of optimum water moderation, and the conditions in Subsection 7.15.2.1.3 are met.

3. The k-effective value for an individual accumulation may be determined from hand calculations using nuclear properties given in the ARH-600 Handbook[9] or the DP-1014 Report,[10] or the k-effective value of an individual accumulation or assembly of fissile material or an array of fissile material may be determined using the KENO-IV Code,[8] provided the upper limit k-effective of at least 2 sigma is used in these criteria.

4. Different computational methods may be used to determine the k-effective values or to demonstrate the criticality safety of arrays provided they demonstrate conservatisms as good or better than those shown in Figures 7-6 and Tables 7-2 and 7-3.

5. In evaluating the subcriticality of individual accumulations or arrays of fissile material, the two-contingency criterion shall be applied: that two independent and unrelated accidents must occur before criticality is credible.

6. If special nuclear materials are to be stored, handled, or used in the presence of heavy water, berrylium, graphite, or diphenyl, possible moderating and reflecting effects of these materials shall be considered.

Bases for these criteria are presented below:

The use of a limiting value for k-effective $\leq$ 0.9 is conservative. For example, the minimum fissile mass required for k-effective of 0.9 must be increased by a factor of 1.6 to 2.0 to achieve a critical accumulation under conditions of optimum water moderation and full water reflection.

The Solid Angle Method is inherently conservative, as illustrated by the comparison of the wide range of critical data points with the solid angle criterion in Figure 7-6.

Determination of k-effective values with the use of hand calculations using nuclear properties given in References 9 and 10 and the KENO-IV Code[8] is conservative. As demonstrated in Table 7-3, calculated results either differ negligibly from the critical data or are in the conservative direction as compared to critical data.

The two-contingency criterion is needed to ensure that failure of only one criticality control will not result in a criticality accident. For example, the application of a safe batch limit (0.45 x the minimum critical mass) is an application of this criterion when double batching is credible due to permissible operating procedures. For fissile materials permitted at the NTR, a safe batch for an accumulation of fissile material would have a k-effective value less than 0.9.

The conditions in Tables 7-2 and 7-3 have been studied in which water is the moderator and the reflector. If super-normal moderators such as heavy water, graphite, beryllium, or diphenyl are present in facilities external to the NTR where fissile material is handled, smaller minimum critical masses than those shown in Table 7-2 may be possible. Smaller minimum fissile material masses required to achieve an accumulation whose k-effective is 0.9 may be smaller than those shown in Table 7-3.

8. INSTRUMENTATION AND CONTROL

8.1 GENERAL

The reactor is equipped with sufficient instrumentation to control operation of the facility, measure operating parameters, warn of abnormal conditions, and scram the reactor automatically if an abnormal condition occurs (Figure 8-1 and Table 8-1). All reactor scram functions cause a loss of energizing currents to electromagnets, which, when deenergized, permit rapid insertion of the spring-loaded safety rods. The energizing currents can be disrupted by contacts in the power switches, scram relays, or by a manual scram switch. The power switches are controlled by logic units which monitor the trip circuits, on a two-out-of-three coincidence basis, for high reactor power from 3 picoammeters and for loss of high voltage for the three Compensated Ion Chambers (CIC) in the picoammeter channels. Another logic unit monitors singly (noncoincidence) the fast reactor period trip, and high log N power trip. All other scrams, except the console manual scram, operate through the scram relays and are initiated by the following signals:

1. Log N amplifier mode switch position
2. Log N CIC loss of "positive" high voltage
3. Primary coolant high core outlet temperature
4. Primary coolant flow low
5. Loss of ac power
6. Reactor cell manual scram.

Safety is also provided by having each scram (except loss of ac power) cause the control rod drives to run to their fully inserted positions. Also, a rod withdrawal permissive interlock is provided that blocks control rod and safety rod withdrawal if a picoammeter is not indicating above a preset minimum level. The rod withdrawal permissive circuit ensures that instrumentation is seeing the neutron source for reactor startups. Additional interlocks associated with the rod drive system include the following:

1. For initial startup, or following a scram, magnets cannot be energized unless all safety and control rods and the neutron source are at their inner limits.

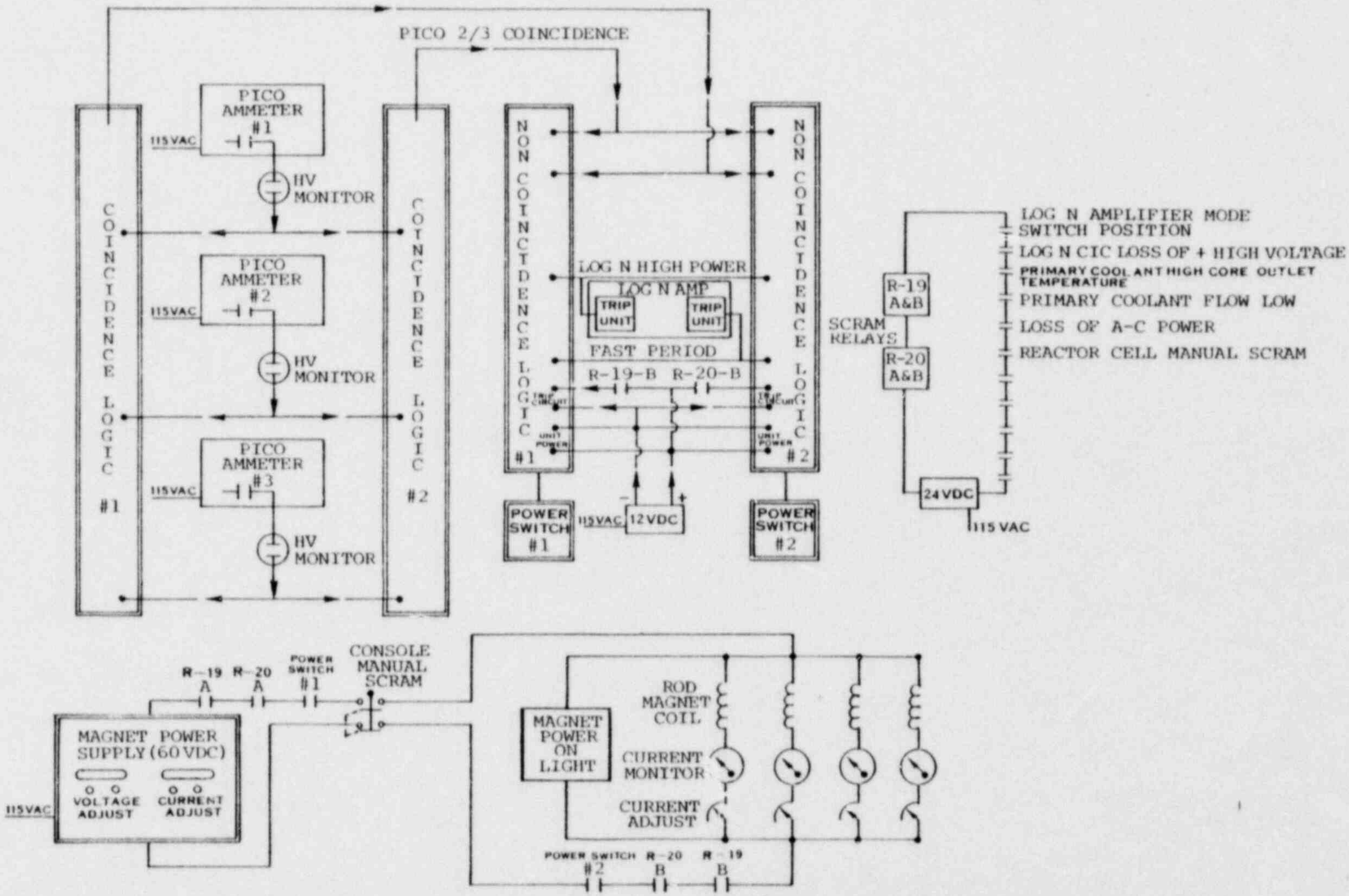


Figure 8-1. NTR Scram Systems

POOR ORIGINAL

2. Safety rods must be drawn one at a time to their outer limits before more than one control rod can be withdrawn.

3. For rod inspection and testing, an interlock is provided that permits the withdrawal of any single rod at one time but no more than one at a time. A selector switch allows any one rod to be withdrawn at a time but must be returned to the FULL IN position to allow normal start-up sequence.

To keep the system as simple as possible, bypasses are not provided in most of the scram circuits. It is felt that simplicity and ease of operation are more important than continuity of operation. If important components become defective, the reactor is shut down until repair or replacement is completed. However, some bypasses are necessary; for example, an automatic bypass has been provided for low primary coolant flow while at powers less than 0.1 kW.

The fail-safe philosophy has been incorporated into the design as much as is practical. In most instances, circuits are completed by energized relays or actuated microswitches to give protection against loss of voltage, poor contacts, or broken wires. Manually operated switches with a spring-return to open circuit (or a more safe position) are used where practical, and include places such as the control rod drive circuits.

8.2 OPERATION CONTROL STATION

The location of the reactor control room is shown in Figures 1-1 and 3-1. The reactor cell is immediately to the north, the south cell to the east, Building 105 hallway on the south, and other laboratories to the west. The shielded doors between the control room and the two cell areas are the only personnel entries to these areas. Flashing warning lights at each doorway are actuated if the doors are opened; however, the radiation level in the south cell must be above a preset minimum for this warning device to actuate at the south cell door. Since the south cell door will be open to perform some experiments, it also has an audible alarm which may be actuated by breaking the beam of light from an electric eye across the doorway. This alarm system alerts the reactor operator to traffic to or from the cell as required.

The reactor control console is a vertical metal structure approximately 6 feet high by 13 feet wide, designed to accommodate racks of standard 19-in. instrument chassis. Attached to the front of the vertical panel is a small sloping benchboard that contains the controls and indicating devices for the rod drives and most of the lights and switches for the alarm system. Also attached to the front of the panel is a horizontal work surface for the convenience of the operator. The vertical panels contain visual readout, power supplies, and recording devices for nuclear process and stack effluent release parameters.

An intercommunication system is installed with stations in the control room, set-up room, office areas, shop, north room, and others, as required. A loudspeaker page system is also available, which can be heard in all NTR areas. Communication between the control room and reactor cell is by a microphone and speaker system.

8.3 SCRAM SYSTEM

The scram system, which is shown in Figure 8-1 and Table 8-1, consists of manual, process, and nuclear scrams.

8.3.1 Manual Scram

The manual scram system consists of a manual button/switch located on the reactor console. When depressed, it directly opens the circuit supplying power to the safety rod magnets, providing a method for the reactor operator to manually shut down the reactor if an unsafe or abnormal condition should occur, and the automatic reactor protection action, as appropriate, does not function.

8.3.2 Process Scrams

The process scram chain consists of relay contacts and switches connected in series between the +24V direct current bus and the coils of scram relays R-19 and R-20. Normally open contacts of these relays are in the circuit supplying power to the safety rod electromagnets and in the circuits for the rod drive motors. Two additional normally open contacts of these relays are used in the

Table 8-1

Scram Systems

Item No.	System	Condition	Trip Point	Function
1	Linear	High reactor power	$\leq$138 kW	Scram (2-out-of-3 or 1-out-of-2)
		Loss of positive high voltage to ion chambers (if ion chambers are used)	$\geq$90% of operating voltage	Scram (2-out-of-3 or 1-out-of-2)
2	Log N	High reactor power	$\leq$138 kW	Scram
		Short reactor period	$\geq$ +5 seconds	Scram
		Amplifier mode switch not in operate position	NA	Scram
		Loss of positive high voltage to ion chambers (if ion chambers are used)	$\geq$90% of operating voltage	Scram
3	Primary Coolant Temperature	High core outlet temperature	$\leq$222°F	Scram
4	Primary Coolant Flow	Low flow	$\geq$15 gpm when reactor power is >0.1 kW	Scram
5	Manual	Console button depressed	NA	Scram
6	Electrical Power	Reactor console key in off position	NA	Scram
		Loss of ac power to console	NA	Scram

process scram chain parallel to the rod-in limit switches; this parallel circuit requires all motor-driven rods to be fully inserted before the scram relays can be energized. Any off-standard condition of any component supplying action to either the switches or to the relay contacts in the scram chain will disrupt power to scram relays R-19 and R-20 and cause them to deenergize. Process scrams are as follows:

1. The log N channel has an amplifier position mode switch which is used for checkout and testing of the instrument. Not having the amplifier mode switch in the operate position will prevent the safety chain from being made up, or, if moved from the operate position during operation, will scram the reactor.

2. Loss of positive high voltage below a predetermined value to the log N Compensated Ion Chamber (CIC) will scram the reactor. Loss of positive high voltage provides assurance that the ion chamber is capable of detecting neutrons.

3. A thermally actuated switch in the core outlet line senses the primary water core outlet temperature. A high outlet temperature will cause the switch to deenergize the scram relays and scram the reactor.

4. The primary flow is measured with a differential pressure transducer sensing the pressure drop across an orifice in the primary water coolant loop. An electric signal from the transducer is indicated at the control console. The reactor will scram when reactor power is greater than 0.1 kW and the primary coolant flow drops below a predetermined value.

5. Loss of ac power to the console will cause the scram relays and the magnet power supply to deenergize and scram the reactor.

6. A manual scram button is located in the reactor cell to scram the reactor from this area, if required. Actuation of this button also deenergizes the scram relays and will scram the reactor.

When the scram relays (R-19 and R-20) are deenergized, the following actions take place:

1. The power being supplied to the safety rod electromagnets is interrupted to allow the spring-loaded safety rods to be inserted.

2. All rod and motor circuits are closed to cause the rod carriages to drive in.

3. The process scram chain is blocked open until the scram condition is corrected and all rods are fully inserted.

In addition to the above actions, the scramming condition will cause an annunciator to actuate, give an audible alarm, and illuminate a push-button lamp on the console which indicates the source of the scram. The audible alarm will continue until an ACKNOWLEDGE push-button switch is actuated; the push-button lamp remains illuminated until the scram condition is corrected and the push-button lamp is depressed. Some conditions cause indicator lamps to illuminate, but do not cause audible alarm; these conditions are not scrams.

8.3.3 Nuclear Scrams

The nuclear scrams consist of four power range channels. A block diagram of the system is shown in Figure 8-1. The power range instrumentation is used to monitor neutron flux (reactor power) and to protect the reactor against excessive power levels or rates of power rise. This instrumentation is required to be operable and connected to the safety system during each startup and the subsequent operating period. The system consists of four independent neutron detection channels; three are monitored by picoammeters and the fourth by a log N and period amplifier. The three picoammeters have trip circuits which operate into a two-out-of three (or one-out-of-two if one channel is inoperative) coincidence logic circuit capable of causing reactor scram. The log N and period amplifier is capable of causing reactor scram on high power or on fast period. Minimum operating range is required to be 150×10^{-8} to 150% full power for the picoammeter channels and 100×10^{-5} to 150% full power for the log N channel.

A gamma-compensated ion chamber is used as a detector in each power range channel. The detectors are positioned in thimbles in the fuel storage tank or at one of the faces of the reflector. The exact location selected for a particular chamber is determined by the intended use of the reactor, sensitivity of the system, and the desired meter reading. The desirability for seeing the neutron source for startup, the minimization of shadowing effects, and the provision of physical protection for the chamber are the primary factors considered when positioning the chambers. The CIC output currents can be interpreted in terms of reactor thermal power through calibrations based on measurement of thermal power as determined by a heat-balance measurement which utilizes the coolant flow through the core and the differential temperature across the core.

High voltage for the three picoammeter CICs is supplied by three dual high-voltage supplies. Each high-voltage power supply is capable of supplying positive and negative regulated 600 volts to all three channels. The positive side of each channel is monitored by a meter relay that will introduce a trip to the coincidence logic if positive high voltage is lost. When in a two-out-of-three situation, loss of positive high voltage on any two power supplies, or loss of high voltage on one power supply with two or more channels connected to it will cause a reactor scram. When in a one-out-of-two situation, loss of positive high voltage on any one power supply will cause a reactor scram.

High voltage for the log N period amplifier CIC is supplied by a power supply in the log N period amplifier. Loss of positive high voltage from this power supply will initiate a scram through the process scram relays.

Three multirange picoammeters are normally used (although operation is permitted with two) to monitor the signals from three (or two) of the CICs. Each picoammeter has 19 ranges covering 9 decades. Sensitivity and range of the systems are such that the flux (with the reactor shut down) from the reactor neutron source will bring all channels well on scale, and maximum reactor power does not exceed the range of the instrument. Each picoammeter amplifier output signal, in addition to driving the picoammeter meter, is connected to an internally mounted trip circuit and externally through a selector switch to a linear power recorder. Each trip circuit is set to trip when the meter reads 138 kW or less. When the instrument reading is less than the trip point, the trip circuit supplies 12 Vdc to a coincidence logic circuit, wired to cause reactor scram if 2 of 3 (or 1 of 2) inputs are tripped. The picoammeter(s) (at least one) also have an interlock at the low end of the scale which prevents rod withdrawal if the picoammeter(s) are not downscaled to a sufficiently sensitive scale for startup. Reactor operation may continue with one picoammeter out of service, provided the trip circuit is set up so that a trip signal from either of the remaining picoammeters will cause reactor scram. If one picoammeter is out of service, the interlock at the low end of the scale for that picoammeter (if applicable) which prevents rod withdrawal may be bypassed. These automatic actions ensure that the picoammeters have the proper start-up sensitivity and the high power scram trip point is always within a decade of operating power during operations which increase power.

The log N and period amplifier receives its signal from the fourth CIC and displays the reactor period and reactor power on front panel meters. This system may be set up to cover the power range from source or reactor critical level, depending on CIC position, to 150% of power. Relay outputs from the period amplifier trip circuit and the log N amplifier trip circuits are connected through the noncoincident logic circuit to initiate a scram for reactor periods of less than 5 seconds or reactor power greater than 138 kW. At powers of less than 0.1 kW, a signal from the log N recorder actuates automatic bypass of the primary coolant low-flow scram. At powers greater than 0.1 kW, a signal from the log N recorder actuates a relay which automatically switches the signal to a single function recorder from the start-up channel (source range monitor) if utilized to the thermocouple pile (millivolts) signal, which is required in the heat balance calculation. The log N power signal is also recorded on a strip chart recorder. The mode (multiposition calibration) switch for the log N amplifier is interlocked to scram the reactor when the switch is not in the OPERATE position.

The diode logic element system consists of two units. One unit performs coincidence logic functions and the other performs noncoincidence logic functions on signals from the nuclear instrumentation system.

The coincidence logic unit contains five independently functioning component boards which can accommodate a total of 16 signals. Four of the component boards are identical, and provide circuits for performing two-out-of-three coincidence logic. The fifth circuit component board (not used) is designed to perform selective two-out-of-four coincidence.

The 12-V trip output from each of the three picoammeters passes through contacts in the meter relays monitoring positive high voltage to the CIC in that channel. The trip outputs from the three channels are converted parallel to the 2/3 coincidence logic unit. A trip on any two picoammeters or loss of high voltage to any two CICs or a loss of voltage on one channel plus a high power trip in another will cause trip outputs from the coincidence logic unit to be sent to the noncoincidence logic unit which causes deenergization of the power switches.

The noncoincidence logic unit contains two independent noncoincidence logic component boards, each of which accommodates nine input signals and provides one output signal. Depending on the input signal levels, each noncoincidence logic component board provides either 16 volts dc or less than 1 volt output to a power switch. For the output level to be 16 volts, all of the inputs must be normal. If any one or more inputs drop to zero, the output signal drops to less than 1 volt.

Input signals to each nine-channel noncoincidence logic board consist of the following:

1. Two 24-V signals that pass through scram relay R19B;
2. Two 24-V signals that pass through scram relay R20B;
3. One 12-V signal from log N high power trip;
4. One 12-V signal from log N fast period trip; and
5. Three 12-V trip signals from one coincidence logic unit.

Output signals from each noncoincidence logic board goes to a power switch which controls the electromagnet excitation current. If either power switch trips, current to all magnets will be interrupted.

Power for the safety rod electromagnets is supplied from a direct current power supply with the capacity of supplying all four electromagnets. Power to each electromagnet is routed through individual power-adjust modules so that minor variations in the electromagnets can be compensated.

8.4 SAFETY-RELATED SYSTEMS

Safety-related systems are listed in Table 8-2. They consists of instrumentation and systems to assist in the operation of the facility, measure operating parameters, or warn of abnormal conditions. The safety-related instrumentation or systems provided include the following:

1. A differential pressure switch measures the pressure difference between the reactor cell and control room. This switch actuates a visual and audible alarm if cell negative pressure drop below a preset level (not less than 0.5 inches of water). The reactor power must not be increased above 0.1 kW unless the cell negative pressure is as noted above. If the cell negative pressure drops below the preset level and the reactor power is above 0.1 kW, then the reactor power shall be lowered to $\leq$ 0.1 kW immediately and corrective action taken, as required. This ensures that the direction of air flow is from the control room into the reactor cell and that potentially contaminated reactor cell air due to reactor operation is released through the ventilation system.

2. Liquid level switches are provided on the fuel loading tank and actuate an alarm circuit when the tank is either low or too high. As long as the level is above the low-level alarm, it can be assured that the core tank is filled with water. The high-level alarm assures that adequate indication is given to the operator during the filling of the fuel loading tank that it will not overflow.

3. Thermocouple in the primary water core outlet line senses the primary water temperature and reads out on a panel meter in the control room. A high-temperature warning alarm is actuated when the set point is reached to indicate a high primary water temperature.

Table 8-2

Safety-Related Systems

Item No.	System	Condition	Set Point	Function
1	Reactor Cell Pressure	Low differential pressure	$\geq$0.5 inch of water	Visible and audible alarm; audible alarm may be bypassed after recognition
2	Fuel Loading Tank Water Level	Low water level	$\geq$4 feet above the fuel loading chute	Visible and audible alarm; audible alarm may be bypassed after recognition
3	Primary Coolant Temperature	High core outlet temperature	$\leq 140^{\circ}$F	Visible and audible alarm; audible alarm may be bypassed after recognition
4	Primary Coolant Core Temperature Differential	Core delta temperature	NA	Provide information for the heat balance determination
5	Radiation Monitors	North room high level	$\leq$100 mr/h Above background	
		South cell high level	$\leq$100 mr/h Above background	
		Reactor cell high level (reactor shutdown)	$\leq$100 mr/h Above background	Visible and audible alarm; audible alarm may be bypassed after recognition. May be temporarily out of service if portable instruments are used during personnel entry and occupancy
		Reactor cell high level (reactor operating)	$\leq$100 R/h	
		Control room high level	$\leq$5 mr/h Above background	

Table 8-2 (Cont'd.)

Safety-Related Systems

Item No.	System	Condition	Set Point	Function
6	Stack Radioactivity	Beta-gamma particulate high level	$\leq 1 \times 10^4$ cpm	Visible and audible alarm; audible alarm may be reset after recognition
		Noble gas high level	$\leq 2 \times 10^{-11}$ amps	
7	Linear Power	Low power indication	$\geq$5% of full scale	Safety or control rods cannot be withdrawn (1-out-of-3 or 1-out-of-2)
8	Control Rod	Rods not in	NA	Safety rod magnets cannot be reenergized; may be bypassed to allow withdrawal of one control rod or one safety rod or one safety rod drive for purposes of inspection, maintenance, and testing
9	Safety Rod	Rods not out	NA	Control rods cannot be withdrawn; safety rods must be withdrawn in sequence; may be bypassed to allow withdrawal of one control rod or one safety rod or one safety rod drive for purposes of inspection, maintenance, and testing

4. A thermocouple pile is provided which indicates the primary coolant core temperature differential. This is utilized in combination with the primary coolant flow rate to provide information for a heat balance determination.

5. An area radiation monitor system is provided, as discussed in Subsection 8.6. The use of this system will assure that the area(s) of the facility in which a potential high-radiation area exists are monitored to assure protection of personnel.

6. An NTR stack radioactivity air monitoring system is utilized as described in Subsection 6.7. Separate detection channels and alarms are used for particulate material and nonfilterable radioactive gases to assure that the releases are acceptable.

7. A low power level rod block and alarm is provided on the linear power system. This rod block and alarm assures that the operator has a linear power channel operating and indicating neutron flux levels during rod withdrawal.

8. Interlocks are provided on the control rods to prevent outward movement unless the safety rods are all in a full-out position. This condition assures that the reactor will be started up by withdrawing the four safety rods prior to withdrawing the control rods. A bypass is provided for testing purposes.

9. Interlocks are also provided on the safety rods. Each safety rod must be withdrawn in sequence to assure the normal method of reactivity control. A bypass is provided for testing purposes which will allow any one safety rod or safety rod drive to be withdrawn.

Other systems provided include the following:

1. A variable area flow meter (Rotometer type), mounted in the control room, to indicate the secondary coolant flow to the tube side of the heat exchanger;

2. A recorder to monitor thermocouples placed at several locations throughout the primary and secondary systems and the graphite pack;

3. A constant air monitor located in the control room to monitor the air activity in the reactor cell. The monitor is checked prior to each initial entry into the reactor cell.

4. A differential pressure switch senses the pressure difference between the core inlet and outlet lines. When core differential pressure, which is an indication of flow, falls below a preset value, an alarm circuit is actuated and indicates a low differential pressure in the core.

5. A source range monitor channel is provided for use during fuel changes or special tests, as required. The source range monitor channel consists of a proportional counter detector, its power supply, a pulse amplifier, a log count rate meter and a recorder. A scalar and audio signal output are also available from the count rate meter and are normally connected, as shown in Figure 8-2. The source range monitor recorder is utilized to record the thermocouple pile (millivolts) signal when power is greater than 100 watts, as discussed in Subsection 8.3.3.

8.5 REACTOR REACTIVITY CONTROL SYSTEMS

Three types of movable neutron poisons are provided to control core reactivity: safety rods, control rods, and manual poison sheets. All these poisons are located about the periphery of the fuel container (Figure 4.1), and all run in guides that extend from the south end of the fuel container through the reflector and shield to the north face of the reactor. The guides place the center of the poisons on a 9.5-in. radius or about 0.6 inch from the outside edge of the active core. The control and safety rods have horizontally mounted drive mechanisms that are supported from the north face of the reactor on a 5-ft-high aluminum support plate located about 4-1/2 feet in front of the north face. The manual poison sheets are inserted or removed manually through access holes provided in the north shield. Each type poison was designed to perform

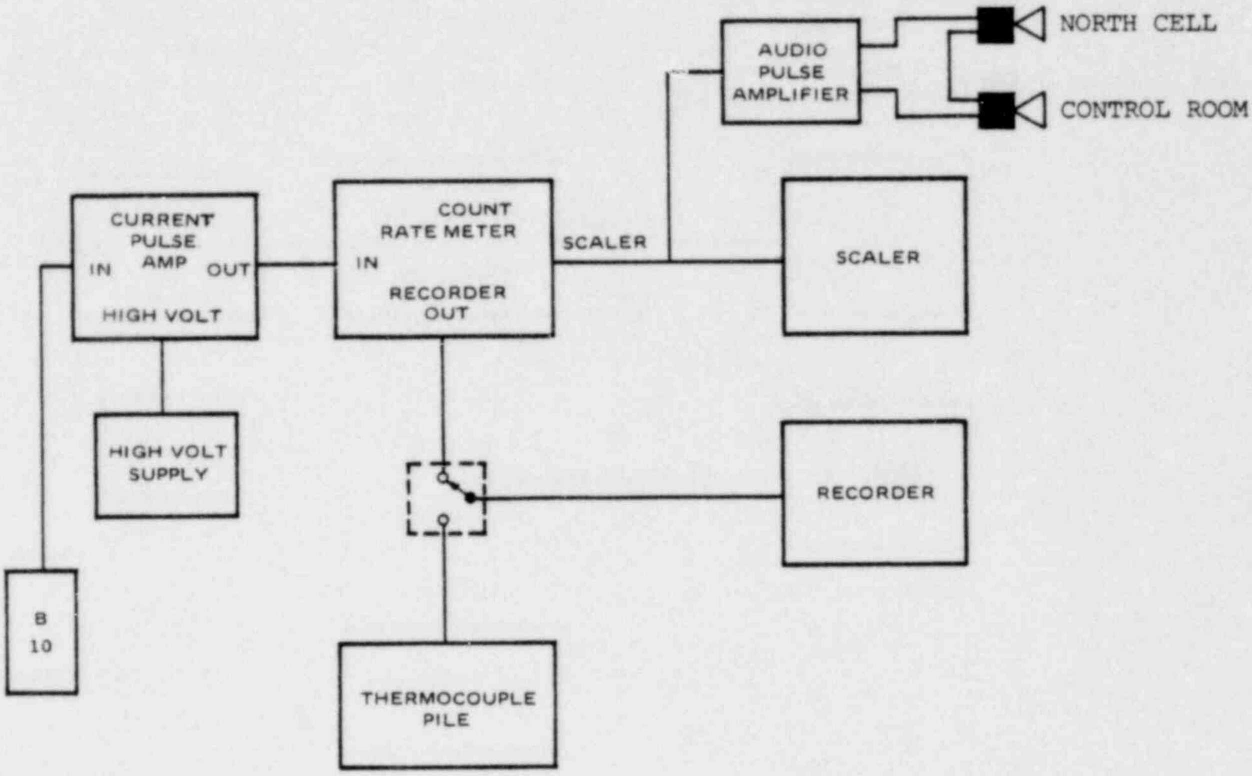


Figure 8-2. NTR Source Range Monitor Channel

a specific function. The four safety rods were designed for rapid insertion to scram the reactor. The control rods (two coarse and one fine) were designed for the precise position control and indication required for analytical work during which the reactor is used as a detector. The manually positioned poison sheets are used to limit the reactivity available to the operator or to increase the shutdown margin. Figure 8-3 shows the control circuits for the safety rod and control rod drives.

Relative positions of the four boron-carbide-filled safety rods are shown in Figure 4-1. The poison section of each rod is 20 inches long and consists of a solid core of 1/2-in.-diam boron carbide cylinders contained in a stainless steel tube. A plug in the north end of the stainless steel tube connects to an extension rod which has a rod stop armature assembly pinned to the other end. Two spiral springs are attached to the extension rod so that withdrawal of the safety rod cocks the springs to store energy. Housings for

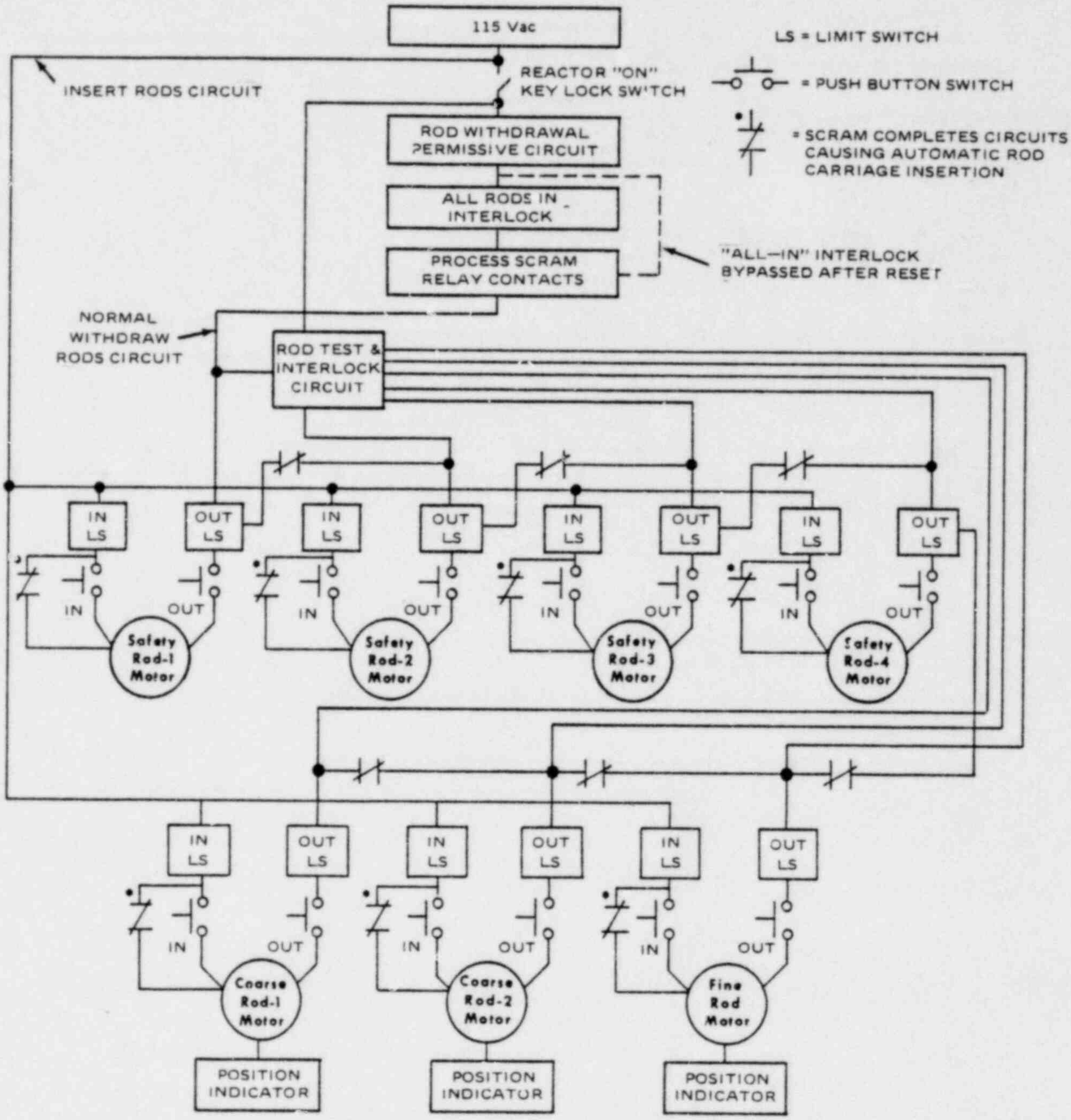


Figure 8-3. Simplified Block Diagram of Rod Drives

the springs are secured to a support bracket attached to the north shield face. Also attached to this support bracket is a long piece of steel angle which passes through and is attached to the aluminum support plate to protect and support most of the hardware for the drive mechanism.

The safety rod is held to the rod drive by an electromagnet that engages the armature attached to the extension rod. The electromagnet is attached to a drive nut that moves horizontally on a lead screw. Rotation of the lead screw is accomplished with an electric motor through a belt and pulley drive. The electric motor is a 1/12 hp, capacitor start and run. 57-rpm output, gear motor connected for instantaneous reversing and is provided with automatic reset overload protection. Power to the motors is from the 115-Vac supply. Remote manual control by the operator is by pushbutton switches at the console. A circuit is provided to run the carriage automatically to the fully inserted position following a reactor scram, provided ac power to the console is maintained.

The sequence of operations to withdraw a safety rod is as follows;

1. Run carriage in to engage armature;
2. Energize the electromagnet; and
3. Run carriage out to withdraw safety rod and cock the spiral springs.

The rods can then be positioned by moving the carriage. Upon initiation of a scram signal, the following sequence takes place:

1. Scram signal deenergizes the electromagnets;

2. Spiral springs cause the armature to separate from the magnet and rapidly insert the safety rods; and

3. Automatic signal runs all rod carriages to the fully inserted positions.

Deceleration of each scrammed safety rod is accomplished by an air dashpot-type shock absorber. The rod-stop armature begins to compress the air in the shock abosrber housing about 4 inches from the full-in position. An orifice is provided to control the release rate of the compressed air and the deceleration rate of the rod.

Four microswitches are associated with each safety rod and drive mechanism. Listed below are the actions initiated by each switch:

1. Drive-Out Limit Switch

   a. Interrupts motor circuit at outer limit of stroke.

   b. Energizes yellow light at console.

   c. Interlocked so that all safety rods must be withdrawn sequentially before control rods can be moved, except when rod inspection and test switch is used.

2. Drive-In Limit Switch

   a. Interrupts motor circuit at fully inserted limit of stroke.

   b. Energizes green light at console.

   c. Interlocked to prevent energizing the electromagnets unless all rods are fully inserted.

3. Safety-Rod-In Position Switch

   Energizes green light at console.

4. Separation Switch

   Operates in series with carriage-out limit switch to energize yellow light, indicating that rods and carriage are out.

The scram mechanisms for the NTR are essentially the same as those that operated satisfactorily on the TTR reactors for many years. The present mechanisms have operated without showing appreciable wear. As required by the administrative procedures, flight time for the rods is measured periodically, and, if it is found that a rod does not meet the required insertion time, the rod will be considered inoperable until repairs are made.

As a result of the lack of symmetry in the arrangement of the nuclear poisons around the core and the possibility of strong shadowing effects, the reactivity worth of the individual safety rods vary. In the normal core, the reactivity worth of the most effective safety rod is about 1$. The minimum worth of all four safety rods is 2$. If it is assumed the entire worth of the rod (for conservation, use 1.5%) is realized in the first 18 inches of rod movement at the withdrawal speed of 1 in./sec, the average reactivity addition rate by withdrawal of the most effective safety rod is only 8.3 ¢/sec, which is a reasonable addition rate for manual control. Actually, the full stroke of the safety rod is approximately 28 inches and the rods are interlocked so that each rod must be fully withdrawn before the next one can be started out. Thus, the reactivity in two safety rods cannot be added to the reactor in less than approximately 1 minute by normal withdrawal.

Figure 4-1 shows the location of the two coarse control rods with respect to the core and other neutron poisons. The poison section of each rod is 16 inches long and consists of a solid core of 1/2-in.-diam boron carbide cylinders contained in a stainless steel tube. A plug in the north end of the tube is connected to an extension rod which is attached to a yoke that is positioned by the drive mechanism.

This yoke is fastened to a lead screw that runs through a sprocket and nut assembly connected through a chain drive to a gear motor. The gear motor is identical to the one described above for the safety rods. Power to these motors is supplied by the 115-Vac supply. Pushbutton switches at the console permit manual control. As with the safety rods, scram provides an automatic signal which takes control away from the operator and runs the rods to their fully inserted position, provided ac power to the console is maintained. Position indicators are provided to indicate the position of each rod, to better than the nearest inch, over the full stroke for rod movements in either direction.

Two limit switches on each control rod drive mechanism perform the following functions:

1. Rod-In Limit Switch

   a. Energizes green light at the console.

   b. Interlocked to prevent energizing the electromagnets unless all control rods (in addition to the safety rods) are fully inserted.

   c. Interrupts the motor circuit at the full in position.

2. Rod-Out Limit Switch

   a. Energizes yellow light at the console.

   b. Interrupts motor circuit at the outer limit of the stroke.

Location of the fine control rod with respect to the other poisons is shown in Figure 4-1. The poison section is 18 inches long and consists of a solid core of 0.365-in.-diam boron carbide cylinders contained in a stainless steel tube that extends through the north face of the reactor to the drive mechanism. An aluminum rod is used to fill the remainder of the tube between the boron carbide cylinders and the drive mechanism.

The stainless steel tube containing the poison connects to a nut block that travels on a lead screw. The lead screw is rotated through a right angle gear box by a gear motor. The gear motor is a 1/20-hp, split capacitor, 173-rpm output with an electrically operated brake. Power to the motor is supplied by the 115-Vac supply. Pushbutton switches on the console are used for remote manual control. The fine control rod is automatically driven to the fully inserted position following a scram, provided ac power to the console is maintained.

Two limit switches actuated by the traveling nut block perform the identical functions discussed above for the control rods. Position indicators are provided to indicate fine rod position to the nearest 0.5 inch over the entire stroke for rod movement in either direction. Readout is in 0.1-in. increments.

The control rods were calibrated using the rising period technique, with the rods in an essentially unshadowed condition. The results indicate the total worth of all the rods is approximately 2.3$. The speed of withdrawal of each coarse control rod drive is 0.138 in./sec. The speed of withdrawal of the fine control rod drive is 0.11 in./sec. If it were assumed that all three rods were withdrawn simultaneously, the average reactivity addition rate would be approximately 2 ¢/sec, which is an amount that is easily manageable.

Six manual poison sheets can be inserted in the graphite around the fuel container (Figure 4-1.).

The sheets consist of 0.031-in.-thick cadmium, 19 inches long, laminated between two sheets of 0.08-in.-thick 6061-T6 aluminum. The width of the cadmium in each sheet is as follows:

1. Full sheet: 2.75 inches
2. 3/4 sheet: 2.06 inches
3. 1/2 cheet: 1.38 inches
4. 3/8 sheet: 1.06 inches
5. 1/4 sheet: 0.69 inches
6. 1/8 sheet: 0.34 inches

All manual poison sheets used are equipped with a spring-loaded latch handle that latches to a special latch plate on the north face of the aluminum box that contains the graphite reflector reactor assembly. This latching assembly provides positive restraint of the manual poison sheets with respect to the reactor assembly.

The manual poison sheets do not have a drive mechanism or any automatic functions associated with them. A sheet can only be inserted or withdrawn by entering the reactor cell to unlock and remove a shield plug in the shield face. The sheet is then positioned by engaging the sheet handle with a special latching tool and physically unlatching or latching it prior to removal or full insertion. The removed sheets are stored in a rack in the reactor cell and are accounted for before reactor startup.

Reactivity worth of individual manual poison sheets in the core with all safety rods inserted were obtained by utilizing a pulsed neutron source. The worth of a full sheet was approximately 1$, and a half-sheet was worth about 0.5$. The worth of all six manual poison sheets is approximately 3.0$. In a typical core configuration (1/2 sheet in slot #1 and 3/8 sheet in slot #5), the worth of the manual posion sheets is approximately 0.[illegible] (subject to change based on experiment worth, etc.). Excluding the transient temperature worth (reactivity addition from the primary coolant temperature change) and the experiment transient worth, the typical excess available from the control rods at the console is 0.3$ (subject to change based on experiment worth, etc.).

8.6 RADIATION MONITORING SYSTEMS

Radiation levels (gamma) are monitored by a five-station remote area monitor. Areas monitored are the reactor cell, south cell, control room, and north room (two stations). Radiation levels are indicated on the control console. Each channel is equipped with an alarm which will actuate visual and audible alarms in the control room and the affected area. The set points of the detectors are as specified in Table 8-2. In addition, the south cell monitor is interlocked with the south cell shutter and door controls to prevent inadvertent exposure to the radiation beam from the reactor.

The system for monitoring airborne radioactivity in the reactor cell ventilation system is discussed in Subsection 6.7. The set points are as specified in Table 8-2.

8.7 EFFLUENT MONITORING AND ENVIRONMENTAL SURVEILLANCE PROGRAMS

Effluent monitoring includes measurements of airborne radioactivity releases from facility stacks and the measurement of radioactive and nonradioactive constituents in water effluents released through the site sanitary and industrial waste water systems. Environmental surveillance encompasses the measurement of radioactivity in air near or beyond the site perimeter and the measurement of both radioactive and nonradioactive constituents in neighboring streams, wells, soils, and vegetation.

8.7.1 Effluent Monitoring Program

The Effluent Monitoring Program has been developed to ensure that VNC site release limits for water are not exceeded and additionally to ensure that releases are maintained as low as reasonably achievable. Release limits for numerous nonradiological constituents have been established by the California Regional Water Quality Control Board (CRWQCB). Radiological release limits have been established by the Nuclear Regulatory Commission (NRC) and by the California State Department of Health Services (CSDHS).

8.7.1.1 Waterborne Effluents

Waterborne effluents released from VNC site facilities can be classified as industrial waste water, or clean water.

Industrial waste water includes process and cooling water which is first piped to a pH adjustment facility before discharge to one of three available 60,000-gal retention basins. Tests for pH and radioactivity are performed on a water sample from each basin prior to discharge into Vallecitos Creek. In addition, samples from all basin discharges are accumulated and analyzed at specified intervals for a variety of constituents.

Clean-water discharges consist of storm runoff and small quantities of water known to contain no radioactivity other than that from natural background. The latter includes condensate from building air conditioning equipment. These waters flow directly to drainage ditches, which enter Vallecitos Creek.

Sanitary wastes are collected and processed in a septic tank before undergoing sand filtration and chlorination. Processed sanitary waste water is discharged by land disposal (irrigation) onto VNC property.

8.7.1.2 Airborne Effluents

Airborne effluents consist of discharges from the NTR stack and other VNC facility stacks. Stack releases are monitored for radioactivity even though multistage filtering is accomplished prior to discharge. Subsection 6.7 discusses the air-monitoring system utilized.

8.7.2 ENVIRONMENTAL SURVEILLANCE PROGRAM

Water samples are obtained within or beyond the site boundary to ascertain to what extent, if any, VNC discharges are detectable in the environment. Receiving waters, ground water, stream bottom sediments, and soils are monitored for constituents which could have been dispersed by water. Air samples are utilized to detect the presence of radioactivity in air, and vegetation samples have been collected and analyzed in the past to assess the accumulation of constituents from both air and water pathways. Figure 8-4 shows some of the sample stations that are utilized. Sample Stations -2, -4, -5, and -FP are utilized for surface water (receiving), bottom sediment, and vegetation sampling. Stations indicated by a number, letter, and number sequence (i.e., 2L1) are wells.

8.8 NEUTRON SOURCE

A reactor start-up neutron source is installed on an electric motor drive mechanism similar to the control rod drives. The source drive has the same controls and indications as a control rod drive, with the exception that continuous position indication is not provided. The same interlocks as those on the control rods are provided (the safety rod magnets cannot be reenergized until the source is full in), except that it is not necessary to pull any safety rods to withdraw the source. Following a scram, the source automatically runs to the fully inserted position. The source travels in a guide tube identical to that used for the control rods, and the limit switches are

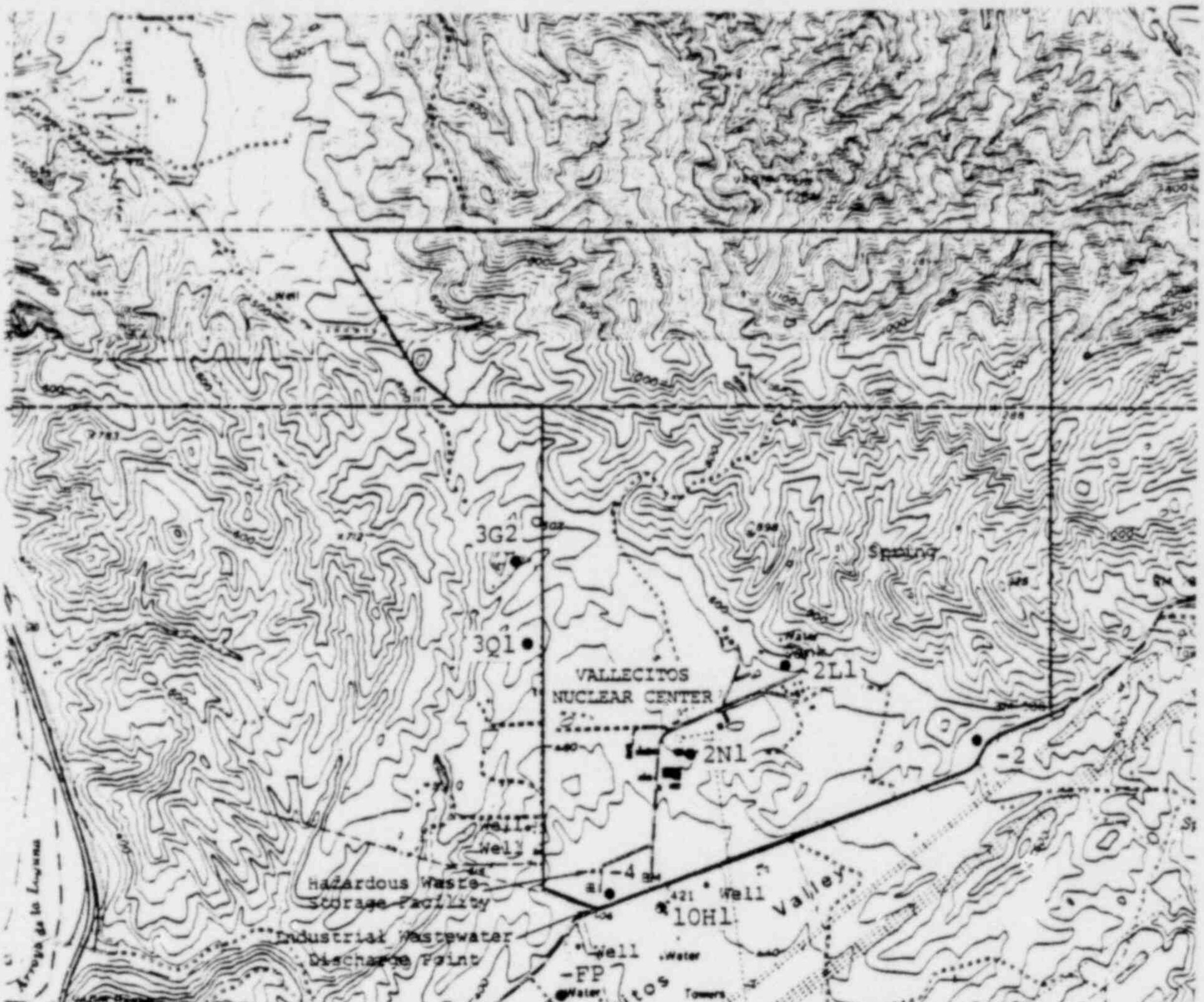


Figure 8-4. VNC Sample Stations

POOR ORIGINAL

adjusted so that the source moves about 30 inches from the in to out positions. A 0.2-Ci radium-beryllium source emitting about 10^6 n/sec is used for a start-up source. It is an R-Monel encapsulation approximately 1/2-in. in diameter and 3-1/2 inches long, attached to an aluminum extension rod that connects to the source drive mechanism. The source-detector arrangement provides at least the minimum neutron flux signal required for the nuclear instrumentation for startup and also gives good indication of subcritical multiplication.

Other neutron sources, including plutonium-beryllium sources, are available for instrument calibrations, functional tests, and experimentation. Use of these sources within the facility is under procedural control.

9. ADMINISTRATION

9.1 OPERATING ORGANIZATION

9.1.1 Summary Description

The establishment of functional levels and assignments of responsibility is the prerogative of the organization authorized to operate the reactor facility.

The Nuclear Test Reactor (NTR) facility organization and interrelationships are shown in Figure 9-1. This figure shows the relationship between the operating organization and the primary supporting organizations. The organization may be modified from time to time to reflect changes in programs and objectives.

The NTR facility has been organized, with responsibilities and the authority to discharge those assigned responsibilities, so that decisions are promulgated at the proper level and with adequate technical advice. Functions performed by one level may be performed by personnel at a higher level, provided they meet the minimum qualifications (i.e., Reactor Operator's license, etc.). The Manager, Reactor Irradiations, is the Facility Manager (subsection level), and has overall responsibility for the operation and administration of the reactor. Reporting directly to the Manager, Reactor Irradiations, is the Manager, NTR (unit level), who is responsible for the safe and efficient operation, maintenance, and repair of the facility. Operation of the reactor may be performed under the direction of a reactor Supervisor. Contributing in a major way to the operating organization, but not reporting to the Facility Manager, is the Nuclear Safety and Quality Assurance organization (NS&QA). Within this organization are nuclear safety engineers, radiological engineers, licensing, safeguards, security, and criticality specialists. This organization also contains health physics monitoring personnel, quality assurance engineers, and quality control technicians.

Also available to the Facility Manager are many other highly specialized technical individuals on and off the Vallecitos Nuclear Center in the Nuclear Energy Business Group of the General Electric Company.

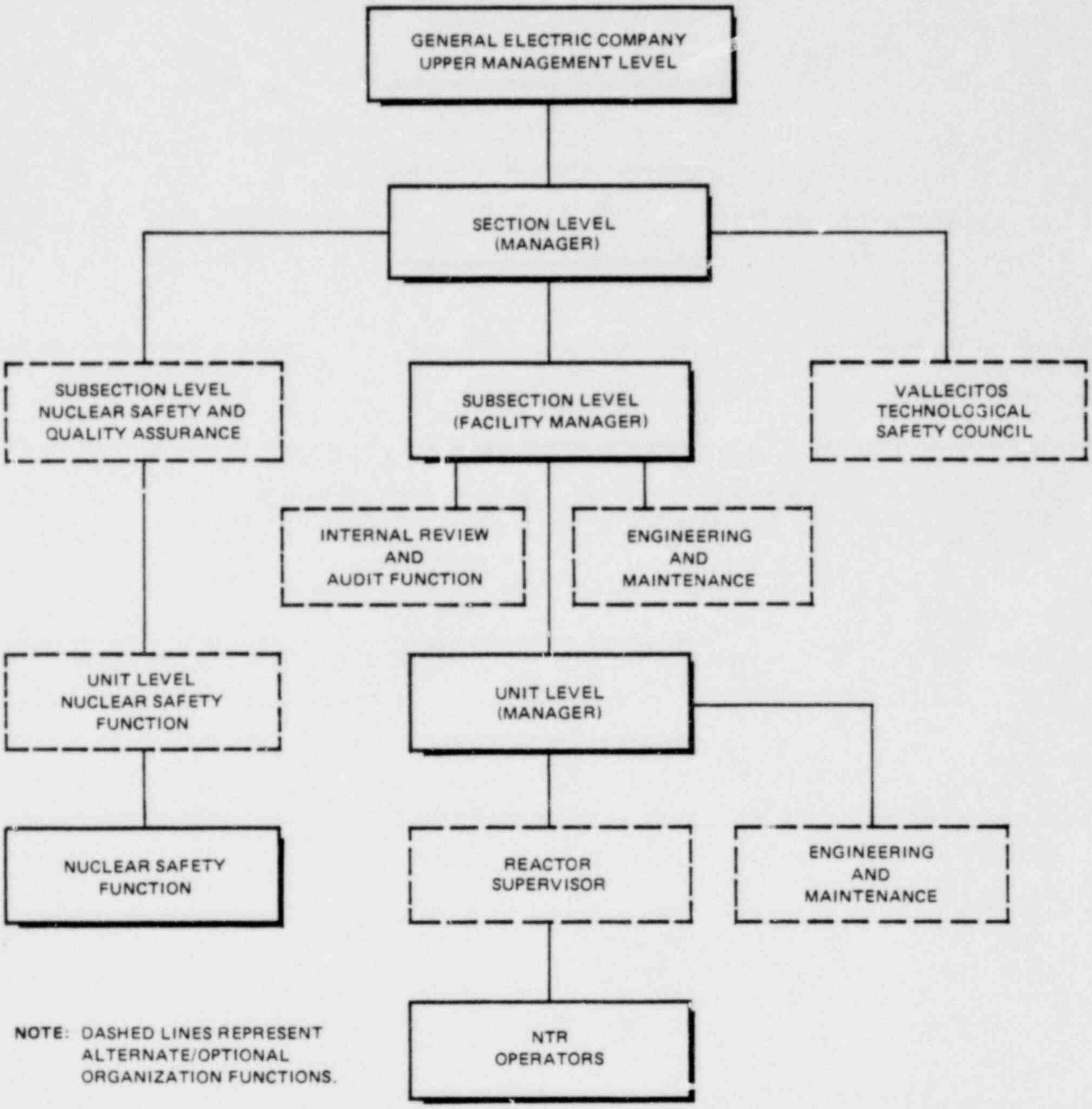


Figure 9-1. Organization Chart

9.1.2 Responsibilities

The responsibilities of selected NTR facility positions are as follows:

1. Manager, Irradiation Processing Operation (IPO) (section level).
   The Manager, Irradiation Processing Operations (section level) has the overall responsibility for the NTR facility's license.

2. Manager, Reactor Irradiations (subsection level)
   The Manager, Reactor Irradiations (Facility Manager), reports to the Manager, Irradiation Processing Operation. He has the over-all responsibility for the safe, reliable, and efficient operation of the NTR. He is responsible for maintaining a competent staff and an effective organization structure as measured by the over-all safety performance. All changes to the facility or facility procedures and all new tests and experiments require the approval of the Manager, Reactor Irradiations, or his designated alternate as defined in the Technical Specifications. He is responsible for an adequate safety review and may utilize resources of other General Electric personnel (or outside consultants) not on his staff.

3. Manager, Nuclear Test Reactor (unit level)
   The Manager, NTR, reports to the Manager, Reactor Irradiations, and is responsible for the safe and efficient operation of the NTR, the direction of the activities of the reactor supervisor, and the NTR engineers(s). He is also responsible for the development, maintenance, and implementation of written operating and maintenance procedures, the coordination of operation maintenance, repairs, modifications, the training and retraining of operating personnel, and all other facility activities, and the safety of assigned and visitor personnel.

4. Reactor Analyst (internal review and audit function)
   The Reactor Analyst reports to the Manager, Reactor Irradiations. He reviews and/or audits reactor operation, maintenance, tests, experiments, and facility and procedure changes. He also investigates all anomalous

conditions or behavior. Primary review and audit responsibilities are designed for compliance with the facility license and Technical Specifications, Code of Federal Regulations, and nuclear safety-related items.

5. Reactor Supervisor
   The Reactor Supervisor (shift supervisors, Operations Supervisor, etc.) reports to the Manager, NTR. He supervises the NTR operation in accordance with written operating procedures, administers planned work, and handles emergencies at the facility. He is responsible for the safety of personnel working at the facility, ensures the security of fissionable materials, regulates entry into radiation and restricted areas, and trains new personnel.

6. NTR Engineer (engineering and maintenance personnel)
   The NTR Engineer (specialist, etc.) may report to the Manager, NTR. He usually provides detailed direction and guidance in the installation and operation of experiment facilities and programs and over-all plant maintenance, repairs, and modifications within the framework of operating procedures.

7. Operating Crew (NTR operators)
   The Operating Crew consists of trainees, as required and qualified, trained technicians, including licensed Reactor Operators, and licensed Senior Reactor Operators, who operate the reactor and experiment facilities in accordance with the operating procedures under the supervision of the Operations Supervisor. Licensed Reactor Operators may direct the activities of trainees, and licensed Senior Reactor Operators may direct the activities of licensed Reactor Operators and trainees, in accordance with the operating procedures.

9.1.3 Qualifications

Operations personnel shall have that combination of academic training, experience, health, and skills commensurate with their level of responsibility which provides reasonable assurance that decisions and actions during normal and abnormal conditions will be such that the plant is operated safely and efficiently in accordance with NRC license requirements and rules and regulations. Minimum qualifications shall include the following:

1. Manager, Reactor Irradiations (subsection level)
   At the time of appointment to the active position, the Manager, Reactor Irradiations (Facility Manager) shall have at least 6 years of nuclear experience. Additionally, he shall have a baccalaureate or higher degree in an engineering or scientific field and have previous managerial experience. Equivalent education or experience may be substituted for a degree. The degree may fulfill 4 of the 6 years of nuclear experience required on a one-for-one basis.

2. Manager, NTR (unit level)
   At the time of appointment to the active position, the Manager, NTR, shall have at least 4 years of nuclear-related experience. Additionally, he shall have a baccalaureate or higher degree in an engineering or scientific field and shall have or immediately pursue the obtaining of a NTR Senior Reactor Operator license. Equivalent education and experience may be substituted for a degree. A maximum of 3 years equivalent full-time academic training may be substituted for 3 of the 4 years of nuclear-related experience required.

3. Reactor Analyst (internal review and audit function)
   At the time of appointment to the active position, the individual assigned the internal review and audit function shall have at least three years of nuclear-related experience. Additionally, he shall have a baccalaureate or higher degree in an engineering or scientific field. Equivalent education and experience may be substituted for a degree. A maximum of 2 years equivalent full-time academic training may be substituted for 2 of the 3 years of nuclear-related experience required.

4. Reactor Supervisor
   At the time of appointment to the active position, the Reactor Supervisor shall have at least 3 years of nuclear-related experience. He

shall have an NTR Senior Reactor Operator license or shall have received sufficient training at the facility or elsewhere to satisfy the requirements of a Senior Reactor Operator's license and shall immediately pursue the obtaining of an NTR Senior Reactor Operator license. A maximum of 2 years equivalent full-time academic training may be substituted for 2 of the 3 years of nuclear-related experience required.

5. NTR Engineer (engineering and maintenance personnel)
   At the time of appointment to the active position, the NTR Engineer shall have at least 1 year of nuclear-related experience. Additionally, he shall have a baccalaureate or higher degree. Equivalent education and experience may be substituted for a degree. One year of equivalent full-time academic training may be substituted for the 1 year of nuclear-related experience required.

6. Operating Crew (NTR operators)

   a. Senior Reactor Operator
      A Senior Reactor Operator shall have, as a minimum, the following qualifications, as determined by the Manager, NTR:

      (1) A sufficient level of experience in NTR reactor operations, experiment setup and operation, and a high level of leadership.

      (2) An NTR Senior Operator's license.

      (3) Mature judgement and a capability for handling diverse problems under rapidly changing conditions.

      (4) Reactor Operator qualifications.

   b. Reactor Operator
      A Reactor Operator shall meet, as a minimum, the following qualifications, as determined by the Manager, NTR:

      (1) A high school diploma or equivalent, with a high degree of mechanical dexterity.

(2) NTR Operator's license.

(3) Sufficient training or experience in related nuclear fields.

c. Trainee

A Trainee shall meet, as a minimum, the following qualifications, as determined by the Manager, NTR:

(1) A high school diploma or equivalent, with a high degree of mechanical dexterity.

(2) Sufficient applicable training or experience.

Senior Reactor Operator and Reactor Operator candidates are required to obtain licenses issued by the U.S. Nuclear Regulatory Commission in accordance with the provisions of 10 CFR 55. Licensed personnel participate in a comprehensive requalification program. In addition, medical examinations are required at least every 2 years to determine the individual's capability to perform assigned duties without undue risk of operating errors or impairment of the ability to cope with emergencies.

9.1.4 Minimum Staffing

The minimum staffing of the reactor when it is not secured shall be as follows:

1. A licensed Reactor Operator or Senior Reactor Operator shall be in the control room with access to the nuclear console.

2. A second person who is familiar with NTR Emergency Standard Operating Procedures shall be at the Site.

3. A Senior Reactor Operator shall be present at the NTR facility or readily available on call at all times the reactor is not secured.

4. A Senior Reactor Operator shall be present at the NTR facility during the following events:

a. The first startup each day.

b. The recovery from an unplanned or unscheduled shutdown or significant reduction in power.

c. Reactor fuel loading or reactor fuel relocation.

d. Manual poison sheet changes.

9.2 PROCEDURES AND CONTROLS

9.2.1 Summary Description

The facility license, Technical Specifications, and Code of Federal Regulations establish the bounds within which the reactor must be operated. VNC Safety Standards as issued by the Manager, IPO, and reviewed and accepted by the Facility Manager or designated alternate supplement the license and federal regulations to ensure further personnel and reactor safety.

In addition, Standard Operating Procedures and Operations Request Forms have been established, as required, to delineate administrative and operational requirements to comply with NRC Regulations and the NTR License. Up-to-date copies of the Standard Operating Procedures and Operations Request Forms, as applicable, are available to all personnel at the facility.

A Change Authorization Procedure has also been established to document and authorize all changes to the facility or facility procedures as they are described in this document.

9.2.2 Required Actions

In the event of an abnormal occurrence, action shall be taken to assure the safety of the plant and personnel and to take appropriate corrective measures. If required, the reactor shall be shut down. If the reactor is shut down because of an abnormal occurrence, the reactor operation shall not be resumed unless authorized by Manager, NTR, or higher management. The NRC shall be notified in accordance with Subsection 9.6.

In the event a safety limit is known to have been exceeded, the reactor shall be shut down and reactor operation shall not be resumed without authorization by the NRC. The NRC shall be notified in accordance with Subsection 9.6.

9.2.3 Vallecitos Nuclear Center Safety Standards

Criteria in the Vallecitos Safety Standards (VSS) have been established for protection against hazards arising from activities conducted under licenses issued by appropriate regulatory authorities and provide guidelines for complying with the several licenses and regulations governing the facility, activities, and materials at VNC. Many of the standards govern the general radiation protection practices for the Site.

9.2.4 Standard Operating Procedures

Standard Operating Procedures (SOPs) have been established to delineate administrative and operational requirements to comply with NRC Regulations and the NTR facility license. The SOPs are in place for the following activities, as required:

1. Normal startup, operation, and shutdown of the reactor and all pertinent systems and components as specified by the Facility Manager or designated alternate involving nuclear safety of the facility;
2. Defueling, refueling, and fuel transfer operations when required;
3. Preventive or corrective maintenance which could have an effect on the safety of the reactor;
4. Off-normal conditions relative to reactor safety for which an alarm is received;
5. Response to abnormal reactivity changes;

6. Surveillance, testing, and calibrations required by the Technical Specifications;

7. Emergency conditions involving potential or actual release of radioactive materials;

8. Radiation protection consistent with 10 CFR 20 requirements;

9. Review and approval of changes to all required procedures; and

10. Security, Opeator requalification, emergency plan, and others, as required by the Facility Manager or his designated alternate.

The SOPs are established by the Manager, NTR, and approved by the Manager, Reactor Irradiations (Facility Manager), or his designated alternate. Independent review is in accordance with Subsection 9.3.

9.2.5 Operations Request Form

Operations Request Forms (ORFs) are issued, as required, to request work, establish temporary procedures or instructions, pass information, document actions, and other items in an effort to ensure the safe, efficient operation of the NTR.

The ORFs are established by the Operations Supervisor and reviewed and approved in accordance with the SOP covering ORFs. Independent review is also in accordance with Subsection 9.3.

9.2.6 Change Authorization

A Change Authorization is required for changes to the facility as defined below and changes to this document. The Change Authorization provides the documented description and safety evaluation required by 10 CFR, Section 50.59, and the review and approval of the change. A change authorization is required for changes, activities, or projects that are judged to involve significant safety considerations or a potential Technical Specification violation or "unreviewed safety question," warranting documented review and approval. A Change Authorization is also required for new types of experiments, or changes to types of experiments.

Charge Authorizations involving experiments (experiment type approval, as discussed in Subsection 7.4) require the following as a minimum:

1. All new types of experiments which could be postulated tc affect reactivity, be postulated to result in unusual radiation exposure to personnel or an unusual release of radioactive materials, shall be reviewed for compliance with the facility license and the Technical Specifications.

2. Changes to approved experiments shall receive appropriate review and approval.

3. Approved experiments are implemented in accordance with written procedures (Standard Operating Procedures or Operations Request Form, as required by governing procedures).

Change Authorizations are administratively controlled by a Standard Operating Procedure.

The Change Authorization is reviewed by the Reactor Analyst and Nuclear Safety & Quality Assurance (NS&QA) to determine that the following criteria are satisfied:

1. The proposed change can be made without prior NRC approval (10 CFR, Section 50.59).

2. The change does not violate any license requirement or federal regulations.

3. Special interim conditions which may exist during the period while the change is being made are analyzed to ensure that hazardous or unauthorized conditions do not exist during the modification or transition period.

More specific criteria and other review responsibilities are delineated in the Change Authorization Standard Operating Procedure.

The Reactor Analyst may request or perform additional analyses to show that the specified criteria are satisfied. The Reactor Analyst may also make recommended changes and comments and suggest other external reviewers.

Personnel in NS&QA provide an independent review of the Change Authorization. These personnel may request or perform additional analyses to ensure the specified criteria are satisfied. Personnel in the NS&QA will review the Change Authorization and:

1. Recommend approval of the proposed change; or

2. Recommend qualified approval of the proposed change; or

3. Recommend disapproval of the proposed change.

The Manager, Reactor Irradiations, or his designated alternate has the responsibility of approving or disapproving proposed Charge Authorizations.

9.2.7 Surveillance Requirements

Surveillance requirements are in place to maintain those systems as required to ensure the safe reliable operation of the facility. The surveillance requirements are implemented by the use of procedures (i.e., Standard Operating Procedures, Operations Request Form, etc.) as discussed in Subsection 9.2. The frequency of surveillance is, as required, based on 23 years of operating experience of the facility and equipment reliability. The surveillance requirements shall include, as a minimum, the following:

1. The potential excess reactivity will be calculated before each startup. Actual critical rod position shall then be used to verify that the measured value is $\leq$ 0.76$.

2. Neutron multiplication will be observed throughout each startup. Safety rod withdrawal shall be stopped if it appears critical will be reached before all safety rods are out.

3. The following restrictions shall be verified at least yearly, except during periods when the reactor is in a nonfueled condition. Prior to refueling, these verifications shall be performed and the normal surveillance schedule shall be resumed.

   a. No more than one safety rod can be moved out at a time.

   b. The rate of withdrawal of each safety rod shall be less than 1-1/4 in./sec.

   c. The rate of withdrawal of each control rod shall be less than 1/6 in./sec.

4. Each manual poison sheet in the core region of the reactor used shall be verified to be properly restrained upon insertion.

5. The potential reactivity worth of experiments shall be assessed before each irradiation. After establishing critical rod position, the potential excess reactivity will be determined which verifies the acceptability of the experiment worth.

6. Types of experiments with the capability for increasing reactivity by a chemical reaction will be designed and reviewed in accordance with Subsection 9.2.6.

7. The temperature coefficient of reactivity shall be verified to be negative above 124°F whenever new fuel is used in the reactor core.

8. Safety rod insertion times shall be measured wherever maintenance that could effect the insertion times is performed on the safety rods; whenever there is reason to suspect the insertion times have changed; and at least yearly, except during periods when the reactor

is in a nonfueled condition. Prior to refueling the reactor, the insertion times shall be measured and the normal safety rod surveillance schedule shall be resumed.

9. A channel check of the nuclear safety channels utilized during reactor operation comparing the channel outputs with a heat balance, shall be performed during each reactor run shortly after reaching equilibrium primary water core differential temperature whenever the power level is maintained above 15 kW.

10. The nuclear safety channels utilized during reactor operations shall be observed during startup. If it is determined that a channel does not have sufficient operating range and trip capability, as required, the reactor shall be shut down immediately. Corrective action shall be taken as appropriate.

11. Prior to the first reactor startup of each day, the reactor cell negative pressure with respect to the control room shall be verified to be acceptable.

12. Checks, tests, calibrations, etc., shall be performed as specified in Tables 9-1 and 9-2.

9.3 REVIEW AND AUDIT PROCESS

9.3.1 Summary Description

An efficient review and audit process is in place at NTR to ensure the proper dissemination of information for the safe, reliable, and efficient operations, maintenance, and modification of the facility. Facility review can be divided into "internal" (as performed by facility personnel), and "external" (as performed by organizations not under facility administrative control). The "independent" review required by the Technical Specifications is an "external" review performed either by the Vallecitos Technological Safety Council (VTSC) or NS&QA.

Table 9-1

Surveillance Requirements of Scram Systems

Item No.	Item	Surveillance	Frequency[a]
1.	Linear System	Channel check	Daily
		Channel test	Daily
		Instrument alarm or trip test	Daily
		Instrument calibration	Quarterly[b]
		Channel calibration (comparison against a heat balance)	Quarterly[b]
2.	Log N System	Instrument test	Daily
		Instrument alarm or trip test	Daily
		Instrument calibration	Quarterly[b]
		Channel Calibration (comparison against a heat balance)	Quarterly
3.	Primary Coolant Temperature	Alarm or trip test of the relay	Daily
		Detector Calibration	Yearly[b]
4.	Primary Coolant Flow	Channel Check	Daily
		Instrument alarm or trip test	Daily
		Instrument calibration	Yearly[b]
5.	Manual	Alarm or trip test	Daily
6.	Electrical Power	Alarm or trip test	Daily

[a] Excluding periods when the reactor is in a nonfueled condition. The normal surveillance schedule, as appropriate, shall be resumed prior to restart.

[b] Prior to placing into service an instrument which has been repaired, the instrument check test, calibration or etc. as appropriate, shall be performed.

Table 9-2

Surveillance Requirements of Safety-Related Systems

Item No.	Item	Surveillance	Frequency[a]
1.	Reactor All Pressure	Alarm or trip test Instrument calibration	Monthly Quarterly
2.	Fuel Loading Tank Water Level	Alarm or trip test	Monthly
3.	Primary Coolant Temperature	Alarm or trip test Instrument calibration	Monthly Yearly[b]
4.	Primary Coolant Core Differential	Channel check Instrument calibration	Quarterly Yearly[b]
5.	Radiation Monitors	Instrument check Alarm or trip test Channel calibration	Daily Monthly Quarterly
6.	Stack Radioactivity	Instrument check Alarm or trip test Instrument calibration	Daily[b] Monthly Quarterly
7.	Linear Power	Channel test	Monthly
8.	Control Rod	Channel test	Quarterly
9.	Safety Rod	Channel test	Quarterly

[a]Excluding periods when the reactor is in a nonfueled condition. The normal surveillance schedule, as appropriate, shall be resumed prior to restart.

[b]Prior to placing into service an instrument which has been repaired, the instrument check, test, calibration, or etc., as appropriate, shall be performed.

In this section, the review and audit responsibilities of the Reactor Analyst and the review and audit responsibilities of NS&QA and the VTSC are described. A comprehensive quality assurance program is also discussed. These reviews and audits are conducted to assure higher management that:

1. Operations comply with the facility license, the Code of Federal Regulations, and established procedures;

2. The operating organization discharges its responsibilities consistent with good safety practices; and

3. The records accurately and adequately reflect actual operation.

9.3.2 Reactor Analyst

The Reactor Analyst has the primary internal review and audit responsibility for compliance with the facility license, Technical Specifications, the Code of Federal Regulations, and the VSS. The Reactor Analyst reviews all SOPs changes to SOPs, and ORFs with any potential nuclear safety or regulatory significance. The review ensures that:

1. All SOPs and applicable ORFs are in compliance with the facility license, Technical Specifications, Code of Federal Regulations, VSS, and established procedures; and

2. The SOPs and applicable ORFs involving experiments, facility modifications, or facility procedures as appropriate, as they are described in this document, are performed in accordance with the Change Authorization (Subsection 9.2.6).

The Reactor Analyst consults with the operating staff, as required, to ensure that:

1. Abnormal occurrences receive proper action;

2. Activities requiring written procedures are not performed without these written procedures; and

3. Potentially reportable events, according to the facility license and Code of Federal Regulations, are brought to the attention of facility management.

4. Operation is in accordance with regulatory requirements, General Electric procedures, and good operating practices.

Other reviews or audits may be performed to ascertain trends not necessarily detected during day-to-day operation and data collection.

9.3.3 Nuclear Safety and Quality Assurance (NS&QA)

The Manager, Nuclear Safety and Quality Assurance, reports to the Manager, IPO, independent of the Manager, Reactor Irradiations. Personnel in NS&QA provide an independent review and audit of the NTR facility for safety and compliance with the facility license, Technical Specifications, and Federal Regulations. The review and audit includes:

1. All proposed procedures required by the Technical Specifications and proposed changes to such procedures;

2. Proposed types of experiments, facility modifications, and facility procedures, as appropriate, as are described in this document, in accordance with the Change Authorization described in Subsection 9.2.6;

3. Proposed changes to the facility operating license, including Technical Specifications and revised bases;

4. Violation of the Federal Regulations, Technical Specifications, facility license requirements, and internal procedures having safety significance;

5. Unusual or abnormal occurrences which are reportable to the NRC, as required by the Federal Regulations or Technical Specifications;

6. Significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety; and

7. Periodic audit of facility operation, maintenance, and administration, to include the following:

   a. The conformance of facility operation to the federal regulations, Technical Specifications, and facility license requirements.

   b. The results of all actions taken to correct deficiencies or increase effectiveness in facility equipment, structures, systems, or methods of operation that affect nuclear safety.

   c. The facility emergency procedures, security plan, requalification program, and their implementing procedures.

The nuclear safety function is administered by a written charter which identifies the scope and policy, subjects reviewed and/or audited, organization, personnel qualifications, responsibilities, authorities, records, including provisions for dissemination, review, and approval of audit findings in a timely manner, and other matters, as may be appropriate.

A comprehensive quality assurance program is required for modification and maintenance of pertinent plant systems and equipment. The objective of the program is to maintain managerial and administrative controls over actions relative to those nuclear safety-related items, as identified in the Standard Operating Procedures. Most of those controls are discussed in Subsection 9.2 and previously in this section. The overall Quality Assurance Program is as described in Appendix C.

9.3.4 Vallecitos Technological Safety Council (VTSC)

The Vallecitos Technological Safety Council, when utilized, is responsible to the Manager, IPO, and advises him on matters affecting the nuclear safety industrial safety and safety-related regulatory compliance aspects of the VNC Site.

The VTSC, when utilized, is comprised of members chosen as representative of all Site activities and technical disciplines. The VTSC review meetings may be held as review activities or license requirements dictate. Activities are reviewed as requested by operating management, the NS&QA management, or at the council's own initiative. Activities which may be reviewed include the following:

1. Results and actions resulting from reportable incident investigations. Occurrences which are not reportable may be reviewed by the VTSC if, in its view, significant safety questions are involved.

2. Proposed new facilities or changes to facilities which may contain potential unreviewed safety questions.

3. The VNC safety standards, as requested by NS&QA or operations management.

4. Change Authorizations, when requested by facility management or the nuclear safety function.

5. Problems of nuclear safety, criticality control, and industrial safety, as related to nuclear operations.

6. Effectiveness and relevance of safety studies, and nuclear safety review and audit activities as they collectively influence the safety conditions at VNC.

7. "Special topics," as requested by nuclear safety or operations management.

8. Any other matter which the VTSC believes to be of importance.

9.4 INCIDENT INVESTIGATION

9.4.1 Summary Description

Incidents at the NTR are investigated in accordance with the policy described below. The purpose of this policy is to ensure a determination of the cause, to provide recommendations, and to establish a mechanism for recommendation followup.

9.4.2 Policy

All incidents which involve actual or potential radiation injury or other serious consequences are investigated, and remedial measures are formulated.

Incidents are classified as Type I or Type II. Type I incidents are those with consequences that include significant radiation exposure, personal injury, contamination spread, property damage, or improper release of radioactive material to the environment. Type II incidents are those for which there was a potential for the occurrence of a Type I incident, but the actual consequences were trivial.

9.4.3 Procedure

After an occurrence is determined to be an investigatable incident, the Manager, Reactor Irradiations, appoints an Investigation Committee and chairman. For a Type I investigation, at least one member shall be from NS&QA, at least three members shall be from components other than the component responsible for the area in which the incident occurred, and the Chairman is at least on the same organizational level as the Manager, Reactor Irradiations, and has no direct responsibility for the NTR schedules or budget.

The Committee has the authority to review all records and interview any personnel associated with the incident. The purpose of the investigation is to accumulate the facts, determine the cause(s), and make recommendations which prevent recurrence or mitigate the consequences of the incident.

The investigation report is addressed to the Manager, Reactor Irradiations, with copies sent to the Manager, NTR, NS&QA, Industrial Safety, the VTSC, and other functions which are affected by the incident and which are also required to respond to the recommendations.

The Site safety organizations may publicize the incident as appropriate. The Manager, Reactor Irradiations, and affected managers respond to the Committee's recommendations, as appropriate, in writing.

NS&QA or Industrial Safety, as appropriate, ensures that the responses to the recommendations are made and provides assistance in implementing the recommendations, as required.

9.5 REACTOR OPERATING TRAINING AND RETRAINING

9.5.1 Summary Description

The Reactor Operator Training and Retraining Program is organized and administered by the Requalification Program Administrator with the concurrence of the Manager, NTR. The Requalification Program Administrator shall posssess a Senior Reactor Operator license so he may officially administer all facets of the program.

9.5.2 Training

The training program provides required training based on individual experience, degree of responsibility, intended position, and assignment. The training program provides personnel with sufficient knowledge and experience to qualify them to operate the reactor facility safely and efficiently in accordance with NTR procedures, reactor license, Technical Specifications, and appropriate NRC rules and regulations. In general, individual training needs are established by the Operations Supervisor who, with the assistance of the Requalification Program Administrator, conducts individualized training.

9.5.3 Retraining

The retraining program is designed to maintain the competence of the NTR operating personnel to handle abnormal events and to comply with the requirements and intent of 10 CFR 55, Appendix A, "Requalification Programs for Licensed Operators of Production and Utilization Facilities." The NTR Requalification Plan (NEDO-12724), approved by the NRC, is described in and is administratively controlled by the NTR Standard Operating Procedures.

9.6 REPORTS

9.6.1 Summary Description

Formal reports are submitted to the NRC, including annual reports, items of interest, as well as other specific reports which are either formally or informally requested by the NRC.

In addition, exceeded limits and abnormal occurrences are reported to the NRC, as specified by the NTR Technical Specifications, the Code of Federal Regulations, Title 10, and other applicable documents. An NTR Standard Operating Procedure, "Reportable Incidents," contains specific reporting requirements and required actions and serves as a guide for NTR personnel in determining reportable incidents.

9.6.2 Notification

Notification shall be made for the items listed below not later than the following working day by telephone and confirmed by telegraph or similar conveyance to the director of the appropriate regional U.S. Nuclear Regulatory Commission Office of Inspection and Enforcement listed in Appendix D of 10 CFR 20. A written report shall follow within 14 days to the appropriate regional USNRC Office of Inspection and Enforcement, with copies to the Director, Office of Inspection and Enforcement, USNRC, Washington, D.C. 20555, and to the Director, Office of Management Planning and Analysis, USNRC, Washington, D.C. 20555.

1. Violation of safety limits.

2. Release of radioactivity from the Site above allowable limits.

3. Operation of the reactor with actual safety-system settings for required systems less conservative than the limiting safety-system settings specified in the Technical Specifications.

4. Operation in violation of limiting conditions for operation established in the Technical Specifications, unless prompt remedial action permitted by the Technical Specifications is taken.

5. A reactor safety system component malfunction which renders, or could render, the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns. [Note: Where components or systems are provided in addition to those required by the Technical Specifications, the failure of the extra components or systems is not considered reportable (other than in the annual report, as required) provided that the minimum number of components or systems specified or required are able to perform their intended reactor safety function.]

To the extent possible, the preliminary telephone notification shall:

a. Describe, analyze, and evaluate safety implications;

b. Outline the measures taken or to be taken to ensure the cause of the condition is determined; and

c. Indicate the corrective action taken or to be taken to prevent repetition of the occurrence.

A written report shall be forwarded within 30 days to the Director of the Regional USNRC Office of Inspection and Enforcement listed in Appendix D of 10 CFR 20 in the event of:

1. Discovery of any substantial errors in the safety analyses or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the Technical Specifications.

2. Discovery of any substantial variance disclosed by operation of the reactor from performance specifications contained in the Safety Analysis Report.

3. A permanent change in the facility management personnel (Manager, NTR, and higher).

9.7 RECORDS

In addition to those required by applicable government regulations, the following records shall be maintained, at a minimum, for the periods specified below The records may be in the form of logs, data sheets, or other suitable forms. The NTR Standard Operating Procedure, "Reactor Records," containing specific retention requirement for records, serves as a guide.

Records to be retained for a period of at least 5 years or for the life of the component, whichever is smaller, include the following:

1. Normal reactor operation records, i.e., supporting documents, such as checklists and log sheets.
2. Unplanned or unscheduled shutdowns and scrams, including the reasons therefore.
3. Principal maintenance activities involving substitution or replacement of reactor equipment or components.
4. Occurrences reported to the NRC, as required by the Technical Specifications.
5. Surveillance, testing, and calibrations required by the Technical Specifications.
6. Experiments performed, including unusual events involved in their handling and performance.
7. Reviews performed for (a) changes made to procedures or equipment, and (b) new tests and experiments not submitted for NRC approval pursuant to 10 CFR, Section 50.59.
8. Meetings and audit reports of the independent review and audit function.
9. Off-site radioactive shipments and receipts.

Records to be retained for at least 2 years from the time of the event include the following:

1. Retraining and requalification of licensed operators.

2. The most recent complete cycle record shall be maintained at all times during which the individual is employed at the facility.

Records to be retained for the lifetime of the reactor facility include the following:

1. Gaseous radioactive effluents released or discharged to the environment beyond the effective control of the licensee as monitored at or before the point of release or discharge.

2. Off-site environmental monitoring surveys.

3. Radiation exposure for all facility personnel monitored.

4. Updated drawings, as required, of the facility and of facility changes.

9.8 EMERGENCY PLAN

9.8.1 Summary Description

The Site Emergency Plan provides guidance and instructions to emergency response personnel for the purpose of protecting personnel, plant, and production in the event of an emergency situation. The plan establishes and identifies an Emergency Control Organization which will assume responsibility for dealing with all emergencies at the VNC, including those involving the NTR.

9.8.2 Procedure

The Emergency Plan for the NTR consists of the sections of the Site Emergency Plan listed below. These broad sections are supplemented, as appropriate, by NTR Standard Operating Procedures.

Site Plan	Description
A-5	Emergency Control Plan - General
C-5	Fire Protection
D-5	Criticality Emergency
E-5	Radiation Emergency

10. SHIELDING

10.1 GENERAL

Reactor shielding is such that personnel radiation exposures throughout the building can easily be maintained well within established limits with the reactor at full power. A list of typical levels in and around the facility, with the reactor at 100 kW, is given in Subsection 10.5. During initial operations under new conditions, radiation dose rates and personnel exposures are closely monitored. If required, modifications are made to the shielding or the procedures to ensure continued compliance to established limits and consister e with As Low As Reasonably Achievable (ALARA) practices.

10.2 REACTOR SHIELD

At present, radiation shielding for the reactor includes the graphite and the cadmium-lined aluminum frame which were discussed previously, local shielding in the vicinity of the reactor, and the thick concrete walls of the reactor cell and the south cell. The arrangement of most of these materials can be seen in Figures 1-1 and 7-1.

The reactor is situated in a high-density concrete alcove (10 feet, 8 inches wide by 10 feet high) in the reactor cell so that the 5-ft-thick alcove walls provide adequate shielding on the east and south sides. On the west side, the fuel storage tank provides 4 feet of water shielding in addition to the 3-ft-thick alcove wall. A lead shield wall, 8 inches thick in front of the reactor with 6-in.-thick extensions on each side, provides shutdown gamma shielding for the control rod drive area. A removable shield slab of reinforced heavy concrete covers the top of the reactor. The 1-ft-thick shield is a four-sided, irregularly shaped, reinforced, heavy concrete slab with a 48-in.-diam hole centered above the reactor core and an 8-in.-diam hole that may be used to gain limited access to the east face of the reactor. A 16-in.-thick stepped concrete plug is available for closing the 48-in. hole when that facility is not in use. The large plug contains a 6-in.-i.d. hole directly above the vertical facility to permit access to this facility without removing the large plug. This slab and plug are sometimes removed to perform tests and experiments during operation, install equipment, or perform maintenance when the reactor is shut down. During periods when this slab or large plug is removed, radiation monitoring is performed and working time or reactor power is limited, if necessary.

10.3 REACTOR CELL

Section 3 contains a detailed description of the reactor cell, including a list of the cell penetrations; the high-density concrete alcove was discussed in Subsection 10.2. The remainder of the cell is constructed of ordinary reinforced concrete and provides the following shielding:

1. 5-ft-thick north and east walls;

2. 4-ft-thick west wall between the NTR and adjacent laboratory cell;

3. 3-ft-thick south wall between the control room and reactor cell, containing the shield door discussed in Section 3; and

4. 3-ft-thick roof.

Radiation levels listed in Subsection 10.5 demonstrate the effectiveness of the cell as a radiation shield. Whenever new operating or maintenance conditions are encountered, radiation surveys are made to determine that existing shielding is adequate and consistent with ALARA practices. Either temporary or permanent improvements in the shield are made if the results of the survey indicate they are necessary.

10.4 SOUTH CELL

The main source of radiation in the south cell is that which comes directly from the reactor and that which is induced in experiments and experimental equipment. As shown in Figure 7-1, high-density concrete walls completely surround the experiment area. The wall between the cell and the control room is 2 feet thick by 7.5 feet high and contains a shield door consisting of about 8 inches of high-density aggregate, 5.25 inches of paraffin, and two 0.125-in. sheets of boral. The portion of the east wall of the cell separating the south cell and the shop area is 3 feet thick. A false ceiling is installed above the south cell and consists of staggered layers of 4-ft by 8-ft by 0.25-in. Masonite to a thickness of 2 inches. Additional shielding has been added in the form of a shield wall extension of the south wall of the south cell. The wall is 50 inches by 59 inches by 8 inches, with a 15-in. by 24-in. by 8-in. hole for movement of experiments, and is constructed of high-density concrete blocks enclosed in a support structure. Shielding material has also been added above the south cell entry ceiling and on the back porch of the NTR adjacent to the penetration through the east wall of the south cell.

Radiation coming from the reactor is reduced by the presence of a 4-ft-thick graphite thermal column. In front of that is approximately a 2-ft-thick high-density concrete block wall and a 4-in. lead brick wall. A thick shutter consisting primarily of lead and borated polyethylene is operated from the control room and shields radiation from the horizontal cavity. An interlock alarm system is provided which:

1. Prevents opening the door if higher than normal levels of radiation are present;

2. Initiates automatic closure of the shutter to reduce radiation levels; and

3. Initiates audible and visual alarms to warn personnel of higher than normal radiation levels.

In addition, a photo-cell alarm is provided at the south cell door access point which will sound whenever the light beam is broken when it is armed.

10.5 NORTH ROOM MODULAR STONE MONUMENT (MSM)

The MSM, which is discussed in Subsection 7.12 (Figure 7-4) provides shielding in the north room from:

1. The north neutron radiography beam; and
2. Radioactive objects during the neutron radiographic process.

The MSM is made of high-density concrete modular blocks (so design changes may be made easily in the future) and houses a borated lead polyethylene beam catcher. Additional lead shield closures may be utilized, as required, to further reduce the radiation from two of the penetrations, as shown in Figure 7-4.

10.6 RADIATION LEVELS

A list of typical radiation levels in the area of the NTR facility, while the reactor is operating at 100 kW, is given below. Unless shielding changes are made, the listed radiation levels are all proportional to reactor power. The values listed include contributions from fast, intermediate, and slow neutrons and gamma rays.

Location	Shutters* Open mRem/h	Shutters* Closed mRem/h
At reactor console	2.5	≤ 1
Hallway south of control room	≤ 1	≤ 1
Building 105 equipment room (2nd floor)	≤ 1	≤ 1
Reactor cell roof directly above reactor (top shield slabs in place)	75	75
South cell roof	50	≤ 2
Outside north or east reactor cell walls	≤ 1	≤ 1
Outside end of removable plug in east wall of the reactor cell	5	5
North Room (center of room)	≤ 1	≤ 1

*North and south cell horizontal cavity shield shutters

11. SAFETY ANALYSIS

11.1 INTRODUCTION

This chapter contains an evaluation of the facility response to certain events that can be reasonably postulated to occur at the NTR and which appear to have safety significance. The results of the analyses show that design features, equipment, and procedures are in place to ensure that the health and safety of the public and plant personnel are not jeopardized by the occurrence of any of the postulated events. The events analyzed include anticipated operational occurrences and potential accidents.

Reactor transients were analyzed by simulating reactor dynamics with a digital computer. The model used is discussed in Subsection 11.2.

Events categorized as anticipated operational occurrences are discussed in Subsection 11.3. Anticipated operational occurrences are the results of single equipment failures, or malfunction, or single operator errors that can reasonably be expected during any planned mode of facility operation. The anticipated operational occurrences analyzed in this chapter are:

1. Loss of electric power;
2. Loss of secondary cooling;
3. Loss of facility air supply;
4. Inadvertent start of primary pump; and
5. Fuel handling errors.

Unacceptable consequences for anticipated operational occurrences are:

1. Release of radioactive material to the environs that exceeds the limits of 10 CFR 20;

2. Radiation exposure of any person in excess of 10 CFR 20 limits; and

3. Violation of an established safety limit.

Events categorized as accidents are discussed in Subsection 11.4. Accidents are defined as postulated events not expected to occur during the course of plant operation that appear to have the potential to affect one or more of the radioactive material barriers. The postulated accidents analyzed in this chapter are:

1. Uncontrolled reactivity increases;
2. Loss of primary coolant flow (pump shaft seizure);
3. Rod withdrawal; and
4. Loss of primary coolant.

Unacceptable consequences for postulated accidents are:

1. Radioactive material release to an extent that exceeds the guideline values of 10 CFR 100; and

2. Violation of a safety limit.

Subsection 11.5 is an evaluation of experiment safety and shows that procedures, limits, and safety equipment are in place to ensure the proposed experiment program can be carried out without undue risk to the health and safety of the public and plant personnel.

There is a close relationship between the safety analyses for anticipated operational occurrences and accidents and the safety limits and limiting safety system settings. Development of proposed safety limits and limiting safety system settings are discussed in Subsection 11.7.

The results of the analyses show that there are no credible events that could cause fuel melt or a significant release of fission products from the fuel. Even if catastrophic nonmechanistic failure of the NTR facilities is assumed, there are no potential consequences more severe than those associated with the accidents analyzed in this section. Compaction of the fuel, while essentially impossible mechanistically, would not cause the reactor to go critical since water loss, increased self-shielding in the fuel, and the geometry change due to flattening of the cylindrical core are all negative reactivity effects. Loss of water shuts down the reactor and no fuel melting occurs, as discussed in Subsection 11.4.6. Also, deformation of the core, which causes the fuel to contact the core can structure, would improve heat transfer and result in lower Loss-of-Coolant Accident (LOCA) temperatures. The only accidents which could possibly cause fuel damage and release of fission products from the NTR fuel are those resulting from large reactivity insertions. Reactor configuration and the reactivity worth of experiments are controlled to ensure that destructive reactivity transients are not credible. Nevertheless, an assessment of the consequences of an assumed fission product release is presented in Subsection 11.6 to demonstrate the capability of the facility, even though such a release is not possible under the 0.76$ reactivity limit.

11.2 TRANSIENT MODEL

The reactor dynamics were simulated with a digital computer. The major features of the model are summarized below.

1. The core was represented by three 5-node channels:

   a. A channel representing average power, flow, and temperature conditions;

   b. A channel with less than average power (90%) but much less than average coolant flow (53%); and

   c. A channel with highest power (130% of average) and highest flow (153% of average).

Each channel corresponded to a flow path from the inlet at the bottom of the core, upwards around the circumference of the core between adjacent fuel disks, to the flow exit at the top of the core.

2. A circumferential power distribution was assumed which had the neutron flux skewed to one of the upper quadrants of the core. Conditions in channels b and c were taken as those in the hottest side of the core and differed only by the relative peaking along the axis of the core (chopped cosine with peak-to-average 1.15). Conditions in the average channel a represented the average of the power distributions on both sides of the core.

3. The basic thermodynamic mass, volume, and energy balance equations were used to calculate the average fuel and water temperatures in each node. New heat-transfer coefficients were computed at each time step during the transient for the temperature and flow conditions existing at the time.

4. Point kinetics were used since spatial neutron coupling is very strong. Six delayed neutron groups were included.

5. Reactivity feedbacks included weighted values from the average channel water temperature and from steam voids in the hot and average channels. Doppler feedback was neglected for this high-enrichment fuel. Water temperature feedback was based on a coolant temperature coefficient of -5.7×10^{-3} (T-124) ¢/°F, where T is the water temperature in degrees Fahrenheit. Average channel void coefficient was -5.71 ¢/% voids, Steam formation in the hot channels was assumed to be worth 5% of average channel voids. Only bulk boiling feedback was included. Steam from subcooled surface boiling was neglected. No credit was taken for expulsion of primary coolant which resulted from thermal expansion of the fuel disks.

6. Although temperature peaking was usually small within the uranium-aluminum alloy fuel, the center temperature was calculated at each node from average and surface temperatures. A parabolic distribution was assumed:

$$\text{Fuel Center Temperature} = 1.5\ (\text{Fuel Node Temperature}) - 0.5\ (\text{Surface Temperature}) \qquad (2)$$

7. For nonboiling nodes, the Seider-Tate relationship was used in the model to estimate the surface heat-transfer coefficient. The results were checked and found consistent with other laminar flow correlations. The average steady-state channel coefficients were calculated to be about 165 Btu/h-ft^2°F.

8. Fuel surface nucleate boiling was simulated in the model, when fuel surface temperature reached the value given by the Jens-Lottes correlation.[7] During this condition, heat-transfer characteristics improve and the surface temperature can be calculated as follows:

$$T_{(Surf)} = T_{(Sat)} + 1.9\ (q/A)^{1/4}/e^{(P/900)} \qquad (3)$$

where

T_{Sat} = coolant saturation temperature, °F,

q/A = surface heat flux, Btu/h-ft^2, and

P = system pressure, psia.

9. Steam blanketing was assumed to occur when a surface heat flux exceeds the critical heat flux. A critical value[7] of 450,000 Btu/h-ft^2 was used in the analysis. More recent work presented in Subsection 11.6 indicates a value of 600,000 Btu/h-ft^2 is justified. During nucleate boiling, the effective heat-transfer coefficient is very large. If the critical heat flux was reached,

the heat-transfer coefficient[23] was conservatively dropped to 10 Btu/h-ft^2-°F.

10. The safety rod insertion time was measured and is less than 300 milliseconds. In the analysis, a time of 200 milliseconds was assumed to include all electronic delays and the time required for the rods to move to the edge of the active core (the first 12 inches of their 30-in. stroke). The remainder of the insertion was fitted to an S-shaped curve of reactivity versus position.

An important aspect of the analysis is the heat-transfer characteristic by which steam is formed during excursions (the steam voids provide the strongest negative reactivity feedback in addition to scram). The general characteristics of the heat-transfer mechanisms have been described here and basic relations used in the heat-transfer analysis are given in Appendix D.

Figure 11-1 illustrates the qualitative hypothetical high-power excursion without scram. For very fast transients which are not possible with the existing 0.76$ reactivity limit, some of the sequences shown may not be the same; however, most mechanisms which appear are illustrated in Figure 11-1. This hypothetical excursion develops according to the following sequence of events.

1. Initially, all channels are in laminar flow with heat-transfer coefficients near 165 Btu/h-ft^2-°F. Figure 11-1 shows three initial fuel temperatures for the three channels described in Subsection 11.2, Item 1. The highest power channel (c) usually produced the highest fuel temperature for transients in which the high-power peak was the dominant factor. As the transient progresses and power increases, fuel and water temperatures rise until the beginning of nucleate boiling.

2. When the fuel surface temperature becomes high enough, nucleate boiling begins on the fuel surface. When this occurs at time, t_1, heat-transfer conditions improve greatly, holding the surface temperature essentially constant, and increasing the rate of rise of channel water temperature. The values of fuel surface temperatures during nucleate boiling conditions were estimated with the Jens-Lottes correlation.[7]

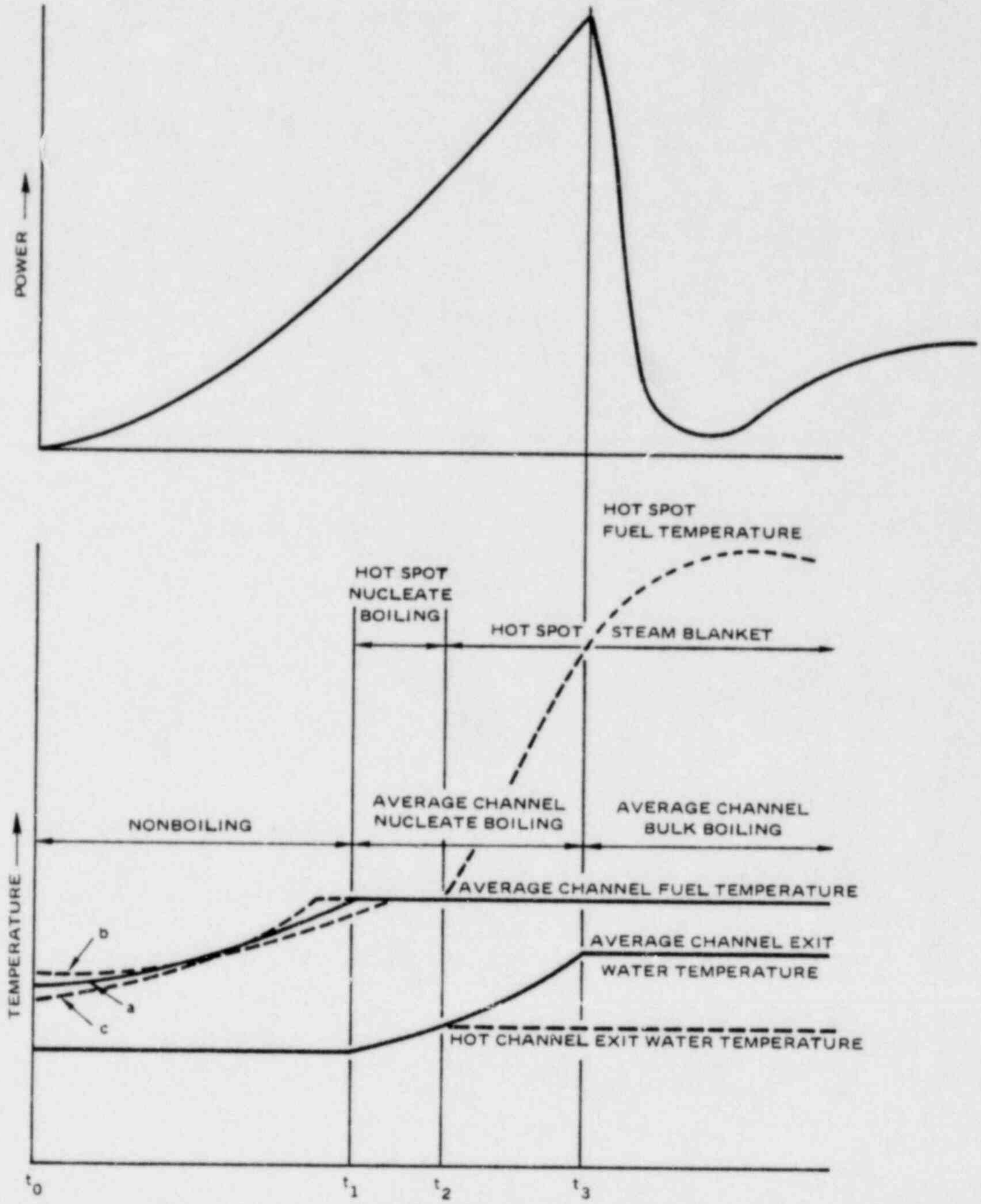


Figure 11-1. NTR Transient

3. In the transient shown in Figure 11-1, average channel exit water temperature reached saturation at time t_3. The fuel temperature remained nearly constant throughout nucleate boiling. The formation of steam produced a large negative reactivity feedback, which turned the power excursion.

4. In this example, the hot spot was steam-blanketed. This phenomenon is triggered if the power rises high enough to produce a surface heat flux greater than the critical value given in Subsection 11.2, Item 9. When this occurs, the surface heat-transfer coefficient drops to about 10 Btu/h-ft^2-°F. The fuel temperature will rise sharply since this condition almost insulates the fuel. As shown in Figure 11-1, this temperature will level off when the power is turned. The temperature will approach a new steady-state value which corresponds to the final power level. This power level is dependent upon the type of accident and the extent of steam formation. The reactivity events typified by this general type of behavior are presented in Subsections 11.4.1 through 11.4.4.

11.3 ANTICIPATED OPERATIONAL OCCURRENCES

11.3.1 Loss of Electrical Power

Other than the battery-operated emergency lights, there are no emergency power supplies, either ac or dc, for the NTR facility. Loss of ac power to the facility means a complete loss of electrical power and results in reactor scram through the following processes:

1. The magnet power supply would be deenergized and cause loss of power to the safety rod electromagnets.

2. The primary pump would stop and cause a low-flow scram signal (if power was greater than 100 watts at the time of loss of ac power).

3. Numerous fail-safe circuits in the safety system would signal the power switches and scram relays to initiate a scram.

The only differences between this scram and a normal scram would be the simultaneous loss of primary coolant flow (although secondary coolant flow is by gravity, it is stopped by a spring-closed, solenoid-activated valve) and the control rod and safety rod drives would not run in automatically. However, the safety rods would be inserted by their spring action (normal scram action) to shut down the reactor. The consequences from a loss of ac power would be no worse than a loss-of-flow scram. As discussed in Subsection 11.4.5, there are no unacceptable consequences, even from the worst case loss-of-flow accident.

11.3.2 Loss of Facility Air Supply

The only items affected by the loss of facility air are the air piston operator for the south cell door, and the radiation shield shutter for the horizontal facility in the south cell. The design specifications for the south cell door require that it be movable manually by one person. Loss of air to the beam shutter would have no effect on its position (i.e., it would remain in the position it was in at the time of air supply failure). Thus, there is no safety concern due to loss of facility air supply.

11.3.3 Loss of Secondary Coolant

Secondary coolant flows by gravity through the tube side of the primary heat exchanger, as described in Subsection 5.3. Loss of secondary coolant or loss of coolant flow when the power level is high enough to produce an appreciable heating rate will cause the reactor to scram from high primary coolant temperature. If the reactor power is not high enough to produce a heating rate that will soon cause a scram, the loss of secondary coolant will soon be evident to the operator by:

1. The slightest changes in temperature, which cause an observable reactivity effect.

2. The temperature monitor system readout at the console.

3. The secondary flow control in the control room.

11.3.4 Primary Pump Inadvertent Start

If the primary pump were inadvertently started, the effect would be to change the reactor inlet temperature. A decrease in inlet temperature will cause reactor power to drop. An increase in inlet temperature will produce a rising power transient — a hot-water transient. This transient is comparable to a cold-water accident for reactors that operate with a negative temperature coefficient of reactivity. Normally, there is no source of energy to produce an increase in the primary coolant temperature; however, the system is designed to accommodate an electric heater. The amount of positive reactivity which could be added is less than 0.10$ (from room temperature to turnover temperature); therefore, the resultant transient would, and could, be controlled by manipulation of the control rods. Although the 5-kW heater has been removed from the system, it could be replaced, if needed.

The worst possible case would be a coolant heatup to 124°F from shut-down power and temperature conditions. The temperature, net reactivity, and power characteristics are shown in Figure 11-2. The electric heater was assumed to be shut off (or regulated) after 1000 seconds, so that the inlet remained at 124°F. The power will continue to rise with the period between 80 and 100 seconds until power increases sufficiently to raise the core coolant temperature above 124°F and reduce the net reactivity.

For such a slow transient, a high-power scram would clearly stop the excursion without fuel damage. If the scram failed, bulk boiling would occur soon enough to prevent the power from reaching a level high enough to produce steam blanketing. It has been shown that a step insertion of 0.76$ of reactivity would not cause fuel damage, even if the reactor failed to scram. Therefore, it can be concluded that a transient caused by the small amount of reactivity from the temperature would also be safely limited.

11.3.5 Fuel Handling Errors

Fuel handling equipment and procedures are discussed in Subsection 6.1. It should be reemphasized here that refueling for reactivity increase is not

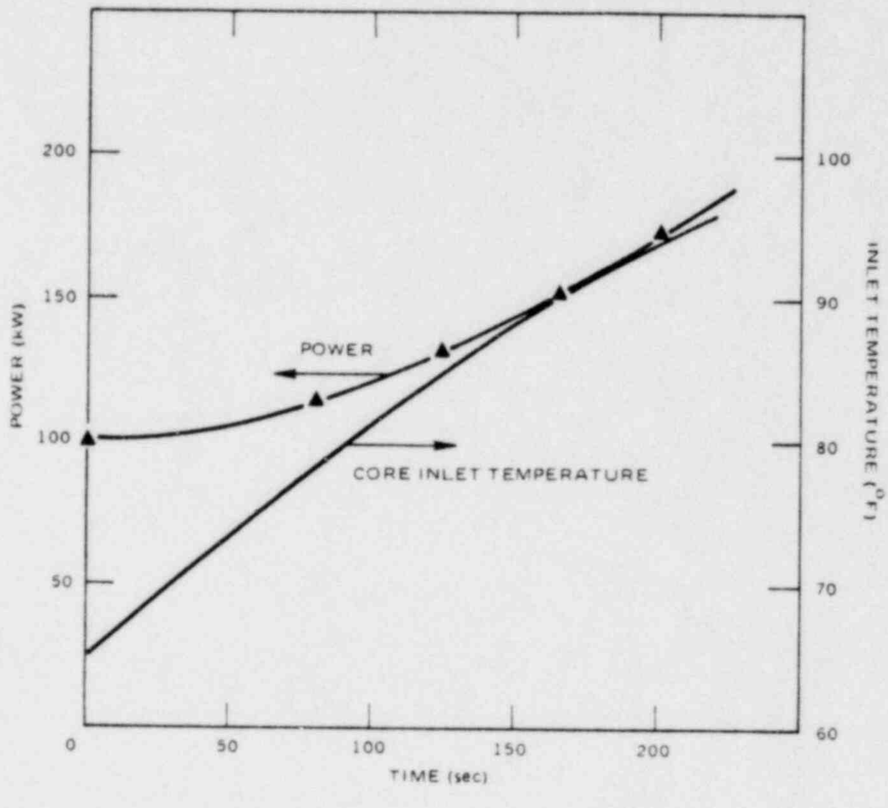


Figure 11-2. Primary Pump Inadvertent Start (from 65°F)

necessary, and fuel handling is very rare. The only fuel handling occurrence in the past 20 years was one core unloading and reloading associated with the core container replacement in 1976. The only fuel handling anticipated in the foreseeable future will involve inspection of the existing fuel assemblies and the present core configuration. The 16 fuel assemblies on hand completely fill the core reel assembly and fill the fuel container to the extent that the only remaining space of appreciable size is in the fuel loading chute. In other words, a fuel assembly, once inside the fuel container, must either be in the provided positions in the reel or in the fuel loading chute. The physical arrangement of the fuel container is such that an element located in the loading chute results in a worse core geometry than the cylinder formed by having all elements in the core support reel. Dropping a fuel element could only cause an accident if the control rods were withdrawn during loading so that the reactor was almost critical before adding fuel. Such an act is contrary to operating procedures and requires errors by the console operator and fuel loaders. The only other means of getting fuel close to the core is by inserting it into either the horizontal or vertical facilities. Use of these facilities is discussed in Section 7.

In addition to the inherent safety feature provided by having all existing NTR fuel elements in their most reactive configuration in the core, the following additional safety features ensure safety during all phases of fuel handling:

1. Reactor design, fuel handling equipment, and administrative controls are such that not more than two elements can be handled at one time.

2. All fuel movement must be performed in accordance with written procedures.

3. The cell high-gamma-level alarm system will be in operation.

4. By using all the manually positioned poison sheets, the core can be made 6.1$ subcritical (Table 4-1). Removal of the graphite plug from the fuel loading chute provides additional negative reactivity of approximately 1.25$.

5. Any movement of source and special nuclear material within the NTR facility must have the approval of the licensed operator on duty.

6. Any storage arrangement used will be analyzed to ensure a subcritical configuration.

11.4 POSTULATED ACCIDENTS

The transient model used to simulate the reactor dynamics is presented in Subsection 11.2.

11.4.1 Idealized Step Reactivity Insertions - with Scram

Transients resulting from step reactivity insertions up to 2.0$ were studied; a range of different initial reactor powers and flows were used. The results for steps with high-power scram occurring at 150 kW are shown in Figure 11-3. Only a very slight fuel temperature increase was observed for steps up to 1.0$. In all cases, peak temperature rose sharply for reactivities above this value.

The transient due to a step reactivity insertion of 1.3$ while the reactor is at 100 kW and at rated flow is shown in Figure 11-4. The sequence of events for this transient follows.

Time (sec)	Event	Peak Fuel Temperature (°F)
0.0	1.3$ step insertion	195
0.0044	Scram circuit tripped	195
0.1460	Nucleate boiling began at hot spot	241
0.1558	Steam-blanketing occurred at hot spot	258
0.2046	Safety rods reached active core	504
0.218	(Power peak 1.04 x 10^5 kW)	652
1.0	Power dropping, temperature rising slowly	841

The transient is too fast for any channel bulk boiling to help the scram reduce power. The relatively high "tail" on the power curve is the result of delayed

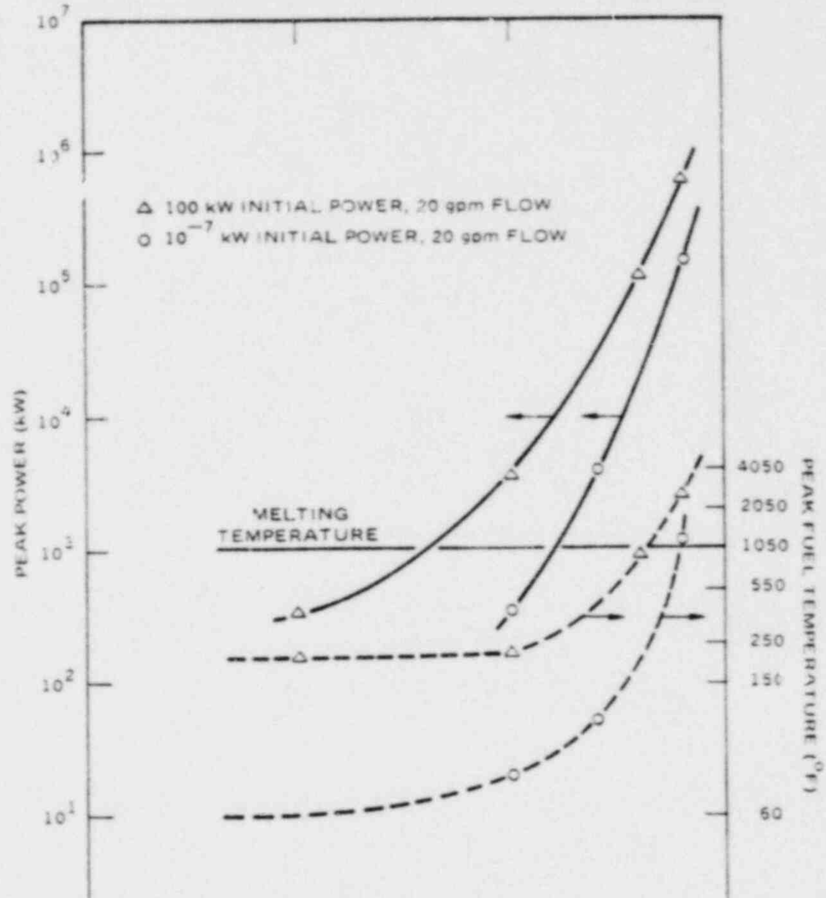
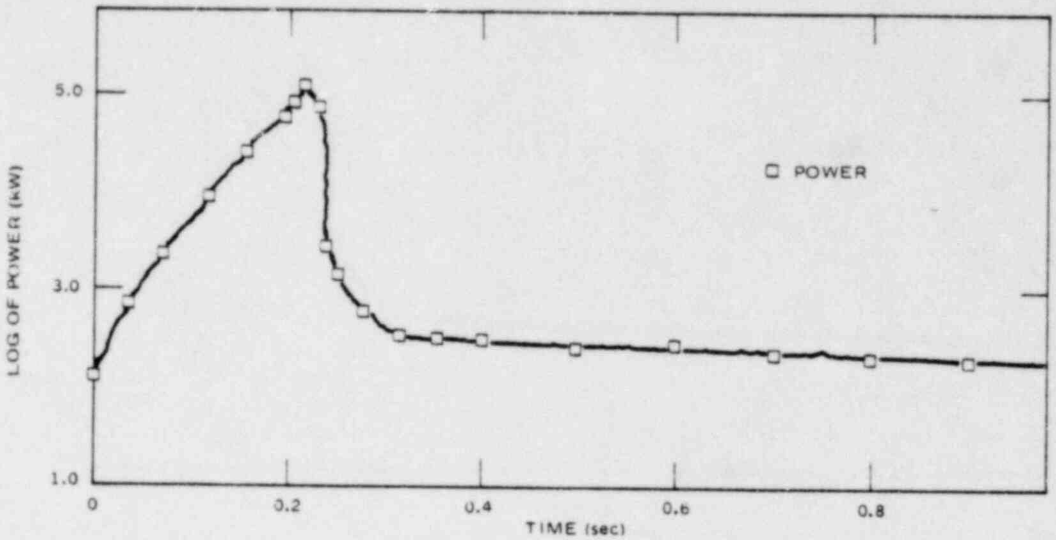
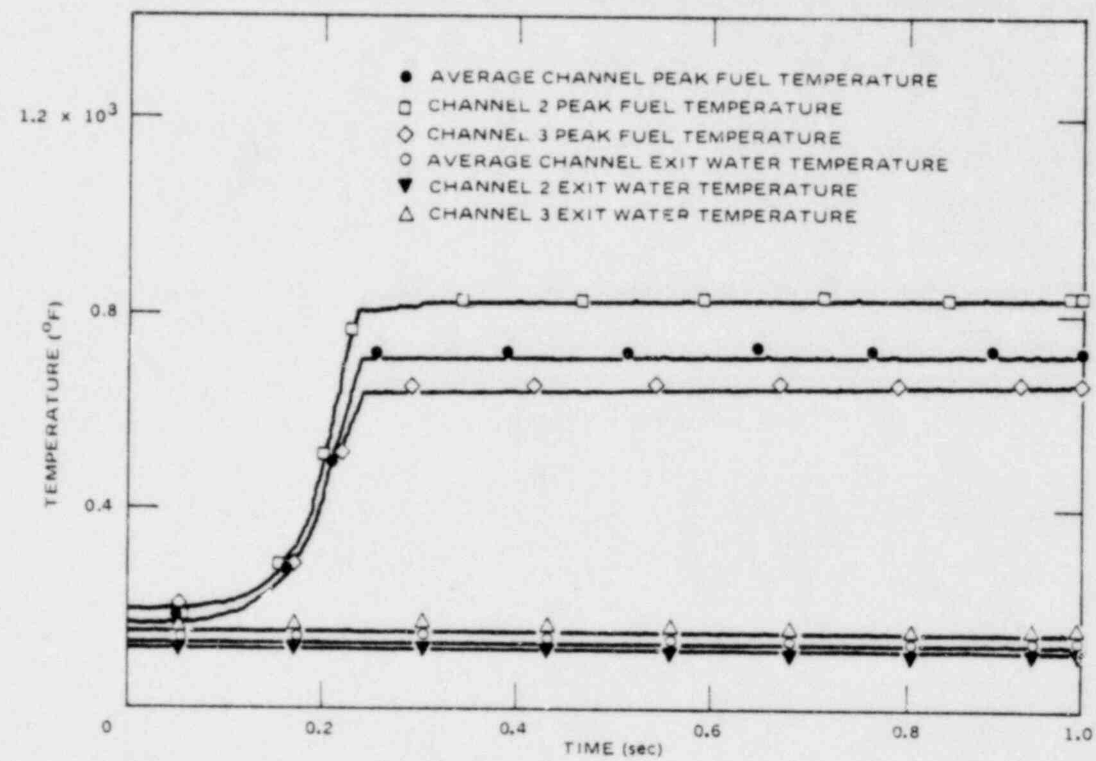


Figure 11-3. Δk-Steps with 150-kW Scram



a. Log of Power



b. Temperature

Figure 11-4. 1.3$ Step from 100 kW with Scram neutron groups which are controlling the rate of change of power. Even after an excursion has reached the steam-blanketed condition and the heat-transfer coefficient has dropped to 10 Btu/h-ft^2-°F, a power level of 100 kW can be maintained without melting at the hot spot. The peak temperature characteristic is very sharp. A peak temperature of only 400°F resulted from a 1.2$ step, compared to approximately 840°F for the 1.3$ step. The results for lower initial power and flow show that fuel temperatures are lower for these other cases.

11.4.2 Idealized Finite Ramp Reactivity Insertions - with Scram

Large reactivity insertions over short periods of time were studied for finite reactivity ramps. Reactivity insertions of 2$ and 4$, with durations from 0.2 to 0.6 second, were analyzed. The results for initial powers of 100 kW with the overpower scram occurring at 150 kW are given in Figure 11-5. For the 4$ insertion, fuel melting is not expected if the duration of the insertion is greater than 0.5 second. For the 2$ case, the minimum acceptable insertion time was 0.24 second. Figures 11-6 and 11-7 show near-limiting cases. In both cases, steam-blanketing and nucleate boiling occurred almost simultaneously so that fuel-surface, heat-transfer conditions were poor throughout the transients, and no bulk boiling was observed. In each case, power dropped below the level at which the hot spot is cooled even with steam-blanketed conditions before peak fuel temperature reached melting. For transients starting from lower power levels, the temperatures will be slightly less than those shown in Figure 11-5 because of the lower initial temperature. The sharp characteristic, however, places the limiting reactivity insertion time at nearly the same value. The consequence of inserting these large amounts of reactivity too fast, or if the scram failed, would be partial core destruction. The primary shut-down mechanisms would be associated with the expansion and dispersion of the fuel.

11.4.3 Reactivity Insertions - without Scram

It may be hypothesized that certain structures (used to support the control and safety rod mechanisms as well as experiments) might fail or move during a seismic event in such a manner as to withdraw the control rods and experiments from the core region and prevent operation of the safety rods. The cadmium

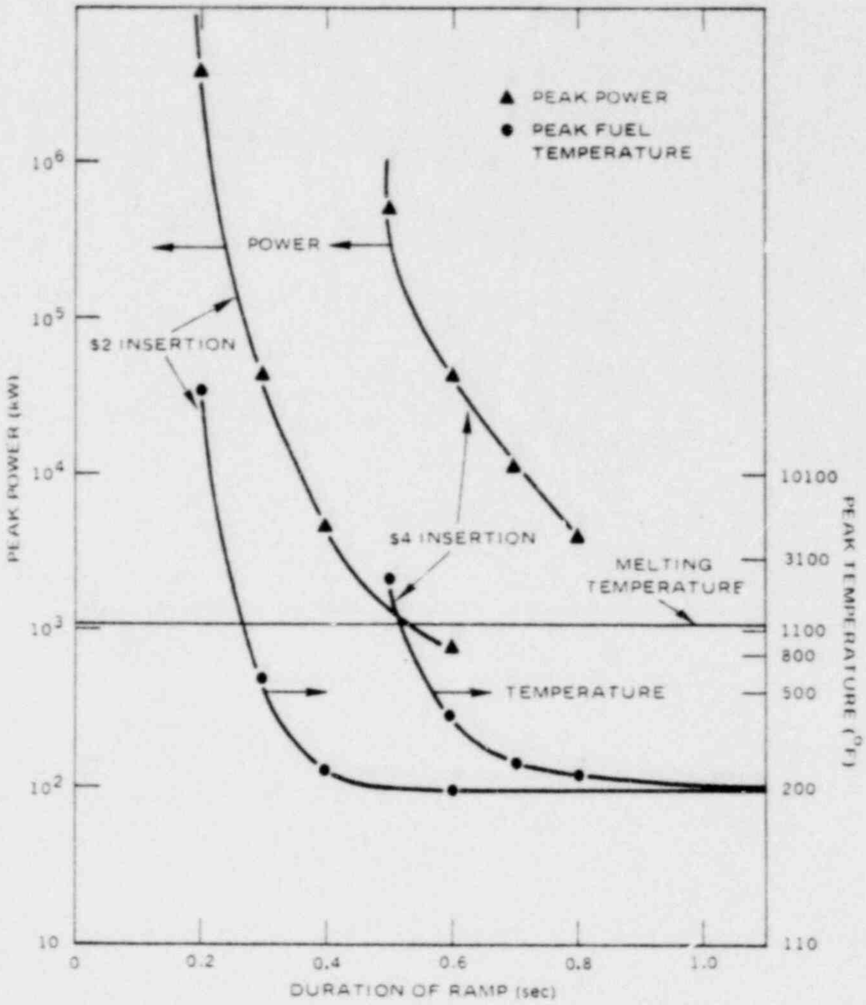
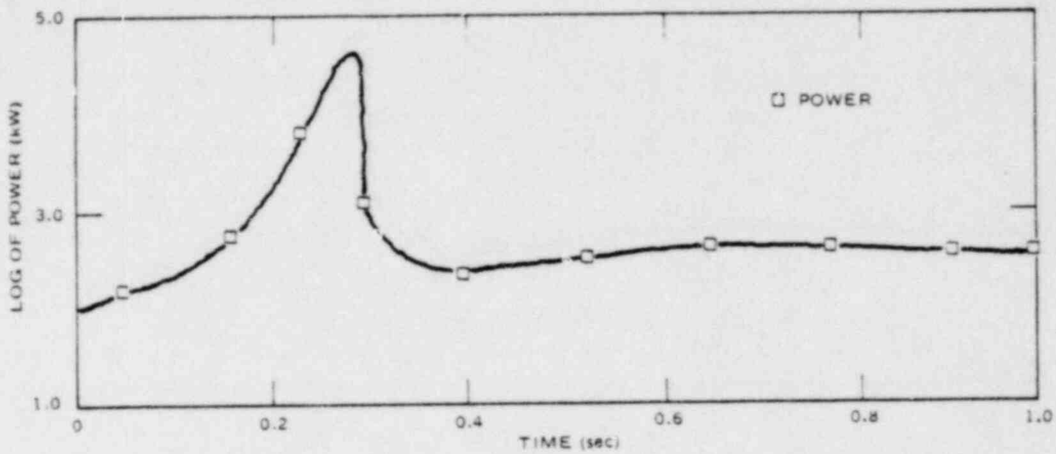
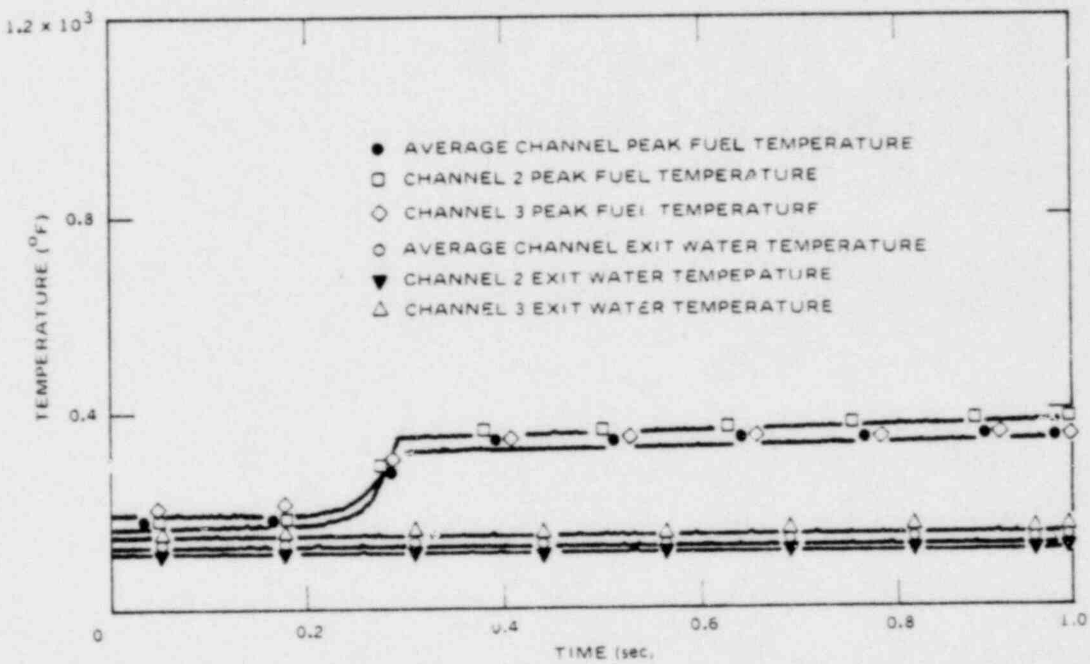


Figure 11-5. 100-kW Finite Ramp Insertion with High-Flux Scram

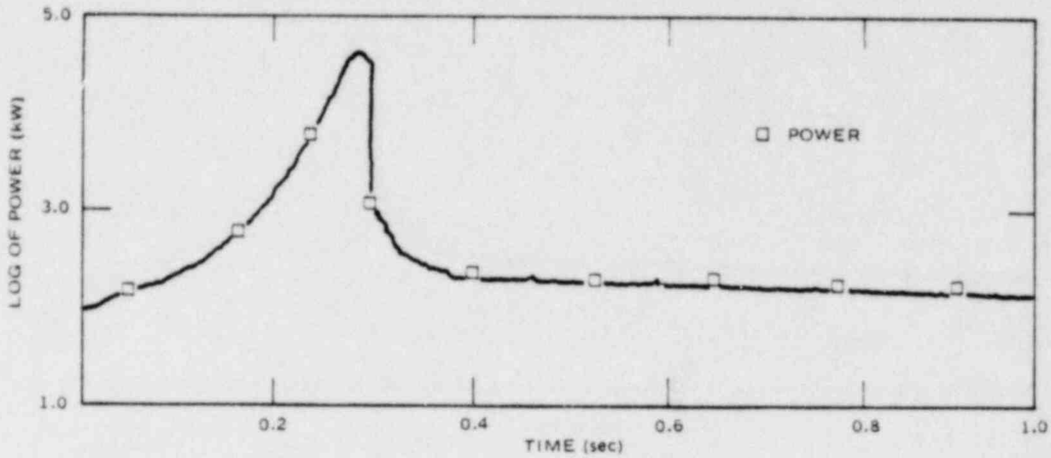


a. Log of Power

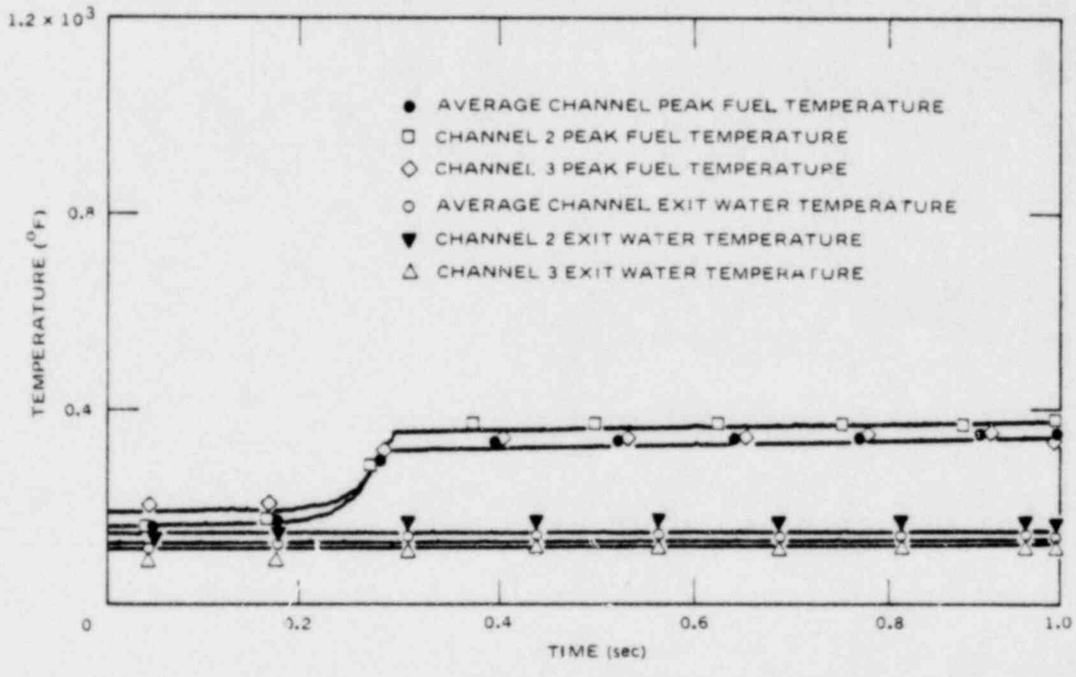


b. Temperature

Figure 11-6. 4$ Ramp in 0.6 Second from 100 kW with Scram



a. Log of Power



b. Temperature

Figure 11-7. 2$ Ramp in 0.3 Sec from 100 kW with Scram poison sheets are manually positioned entirely within the graphite reflector, have no drive mechanisms, and are mechanically restrained so they will not move relative to the core during a seismic event. If the reactivity addition caused by control rod and experiment movement is sufficiently large, a power excursion not terminated by a scram could occur and result in fuel melting. The NTR will be operated in such a manner as to limit the potential excess reactivity to less than that required to cause fuel damage, assuming failure to scram.

From full power, the transient would be stopped by bulk boiling, even if all scrams fail, before fuel damage occurs for sizeable step reactivity insertions. The results of a 0.76$ step reactivity insertion are shown in Figure 11-8. Power peaked at 4×10^3 kW, and bulk boiling began in the hot channel at about 2.3 seconds. The negative reactivity feedback from the voids was enough to drop power low enough that the temperature of the hot spot stayed below 255°F. The core did not steam-blanket and provided good boiling heat transfer. The peak temperature characteristic versus magnitude of the reactivity step was very sharp (Figure 11-9). For small steps, the transient settled out with nucleate boiling characteristics.

To determine the effects of positive reactivity additions from less than full power, all transients were run with an initial power level of 1×10^{-7} kW. Inlet water temperatures ranged from 55 to 90°F and initial positive reactivity steps were varied from 0.60 to 0.80$. Results of the transient analyses led to two conclusions. First, the transients are relatively long, on the order of 40 or more seconds, which leads to the conclusion that the positive reactivity can be introduced in either a step or relatively long ramp without affecting the outcome. Second, the positive reactivity feedback from the temperature coefficient, while not important for the full-power cases because the feedback is very small, is important for the zero-power cases.

Limiting values for the positive reactivity insertions were determined based on the acceptance criteria that the resultant transient was terminated by bulk boiling before any steam-blanketing occurred in the core. The limiting values based on this criterion are shown as the reactivity insertion limit values in Table 11-1 for various inlet water temperatures. Also in Table 11-1 are

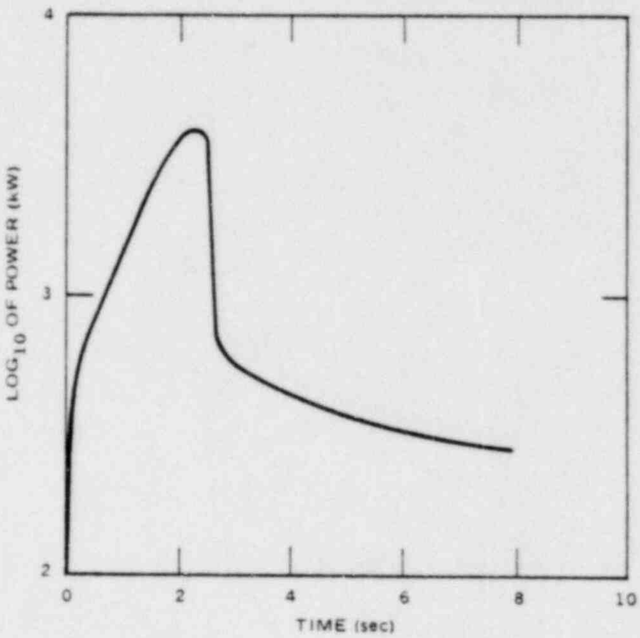


Figure 11-8. 0.76$ Step from 100 kW - No Scram

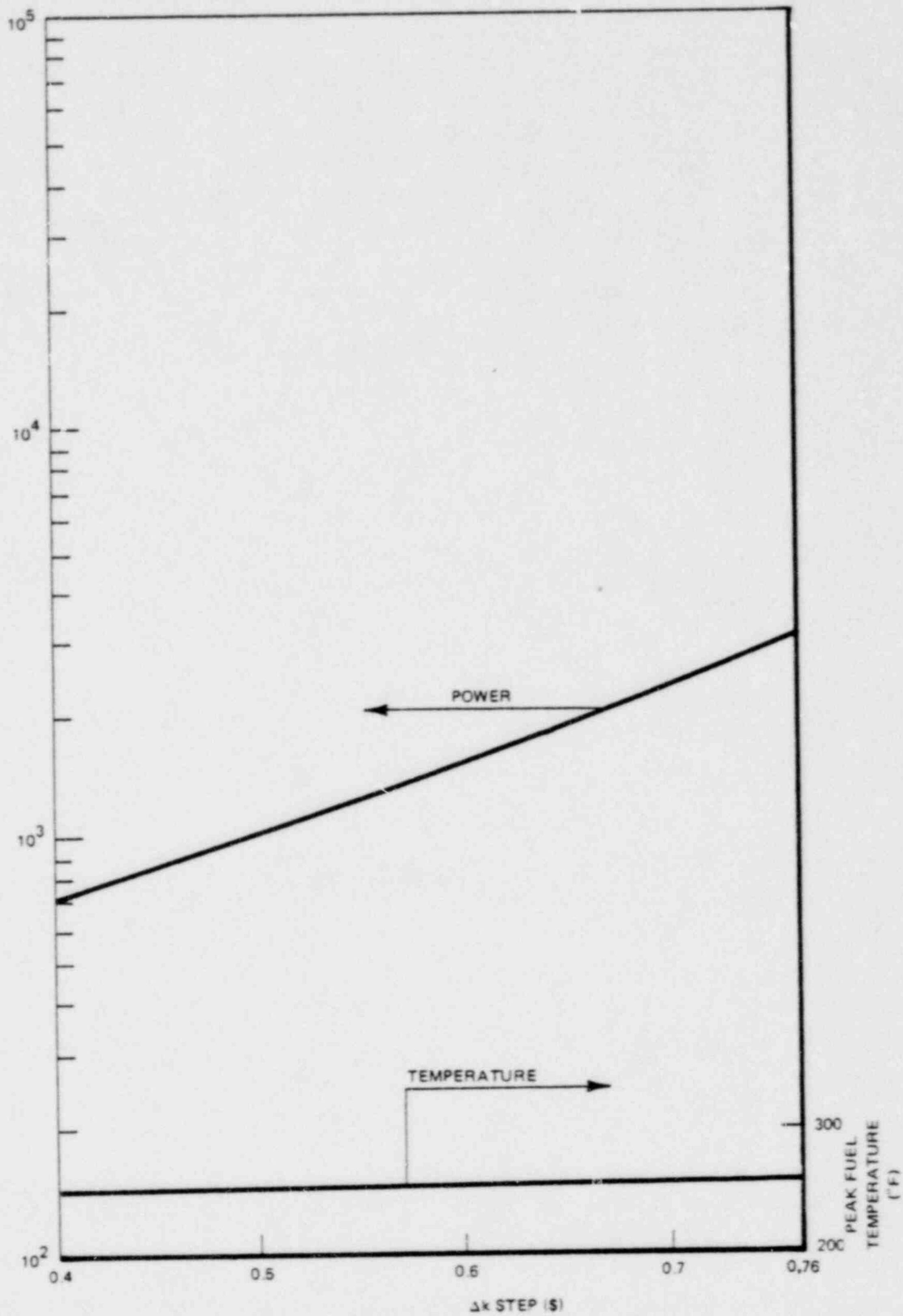


Figure 11-9. ΔK Steps from 100 kW-No Scram

Table 11-1

LIMITING REACTIVITY INSERTION VALUES

BASIS: Transient terminated by bulk boiling before any steam-blanketing in core; initial power 1×10^{-7} kW/no scram.

Reactivity Insertion Limit ($)	Inlet Water Temperature (°F)	Reactivity Addition from Temperature ($)*	Total Reactivity ($)
0.62	55	0.14	0.76
0.66	65	0.10	0.76
0.76	90	0.03	0.79

*Using the temperature coefficient of $d\rho/dt = -5.7 \times 10^{-3}$ (T-124) ¢/°F, where T is the water temperature in °F; the reactivity added by increasing the water temperature from T to 124°F is equal to 2.85×10^{-3} $(T-124)^2$ ¢

the maximum values of additional reactivity available from the temperature coefficient, which is positive at temperatures less than or equal to 124°F. As can be seen from the total reactivity values, limiting the total excess reactivity available from the temperature coefficient, control rods and experiments to 0.76$ or less ensures that there are no mechanisms available which will cause fuel damage.

Reactor power and peak fuel temperature versus time for a 0.66$ step insertion from 1×10^{-7} kW and 65°F inlet water temperature is given in Figure 11-10. While the time scale is different for other limiting reactivity insertions, the peak fuel temperature is virtually identical; it remains in the 240-250°F range during bulk boiling.

It should be stressed that these transient calculations are extremely conservative, since no credit is taken for the negative reactivity feedback from subcooled voids during nucleate boiling. With the large negative void coefficient of the NTR, it is felt that all the transients presented here would

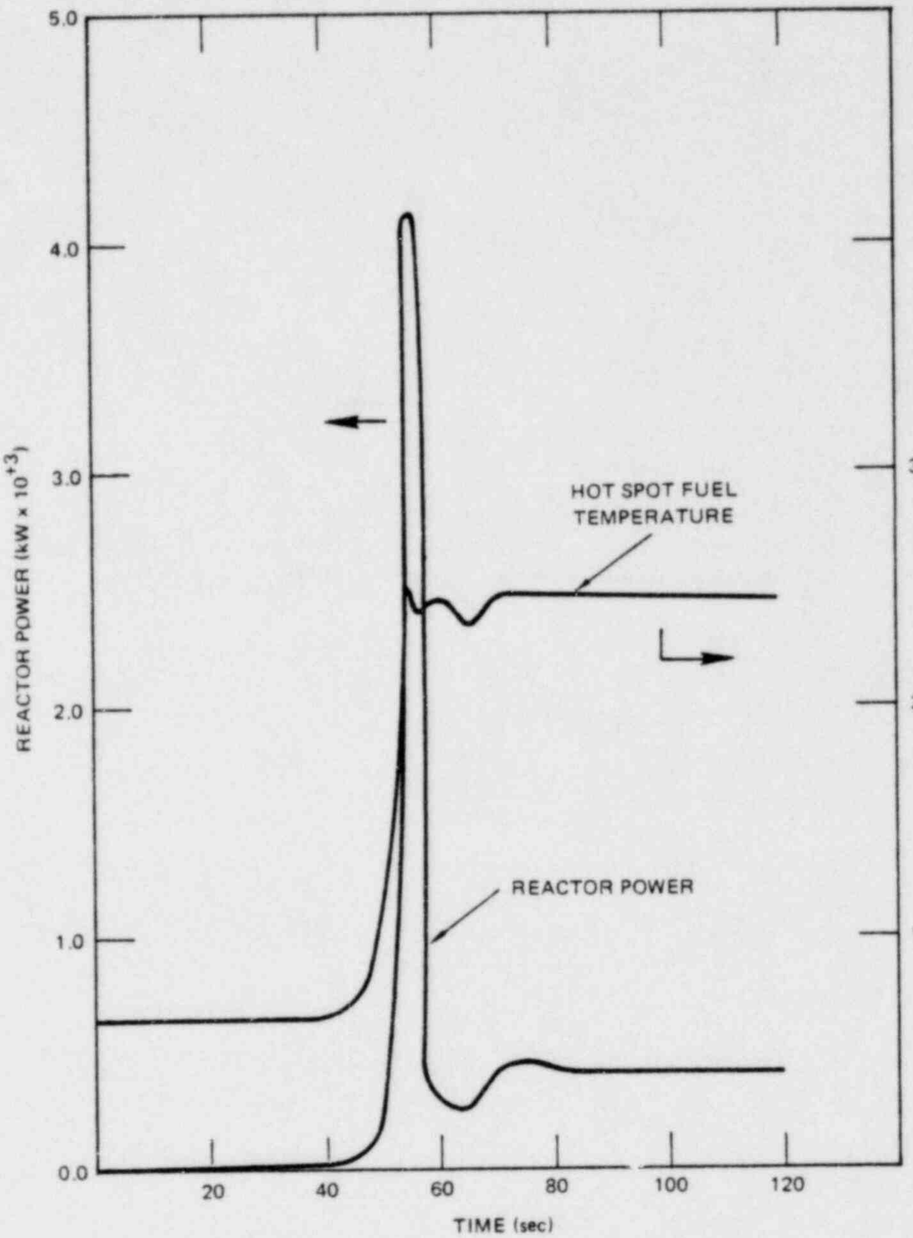


Figure 11-10. Reactor Power and Hot Spot Fuel Temperature Versus Time, 0.66$ Step from Source Level, 65°F Coolant Inlet Temperature - No Scram

terminate before bulk boiling and realistic limits for reactivity insertions would be 0.90-1.00$.

The discussion above described the sequence of events for the first 2 minutes of the hypothetical event. A discussion of several possible sequences of events from the 2-min point in time out to the final state for the reactor follows.

The diagram presented in Figure 11-11 shows the various possible states for the NTR following the initial reactivity transient. If no operator intervention is taken, the final state of the reactor will always be State A (reactor shutdown caused by loss of coolant). The extremely conservative loss-of-coolant analysis presented in Subsection 11.4.6 demonstrates the loss of coolant for the NTR has no significant consequences.

The performance of the reactor in the near term after the postulated seismic event depends on the extent of damage to the remainder of the reactor system. The most significant items are: (a) the primary system piping, (b) the primary pump, (c) the secondary water supply system, and (d) the electrical supply to the reactor system.

It is highly unlikely the primary system of the reactor would still be intact after a seismic event severe enough to result in the reactivity addition by the massive structural failure postulated here. If the primary system failed at the same time as the reactivity addition, the reactivity transient would not be significantly altered. The loss of coolant from the reactor results in shutdown by voiding the reactor core (Figure 11-11, State A).

We may also assume loss of electric power because: (a) it is highly improbable that electric power to the site (including the NTR) would survive the event postulated here; and (b) even in such an improbable circumstance, site emergency procedures call for the termination of all utility services to any buildings or facilities believed to have suffered damage.

As the loss of electrical supply automatically deactivates the primary system pump and automatically closes off flow to the secondary system, the structural

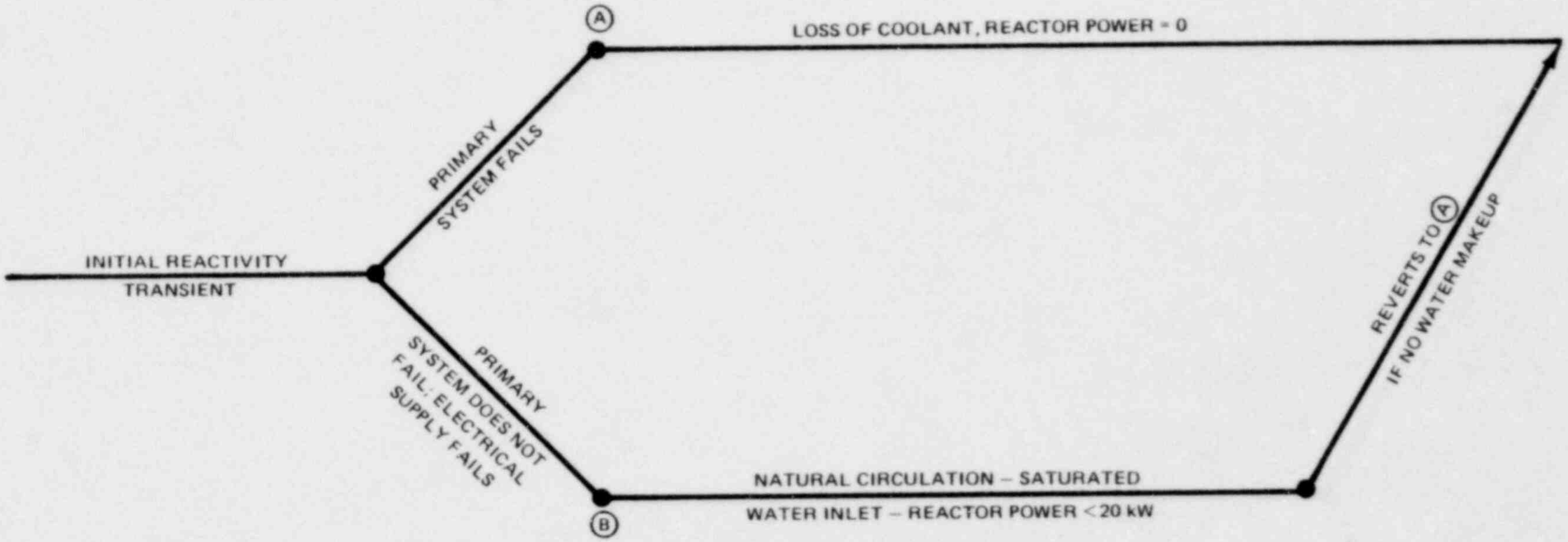


Figure 11-11. Possible Reactor States Following the Postulated Seismic Event

fate of the secondary system becomes a moot question, and we need consider only the possibility of the primary system somehow surviving the event. If the primary system does not fail, or leaks at a very slow rate, the system will arrive at State B. For this state, the reactor will operate in a natural circulation mode at less than 3% of the pumped flow rate and at a power of less than 20 kW. Since there is no secondary cooling, the 20 kW of reactor power must be dissipated by the heat loss from the uninsulated reactor primary piping and by evaporation or boiloff of the primary coolant. If it is assumed that all of the heat is lost by boiloff, the rate of coolant loss is less than 4.5 lb/h. There are approximately 1000 of the 1800 gallons of water in the fuel storage tank which could drain into the reactor core can through the fuel loading chute to make up for the boiloff. If no water were made up to the system and no action were taken to shut down the reactor, it would operate for 70 days or more at a power level of 20 kW, or less. The loss of coolant by boiling will be a less severe event than the loss of coolant event described in Subsection 11.4.6 for two reasons. First, the reactor power is lower (20 kW, rather than 100 kW) and, if a primary system leak was not developed, the loss of coolant is not complete. In fact, the slow loss of coolant will result in a slow decrease in power and only a partial loss of coolant will occur. The core can be voided from the top by nearly 20% before any single fuel element is totally uncovered and the surface heat flux would be so low that it could quite easily be cooled by convection to the steam and radiation to the inner surface of the core can.

As the maximum fuel temperature for a loss of coolant occurring at a power level of 100 kW is 620°F, no fuel melting, and hence, no fission product release, will occur from this accident.

11.4.4 Rod Withdrawal Accidents

The safety system and rod withdrawal procedures are designed to provide adequate control of the reactor at all times. Even if interlocks fail and the operator deviates from normal procedures so that the rate of power increase is not controlled by normal manual control rod movements, the reactor period and neutron flux level monitors would scram the reactor. If the reactor did not scram, the analysis in Subsection 11.4.3 is applicable. It is shown in the transient analysis the reactivity can be introduced in either a step or relatively long ramp without affecting the outcome. This analysis indicates that the transient which results from the total reactivity addition of the control rods, experiments, and temperature effect without scram (and the potential excess reactivity is $\leq$0.76$) does not melt fuel. Therefore, the transient which would be caused by the withdrawal of all the rods can be accommodated.

11.4.5 Reactor Loss-of-Flow Accident

To analyze the effects of a sudden loss of primary coolant recirculation pumping, it was assumed that the worst loss-of-flow accident (instantaneous seizure of the rotor in the single recirculation pump in the system) occurs. For such an accident, it is estimated that the pump flow will coast down to a natural circulation value within 0.1 second. The accident is assumed to occur while the reactor is operating at 100 kW. Although the transient would be terminated by the low-flow scram, in this analysis, it will be assumed that this scram does not function. After the flow has decreased to the natural circulation rate, the coolant temperature and the natural circulated flow rate will increase. This trend will continue until either (a) bulk boiling at the hot spot produces enough voids to stop the power rise by reactivity feedbacks, or (b) the average coolant temperature goes high enough to allow the negative temperature coefficient to halt the power rise. The initial core average coolant temperature is 110.6°F, and the initial excess reactivity is assumed to be zero. As the coolant temperature increases, the excess reactivity also increases to a turnaround temperature of 124°F, at which point the temperature coefficient becomes negative. Meanwhile, reactor power is on the rise, but will begin to slow down as the coefficient goes negative. The final steady-state operating point will correspond to a power and flow combination which gives the same reactivity contribution from temperature as for initial steady-state operation. Using the coolant temperature coefficient (Section 4, Equation 1), this final coolant temperature level is 138°F. Thus, there is no bulk boiling in the average channel. The heat flux is far below the heat flux necessary to initiate film blanketing. Moreover, the fuel plate surface tempera re has been limited to a value well below the melting point, as a result of local surface boiling.

The times in the preceding discussion are all referenced to the time of initial flow reduction. Peak reactor power during the loss-of-flow condition is 101.2 kW and occurs 17 seconds after instantaneous reduction of flow. Nucleate boiling begins in the outlet of the hot channel at 17.5 seconds. Maximum fuel temperature during the transient is 238°F. Bulk boiling begins in the outlet of the hot channel at 48 seconds. Equilibrium reactor power of 16 kW is reached at approximately 160 seconds. Maximum equilibrium fuel temperature is 226°F. After the very small power increase, the burnout ratio increases as power decreases during the transient and ends at an equilibrium value of approximately 26.

11.4.6 Reactor Loss-of-Coolant Accident

The reactor loss-of-coolant accident involves the total loss-of-coolant inventory in the core as the result of a rupture in the primary system, combined with a failure to scram. The accident is postulated to occur as follows:

1. Primary system ruptures at some point below the core entrance so that gross removal of core coolant supply occurs.

2. As the water in the core is removed, the fuel is uncovered; the uncovering of the fuel acts to shut down power generation to a decay heat level.

The rupture is taken as being large enough to cause a very rapid coolant loss so that all water is lost and the core power is down to the decay heat level very shortly after the accident. It is assumed there is no post-incident cooling system in the reactor and, as a result, the only cooling of the fuel plates occurs by any natural convection air currents that may be set up and by radiation heat transfer from the core to the graphite. For simplicity and conservatism, convective heat removal by natural air currents is neglected. It is further assumed that no heat escapes from the graphite stack to the outside environment.

The initial power level of the reactor is taken to be 100 kW, and the subsequent decay heating rates are given in Figure 11-12 as a fraction of the initial power. A power-peaking factor of 1.30 was assumed, which includes

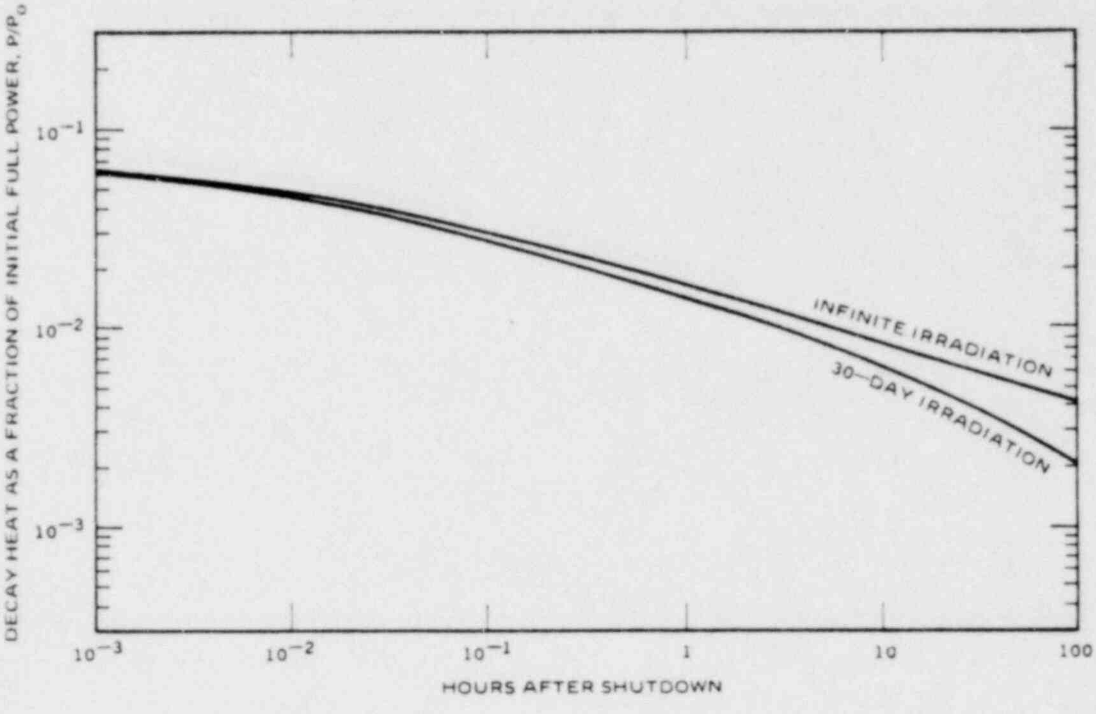


Figure 11-12. Decay Heat Rate

both the normal axial peaking and severe azimuthal skewing. The calculation was performed using a version of the Transient Heat Transfer (THT) computer program.[24] The nodal structure for the problem is shown in Figure 11-13. Axial heat transfer was neglected.

The peak fuel temperature and the volume-averaged graphite temperature are shown in Figure 11-14 as a function of time after coolant loss. The fuel temperature reaches a maximum of about 570°F 100 minutes after coolant loss and then begins to decline. The rise of the graphite temperature is almost imperceptible - only 15°F in 3 hours.

The analysis was repeated using a higher peaking factor. The maximum fuel temperature for a loss-of-coolant accident with a 1.58 peaking factor is 620°F at 1-1/2 hours. Reactor power at the time of the peak fuel temperature is 1.5 kW. It has been shown that this power could be tolerated indefinitely without increasing graphite temperatures to over 150°F, assuming a natural convection heat-transfer coefficient of 0.6 Btu/h-ft^2-°F on the exposed surface of the reactor. Therefore, a second fuel temperature peak greater than 150°F is not possible.

11.5 EXPERIMENT SAFETY ANALYSIS

11.5.1 Introduction

Descriptions of the NTR experiment safety programs and associated facilities, equipment, and procedures are presented in Section 7. As stated, before any experiment may be conducted, there must be a review and approval of a written description and safety analysis.

The purpose and requirements for experiment safety analyses are described in Subsection 11.5.2. Considerations that will be addressed are identified and discussed.

The potential mechanical and radiological consequences of postulated accidents involving explosive material at the NTR facility are described in Subsection 11.5.3. The details of the accident analysis are contained in Appendix B.

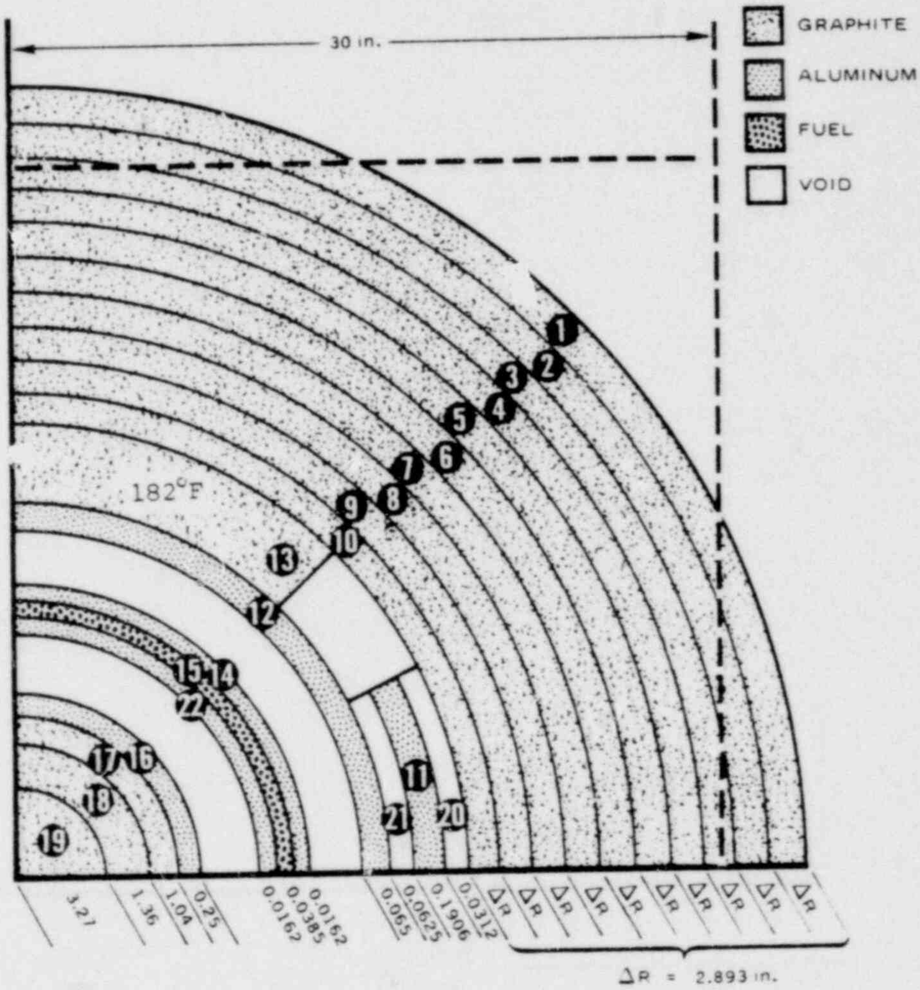


Figure 11-13. Node Structure for THT Analysis

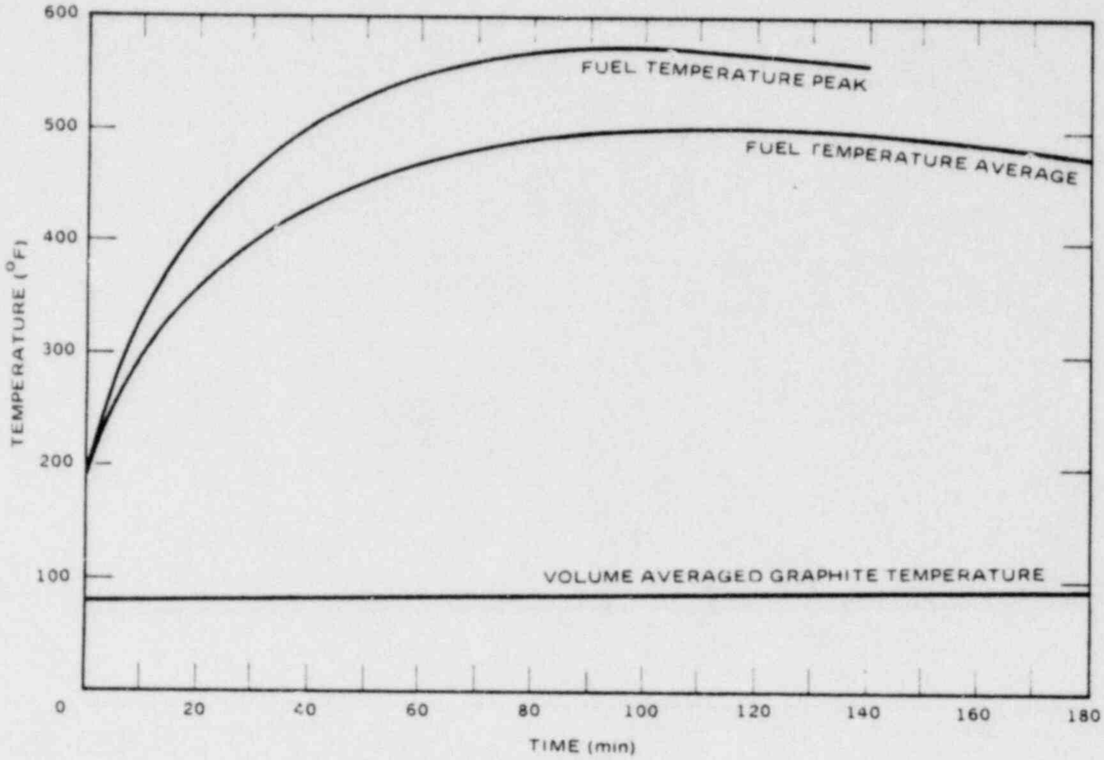


Figure 11-14. Fuel and Graphite Heatup Following Loss of Coolant (Power Calculated to 200 seconds)

The limitations that will apply to all types of experiments are discussed. Adherence to these limitations is mandatory and will provide assurance of safe performance of experiment programs within imposed regulatory restrictions.

11.5.2 Safety Analysis

The purpose of the safety analysis for experiments is to ensure that consideration is given to any feature of the design or conduct of an experiment, including intended functions and possible malfunctions, which could create, directly or indirectly, a radiological exposure hazard. When applicable, the analysis will consider:

1. Any interaction of an experiment with the reactor system that has potential for causing fission product release from the fuel.

2. Any interaction that could adversely affect an engineered safety feature or control system features designed to protect the public from fission product release.

3. Inherent features of an experiment that could create beams, radiation fields, or unconfined radioactive materials.

4. Potentially adverse interaction with concurrent experimental and operational activities.

The safety evaluation for each experiment utilized in experimental facilities will consider:

1. The physical conditions of the design and conduct of the experiment.

2. The content of the material.

3. The administrative controls employed to evaluate, authorize, and carry out the experiment.

A description of specific items that will be addressed, when applicable, follows.

11.5.2.1 Reactivity Effects

The principal concern is with a net positive reactivity effect, whether it is caused by the insertion of an experiment with a positive effect, or by the removal of an experiment having a negative reactivity effect. Every experiment or type of experiment, as appropriate, will be evaluated for:

1. Its potential reactivity worth.

2. The rate of change of reactivity of unsecured experiments and movable experiments.

11.5.2.2 Thermal and Hydraulic Effects

An experiment will be evaluated to ensure its thermal limits are not e ceeded and for its actual and potential thermal effects on reactor components and coolant. This evaluation will be made for the reactor at the extremes of its operating margin, as defined by limiting safety system settings.

The experiment design will be evaluated to ensure it will not adversely affect flux shape or reactor coolant flow considerations that were used to define or are implicit in the reactor safety limits.

11.5.2.3 Mechanical Stress Effects

Materials of construction and fabrication and assembly techniques utilized in experiments will be evaluated, as appropriate, to provide assurance that no stress failure will occur from manipulation and conduct of the experiment or as a result of unintended but credible changes of, or within, the experiment. Every experiment or type of experiment, as appropriate, will be evaluated with respect to storage and possible uncontrolled release of any mechanical energy.

11.5.2.4 Material Content of Experiments

Certain kinds of materials which may be used in experiments possess properties which could have significant afety implications. Limitations on the amounts of such materials can limit the consequences of experiment failures. The material content of an experiment will be analyzed and limited, as required, utilizing the following classifications as a guide:

1. Radioactive material
2. Trace elements and impurities
3. High cross section materials
4. Highly reactive chemicals (explosives)
4. Corrosive chemicals
6. Radiation-sensitive materials
7. Flammable material
8. Toxic material
9. Cryogenic liquids
10. Unknown materials

11.5.2.5 Administrative Controls

Administrative controls are in place to ensure a written description and safety analysis are generated for each experiment type. Each experiment type must be reviewed by a technically competent independent review unit and approved by the Facility Manager or his designated alternate. An experiment type includes repetitive experiments that involve common safety considerations and a similar reactor setup. Acceptance criteria for an experiment include compliance with regulatory requirements (including 10 CFR 20 and Technical Specifications), GE procedures, and good safety practices. The independent review unit and its modulus operandi is discussed in Section 9.

Administrative controls applicable to all experiments are listed below:

1. Every experiment must have prior explicit written approval by the Reactor Supervisor or an NTR Licensed Senior Reactor Operator, as determined by the experiment type.

2. Every person not part of NTR Operations who is to carry out an experiment must be certified by the Reactor Supervisor or an NTR Licensed Senior Reactor Operator, as determined by the experiment type, as to the sufficiency of his knowledge and training in procedures required for the safe conduct of the experiment.

3. Detailed written procedures must be provided for the use, or operation of, each experiment facility and each experiment type.

4. The Licensed Reactor Operator at the console must be notified just prior to moving any experiment (or series of experiments as specified by procedures) within the NTR facility.

5. Every experiment removed from the reactor must be subject to a radiation and contamination monitoring procedure, as applicable, that anticipates levels greater than those predicted.

11.5.3 Consequences of Accidental Explosions

The facilities, equipment, and procedures used for experiment programs that involve explosive material are described in Section 7. To provide safe limits for the amounts of explosives permitted in the NTR handling and radiography areas, separate Design Basis Accidents (DBAs) were defined for the south cell, the north room, and the set-up room. In general, these DBAs assumed a highly improbable accidental detonation of all explosive devices in the particular area and the consequences are evaluated in terms of both radiological and mechanical effects.

11.5.3.1 Radiological Consequences

The radiological consequences of an accidental detonation of an explosive device are essentially nonexistent. Induced activities in explosive materials, structural materials containing the explosive, or structures used in neutron radiography are extremely small considering thermal neutron fluxes of 2×10^6 nv and normal exposure times of 10^3 seconds. However, if sufficient other sources of radioactive materials are present in the immediate area and become dispersed or airborne during the accidental detonation, the radiological consequences could be serious. Operations at the NTR include neutron radiography of plutonia fuel pins and capsules containing significant amounts of fission products. Evaluation of the DBAs indicates that while it is virtually impossible to involve these materials in the accident, it is prudent to exclude these large sources of radioactive material from any area in which explosive devices are being handled.

Small amounts of radioactive materials (e.g., uranium contained in fission chambers or irradiated samples used in various experimental programs) may be safely stored in the south cell or the north room during the neutron radiography of explosives. By limiting these quantities to 10 curies of radioactive materials and to 50 grams of uranium, the health and safety of the general public will in no way be compromised. Storage locations are at least 5 feet from any explosive handling position and are normally either in concrete block caves or small lead casks. While accidental detonation of explosive devices might cause minor damage to the storage structures, the probability of releasing even a small percentage of the radioactive material from their contents is negligible. Assuming a 1% release and stable atmospheric conditions (inversion), maximum site boundary doses are less than 20 mRem to the thyroid and 1 mRem to the whole body under this most pessimistic combination of circumstances. No radioactive materials other than those produced by neutron radiography are permitted in the set-up room if explosive devices are present.

11.5.3.2 Mechanical Consequences

The primary safety criterion is that complete simultaneous detonation of all explosive devices in a particular area will not increase the probability or consequences of accidents previously analyzed or create the possibility of a different type of accident not previously analyzed. While minor structural damage and possible injury to personnel will occur in the immediate area, damage to the reactor core, graphite pack, or control system is not expected, and injury to personnel is minimized. Damage to the reactor is prevented by limiting the amount of explosive material allowed in the particular areas (south cell, north room, and set-up room) and by design and construction of an additional shield structure (south cell). Potential injury to personnel is minimized by strict adherence to safe explosive handling procedures. The mechanical safety analyses are discussed in detail in Attachment B-1 (Appendix B[1]) and show that neutron radiography of explosives can be accomplished safely in the reactor facility by limiting both the total quantity of explosive materials in pounds of equivalent TNT and the distance of the explosive material from sensitive components and structures.

Summarizing Attachment B-1 gives the following limits:

South Cell $W = ({}^{D}/2)^2$

Where W is the weight of explosive in pounds of equivalent TNT; D is the distance from blast shield in feet; and $W \leq 9$ pounds, $D \geq 3$ feet.

North Room (without MSM) $W = D^2$

Where W is the weight of explosive in pounds of equivalent TNT; D is the distance from north room wall in feet; and $W \leq 16$ pounds, $D \geq 1$ foot.

North Room (with MSM) $W \leq 2$ pounds of equivalent TNT

Set-up Room $W \leq 25$ pounds of equivalent TNT

11.5.3.3 TNT Equivalence

The equivalence of an explosive material to TNT on a gram basis is determined by ratioing various parameters of the explosive to those of TNT. These parameters include brisance, ballistic mortar, trauzel test, and detonating velocity, and are described in "Properties of Explosives of Military Interest," AMCP 706-177. This report contains pertinent data on many types of explosives and is used as a primary reference document. The equivalent grams of TNT for an explosive being handled or radiographed is determined by the following:

$$\text{Gram equivalent TNT} = \text{grams of explosive} \times \frac{\text{Parameter of explosive}}{\text{Parameter of TNT}}$$

where the ratio of parameter is chosen to be the highest value of the brisance, ballistic mortar, trauzel test, or detonating velocity ratios.

If data are not available on the explosive, or the composition is proprietary, a factor of 2 is used for the parameter ratio, which is conservative and higher than any value found in AMCP 706-177.

11.5.3.4 Reactivity Effects

There are no reactivity effects directly associated with neutron radiography of explosive or other materials. Objects undergoing inspection are located at relatively large distances from the reactor and have no effect on core reactivity. Even the large shutter in the south cell may be moved during reactor operation without affecting core reactivity. Some minor reactivity effects are associated with the neutron radiography beam preparation devices. Under normal circumstances, shock waves from accidental detonation of explosives will be attenuated sufficiently to make movement of the beam preparation device highly improbable. It is also noted that the reactivity added during removal or expulsion of the beam preparation device from the core region is included in the total amount that would be available, as discussed in Subsection 11.4.3. Therefore, the consequences would be less severe than those analyzed, which assumed 0.76$ step insertion both with and without scram.

11.5.4 Experiment Limitations

Safety-oriented limits and restrictions applicable to experiment facilities and experiment programs follow. The limits and restrictions presented are derived from the reactor and experiment safety analyses, over 23 years of experience in conducting experiments at the NTR, and sound engineering practice. The majority of these limits are contained in the Technical Specifications. Adherence to the limits and restrictions below is mandatory and provides assurance that:

1. There is no anticipated mode of experiment operation that will endanger the health or safety of the general public or plant personnel.

2. No experiment will be performed that involves a technical specification change or an unreviewed safety question (as defined in 10 CFR, Section 50.59).

3. A proposed experiment type will be evaluated in detail and its execution controlled so as to reduce any radiation exposure to the public and plant personnel to the lowest practicable level.

11.5.4.1 General Experiment Requirements

1. A written description and analysis of the possible hazards involved for each type of experiment shall be evaluated and approved by the facility manager or his designated alternate before the experiment may be conducted. Records of such evaluation and approval shall be maintained.

2. No irradiation shall be performed which could credibly interfere with the scram action of the safety rods at any time during reactor operation.

3. Experimental capsules to be utilized in the experimental facilities shall be designed or tested to ensure that the pressure transients,

if any, produced by any possible chemical reaction of their contents and leakage of corrosive or flammable materials will not damage the reactor.

4. No experimental objects shall be inside the core tank when the reactor is operating at a power greater than 0.1 kW.

5. Experimental objects located in the fuel loading chute shall be secured to prevent their entry into the core region.

11.5.4.2 Reactivity Limits

1. Requirements pertaining to the reactivity worth of experiments are as follows:

   a. The sum of the potential reactivity worths of all experiments which coexist plus the reactivity available from control rods and coolant temperature shall not exceed 0.76$.

   b. No experimental object shall be moved during reactor operation unless its potential reactivity worth is known to be less than 0.5$ and the operation is performed with the knowledge of the licensed operator at the console. All remotely controlled mechanisms for moving an object into the reactor core shall be energized from the reactor console; however, movement of the object may be initiated from another location.

   c. The potential reactivity worth of any component which could be ejected from the reactor by a chemical reaction shall be less than 0.50$.

2. The potential reactivity worth of experiments shall be assessed before irradiation. If the assessment warrants, the reactivity worth of the experiment shall be measured and determined acceptable before reactor full-power operation.

11.5.4.3 Explosive and Flammable Material Lists

1. The maximum amounts of explosives permitted in the NTR facilities are as follows:

   a. South cell: $W = (D/2)^2$ with $W \leq 9$ pounds and $D \geq 3$ feet;

   b. North room (without MSM): $W = D^2$ with $W \leq 16$ pounds and $D \geq 1$ foot;

   c. North room (with MSM): $W = 2$ pounds;

   d. Set-up room: $W = 25$ pounds

   Where W = Total weight of explosives in pounds of equivalent TNT
   D = Distance in feet from the south cell blast shield or the north room wall.

2. A maximum of 10 curies of radioactive material and up to 50 grams of uranium may be in storage in a neutron radiography area where explosive devices are present (i.e., in the south cell or north room). The storage locations must be at least 5 feet from any explosive device. Radioactive materials other than those produced by neutron radiography of the explosive devices and imaging systems are not permitted in the set-up room if explosive devices are present.

3. Unshielded high-frequency generating equipment shall not be operated within 50 feet of any explosive device.

4. The cumulative radiation exposure for any explosive device shall not exceed 3×10^{12} n/cm^2 from thermal neutrons and 1×10^4 roentgens from gamma.

5. The maximum possible chemical energy release from the combustion of flammable substances contained in any experimental facility shall not exceed 1000 kW-sec. The total possible energy release from chemical combination or decomposition of substances contained in any experimental capsule shall be limited to 5 kW-sec, if the rate of the

reaction in the capsule could exceed 1 watt. Experimental facilities containing flammable materials shall be vented external to the reactor graphite pack.

11.5.4.4 Special Nuclear Materials

The materials authorized by Special Nuclear Materials License No. SNM-960 or TR-1 are not permitted in the NTR experimental facilities, but are permitted in other areas.

Special Nuclear Materials (SNM) are limited to the amounts authorized under Subsections 7.15 and 11.6 - Criticality and Experiment Design Basis Accident, respectively.

11.6 EXPERIMENT DESIGN BASIS ACCIDENT

11.6.1 Introduction

The material quantity limits, clad requirements, operating limits, and required safety equipment for irradiation experiments at the NTR has been developed based on the radiological criteria given in Regulatory Guide 2.2.[25] This analysis specifically addresses the limits for singly and doubly clad plutonium-fueled capsules and shows the capability of the facility and site to accommodate a radio tive material release. The limits are dependent on the physical form of the plutonium and the filtration efficiency of the NTR stack HEPA filter system.

11.6.2 Accident Description

Regulatory Guide 2.2, Part c.2.a ("Material Content of Experiments") describes the release event as

> ". . . .a single-mode nonviolent failure of the encapsulation boundary that releases ll radioactive material into the immediate environment of the experiment or to the rea tor building as appropriate. . . ."

and in addition it states that

> "The analysis should establish the most probable trajectory of the material, if any, into restricted and unrestricted areas. Credit for natural consequence-limiting features such as solubility, absorption, and dilution and for installed features such as filters may be taken provided each such feature is specifically identified and conservatively justified by specific test or physical data or well-established physical mechanisms."

Therefore, the design basis accident for an experiment in the NTR is described as follows:

1. Experiment material is Pu-239. Limits have been established for both single and double encapsulation and for both loose powder form and

sintered oxide pellet form resulting in four different cases with independent limits.

2. The most probable trajectory of the released material is from the experiment location to the reactor cell area. Since the event is a single-mode nonviolent failure, the NTR ventilation system will be considered to be operational. The airborne material will be exhausted from the reactor cell area through the HEPA filter bank and out the NTR stack. Experiments with a potential for release that would not be released via the reactor cell will be provided with a local close-capture system to ensure release is through the HEPA filters. For purposes of this evaluation, the NTR HEPA filter system filtration efficiency for 0.3-μm-diam particles is conservatively assumed to be only 99%.

3. The release fractions of Pu fuel and fission products to the environment are assigned as follows:

	Powder (%)	Pellet (%)
Release from capsule to reactor cell:		
Pu-239	100	0
Noble Gas	100	100
Iodine	100	25
All Remaining Fission Products	100	0
Release from reactor cell to stack:		
Pu-239	1	1
Noble Gas	100	100
Iodine	100	100
All Remaining Fission Products	1	1

4. Dose Limits:

	Single Encapsulation	Double Encapsulation
2-hour Fence-Post Man	0.05 Rem	0.5 Rem or 1.5 Rem to Thyroid
Operator, During Evacuation	0.5 Rem	5 Rem or 30 Rem to Thyroid

A more complete list of critical organ dose limits is given in Table 11-2.

5. The unrestricted area exposure will result from the diluted-dispersed cloud of isotopes released from the NTR stack, which reaches the nearest site boundary under type F meteorological conditions at 1 m/sec over a 2-hour period.

6. The restricted area exposure will result from the submersion in and inhalation of 3.06×10^{-3}% of the isotopes released from the NTR stack at a flow rate of 1000 cfm* for a period of 5 minutes. The bases for this postulated exposure are as follows:

   a. It is assumed that the total release which will occur will be uniformly distributed over the two hours following the experiment failure.

   b. The fission products from this release will cause high-activity alarms on the stack monitors.

   c. The NTR operator will respond to the stack alarms and announce an area evacuation over the building public address system if the stack noble gas monitor indicates a concentration of $>6 \times 10^{-3}$ μCi/cc (6×10^{-10} amps). The basis for this action point is that a 5-min exposure to 6×10^{-3} μCi/cc of Kr-87 would be roughly equivalent to the maximum allowable quarterly average of 520 MPC-hours, i.e.,

$$\frac{6 \times 10^{-3}\ \mu\text{Ci/cc}}{1 \times 10^{-6}\ \mu\text{Ci/cc/MPC}} \times \frac{5\ \text{min}}{60\ \text{min/h}} = 500\ \text{MPC-hours}$$

   d. Evacuation to an upwind location will remove personnel from the stack concentration of released isotopes. On-site exposures can be controlled by use of the site alarm system and evacuation procedures.

*The stack flow rate may be as high as 3000 cfm, but 1000 cfm is used for conservatism in calculating the concentration at the stack and the resulting exposure.

Table 11-2

MAXIMUM ALLOWABLE ORGAN DOSES FOR SINGLY CLAD EXPERIMENTS

Organ	Unrestricted Area (Boundary) (Rem)	Restricted Area (NTR Stack) (Rem)
Total Body	0.05	0.5
Kidneys	0.15	1.5
Liver	0.15	1.5
Bone	0.3	3.0
Lungs	0.15	1.5
Thyroid	0.15	3.0
Stomach	0.15	1.5
Small Intestines	0.15	1.5
Upper Large Intestines	0.15	1.5
Lower Large Intestines	0.15	1.5
Skin of Body	0.3	3.0

11.6.3 Calculation Method

Computer Code DOSE77 is used to calculate organ doses resulting from the inhalation and submersion in a cloud of radioactive materials. RIBD, an optional part of DOSE77, calculates the fission product inventory which results from irradiation of fissionable material. The input required for DOSE77 to calculate the fission product inventory and organ doses resulting from exposure to the released isotopes from a Pu-239 experiment in the NTR is:

1. Capsule operating power = 60 watts (for 1 gram Pu-239 at a thermal neutron flux of 10^{12} n/cm^2-sec).

2. Capsule operating time = 1 day (arbitrary selection).

3. U-235 thermal fission cross section = 572 barns (Note: The code inputs the Pu-239 fission cross section as a function of the U-235 cross section).

4. The initial ratio of Pu-239 fissions to U-235 fissions = 10^6.

5. Thermal neutron flux = 1×10^{12} n/cm^2-sec.

6. Decay time after shutdown = 2 minutes (arbitrary).

7. Meteorology type is Pasquill Type F with a wind speed of 1 m/sec (for the boundary dose).

8. Diameter of radioactive particles = 1 micron.

9. Effective release height is at ground level.

10. The breathing rate of the exposed subject is 347 cc/sec.

11. The dose commitment time = 50 years.

12. The distance from the nearest boundary point to the NTR stack is 510 meters.

13. The fraction of the uniform 2-hour release which is inhaled in 5 minutes of exposure to the NTR stack concentration of nuclides is:

$$\text{Fraction of release inhaled} = \frac{(300\ \text{sec})(347\ \text{cc/sec})}{(120\ \text{min})(1000\ \text{ft}^3/\text{min})(2.83\text{x}10^4\ \text{cc/ft}^3)} = 3.06 \times 10^{-5}$$

14. The various release or inhalation quantities and fractions used for the various DOSE77 cases are:

Boundary Evaluation - 2-hour release quantities

	Powder Form	Pellet Form
Noble Gas	100%	100%
Halogens (except Iodine)	1%	0%
Iodines	100%	25%
Volatile Solids	1%	0%
All Remaining Fission Products	1%	0%
Pu-239	6.17E-4 Ci	0%

Restricted Area (Stack Concentration) Evaluation - 5-min inhalation and exposure

	Powder Form	Pellet Form
Noble Gas	3.06E-3%	3.06E-3%
Halogens (except Iodine)	3.06E-5%	0%
Iodines	3.06E-3%	7.65E-4%
Volatile Solids	3.06E-5%	0%
All Remaining Fission Products	3.06E-5%	0%
Pu-239	1.89E-8 Ci	0%

15. The soluble forms of the isotopes were used to evaluate the doses to most organs, but insoluble forms were used in a few cases to check the doses to the lungs, stomach, small intestines, upper large intestines, and lower large intestines.

11.6.4 Results

DOSE77 was run by using a unit quantity of Pu-239 (1 gram) for the various cases of interest. The cases are:

1. The 510-meter-boundary doses due to the mixed fission products released from both:

   a. the power form, and

   b. the pellet form of fuel.

2. The 510-meter-boundary doses due to the Pu-239 released from the powder form of fuel.

3. The doses resulting from exposure to stack concentrations of fixed fission products from both:

   a. the powder form and,

   b. the pellet form of fuel.

4. The doses resulting from exposure to stack concentrations of Pu-239 from the powder form of fuel.

The organ doses from these runs are tabulated in Table 11-3. These doses were then compared with the maximum allowable doses given by the Regulatory Guide 2.2, Code of Federal Regulations 10 CFR 20, and the International Commission on Radiological Protection (ICRP 9).[26] These maximum allowable organ doses are shown in Table 11-2.

The maximum allowable quantities of Pu-239 (in grams) allowed in both single- and double-clad experiments in both solid and powder forms, assuming a 24-hour continuous irradiation at a thermal neutron flux of 1×10^{12} n/cm^2-sec, were calculated as the ratio of the maximum allowable dose to the highest appropriate

Table 11-3

ORGAN DOSE SUMMARIES FOR BOUNDARY AND RESTRICTED AREA EXPOSURE TO NTR EXPERIMENT ISOTOPES

(1 gram Pu-239, Experiment Fission Power = 60 watts, Irradiation Time = 24 hours.)

	Soluble[a] Isotopes 50-Year Organ Dose (Rem)			
	Pellet-Form Capsule		Powder-Form Capsule	
Organ	Boundary (MFP Only)	At Stack (MFP Only)	Boundary (MFP+Pu-239)	At Stack (MFP+Pu-239)
Total Body				
Inhalation	2.92E-4	1.17E-2	3.59E-2	1.45E0
Submersion	4.54E-3	2.41E-1	9.89E-3	4.72E-1
Kidneys	1.44E-3	5.85E-2	1.53E-1	6.19E0
Liver	8.63E-4	3.50E-2	4.77E-1	1.93E-1
Bone	4.37E-4	1.76E-2	7.42E-1	3.00E-1
Lung[a]	3.54E-3	1.40E-1	2.55E-1	1.03E1
Thyroid	1.61E-1	6.58E0	6.47E-1	2.63E-1
Stomach	5.84E-4	2.47E-2	2.39E-3	1.01E-1
Small Intestine[a]	2.62E-4	1.10E-2	1.37E-3	5.72E-2
Upper Large Intestine[a]	1.02E-4	4.15E-3	9.27E-4	3.77E-2
Lower Large Intestine[a]	2.16E-4	8.77E-3	2.18E-3	8.81E-2
Skin, Submersion	7.59E-3	4.74E-1	1.54E-2	8.13E-1

[a]Most organ doses are calculated for soluble forms of the isotopes; the lung, small intestine, upper large intestine and lower large intestine dose are calculated for insoluble forms.

calculated organ dose for one gram of Pu-239. These limits are given in Table 11-4. For example:

> The maximum allowable thyroid dose for an on-site employee from failure of a single-clad experiment is 3 Rem, and the calculated thyroid dose to an on-site employee resulting from the failure of a singly clad, pellet-form NTR experiment containing one gram of Pu-239 and irradiated in the center of the reactor (experiment fission power = 60 W) is 6.58 Rem. Therefore, the Pu-239 limit for a singly clad experiment operated at a 10^{12} n/cm^2-sec flux for one day is $\frac{3 \text{ Rem} \times 1 \text{ gram}}{6.58 \text{ Rem}} = 0.46$ gram; since the double encapsulation dose limit is 10 times the single-clad limit, the Pu-239 limit for a doubly clad experiment operated at a 10^{12} n/cm^2-sec flux for one day is 4.6 grams.

In the case of a pellet-form experiment, the thyroid dose due to the iodine isotopes is the limiting factor for establishing capsule operating limits. Therefore, pellet-form experiment limits are flexible as long as the iodine inventory and resulting thyroid dose at the stack do not exceed those which result from a 24-hour irradiation of 0.46 gram (for a single-clad) of Pu-239 at a thermal neutron flux of 10^{12} n/cm^2-sec (at a power of approximately 28 watts); i.e., if the flux and/or irradiation time is reduced, the fuel quantity can be increased, as presented later. The same criteria apply to solid-form uranium-fueled experiments.

In the case of powder-form experiments, over 99% of the limiting liver dose results from the plutonium and less than 1% is due to iodine isotopes. Therefore, powder-form capsules are limited by the quantity of fuel which is released from the stack. Again, the same criteria apply to powder forms of uranium-fueled experiments, and uranium weight limits are higher than plutonium due to, among other things, the specific activity difference between the two fuel types.

The uranium limits in experiments are established using the same methods described for the plutonium evaluation. The uranium and plutonium experiment

Table 11-4

SINGLE-CLAD NTR EXPERIMENT Pu-239 LIMITS BASED ON CALCULATED ORGAN DOSES AND ALLOWABLE DOSES

	Calculated Limits, Grams Pu-239			
	Based on Pellet Form		Based on Powder Form	
Organ	Boundary	Stack	Boundary	Stack
Total Body	10.35	1.98	1.09	0.26
Kidneys	104.17	25.64	0.98	0.24
Liver	173.81	42.86	0.31	0.078[a]
Bone	686.50	170.45	0.40	0.10
Lungs	42.37	10.71	0.59	0.15
Thyroid	0.93	0.46[b]	0.23	0.11
Skin	39.53	6.33	19.48	3.69

Note: The gastrointestinal doses were all generally small compared to the limits.

[a] Powder-form capsule (single-clad) limit = 0.078 g (∿5 mCi) Pu-239

[b] Pellet-form capsule (single-clad) limit = 0.46 g (∿30 mCi) Pu-239

limits are shown as both quantity limits and one-day operating power limits in Table 11-5, assuming a 24-hour continuous irradiation at a thermal neutron flux of 1×10^{12} n/cm^2-sec. The operating power is calculated from the quantity limits with the following equation:

$$P = (8.3 \times 10^{10}) (g) (\sigma_f) (\phi)$$

where

P = operating power of experiment (watts)

g = mass of fissile material(grams)

σ_f = thermal fission cross section of fissile material (cm^2)

ϕ = thermal neutron flux (n/cm^2-sec)

For the NTR experiment, the values used are

$U^{235}\ \sigma_f = 572 \times 10^{-24}$ cm^2

$Pu^{239}\ \sigma_f = 742 \times 10^{-24}$ cm^2

$\phi = 1 \times 10^{12}$ n/cm^2-sec

Figure 11-15 gives gram limits for a single clad, powder-form U-235 experiment in the NTR test facilities as a function of the thermal neutron flux and the continuous irradiation time of the experiment. These curves were generated by plotting the calculated limits resulting from 15 different power/time cases run with the RIBD/DOSE77 codes. These curves expand the single-condition limit to a large range of limits, depending on irradiation conditions. Double clad and/or solid-form U-235 limits can be derived from these curves as a direct ratio of the limits given for the specific cases in Table 11-5.

Table 11-5
Pu-239 AND U-235 EXPERIMENT LIMITS

	Pu-239 Limits		U-235 Limits	
Capsule Form	Quantity	24-hour Power	Quantity	24-hour Power
Single-Clad				
Solid Pellet	0.46 g	28 watts	0.60 g	28 watts
Powder	0.078 g	5 watts	0.15 g	7 watts
Double-Clad				
Solid Pellet	4.6 g	280 watts	6.0 g	280 watts
Powder	0.78 g	50 watts	1.5 g	70 watts

Note: Assumes a 24-hour continuous irradiation at a thermal neutron flux of 1×10^{12} n/cm^2-sec.

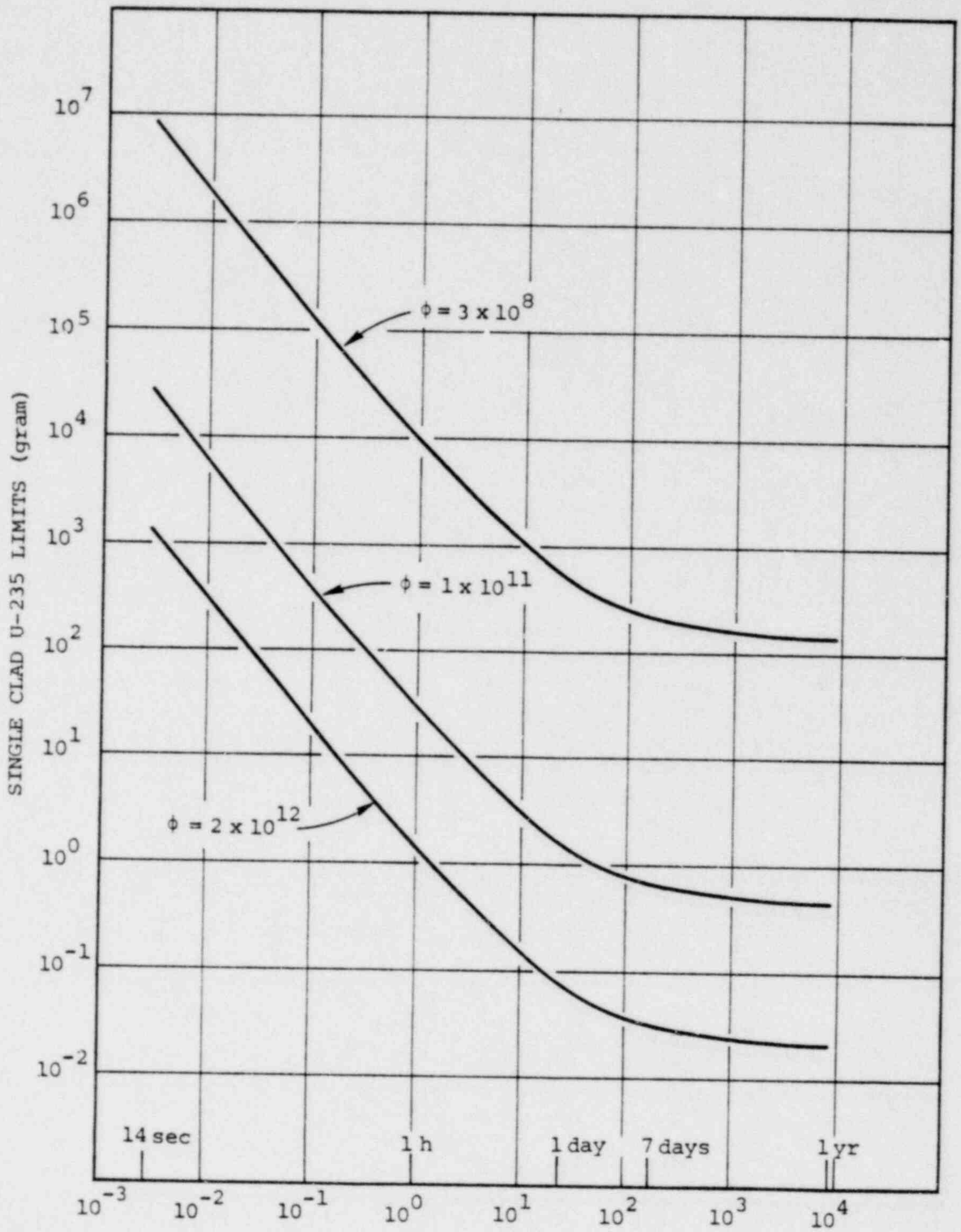


Figure 11-15. NTR Single-Clad, Powder-Form U-235 Experiment Quantity Limits as a Function of Thermal Neutron Flux and Irradiation Time

For example, the limit for a doubly clad, solid pellet of U-235 irradiated in a flux of 1 x 10^{11} n/cm^2-sec for 1 hour would be:

$$(28.6 \text{ grams}) \frac{6.0 \text{ g}}{0.15 \text{ g}} = \underline{\underline{1,144 \text{ grams}}}$$

The 15 conditions and resulting thyroid doses per gram of U-235 for the restricted area exposure are:

Thyroid Dose per Gram U-235 (Rem)

	Irradiation Time				
Flux	10 seconds	1 hour	1 day	7 days	1 year
$3x10^8$	3.196E-7	3.145E-4	5.992E-3	1.352E-2	2.018E-2
$1x10^{11}$	1.066#-4	1.049E-1	1.998E0	4.508E0	6.732E0
$2x10^{12}$	2.132E-3	3.097E0	3.995E-1	9.016E-1	1.355E-2

The U-235 limits for these cases, based on a 3-Rem thyroid dose limit are:

U-235 Limits (Grams)

	Irradiation Time				
Flux	10 seconds	1hour	1 day	7 days	1 year
$3x10^8$	9.385E-6	9.539E-3	5.007E-2	2.219E-2	1.487E-2
$1x10^{11}$	2.814E-4	2.860E-1	1.502E0	6.655E-1	4.458E-1
$2x10^{12}$	1.497E-3	1.431E0	7.509E-2	3.327E-2	2.214E-2

11.6.5 Summary

The NTR experiment design basis accident analysis is based on the single-mode failure of a capsule which contains Pu-239 and/or U-235. Based on the conservatively assumed 5-min restricted area exposure of an on-site employee to the stack concentration of the released isotopes (averaged over 2 hours), the on-site organ dose determines the maximum quantity of Pu-239 and/or U-235 allowable in a capsule. The limits are dependent on the physical form of the fuel and the number of clads used for fuel containment.

The limit for a power-form plutonium experiment is determined by the plutonium release, i.e., greater than 99% of the controlling organ dose (liver) results from the released Pu-239. All other cases (solid Pu or any form of uranium) are limited by the thyroid dose caused by the released iodine isotopes. Therefore, the fueled experiment limits for all but powdered forms of plutonium are flexible with respect to the total quantity of material, but must be controlled with respect to operating power and time based on the iodine inventory.

11.7 REACTOR SAFETY LIMITS

11.7.1 Introduction

Safety limits for operation of the NTR are developed in this section. The safety limits presented also provide the basis for determining and specifying the Limiting Safety System Settings (LSSS) for important process variables.

Safety limits are developed for the reactor power, the only important measurable process variable with safety significance for reactor operation. Other process variables, namely core coolant inlet temperature and reactor primary flow rate, have no significant effect upon the safety criterion over the entire range of core flow conditions, including natural circulation.

In this section, the safety criterion of Departure from Nucleate Boiling is discussed. The critical heat flux relationship and the thermal-hydraulic computer model used in the NTR safety limit analysis are described. The resultant safety limit curves are presented. Instrument uncertainties are applied to the safety limit curves to provide the LSSS for steady-state reactor operation.

11.7.2 Criterion for Development of Safety Limits

Departure from Nucleate Boiling (DNB) has been selected as the most relevant criterion for development of safety limits for operation of the NTR. DNB is that stage of the boiling phenomenon when sufficient liquid is unable to reach the heating surface due to the rate at which vapor is leaving the surface. This restriction of the liquid flow causes an abrupt surface temperature rise above the saturation temperature in a heat-flux controlled situation.

The safety limits for the reactor power are chosen to restrict the actual heat flux in the hottest fuel element coolant passage below the DNB surface heat flux to preclude any subsequent fuel cladding damage due to a rise in surface temperature. The Departure from Nucleate Boiling Ratio, DNBR, is the ratio between the surface heat flux at DNB and at operating conditions; thus

$$\text{DNBR} = \frac{\text{DNB surface heat flux}}{\text{operating surface heat flux}}$$

It was necessary to use two different correlations to evaluate the DNB for the NTR. The steady-state DNB condition is found to occur with saturated bulk boiling in a substantial portion of the core and is accompanied by a significant void fraction. The postulated reactivity transients presented in Subsection 11.4 reach a DNB heat flux with the core coolant significantly subcooled.

11.7.3 Analysis for Development of Safety Limits

11.7.3.1 Steady-State Critical Heat Flux Relationship

The steady-state safety limit analysis required a DNB heat flux correlation which is applicable to low-velocity, low-pressure saturated boiling with a significant void fraction. As cited in Reference 27, Macbeth[28] developed an empirical correlation of experimental data which presents the critical heat flux[29] as a linear function of the mass quality at the hottest surface location. This correlation, which accounts for steam quality, is superior to other correlations which ignore the effect of void fractions and consider only other physical properties of the coolant. The Macbeth correlation states that the critical heat flux is proportional to the mass velocity in the low mass velocity region. The NTR core operates in the low mass velocity region for all operating conditions. The optimized correlation is

$$\text{DNB} = \frac{(H_{fg})\ (G \times 10^{-6})^{0.5}}{(13[illegible])} (1 - X_{max})$$

where

DNB = departure from nucleate boiling critical heat flux (Btu/h-ft^2)

H_{fg} = latent heat of vaporization (Btu/lb)

G = mass velocity (lb/h-ft^2)

X_{max} = maximum quality = $\frac{\text{mass of vapor}}{\text{mass of vapor and mass of liquid}}$

The critical heat flux is calculated for the hottest location of the hottest channel.

11.7.3.2 Transient Critical Heat Flux Relationship

The DNB correlation used to evaluate the reactor safety under transient conditions must be applicable to subcooled boiling. Macbeth[28] developed the following empirical correlation for the DNB critical heat flux under subcooled, low-pressure, low-flow conditions:

$$DNB = (0.247)\,(H_{fg})\left(\frac{\rho v}{\rho \ell}\right)^{0.024}\left(\frac{DG}{L}\right) +$$

$$+\ (0.00213)\,(\rho \ell)^{1/2}\,(H_{fg})^{1/6}\,(10^6)^{1/3}\left(\frac{DG}{L}\right)^{2/3}\Delta Hi$$

where

H_{fg} = latent heat of vaporization (Btu/lb)

ρv = density of the vapor (lb/ft^3)

$\rho \ell$ = density of the liquid (lb/ft^3)

D = hydraulic diameter (inches)

G = mass velocity (lb/h-ft^2)

L = overall length (inches)

ΔHi = enthalpy difference between saturation temperature and channel temperature (Btu/lb).

For a typical hypothesized transient, as presented in Subsection 11.4, the hottest surface location within the NTR core will attain DNB above a heat flux of 600,000 $Btu/h\text{-}ft^2$.

The postulated accidents presented in Subsection 11.4 were analyzed using a DNB heat flux of 450,000 $Btu/h\text{-}ft^2$).

11.7.3.3 Thermal-Hydraulic Computer Model

The computer model CORLOOP[30] was developed for the NTR natural circulation analysis. It is also used for forced convection analysis. CORLOOP includes a multi-channel core model and a circulation loop which includes the core, a heat exchanger, and a pump. The core model is illustrated in Figure 11-16; the circulation loop is illustrated in Figure 11-17. Situations with the secondary coolant flow to the heat exchanger on or off and the primary coolant pump on or off were analyzed using the program. When the primary pump is off, the core is cooled by natural circulation. The core model is adapted to the NTR from the GETR multi-channel core model, CORFLO[31].

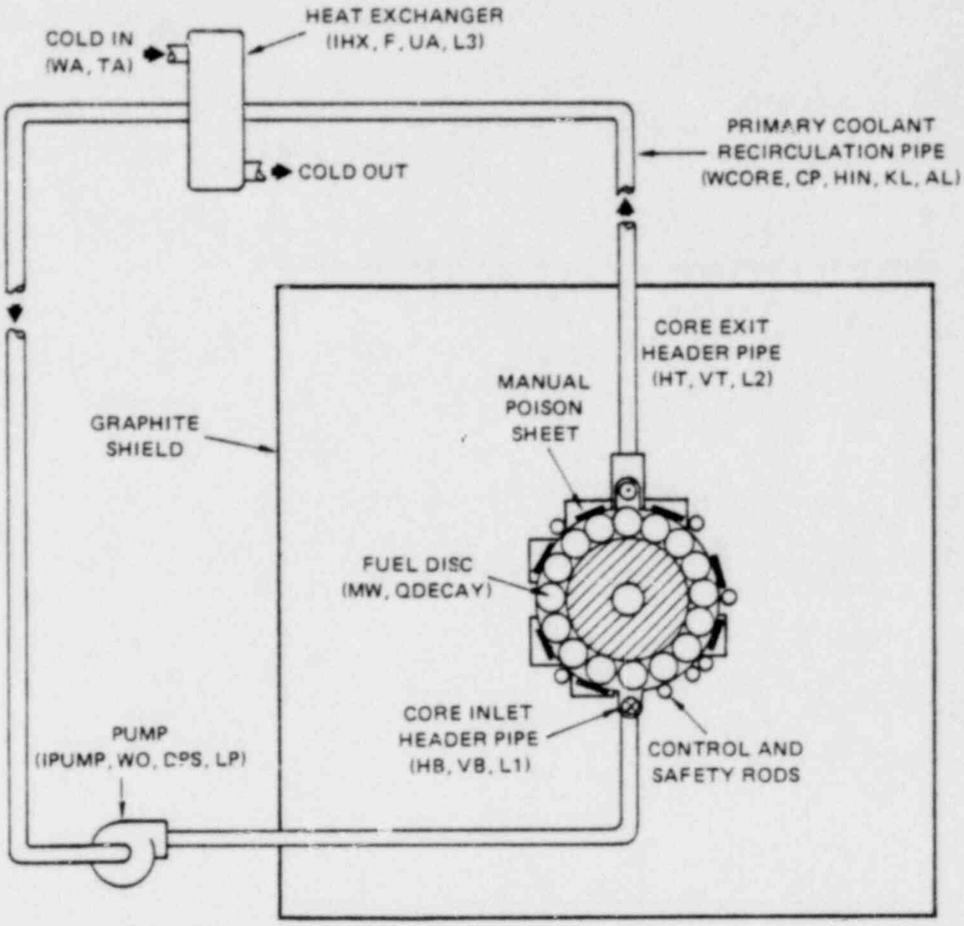
CORLOOP represents the parallel flow channels between the vertical fuel discs in the NTR core as four channels with six nodes per channel. This provides an adequate grid for determining the values for the measurable process variables. The CORLOOP program has been checked against hand calculations for steady-state conditions, and has been verified against actual operating conditions for the NTR.

11.7.4 Safety Limits

The safety limit for the NTR was determined for two different kinds of events. The first analysis considers the steady-state high-power operation of the reactor for various boundary conditions. The second type of analysis considers the behavior of the reactor during various postulated transient events. This second analysis involves the indirect application of the safety limit concept. For the transient analysis, a scram trip point is assumed for important process variables, mainly reactor power, and a value is chosen for the DNB heat flux. After the transient analysis is performed, the integrity of the reactor fuel is evaluated, and the validity of the safety limit is determined.

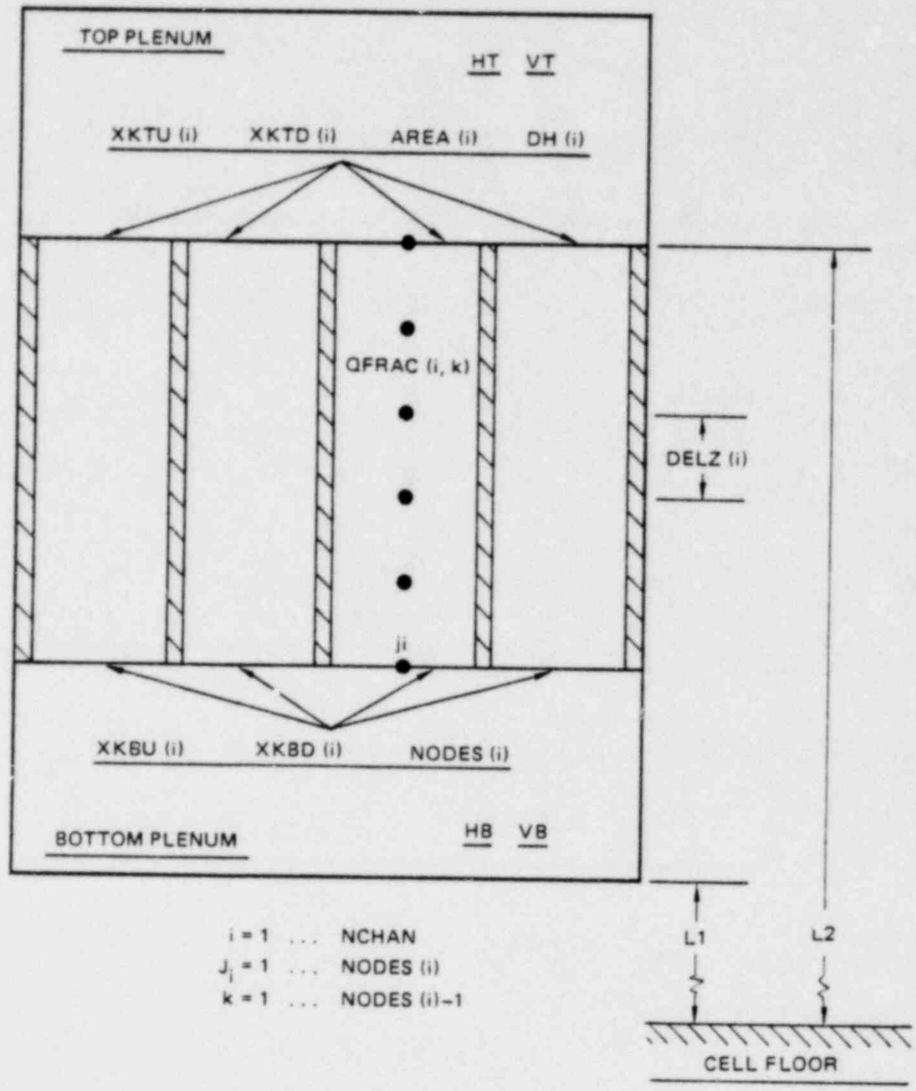
The steady-state safety limits for the NTR were determined using the CORLOOP computer program. The analysis shows that the critical heat flux for the NTR is a strong function of the reactor power. Figure 11-18 shows, as one would expect, that the departure from nucleate boiling ratio approaches unity as the reactor power increases. Figure 11-19 shows the trend of increasing void fraction with increasing reactor power. The analysis shows, however, that the critical heat flux for the NTR is not significantly affected by the core flow rate, or the core inlet temperature (bottom plenum temperature), as shown in Figures 11-20 and 11-21. The reactor power, therefore, is the only important measurable process variable to be limited. The safety limit for the reactor power assures that the actual heat flux never approaches the DNB heat flux.

In the analysis, the actual heat flux has been determined from the CORLOOP computer model. The DNB heat flux has been determined from the Macbeth correlation for pool boiling conditions. At a reactor power of 430 kW, the DNBR reaches unity. At a reactor power of 190 kW, the DNBR = 1.5. As shown in Figure 11-20, a 30% decrease in DNBR corresponds to more than a 100% increase in reactor power. The analytical uncertainties present in the results represent the RMS error of the empirical correlation, the physical differences in the flow conditions between the NTR core and the experimental apparatus used in two-phase flow research, and the assumptions incorporated in the four-channel computer model of the core. A safety limit which corresponds to a minimum allowable value of DNBR = 1.5 provides a conservative and satisfactory margin to more than compensate for any analytical uncertainties. The steady-state safety limit for reactor power is 190 kW, as shown in Figures 11-20 and 11-22.



(For definitions of variables listed above, see Reference 30, CORLOOP Multi-Channel Core and Loop Model by A. I. Yang)

Figure 11-16. Multi-Channel Core Model of NTR (CORLOOP)



(For definition of variables listed above, see Reference 30, CORLOOP Multi-Channel Core and Loop Model, by A. I. Yang.)

Figure 11-17. Schematic Diagram of the NTR Circulation Loop Model (CORLOOP)

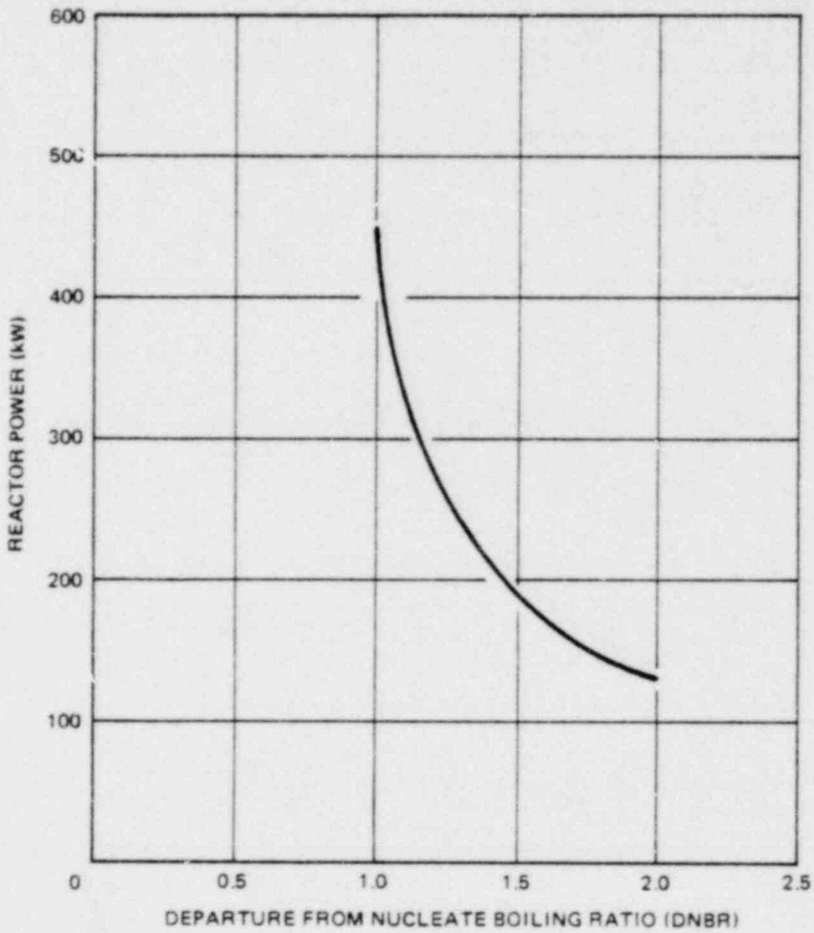


Figure 11-18. Reactor Power Versus DNBR = Departure From Nucleate Boiling Ratio

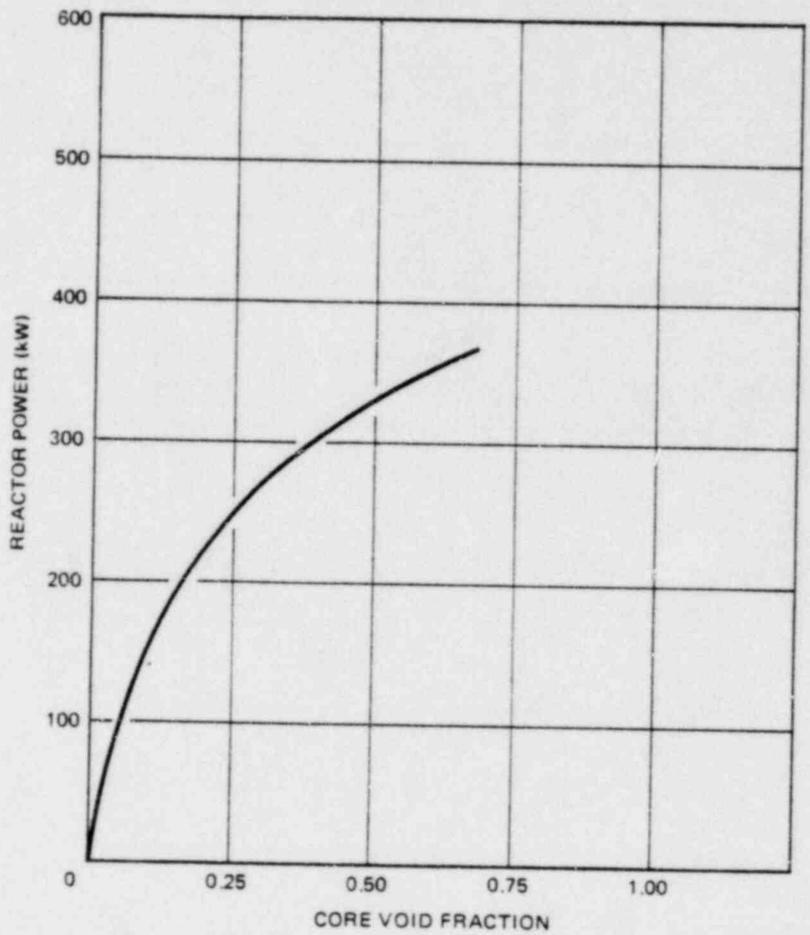


Figure 11-19. Reactor Power Versus Core Void Fraction

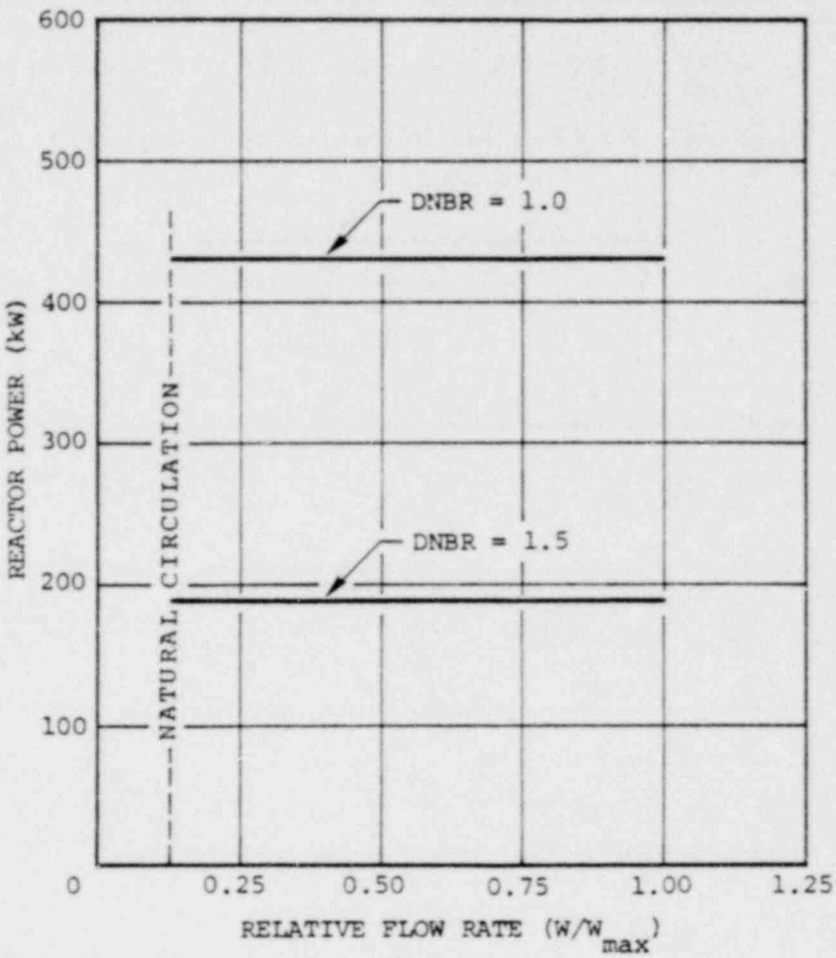


Figure 11-20. Reactor Power Versus Relative Flow Rate

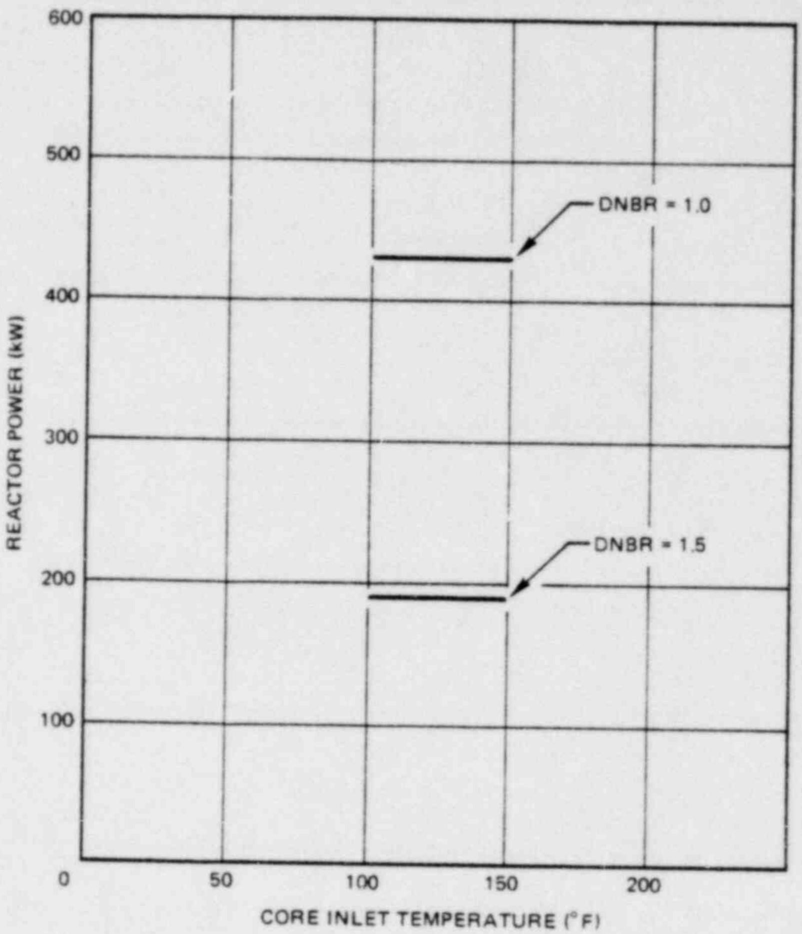


Figure 11-21. Reactor Power Versus Core Inlet Temperature

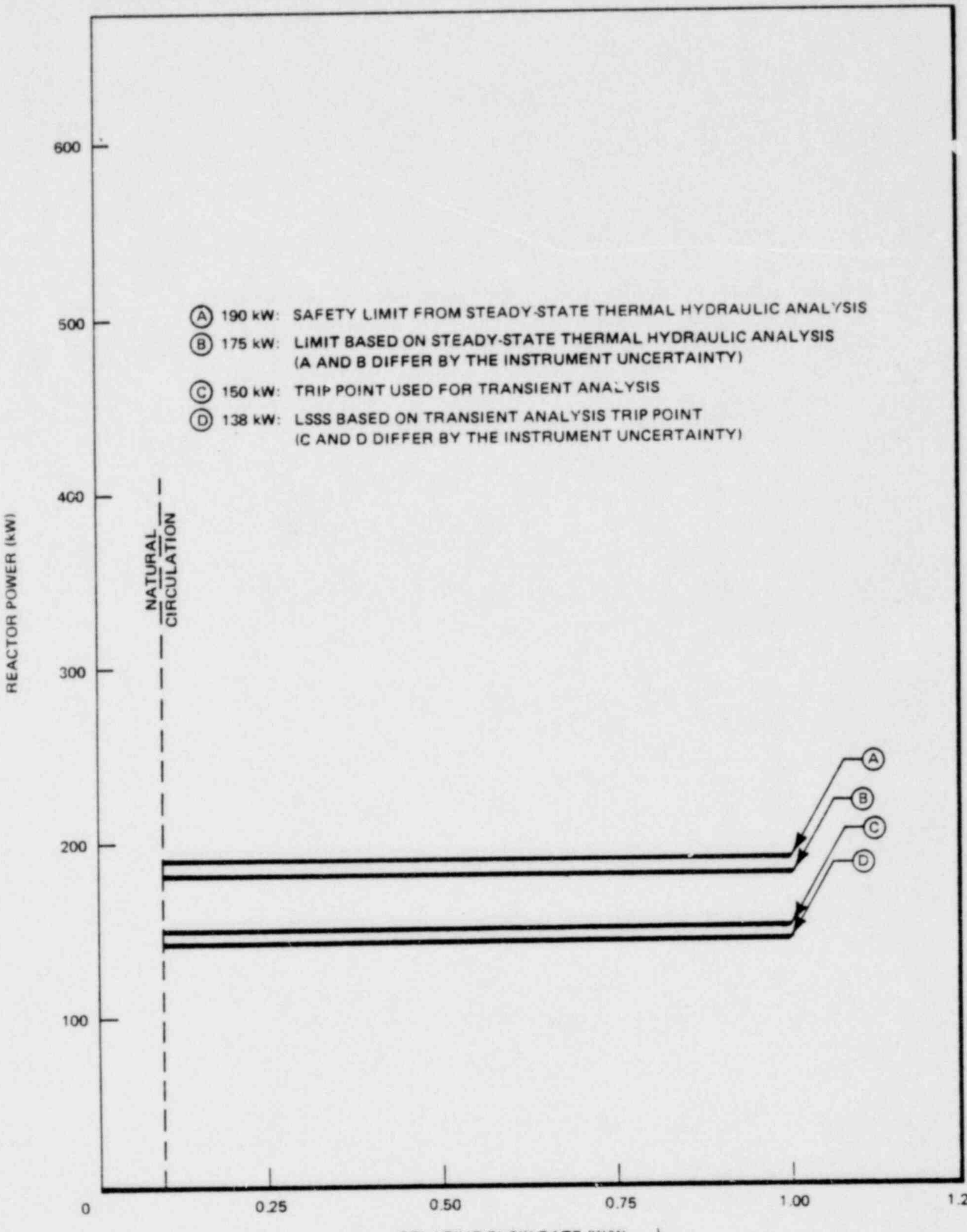


Figure 11-22. LSSS and Safety Limit for Reactor Power in Terms of Relative Core Flow Rate

The curves presented in Figures 11-20 and 11-22 do not extend below a relative flow rate of ~0.12. This flow rate is the value which would exist if the reactor is operated at 190 kW with the pump turned off. Steady-state operation below this flow rate at a power level of 190 kW or greater is not possible. Likewise, the steady-state operation of the reactor with inlet temperatures of less than 100°F, or greater than 150°F, is not possible at these power levels with reasonable secondary coolant inlet temperatures. Values of reactor power, flow rate, and core inlet temperature which fall outside these bounds do not represent steady-state conditions and should be evaluated on the basis of the transient safety limits and analyses.

The transient analysis presented in Subsection 11.4, which required a reactor scram, were all performed assuming a scram occurred at 150 kW, a scram delay time of 0.200 seconds, and a DNB heat flux of 450,000 Btu/h-ft^2. None of the anticipated abnormal occurrences or postulated accidents resulted in fuel damage using these values. The transients were not reevaluated using the more realistic DNB heat flux of 600,000 Btu/h-ft^2 presented in Subsection 11.6.4.2.

. . Instrument Uncertainties

The instrument uncertainties are presented in Table 11-6 for each of the measured variables under consideration. These uncertainties, determined when the process variables were at their normal values and assumed unchanged over all acceptable LSSS, are both the systematic and random types. In general, systematic uncertainties include biases in calibrations, "standards," signal transmitters, and recorders. Random errors include drift of instrument settings, signal-to-noise ratio of instrument electrical output, instrument instability, and operator-to-operator variation in interpretation within least count.

The uncertainty values for the three measurable process variables used in the heat balance for reactor thermal power determination were determined by extracting the square root of the sum of the squares of the individual uncertainties in the contributing measurements.

Table 11-6

UNCERTAINTIES IN THE PRESENT METHODS FOR MEASURING IMPORTANT PROCESS VARIABLES

Reactor Power (Flux Monitoring)		
Compensated Ion Chamber	negligible	
High-Voltage Power Supply	negligible	
Picoammeter Setting Accuracy	±0.25%	
Picoammeter Calibration Accuracy	±4%	
Picoammeter Long-Term Drift	±0.25%	
Net Reactor Power Uncertainty		±4.0%
Reactor Coolant Core Inlet and Outlet Differential Temperature		±3%
Reactor Primary Flow		±1.0%
Overall Instrument Uncertainty for Reactor Power =		±8.0%

11.7.6 Limiting Safety System Settings (LSSS)

The LSSS have been chosen to ensure that reactor scram is initiated in time to prevent exceeding the safety limit for reactor thermal power during normal operation and anticipated abnormal occurrences, or violation of safety criteria during postulated accidents. The safety margin (the difference between the safety limit and the LSSS) includes systematic and random types of instrument uncertainties, and, for transient events, also includes the effect of safety system delay times. The LSSS appears as Curve D in Figure 11-22. The limiting safety system setting for the reactor power is 138 kW over the entire range of core flow conditions, including natural circulation. The value of 138 kW is derived from the trip point of 150 kW used in the transient analysis of the postulated accidents and is used rather than the 190 kW steady-state safety limit because it is more restrictive.

Any quasi-steady event comprising changes in any process variables may be analyzed using the safety limit curves regardless of the rationale used for postulating the event. The most severe anticipated off-normal, quasi-steady event is one in which the reactor power is at its least favorable value of 150 kW. For this highly unlikely operating condition, the DNBR = 1.8.

As a result of certain postulated accidents, the reactor power may exceed the specified safety limit without causing damage to the reactor fuel. The amount by which the safety limit may be exceeded is a time-dependent variable, with each case evaluated individually. Application of the limiting safety system settings for reactor power ensures that no damage to the fuel will occur for any transient resulting from the postulated accidents.

Curve A in Figure 11-22 shows the safety limit based upon the steady-state thermal hydraulics analysis. Curve B shows the safety limit curve adjusted to account for instrument uncertainties. Curve C in Figure 11-22 shows the scram trip point used in the transient analysis. Curve D, the LSSS curve, represents Curve C adjusted to account for instrument uncertainties.

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APPENDIX A
STACK RELEASE LIMITS FOR THE NUCLEAR TEST REACTOR

1. INTRODUCTION

The basis for establishing radioactive release limits from a facility stack is derived from the potential exposure rate in unrestricted areas from ionizing radiation which the released effluent causes. The exposure to radiation can occur from the direct radiation from airborne or deposited material; inhalation and subsequent organ irradiation from the airborne material; and intake of radioactive material and subsequent organ irradiation through the food chain (iodine-grass-cow-milk-consumer).

The release limit of radioactive airborne effluent is determined by first establishing an allowable rate of radiation exposure in unrestricted areas, calculating a representative factor for the dilution and dispersion of the effluent as it is transported from the facility, applying appropriate reduction factors, and then calculating the various isotopic group release rates which would give the established unrestricted area dose rate.

The allowable dose to persons in an unrestricted area is set by the Nuclear Regulatory Commission (NRC) in the Code of Federal Regulations, Title 10, Section 20.105. The dose limits are:

0.5 Rem/year
2 mRem/hour, and
100 mRem/7 consecutive days

where the first limit is an annual whole-body limit and the last two are short-term dose limits. It is further required in 10 CFR, Section 20.106 that the concentrations of radioactive materials released to unrestricted areas do not exceed the limits specified in Appendix B, Table II of 10 CFR 20 unless permitted by license. The concentrations can be averaged over a period not greater than 1 year.

Finally, 10 CFR, Section 20.1(c) requires that licensees make every reasonable effort to maintain exposures and releases of radioactive materials in effluents to unrestricted areas as low as is reasonably achievable.

The NRC can impose additional restrictions on radioactive effluent release rates if the exposure from all sources (air, water, or food) to a suitable sample of the exposed population group, averaged over a period not exceeding 1 year, would otherwise exceed the daily intake resulting from continuous exposure to air or water containing one-third the concentrations specified in Appendix B, Table II, of 10 CFR 20.

The stack release limits are traditionally established for four general categories (i.e., noble gas, halogens, beta-gamma particulate, and alpha particulate). The limits are either based on actual release mixtures or on the most restrictive isotope in each category, whether it is present or not.

The same basic categories will be used for the Nuclear Test Reactor (NTR) stack limits, except that noble gas will be included in an "all other" category to account for short half-life particulates. For the NTR, the limits will be based on the most restrictive isotope for each group (i.e., the isotope that has been assigned the lowest maximum permissible concentration allowable in air for continuous inhalation by the general public). Continuous exposure to 1 Maximum Permissible Concentration (MPC) of an isotope will be roughly equivalent to a whole-body exposure of 500 mRem/year. The MPC values are listed in Table II, Column 1, Appendix B, of 10 CFR 20.

Two release limits for each isotope group will be derived. One is the annual average release limit, the second is the short-term limit. The annual average limit is the rate at which continuous releases for a period of 1 year would not exceed the allowable annual average concentration at the site boundary. The release rate can vary, both high and low, as long as the average does not exceed the limit. However, because of the short-term exposure limits of 2 mRem/h and 100 mRem/7 consecutive days, short-term release limits must be established, while remembering that releases which exceed the annual average limit must be accounted for by an equivalent reduction in releases during the rest of the year.

Both release limits are derived from MPC values (which equate to annual doses of roughly 500 mRem) and calculated dilution-disperion (X/Q) factors. The annual average release limit is calculated from an MPC value which is reduced by appropriate factors to account for other isotopes, other stacks on site, and reconcentration of certain isotopes in food chains. The annual average X/Q value is computer-calculated using 2 years worth of hourly average meteorological data and a revised version of computer code RALOC, which averages the X/Q values for each of 16 compass sectors over the period. The highest resulting average X/Q value is then used in calculating release limits. The short-term release limit is similarly calculated, only using higher unrestricted area concentration values (equivalent to 2 mRem/h) and less reduction factors. The short-term release condition (i.e., in excess of the annual average release limits) must be closely monitored in terms of isotopic releases and meteorological conditions to avoid exceeding fence-post man doses of 2 mRem/h and 100 mRem/week.

2. DERIVATION OF LIMITS

2.1 Allowable Dose to Fence-Post Man

The annual average dose limit to a hypothetical continual resident at the site boundary is 0.5 Rem/year due to all sources from the site. If this hypothetical resident were exposed to an average concentration of an airborne radioisotope equivalent to the MPC value in Appendix B, Table II, of 10 CFR 20 for 1 year, the cumulative theoretical whole-body dose equivalent should be 500 mRem. The specific organ dose due to localized concentration of the isotope may be different from 500 mRem. Mathematical round-off and biological differences would also cause variations in the actual dose.

If a release limit for a single isotope from a single stack were established based on exposure to an average concentration of one MPC at the site boundary, the maximum average release rate could be calculated by dividing the MPC value by the calculated annual average dilution-dispersion coefficient. However, the NTR effluent contains multiple isotopes, albeit in small quantities, and the other operating facilities on site might release mixed fission products and activation product isotopes from their stacks. Therefore, to adequately control the total potential exposure from site stack effluents, the particular isotopic limits for each facility are grouped into categories and reduced by an appropriate factor to account for releases of multiple isotopes from all of the other site-originating sources.

2.2 Annual Average Limits

The annual average dilution-dispersion factor to be used in deriving NTR stack limits is calculated by computer code NUWRALOC (Attachment A-1). The code was run using the actual compiled meteorological history available for the years 1976 and 1977. A wind direction, speed and stability condition for each hour during the year is recorded and used to calculate an hourly X/Q value in one of the 16 equal sectors (22.5°) at the site boundary. After an X/Q value is calculated for each hour of the year, the average value for each sector is computed by summing all X/Q values per sector and dividing by the total number of valid values obtained for all sectors during the year. This gives an annual average X/Q value for each sector, and, of these 16 values, the highest value is chosen as the worst case annual average dilution-dispersion coefficient. The worst case annual average dilution-dispersion coefficient for a ground release from the NTR to the site boundary (based on 1976 and 1977 conditions) is 3.48×10^{-11} (μCi/cc)/(μCi/sec). This value is the annual average X/Q value for the east-southeast sector, which has a nearest boundary distance (as recommended by Regulatory Guide 1.145)[1] of 622 meters from the NTR stack and had wind in that direction approximately 7% of the time, or 1147 hours during the 16,184 hours of the years 1976 and 1977 for which acceptable meteorological data were available (approximately 92% of the 2-yr period).

The worst case isotopes from each category used to establish the release limits are:

Group	Isotope*	Appendix B, Table II, MPC (μCi/ml)
Halogen, >8 day T 1/2	I^{131} (Sol)	1×10^{-10}
β-γ particulate, >8 day T 1/2	Sr^{90} (Sol)	3×10^{-11}
α particulate, >8 day T 1/2	Pu^{239} (Sol)	6×10^{-14}
All others (including noble gas)	Kr^{87}	2×10^{-8}

*The few isotopes having more restrictive MPC values in some of the categories are ignored because of their relatively low abundance around a research reactor.

[1] Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, August 1979.

The following reduction factors are applied to the MPC values for calculating release rates:

Reason	Reduction Factor
Releases from other stacks on site	2
Releases of other isotopes in a different group from the NTR stack	**
Reconcentration of halogens and particulates >8 day T 1/2 through food chain	700

The release limits for the "all other" group of isotopes are calculated as follows:

$$\text{Release limit, } \mu\text{Ci/sec} = \frac{(\text{MPC, } \mu\text{Ci/cc})}{(2)(\text{X/Q, Sec/cc})}$$

$$= \frac{1}{2} \times \frac{1}{3.48 \times 10^{-11}} \times \text{MPC}$$

$$= (1.44 \times 10^{10}) \text{ (MPC)}$$

and the halogen and particulate >8 day T 1/2 limit is divided by 700, or

$$\text{Release limit, } \mu\text{Ci/sec} = \frac{1.44 \times 10^{10} \text{ MPC}}{700}$$

$$= (2.05 \times 10^{7}) \text{ (MPC)}$$

**The annual average releases of each group are ratioed to their respective limits and the sum of these ratios must total unity or less.

Table 2-1

ANNUAL AVERAGE RELEASE LIMITS

Isotope Group	Representative Isotope	MPC (μCi/cc)	Release Limit (μCi/sec)
Halogen, >8 day T 1/2	I-131	1×10^{-10}	2.05×10^{-3}
Particulate, >8 day T 1/2			
Beta-gamma	Sr-90	3×10^{-11}	6.15×10^{-4}
Alpha	PU-239	6×10^{-14}	1.23×10^{-6}
All Other (including noble gas)	Kr-87	2×10^{-8}	288

2.3 Short-Term Release Limits

The short-term release limit shall be established so that the dose received by a fence-postman shall not exceed 2 mRem in 1 hour. The concentration equivalent to this dose rate can be simplistically and conservatively approximated by considering the dose rate which results from exposure to a concentration of an isotope equal to 1 MPC to be 500 mRem/year. Then:

$$\text{2 mRem/h concentration limit} = \frac{\text{2 mRem/h}}{\text{500 mRem/yr}} \times (1 \times \text{MPC})$$

$$\times\ 8760\ \frac{\text{h}}{\text{yr}} = 35 \times \text{MPC}$$

Regulatory Guide 1.111 allows the use of annual average X/Q values for intermittent, short-duration releases if it is established that releases will be random in time. Therefore, the same annual average X/Q value used for establishing the continuous release limit will be used for the short-term release limit:

$$X/Q = 3.48 \times 10^{-11}\ \frac{\mu\text{Ci/cc}}{\mu\text{Ci/sec}}$$

For the short-term limits, the "other stacks" reduction factor will not be applied because of the short duration of the release. Short-term releases lasting longer than several hours will be sampled and isotopically analyzed so that more realistic assessments of the off-site impact can be made considering the actual meteorological conditions during the release.

For the "all other" group, the short term release limit is:

$$\frac{35 \times \text{MPC } \mu\text{Ci/cc}}{(3.48 \times 10^{-11} \frac{\mu\text{Ci/cc}}{\mu\text{Ci/sec}})} = (1.01 \times 10^{12}) \text{ (MPC) } \mu\text{Ci/sec}$$

and the halogen and particulate 8 day T 1/2 short-term release limit is:

$$\frac{(1.01 \times 10^{12}) \text{ (MPC)}}{(700)} = (1.44 \times 10^{9}) \text{ (MPC) } \mu\text{Ci/sec}$$

Table 2-2

SHORT-TERM RELEASE RATE LIMITS*

Isotope Group	MPC (μCi/cc)	Release Limit (μCi/sec)
All other (including noble gas)	2×10^{-8}	20,200
Halogen, >8 day T 1/2	1×10^{-10}	0.144
Particulate, >8 day T 1/2		
Beta-gamma	3×10^{-11}	0.0432
Alpha	6×10^{-14}	8.6×10^{-5}

*Short-term releases exceeding the annual average release limits will be restricted to time periods short enough to avoid off-site doses of 100 mRem/week. In addition, these releases will be carefully monitored and sampled to establish the isotopic mix to avoid releases of multiple, restrictive isotopes during adverse meteorological conditions.

2.4 Summary of Limits

Stack activity release rates shall not exceed the limits listed in Table 2-3.

Table 2-3

STACK ACTIVITY RELEASE RATE LIMITS

Isotope Group	Annual Average Release Rate Limits (µCi/sec)	Short-Term Release Rate Limits (µCi/sec)
Halogen, >8d, $T_{1/2}$	$(2.0 \times 10^7) \times (MPC_a)$	$(1.4 \times 10^9) \times (MPC_a)$
Particulate, >8d, $T_{1/2}$		
Beta-gamma	$(2.0 \times 10^7) \times (MPC_a)$	$(1.4 \times 10^9) \times (MPC_a)$
Alpha	$(2.0 \times 10^7) \times (MPC_a)$	$(1.4 \times 10^9) \times (MPC_a)$
All other (including noble gas)	$(1.4 \times 10^{10}) \times (MPC_a)$	$(1.0 \times 10^{12}) \times (MPC_a)$

Note: MPC_a is the Maximum Permissible Concentration in air allowed in an unrestricted area. These values are listed in Appendix B, Table II, of 10 CFR 20.

Table 2-4

RELEASE LIMITS
BASED ON THE RESTRICTIVE ISOTOPE OF EACH GROUP

Isotope Group	Restrictive Isotope	Annual Average Release Limit (µCi/sec)	Short Term Release Limit (µCi/sec)
Halogen, >8d, $T_{1/2}$	I-131	2.0E-3	1.4E-1
Particulate, >8d, $T_{1/2}$			
Beta-gamma	Sr-90	6.1E-4	4.3E-2
Alpha	Pu-239	1.2E-6	8.6E-5
All other (including noble gas)	Kr-87	288	20,200

The annual average meteorological conditions recorded at VNC (during 1976 and 1977) yield calculated dilution factors from the NTR to the Site boundary (based on a stack flow rate of 3000 cfm) of at least 20,000.

3. COMPLIANCE (SAMPLING-MONITORING)

Compliance with these airborne effluent release limits is demonstrated by a combination of stack sampling and continuous monitoring. The stack releases are maintained As Low As Reasonably Achievable (ALARA) by a bank of absolute (HEPA) filters on the inlet to the ventilation exhaust fan which precedes the exhaust stack. Gaseous and particulate activity which escapes or passes through the filter bank is continuously sampled by an isokinetic probe located on the stack exit. The sample of stack effluent is piped to the monitoring system located in the Building 105 shop area on the first floor of the building.

3.1 Description of System Components

3.1.1 Sample Vacuum

The vacuum necessary for drawing a sample of the stack effluent through the probe, piping, and monitoring system is supplied by the building central vacuum system. At the heart of this system is a 7.5-hp Hoffman blower (~200 cfm capacity). Vacuum ducting is installed throughout Building 105 for use in stack and room-air sampling. Most sample stations, including the NTR stack sampler, have the sample flow rate controlled by a flow regulator. The stack sample flow rate is 1 cfm.

3.1.2 Noble Gas Monitor

A "standard" 16.3-liter Kanne Chamber is used for monitoring the noble gas in the effluent sample. The Kanne Chamber is the last detector in the sample stream. Radioactive gases flowing through the active volume of the chamber ionize the air in the chamber, causing an electron current in a picoammeter proportional to the concentration of the given radioisotope. An average conversion factor suitable for routine releases used with the NTR noble gas monitor is:

$$1 \times 10^{-7} \text{ amperes} = 1\ \mu Ci/cc$$

The output signal is recorded by a two-point recorder. The daily noble gas releases are calculated, tabulated, documented, and submitted monthly for review by the VNC Specialist-Environmental Protection.

3.1.3 Halogen Sample

An activated charcoal cartridge is installed inline between the noble gas monitor and the particulate monitor. This charcoal cartridge absorbs halogens from the sample stream with a known efficiency. The cartridge is changed routinely (normally at weekly intervals) and analyzed for I-131 by the VNC counting lab. The I-131 concentration and integrated release quantity is formally reported monthly to the VNC Specialist-Environmental Protection. The Reactor Supervisor or his designated alternate will be promptly notified if the concentration exceeds a "routine" notification level.

3.1.4 Particulate Monitor

Airborne particulates in the sample line are collected on a filter. The collected beta-gamma-emmiting particulates are continuously monitored by an end window G-M tube, which is mounted in a lead shielding pig. The signal from the G-M tube is displayed on a log count rate meter and a two-point recorder.

The filter is changed routinely (normally at weekly intervals) and analyzed by the VNC counting lab for gross alpha and gross beta activity. The calculated average concentrations and integrated release quantities are formally reported monthly to the VNC Specialist-Environmental Protection. The Reactor Supervisor or his designated alternate will be promptly notified if the concentration exceeds a "routine" notification level.

3.2 Release Reporting

The releases from the NTR are documented and reported, as required, by the VNC Specialist-Environmental Protection.

High-level, short-term releases are monitored during reactor operation using the output of the noble gas and beta-gamma particulate monitors. The alarm levels on these monitors are set conservatively low to warn well before short-term release limits are approached.

Short half-life isotope releases are not distinguished from long half-life isotopes. However, release rate limits and monitoring instrument alarm settings are based on long half-lived isotopes, which are the more restrictive, and the short half-life isotope limits need not be considered unless an event causes release rates in excess of long half-life limits. In this case, quick transport and counting of sample filters would be performed to attempt to take credit for the less restrictive limits. A release of this magnitude (requiring short-term limits) would be a rare, unlikely occurrence based on the nature of the reactor and experiments performed.

The relative counting efficiencies based on the ranges of beta energies normally encountered in both the short and long half-life isotope groups do not vary enough to cause gross errors in the interpretation of the particulate monitor. The beta energies of short half-life daughters of noble gases are generally higher than the beta energy of Sr-90 or the Cl-36 calibration source and therefore are monitored with as high or higher efficiency than the isotope (Sr-90) used for setting the particulate monitor alarm point.

3.3 Alarm Set Points

The beta-gamma particulate monitor and the noble gas monitor are equipped with alarm capabilities. The purpose of these alarms is to alert operations personnel to abnormal releases of radioactive effluents from the stack. The normal operation releases are well below the annual average release limits; therefore, the alarm set points will be conservatively selected at, or slightly below, the annual average release limits for the most restrictive isotopes (Sr-90 and Kr-87). These release levels are orders of magnitude below the short-term release limits.

Table 3-1

Bases

Stack flow rate	3000 cfm
Sample flow rate	1 cfm
Sr-90 MPC_a	3×10^{-11} μCi/cc
Kr-87 MPC_a	2×10^{-8} μCi/cc
H-3 MPC_a	2×10^{-7} μCi/cc
β-γ detector efficiency	4%
Kanne Chamber Calibration Factors	
H-3	6.7×10^{7} μCi/cc/amp
All other (Kr-87)	1×10^{7} μCi/cc/amp

Beta-Gamma Particulate

Annual average release limit = 6.1×10^{-4} μCi/sec (see Table 2-4)

Since the sample flow rate is 1/3000 of the stack flow rate, the activity build-up rate on the sample filter is:

$$\frac{6.1 \times 10^{-4}\ \mu\text{Ci/sec}}{3000} = 2.05 \times 10^{-7}\ \mu\text{Ci/sec}$$

Based on the detector efficiency of 4%, the equivalent rate of rise on the monitor is:

$$(2.05 \times 10^{-7}\ \mu\text{Ci/sec}) \times (2.22 \times 10^{6}\ \text{DPM}/\mu\text{Ci}) \times (0.04\ \text{C/D}) \times (3600\ \text{sec/h}) = 65.5\ \text{cpm/h}$$

The sample filters are changed routinely (normally at weekly intervals); in order to receive an alarm within the sample collection period at the Sr-90 annual average release limit, the alarm point may be set as high as:

65.5 cpm/h x 168 hours = 11,004 cpm

The alarm point is conservatively set $\leq 1 \times 10^4$ cpm.

At this alarm point, a short-term limit release rate of Sr-90 would be detected in:

$$\frac{(1 \times 10^4 \text{ cpm})}{\dfrac{(4.32 \times 10^{-2} \ \mu\text{Ci/sec})}{3000} \times (2.22 \times 10^6 \text{ DPM}/\mu\text{Ci})}$$

$$\times \frac{1}{(0.04 \text{ C/D}) \times (3600 \text{ sec/hr})} = 2.17 \text{ hours}$$

Note: Self-shielding by dust on the stack sample filter has been considered to be a negligible attenuation factor for the beta isotopes which would be present and detected by the GM probe in the monitor.

Noble Gas

Annual average release limit = 288 μCi/sec

At 3000 cfm, this is equivalent to:

$$\frac{(288 \ \mu\text{Ci/sec})\ (60 \text{ sec/min})}{(3000 \text{ ft}^3/\text{min})\ (2.83 \times 10^4 \text{ cc/ft}^3)} = 2.04 \times 10^{-4} \ \mu\text{Ci/cc}$$

The alarm point for Kr-87 is:

$$\frac{2.04 \times 10^{-4} \ \mu\text{Ci/cc}}{1 \times 10^7 \ \dfrac{\mu\text{Ci/cc}}{\text{amp}}} = 2.04 \times 10^{-11} \text{ amps}$$

Since H-3 has a different calibration factor, the H-3 alarm point is determined:

Release limit = $(1.44 \times 10^{10}) \times (2 \times 10^{-7}) = 2.88 \times 10^{3}$ µCi/sec

which is equivalent to:

2.04×10^{-3} µCi/cc (at 3000 cfm)

The alarm point is:

$$\frac{2.04 \times 10^{-3}\ \mu\text{Ci/cc}}{6.7 \times 10^{7}\ \frac{\mu\text{Ci/cc}}{\text{amp}}} = 3.04 \times 10^{-11}\ \text{amps}$$

Therefore, tritium release will be conservatively controlled the by Kr-87 alarm points.

<u>Summary of Stack Monitor Alarm Points</u>*

Beta-gamma particulate alarm	$\leq 1 \times 10^{4}$ cpm
Noble gas alarm	$\leq 2 \times 10^{-11}$ amps

*See Table 3-1 for bases

ATTACHMENT A-1

DISCUSSION OF RADLOC CODE USED FOR CALCULATING THE ANNUAL AVERAGE DILUTION-DISPERSION COEFFICIENT

A modified version of the BWRSD computer code RALOC (Radiological Consequences of a Loss-of-Coolant Accident) code[2] was used to calculate the annual average dilution-dispersion coefficient (X/Q) used for determining the NTR stack release limits. The modified version is called NUWRALOC. It was developed at VNC to better fit the model suggested in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (9/23/77). A total of 16,184 acceptable hourly meteorological conditions from 1976 and 1977 (approximately 92% of the total time) were used as input to the code. The code calculates X/Q values for each hourly period and categorizes them in the appropriate 22½-deg compass sectors around the NTR. The distance to the closest boundary point of each sector from the NTR was scaled from a topographic map of the Site. These distances are part of the input to the computer code which calculates the X/Q values. The input values used, in addition to the hourly meteorological values, are as follows:

1. Distances (in meters) to the 16 sectors nearest boundary points, starting at the north sector and proceeding clockwise: 2302, 2390, 1926, 1615, 955, 622, 522, 519, 515, 587, 756, 636, 622, 634, 749, and 1109.

2. Building 105 cross-sectional area, for building wake effect: $= 281\ m^2$.

3. Stack height above ground level: 0 m.

4. Pasquill-type meteorological condition designations.

5. The sector average X/Q values are used instead of centerline values.

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[2] F. S. Hong, *RALOC Code; Radiological Consequences of a Loss-of-Coolant Accident (LOCA)*, March 1976 (NEDO-21209)

6. No plume depletion was accounted for.

7. Continuous release conditions were used.

The code sums all of the hourly X/Q values, including zero values when the wind blows in another direction, for each sector and divides the sum by the total number of good hours used out of the year to determine an annual average X/Q value for each sector. The largest of the 16 annual average X/Q values is then selected as the worst case for use in calculating stack release limits.

APPENDIX B
EVALUATION OF THE CONSEQUENCES OF ACCIDENTAL EXPLOSIONS

1. INTRODUCTION AND GENERAL CONCLUSIONS

1.1 INTRODUCTION

The use of explosive material within a reactor facility has been recognized as a significant safety concern by the General Electric Company (GE). Careful evaluation of the consequences of accidental detonation of such devices has been performed. In response to a letter of June 23, 1971 from the Nuclear Regulatory Commission (NRC), GE submitted an evaluation of the consequences of accidental explosions and proposed facility operation restrictions and safety controls for review and approval. Following is a summary of that submittal, plus some updated information.

1.2 GENERAL CONCLUSIONS

Because of the many safety features provided and the strong administrative controls applied to the operation of the facility, the possibility of an accident involving explosive material is considered remote. On the basis of the descriptive and analytical information provided, and the proven performance of the facility over an extended operating period, it is concluded the operating and safety methods of the Nuclear Test Reactor (NTR) facility provide the reasonable assurance required by the regulations that the health and safety of the public will not be endangered as a consequence of explosive material handling and inspection at the facility.

2. FACILITY AND PROCESS DESCRIPTION

2.1 GENERAL

The Nuclear Test Reactor (NTR) facility is located at the east end of Building 105. The reactor and its control mechanisms are located within a concrete-shielded room designated as the reactor cell. Operation of the reactor is from a console located in the control room. Figure 2-1 illustrates the floor plan of the NTR and adjoining related facilities. The south cell and north room areas are utilized for neutron radiography exposure facilities. The set-up room is utilized for preparation of material for neutron radiography. Darkroom and office areas are provided to service the facility. The explosive storage magazine is located east of the facility across the access road, behind a dirt embankment. The magazine is utilized as a temporary storage facility for explosive materials which are to be processed at the facility and returned to customers.

2.2 NEUTRON RADIOGRAPHY OF PYROTECHNIC AND EXPLOSIVE DEVICES

Neutron radiography has practical application in the nondestructive quality control inspection of pyrotechnic and explosive devices, particularly for military and aerospace programs. Inspection of pyrotechnic and explosive material is performed on finished or test sample devices. No work is performed which involves loose powder handling, explosive device repair, or modifications. Thermal neutrons are the main component of the reactor flux in the process area. The integrated thermal flux will not exceed 3×10^{12} n/cm^2 and the integrated gamma exposure will not exceed 1×10^4 roentgens. Investigations by Urizar,[1] et al., have shown that these levels should have no effect on explosive materials undergoing neutron radiography.

The exposure of material is controlled by the use of automatic shielding shutters, which close off the access to the reactor beam tube.

[1] M. J. Urizar, E. D. Loghran, and L. C. Smith, A Study of the Effects of Nuclear Radiation on Organic Explosives (TID-12491).

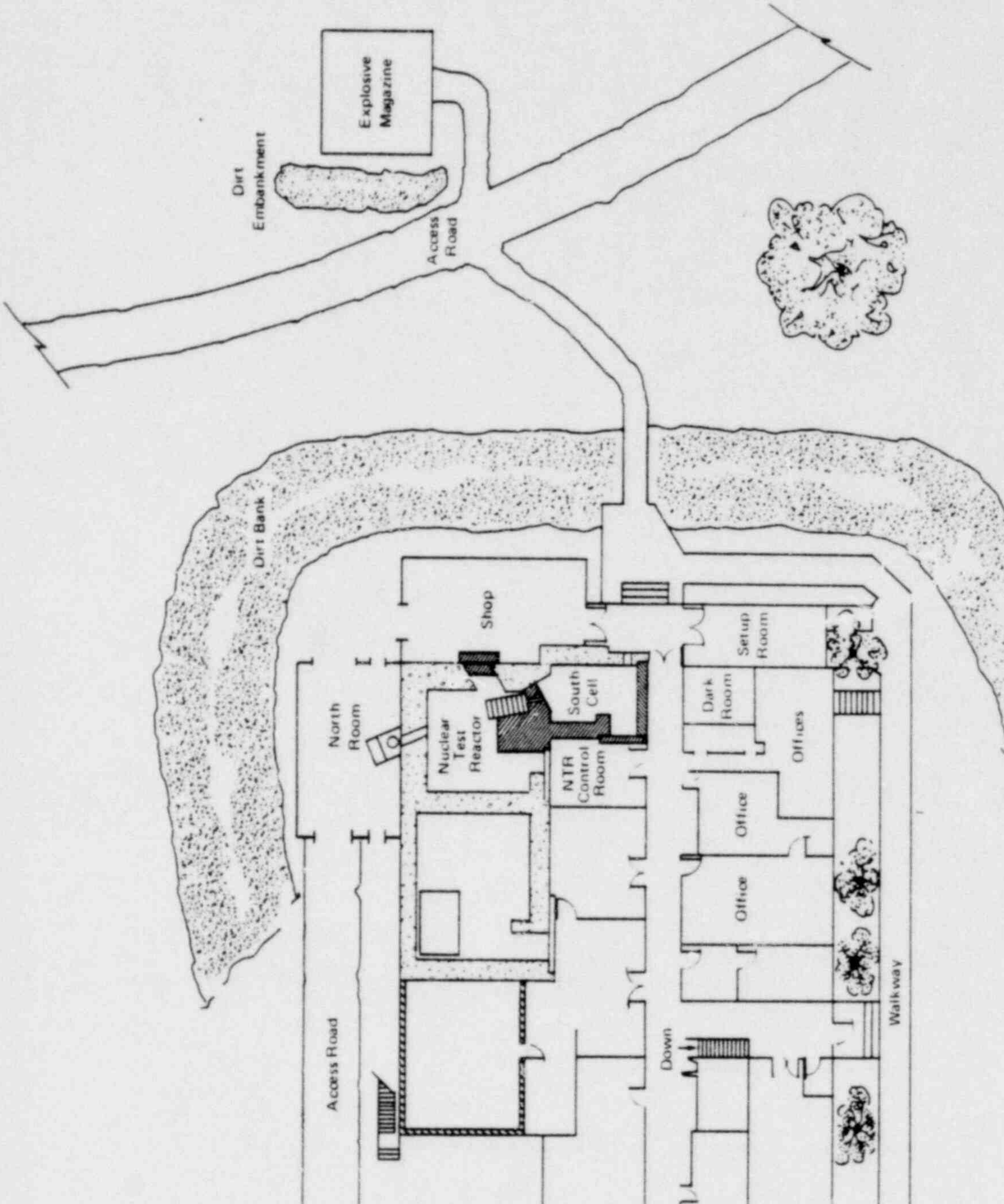

Figure 2-1. Building 105 Floor Plan

2.3 NEUTRON RADIOGRAPHY FACILITIES

The neutron beams emitted from the horizontal facility of the reactor are used for neutron radiographic inspection. The south beam enters the south cell where facilities are available to perform neutron radiographic inspection of explosive material. The north beam enters the North Room where it is also available for neutron radiographic inspection of explosive devices. These two facilities represent the only locations where neutron radiographic inspection of explosive material is performed.

The neutron beam from either the north or south beam ports is generated and shaped, using beam preparation devices which occupy part of the horizontal facility as well as the beam port penetration through the graphite pack and thermal column. The only portion of the beam preparation devices which could affect the core reactivity is that portion which is in the core section of the horizontal facility. The devices are normally constructed of graphite, lead, and polystyrene. The potential reactivity worth is normally very small, but in any case is included in the potential excess reactivity and is controlled accordingly.

2.4 STORAGE FACILITIES

Explosive material received at the facility which is to be stored would normally be stored in the explosive magazine. The magazine facility is located east of Building 105, across the rear access road. A dirt barricade is located between the magazine and the facility. Most material received at the NTR is processed as soon as possible after receipt and therefore would routinely be considered to be work-in progress. See Subsection 2.5.

The magazine is a 5-ft-high by 5-ft-wide by 3-ft-deep plywood box structure constructed of 2-in.-thick plywood. The magazine is located inside a utility shed, which is mounted on a plywood base.

The magazine is rated as Class II and is constructed to the requirements of the State of California Industrial Safety Orders. As a Class II magazine, the maximum explosive amount which may be stored in the facility will be 10 pounds of Class A and B, with a total maximum of 100 pounds, including Class C materials.

The magazine is locked at all times except during the transfer of material, housekeeping, inspection, or when no explosives are present. The magazine key* is under the custody of licensed reactor operators when not in use (normally stored in the security-type safe). A log book is located at the magazine to record all receipts and shipments of explosives and maintains a current record of the amount of explosives located within the magazine. Inventories of magazine contents and housekeeping inspections are made on a routine basis as required. The magazine is located within a restricted area and security personnel check the facility on a routine schedule when explosives are present. Appropriate warning signs are posted around the magazine to prohibit smoking and radio transmissions.

2.5 SET-UP ROOM

The set-up room is located outside the Building 105 east exit door and is primarily a preparation room for neutron radiography. Explosive devices which are scheduled to be inspected are brought to this location from the magazine unless delivered directly by customers or Site transportation services. All explosive material in the set-up room is work-in-progress and the room will not be used as a storage facility for explosives. The total explosive material inventory present in the room (or facility, as appropriate) is recorded for the information of all personnel.

The set-up room contains electrically grounded work benches and the floor is painted with electrically conductive paint. Nonsparking tools are utilized when working near and with explosives. Appropriate restriction and warning signs are posted at all room entrances.

2.6 TRANSPORTATION ROUTES

Four routes of delivery are normally available into the facility. Route No. 1 is a direct delivery to the magazine by means of an access road. Route No. 2 is a direct path delivery from the magazine to the setup room. Route No. 3 is the walkway along the south side of Building 105 to customer parking spaces. Route No. 4 is directly into the north room through the delivery entrance.

*or combination if a combination lock is used.

While explosive material is in process, it will be transported down the hallway between the set-up room and the reactor control room. The material will pass through the south portion of the control room into the south cell for neutron radiography. The south portion of the control room may also be utilized as a staging area for preparing the next neutron radiographic exposure while one is already in process.

Explosive material to be inspected in the north room may be transported through the shop area or along the east walkway around the building to the north room. All routes are premarked, as required, with appropriate warning signs for the material being moved.

2.7 RADIOGRAPHIC PROCESS AREAS

The north room and the south cell are designated as the two process areas for explosive material. The north room is located at the northeast corner of Building 105. It is a sheet-metal building approximately 25 feet by 41 feet by 35 feet high. The neutron beam from the reactor is isolated from the room by a shutter assembly and the Modular Stone Monument (MSM) which is discussed below. When a radiographic exposure is to be made, the shutter control is actuated, the shutter opens, and the neutron beam passes through the 24-in.-diam reactor cell wall penetration (containing additional shielding for beam improvement) to the object in the MSM being processed. The MSM is a dual neutron radiography facility, allowing the capability of performing neutron radiography on unirradiated or irradiated objects (Figures 7-2 and 7-5*). The design involves six concrete blocks that make up the shield and structural unit. A 12-in.-i.d stainless steel pipe capped off the bottom penetrates into the ground beneath the MSM for 20 feet. This penetration allows neutron radiography of long objects to be performed by lowering them into the pipe.

Irradiated objects normally arrive at the NTR in large casks which are placed on top of the MSM using the overhead crane. The objects can then be lowered down into the MSM in front of an imaging foil and the neutron radiography is then performed.

*Section 7, NEDO-12727

Unirradiated objects are moved into a facility on the north end of the MSM, usually by a trolley arrangement. The imaging system is placed behind the object and neutron radiography or irradiation is performed. Appropriate radiation shielding, radiation alarms, air monitors, warning signs, communication systems, and locks are provided in the area for safe operation of the facility.

The south cell is a room approximately 17 feet long by 9 feet wide by 8 feet high which serves as a neutron radiography process area. The cell walls are thick shielding structures to minimize radiation exposure to personnel in the adjoining control room. Objects may be moved into the cell, usually by a trolley arrangement, either from the control room or through a penetration in the south cell east wall. The cell is serviced by radiation alarms, a visual personnel monitoring system and an electric eye entry warning system. The cell is isolated from the reactor neutron beam by a shutter mechanism. When the object has been positioned and the imaging system installed, the shutter mechanism can be opened for the radiographic exposure.

3. EXPLOSIVE MATERIALS DESCRIPTION

3.1 CLASSIFICATION OF EXPLOSIVE MATERIAL

Explosive materials are Class A, Class B, and Class C, as described in Title 49, Parts 172 and 173, of the Code of Federal Regulations, regarding transportation of explosives and other dangerous materials. These classifications are assigned to packaged material and are not normally associated with the unpackaged explosive. Thus, explosive material may be noted as Class C on shipping containers, yet may be considered to be Class A when removed from that container. Explosive material classifications A through C relate to the consequences of initiation and not to initiation sensitivity. As a result, many Class A explosives are safer to handle than Class B or C because of their insensitivity to initiation. Facility handling procedures are based upon the assumption that all explosive material is initiation-sensitive.

3.2 TYPES OF EXPLOSIVE MATERIAL HANDLED

All explosive material handled at the facility is normally in the form of encapsulated powder or pellets. This material is usually installed in a finished manufactured device ready for quality control inspection before use. Sample portions of explosive material and some bulk forms are also inspected. No loose-powder-loaded device modification or detonation testing will be performed at this facility.

Normally, all explosive material will be in the form of squibbs, initiators, bolts, blade charges, delays, and fuse material. The major portion of the devices are manufactured for NASA and military applications.

Explosive material received at the facility is reviewed to establish its characteristics, amount, and related safety precautions concerning the device it is loaded within. Following the safety review, the material is carefully uncrated and each piece inspected to assure all safety devices are installed, as required. Only grounded facilities and nonsparking tools will be used in unpackaging these materials. Strict enforcement of facility explosive handling procedures is mandatory during all explosive material work.

The major portion of the explosive material handled at the facility is insensitive to initiation. Drop tests are performed by customers at their facilities and other safety-related information such as explosive type and amount are a part of the safety review. Some of the explosive devices may be held in the hand when detonated without injury to the holder. The main hazard to personnel in the immediate area is from fine metallic fragments which may become missiles upon detonation of the device. Since strict explosive material handling procedures are enforced at all times, the accidental detonation of a device is highly improbable.

A large portion of the explosive material handled is in the form of fuses or propellants. Should accidental detonation of these devices occur, there would only be smoke and sound emission. This type of material represents a minimal hazard to personnel and the facility.

Personnel handling ordnance devices are provided with special safety equipment, as required. The operation of unshielded, high-frequency generating equipment within 50 feet of any explosive device is prohibited, and radio transmissions are not allowed within 15 feet of explosive material. Appropriate warning signs are posted conspicuously to control transmission, smoking, and other unsafe acts.

3.3 EXPLOSIVE MATERIAL PROCEDURES AND CONTROLS

The Vallecitos Nuclear Center (VNC) maintains written safety standards governing industrial safety controls on explosive material. Detailed procedures are maintained in the area of routine operations, safety, and emergencies at the facility.

Audit functions are performed, as required, by Nuclear Safety and Quality Assurance personnel and the Specialist, Industrial Safety and Fire Prevention. Additionally, facility inspections are made by insurance inspectors and customer Quality Assurance inspectors to ensure a high degree of safety at the facility.

The NTR facility has a 15-year record of safe explosive material handling. During this period there have been no accidents or injuries associated with explosive material. A continuing program of personnel training provides the NTR with highly trained and disciplined personnel.

4. SAFETY ANALYSIS

4.1 DESIGN BASIS ACCIDENTS

To provide safe limits for the amounts of explosives permitted in the NTR handling and radiography areas, separate Design Basis Accidents (DBA) were defined for the south cell, the north room and the set-up room. In general, the DBA assumed a highly improbable accidental detonation of all explosive devices in the particular area and evaluated the consequences in terms of both radiological and mechanical effects.

4.2 RADIOLOGICAL CONSEQUENCES

The radiological consequences of an accidental detonation of explosive devices are essentially nonexistent. Induced activities in explosive materials, structural materials containing the explosive, or structures used in neutron radiography are extremely small, considering thermal neutron fluxes of approximately 2×10^6 nv and normal irradiation times of 10^3 seconds. However, if sufficient other sources of radioactive materials are present in the immediate area and become dispersed or airborne during the accidental detonation, the radiological consequences could be serious. Operations at the NTR include neutron radiography of plutonia fuel pins and capsules containing significant amounts of fission products. Evaluations of the DBA indicate that, while it is virtually impossible to involve these materials in the accident, it is prudent to exclude these large sources of radioactive material from any area in which explosive devices are being handled.

Small amounts of radioactive materials (such as uranium contained in fission chambers and irradiated samples used in various experimental programs) may be safely stored in the south cell or the north room during the neutron radiography of explosives. By limiting these quantities to 10 curies of radioactive materials and 50 grams of uranium, the health and safety of the general public will in no way be compromised. Storage locations are at least 5 feet from any explosive handling position and are normally either concrete block caves or small lead casks. While accidental detonation of explosive devices might cause minor damage to the storage structures, the probability of releasing even a small percentage of the radioactive material from their contents is negligible. Assuming a 1% release and stable atmospheric conditions (inversion), maximum Site boundary doses are less than 20 mRem to the thyroid and 1 mRem to the whole body under this most pessimistic combination of circumstances. No radioactive materials other than those produced by neutron radiography are permitted in the set-up room if explosive devices are present.

4.3 MECHANICAL CONSEQUENCES

The primary safety criterion is that complete simultaneous detonation of all explosive devices in a particular area will not increase the probability or consequences of accidents previously analyzed or create the possibility of a different type accident not previously analyzed. While minor structural damage and possible injury to personnel will occur in the immediate area, damage to the reactor core, the graphite pack, or control systems is not expected and injury to personnel is minimized. Damage to the reactor is prevented by limiting the amount of explosive material allowed in the particular areas (south cell, north room, and setup room) and by design and construction of an additional shield structure (south cell). Potential injury to personnel is minimized by strict adherence to safe explosive handling procedures. The mechanical safety analyses are discussed in detail in Attachment B-1 and show that neutron radiography of explosives can be accomplished safely in the reactor facility by limiting both the total quantity of explosive materials in pounds of equivalent TNT and the distance of the explosive material from sensitive components and structures.

A summary of Attachment B-1 gives the following limits:

1. South cell $W = (D/2)^2$

   where W is the weight of explosive in pounds of equivalent TNT, D is the distance from blast shield in feet, and $W \leq 9$ pounds, $D \geq 3$ feet.

2. North room (without MSM) $W = D^2$

   where W is the weight of explosive in pounds of equivalent TNT, D is the distance from the north room wall in feet, and $W \leq 16$ pounds, $D \geq 1$ foot.

3. North room (with MSM) — W ≤ 2 pounds of equivalent TNT. Since the distance is stationary, there is no value for D (Figure 7-4*).

4. Set-up room — W ≤ 25 pounds of equivalent TNT

4.4 TNT EQUIVALENCE

The equivalence of an explosive material to TNT on a gram basis is determined by ratioing various parameters of the explosive to those of TNT. These parameters include brisance, ballistic mortar, Trauzl test, and detonating velocity, and are described in "Properties of Explosives of Military Interest," AMCP 706-177. This report contains pertinent data on many types of explosives and is used as a primary reference document. The equivalent grams of TNT for an explosive being handled or radiographed is determined by:

$$\text{Gram equivalent TNT} = \text{grams of explosive} \times \frac{\text{parameter of explosive}}{\text{parameter of TNT}}$$

where the ratio of parameters is chosen to be the highest value of the brisance, ballistic mortar, test, or detonating velocity ratios.

If data are not available on the explosive or the composition is proprietary, a factor of 2 is used for the parameter ratio, which is conservative and higher than any value found in AMCP 706-177.

4.5 REACTIVITY EFFECTS

No reactivity effects are directly associated with neutron radiography of explosive or other materials. Objects undergoing inspection are located at relatively large distances from the reactor and have no effect on core reactivity. Even the large shutter in the south cell may be moved during reactor operation without affecting core reactivity. As discussed in Subsection 2.3 of this appendix, some minor reactivity effects are associated with the neutron radiography beam preparation devices. Since these devices are considered to be experiments, their reactivity worth is controlled in accordance with Subsection 7.2, NEDO-12727.

*Section 7, NEDO-12727.

Under normal circumstances, shock waves from accidental detonation of explosives will be attenuated sufficiently to make movement of the beam preparation device highly improbable. It is also noted that removal or expulsion of the beam preparation device has already been taken into account in the transient experiment value which is utilized and has been analyzed in the Safety Analysis Report.

APPENDIX B'

ATTACHMENT B-1

ENGINEERING ANALYSIS FOR NTR NEUTRON RADIOGRAPHY OF EXPLOSIVES

1. INTRODUCTION AND SUMMARY

The following analyses of blast effects due to highly improbable accidental detonation of explosives during NTR neutron radiography operations provide the basis for establishing limits on the quantity of explosives which may be present in the facility. The safety criteria for the various NTR operating areas are as follows:

South cell	No damage to reactor, thermal column, or graphite pack.
North room	No damage to graphite pack, control rod mechanisms, or support structure.
Set-up room	No damage to south cell or reactor cell.

Based on the determination of the critical impulse capacities of the sensitive reactor structures, the maximum amounts of explosives permitted in the NTR facilities are as follows:

South cell	$W = (D/2)^2$; $W \leq 9$ pounds, $D \geq 3$ feet
North room (without MSM)	$W = D^2$; $W \leq 16$ pounds, $D \geq 1$ foot
North room (with MSM)	$W \leq 2$ pounds
Set-up room	$W \leq 2$ pounds

where W is the total weight of explosive in pounds of equivalent TNT and D is the distance in feet from the south cell blast shield or the north room wall.

While the actual quantities of explosives handled and inspected by neutron radiography are normally much less than the safe quantities shown above, these limits will not increase the probability or consequences of accidents previously analyzed for the NTR.

2. IMPULSE CAPACITIES

2.1 GENERAL

To determine safe quantities of explosives permitted in the NTR neutron radiography and explosive handling areas, the impulse capacities of structures related to safety criteria in Section 1 must be analyzed. These structures are the blast shield in the south cell, the control rod support plate in the reactor cell, and the south cell shield wall, the latter of which is shown in Figure 2-1.

The impulsive loading delivered to surfaces exposed to blast fronts is generally expressed in terms of impulse per unit area, i.e.,

$$I' = \frac{1}{A} \int_0^t F dt$$

so that the momentum imparted to a structural component will be the product of the areal impulse density of the r flected blast front and the exposed area of the component, i.e.,

$$I'A = \int_0^t F dt = mv_o ,$$

Since

$$(I'A)^2 = m^2 v_o^2 \rightarrow v_o^2 = 1/2 \left(\frac{I'A}{m}\right)^2 ,$$

the component gains a kinetic energy of

$$K = 1/2\ mv_o^2 = 1/2\ m \left(\frac{I'A}{m}\right)^2 .$$

The resistive forces restraining the affected component must be capable of dissipating this kinetic energy within an envelope which precludes destructive transfer of kinetic energy to adjacent sensitive components.

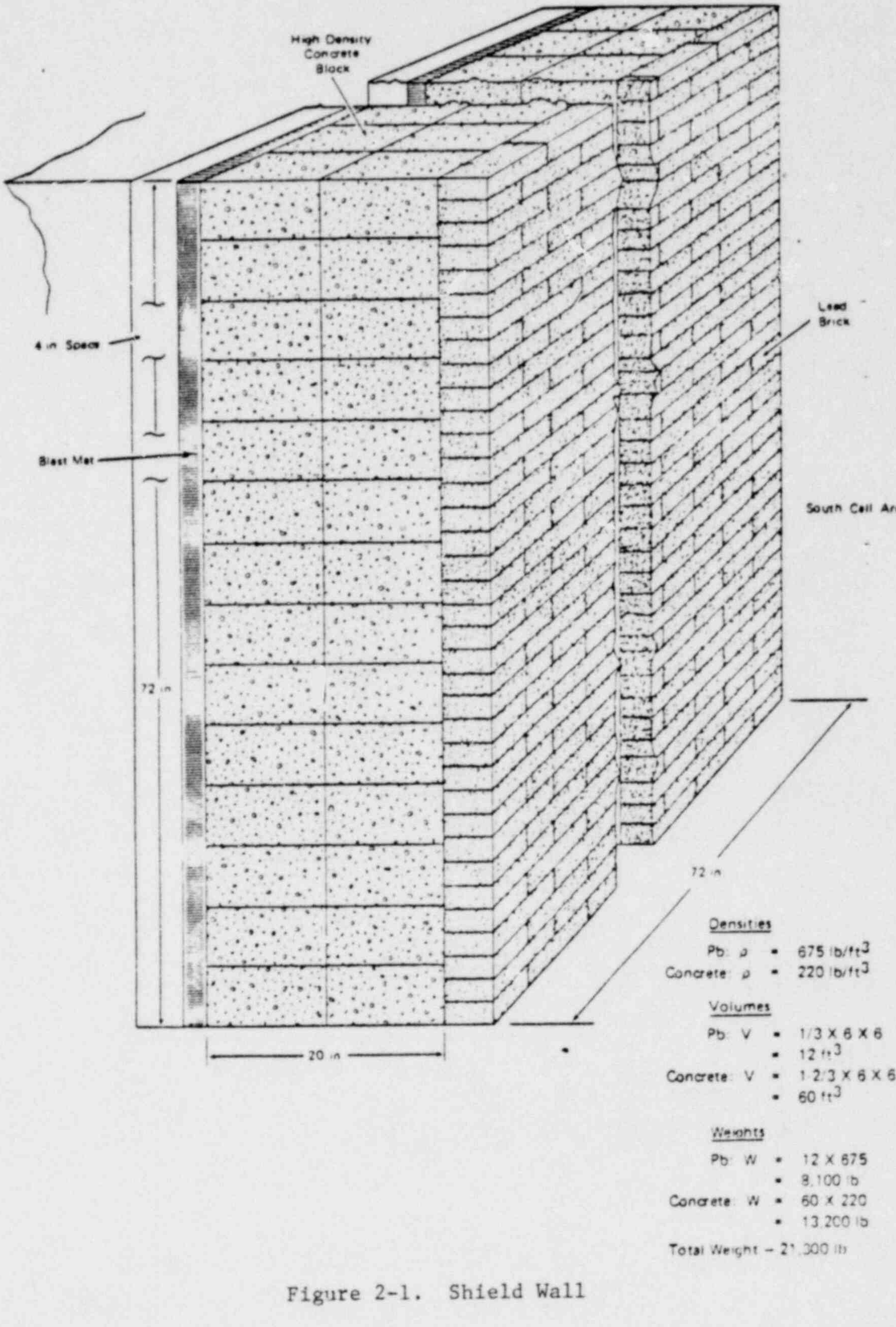

Figure 2-1. Shield Wall

2.2 SOUTH CELL BLAST SHIELDS

To prevent possible damage to the reactor, the thermal column, or the graphite pack, an energy-absorbing shield or blast shield was installed between the biological shield and thermal column in 1967 to allow safe neutron radiography of explosive devices. The kinetic energy imparted to the shield wall will be

$$K = 1/2 \frac{(I'A)^2}{m} = 1/2 \frac{g}{W} (I'A)^2 \ .$$

The blast shield is a 67-in. by 69-in. vertical grid containing 14 1-in. by 1-in. by 3.1875-in. angles in a 3-in. by 2-in. by 0.25-in. angle frame. It is positioned 4 inches from the face of the NTR thermal column and secured in place by 14 0.75-in. anchor bolts in a friction hook arrangement between the frame and concrete walls.

The resultant translation of the wall will be accomplished against a resistive force at least as great as the friction force developed at the wall bottom. (The torque on the blast grid friction hooks ensures the actual resistive forces are much greater). Since the inner face of the blast grid is separated from the reactor face by 4 inches, the work done by the restraint during the 4-in. traversal would be

$$E = F_f \, S = \mu_f \, WS$$

where

F_f = Resistive friction force;

S = Displacement of shield wall;

μ_f = Frictional coefficient; and

W = Weight of shield wall.

Therefore,

$$K \le \mu_f \, WS \text{ or } 1/2 \, \frac{g}{W} \, (I'A)^2 \le \mu_f \, WS \text{ and}$$

$$(I'A)^2 \le \frac{2 \, \mu_f \, W^2 \, S}{g} \text{ or } I' \le \frac{W}{A} \left(\frac{2 \, \mu_f \, S}{g} \right)^{1/2}$$

where A is the shield wall front area.

With

$$W = 21{,}300 \text{ lb, } A = 5{,}184 \text{ in.}^2, \ \mu_f \ge 0.2 \text{ and } S = 4 \text{ in.,}$$

$$I' = \frac{21{,}300}{5{,}184} \left(\frac{2 \times 0.2 \times 4}{32.2 \times 12} \right)^{1/2} \frac{lb_f}{in.^2} = \text{sec}$$

$$I'_{allow} = 0.265 \text{ psi-sec.}$$

2.3 North Room

The two major areas of concern regarding potential reactor damage as a result of blast effects arising from detonation of an explosive charge in the north room are:

1. Possible projection of missiles toward the support plate housing the reactor control rod cluster.

2. Possible blast wave impact loadings on the control rod support plate which might result in sufficient deformation to preclude proper functioning of the control rod mechanisms.

The shock wave arising from an explosion at the neutron radiography station in the north room (without utilizing the MSM) would be propagated through the 24-in.-diam beam penetration in the 5-ft-thick concrete wall separating the reactor cell from the north room. An air control flange covers the beam penetration at the inner face of the reactor cell north wall. This flange is designed with a hinge mounting to allow it to swing open under a significant external overpressure condition, thereby precluding its becoming a missile source.

The problem of determining safe quantities of explosives in the north room then becomes one of restricting charge weight and distance parameters to values such that the propagated shock wave loading on the control rod support structure does not exceed its impulsive load capacity.

The 0.75-in.-thick control rod support plate is taken to be the critical reactor component since its entire frontal area is exposed to the blast wave propagated through the beam penetration and across the reactor cell. Deformation and failure of this support plate would jeopardize the mechanical integrity of the control rods.

Structural damage to the support plate would occur as a result of an impulse loading exceeding the resistance of the support plate. The top, bottom, and sides of the support plate are rigidly supported relative to the center and their inertia is sufficient to preclude their moving significantly relative to the center. For deformation of the support plate center to occur under an impulse loading, the loading must be sufficient to accelerate the central material particles to velocities such that stresses in the membrane exceed the dynamic yield strength for aluminum.

The critical areal impulse density associated with this critical velocity is given by Rinehart[1] as

$$I'_c = \rho \, \delta \, V_c$$

[1] J. S. Rinehart and J. Pearson, Behavior of Metals Under Impulsive Loads, Dover Publications, New York, 1965.

where

ρ = density of the membrane material,

δ = thickness of the membrane, and

V_c = critical velocity.

A stress wave associated with an impulsive load propagates through the material such that

$$\sigma = \rho\ C\ V$$

where

σ = stress;

C = velocity of sound in the material, and

V = relative particle velocity associated with the stress wave.

These two relations may be combined to express the critical impulse in terms of the dynamic yield strength for the material, i.e.,

$$I'_c = \delta\ \sigma_y / C$$

and since C is equivalent to $(E/\ )^{1/2}$, where E is the modulus of elasticity, the resultant critical impulse loading becomes

$$I'_c = (\rho/E)^{1/2}\ \delta\ \sigma_y$$

where

$\rho = 5.22$ slugs/ft^3,

$E = 1.58 \times 10^9$ lb$_f$/ft^2, and

$\sigma_y = 2.02 \times 10^7$ lb$_f$/ft^2

for aluminum, and since δ = 0.75 inch or 0.0574 feet, the critical load becomes

$$I_c = \left(\frac{3.22 \text{ slugs/ft}^3}{1.58 \times 10^9 \dfrac{\text{slug/ft}}{\text{sec}^2\text{-ft}^2}} \right) \left(0.0574 \text{ ft} \times 2.02 \times 10^7 \text{ lb}_f/\text{ft}^2 \right)$$

i.e.,

$$I'_c = 72 \text{ psf-sec or } 0.504 \text{ psi-sec.}$$

Because of uncertainties arising from the fact that the supports other than at the support plate bottom are not truly rigid and the fact that the numerous penetrations of the plate provide opportunities for stress concentrations, an over-all safety factor of 10 will be applied in calculating safe TNT charge weights at the object position in the north room, i.e.,

$$I'_c = 0.0504 \text{ psi-sec.}$$

The following pertains to an accidental explosion where the explosive device is confined in the Modular Stone Monument (MSM). Since the explosive is confined, it is conservatively assumed that all the blast energy is directed through the port. The degree of conservatism in this assumption is difficult to ascertain without a more extensive analysis. It is our judgment that this might be a factor of 2 or 3, but is unlikely to be an order of magnitude greater.

Having assumed all the blast energy is directed through the port, the analysis assumes that the wave follows a spherical divergence. The scaling laws were used to predict the total impulse from normal reflection at the control rod support plate. This was checked by comparison to the curve in Figure 3-1 and gave identical results for small charge weights and slightly more conservative results for larger charge weights.

The allowable loading of the plate was evaluated by treating the plate as a simply supported rectangular plate subjected to a uniform impulse. Since the energy absorption capacity of the plate is small compared to the energy in the blast wave, the plate must reflect essentially all of the blast energy and the impulse is computed based upon this total reflection. The maximum stress (membrane plus bending) and the plate deflection were set at reasonable limits to assure integrity. The resulting allowable specific impulse of 0.041 lb.sec/in.^{2} is within 20% of the allowable selected 0.05 lb.sec/in.^{2}. The approach taken to select this value was used instead of using one-tenth the critical impulse (based upon causing material scabbing).

2.4 SET-UP ROOM

Laing[2] has investigated the response of 24-in. thick reinforced concrete dividing walls separating explosive storage bays. In the case of walls 20 feet long and 12 feet high (approximate dimensions of the 24-in. reinforced concrete separating the south cell from the set-up room), Laing shows that a TNT-equivalent charge weight of 800 pounds detonated at a distance of 3 inches from the wall would bring the wall to incipient failure.

The set-up room is located a minimum of 15 feet from the center of the south wall of the south cell and TNT-equivalent charge weights will be restricted to a total of 25 pounds in the set-up room. In view of these circumstances, no damage to the south cell or the reactor would be expected as the result of an accidental explosion of 25 pounds of TNT in the set-up room.

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[2] E. B. Laing, Design of Ammunition Maintenance Facility, Annals of the New York Academy of Sciences, Vol. 152, 1968, p. 556.

3. SAFE CHARGE WEIGHTS AND DISTANCES

3.1 GENERAL

Considerable data are available on blast parameters for TNT explosions in free air at sea level. Furthermore, the 2W method for determining surface-reflected shock front parameters from free air data is widely accepted as valid. For this analysis, the allowable TNT weight will be taken as one-half the weight whose free air explosive output delivers critical areal impulse densities to the exposed surface of vulnerable components.

By using the data from Figure 3-1 obtained from Ammann and Whitney,[3] and by knowing the value of the critical impulse, the allowable TNT weights and corresponding separation distances are determined.

3.2 SOUTH CELL BLAST SHIELD

For the critical impulse value of 0.265 psi-sec, the following data were determined from Figure 3-1.

Allowable Weight (lb)	Free Air Weight (lb)	$1/W^{1/3}$ (psi-ms/lb$^{1/3}$)	Z (ft/lb$^{1/3}$)	R (ft)
2.5	5.0	155	1.65	2.8
4.0	8.0	132	1.85	3.7
6.4	12.8	113	2.12	5.0
9.0	18.0	101	2.25	5.9

A close approximation to these data takes the form of

$$W = (D/2)^2$$

[3] Ammann and Whitney, Industrial Engineering Study to Establish Safety Design Criteria for Use in Engineering of Explosive Facilities and Operations, a report submitted to Picatinny Arsenal, Dover, New Jersey, 1963.

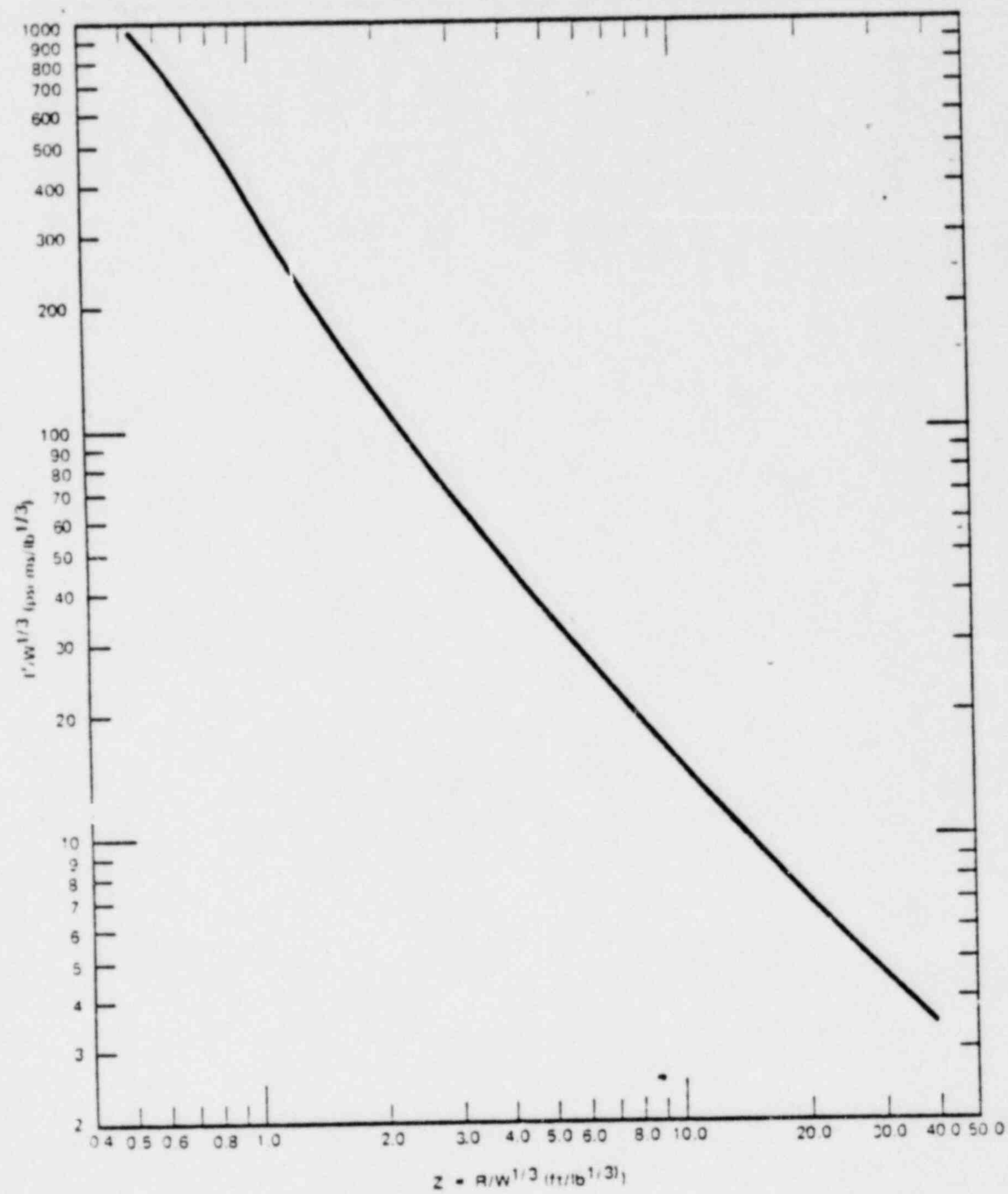


Figure 3-1. Explosive Criteria

and is conservative at all distances. Normal operating procedures for the south cell indicate that the additional limits of D $\geq$3 feet and W $\leq$9 pounds are appropriate.

3.3 NORTH ROOM (WITHOUT MSM)

Although extremely conservative, the areal impulse density of the blast wave emerging from the 24-in.-diam beam port into the reactor cell will be assumed to be attenuated only by an expansion of its frontal area to a value equal to the area of the control rod support plate ($\sim$50 ft^2); i.e., credit will be taken for attenuation of the blast wave areal impulse density by a factor of 16 in undergoing the large, abrupt expansion at the cell north wall and propagation across the 12-ft space separating the wall from the control rod support structure. It should be noted this is a factor of 2 less than the geometric attenuation associated with an explosion occurring at the inner face of the wall.

No credit will be taken for attenuation of the blast wave in its traversal of the 5-ft-long beam port through the wall so that the criteria areal impulse density at the mouth of the north room penetration becomes

$$I_i = 16\ I'_c = 16 \times 0.05 = 0.8 \text{ psi-sec.}$$

Again, using Figure 3-1, the following data were determined:

Allowable Weight (lb)	Free Air Weight (lb)	$I/W^{1/3}$ (psi-ms/lb$^{1/3}$)	Z (ft/lb$^{1/3}$)	R (ft)
1.35	2.7	575	0.75	1.0
4.75	9.5	378	0.96	2.0
0.75	19.5	296	1.1	3.0
17.0	34.0	247	1.23	4.0

A close approximation to these data takes the form of

$$W = D^2$$

and it is conservative for all distances. Normal operating procedures for the north room without the MSM indicate the additional limits of $D \geq 1$ foot and $W \leq 16$ pounds are appropriate.

3.4 NORTH ROOM (WITH MSM)

This analysis summarizes the safe charge weight for a confined explosion in the north room MSM.

3.4.1 Determination of Pressure and Impulse at the Control Rod Support Plate

Using scaling laws where d = 12 ft

$$d_r = \frac{d}{W_S^{1/3}} (2 \times 10^6)^{1/3} = \frac{1510}{W^{1/3}}$$

Duration of positive pressures

$$t_s^+ = \frac{t_{rs}^+ \ W^{1/3}}{(2 \times 10^6)^{1/3}} = \frac{t_{rs}^+ \ W^{1/3}}{126} \qquad \text{Subscript s} \equiv \text{static}$$

$$t_d^+ = t_{rd}^+ \ \frac{W^{1/3}}{126} \qquad \text{Subscript d} \equiv \text{dynamic}$$

As an approximation, the incident impulses associated with the static over pressure and dynamic pressure is

$$I_{so} = \frac{1}{2} p_{so} \ t_s^+$$

$$I_d = \frac{1}{2} p_d \ t_d^+$$

The total impulse during impact (assumes most of the energy reflected)

$$I_t = 2\,(I_{so} + I_d) = p_{so}\,t_s^+ + p_d\,t_d^+$$

3.4.2 Allowable Impulse on Plate

The applied impulse will give velocity and kinetic energy to the plate.
Let V = plate velocity

$$(I_t A) = MV = \rho_m hAV$$

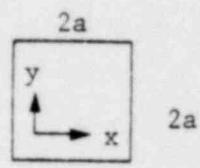
and

$$KE = U_K = \frac{1}{2} MV^2 = \frac{1}{2}\,\frac{(I_t A)^2}{M} = \frac{1\,(I_t A)^2}{2\,\rho_m hA}$$

$$U_K = \frac{I_t^2\;A}{2\,\rho_m h}$$

To determine allowable U_K, assume the deformation shape is

$$u = u_o \cos\frac{\pi x}{2a}\,\cos\frac{\pi y}{2a}$$



u = deflection

If p_e = equivalent static pressure which would cause deflection u

$$U = \iint p_e\,u\,dxdy = 4\int_o^a\!\!\int_o^a p_e\,u_o \cos\frac{\pi x}{2a}\,\cos\frac{\pi y}{2a}$$

$$= 4\,p_e\,u_o\,\frac{a^2}{\pi^2} = 715\,p_e\,u_o \text{ for } a = 42''$$

Table 3-2

SAFE CHARGE WEIGHT

W_c	W_s (lb)	$W^{1/3}$	d_r (ft)	P_{so} (P_{do}) (psi)	P_{refc} (psi)	t^+_{rs} (t^+_{rd}) (sec)	t^+_s (t^+_d) (sec)	I_s (I_d) (lb.sec/in.^{2})	I_t (lb.sec/in.^{2})	I^1 Fig. 2-1 Ref. 1
0.5	1	1	1510	5 (0.4)	11	0.32 (0.40)	0.0025 (0.0032)	0.0063 (0.0006)	0.0138	0.013
1	2	1.26	1200	7.5 (1.0)	17	0.26 (0.36)	0.0026 (0.0036)	0.0097 (0.0018)	0.0230	0.024
2	4	1.59	950	10.5 (3.0)	27	0.23 (0.35)	0.0029 (0.0044)	0.0152 (0.0066)	0.0436	0.033
3	6	1.82	840	14.0 (4.0)	36	0.21 (0.34)	0.0030 (0.0049)	0.0210 (0.0098)	0.0616	0.044
5	10	2.15	700	20 (8.0)	60	0.18 (0.34)	0.0031 (0.0058)	0.0310 (0.0232)	0.1084	0.064

Consider plate as simply supported

Consider bending only

Bending stress at center

$$f_b = 0.221 \, p \frac{a^2}{h^2} (1+\nu)$$

For a = 42 h = 0.75

$$\underline{f_b = 3620 \, p^b}$$

$$U_{bo} = 0.0487 \frac{pu^4 (1-\nu^2)}{E_h{}^3} \qquad E = 10 \times 10^6$$

$$\nu = 0.3$$

$$U_{bo} = 0.52 \, p^b \qquad (f_b = 6950 \, U_{bo})$$

Consider membrane action

$$f_m = 0.396 \left(\frac{Ea^2}{h^2}\right) (p^m)^{2/3}$$

$$U_{mo} = 0.802 \, a \left(\frac{p^{m^3} a}{E_h}\right)^{1/2}$$

$$f_m = 1250 \, p_m{}^{2/3}$$

$$U_{mo}^3 = 0.214 \, p_m \qquad (f_m = 3480 \, U_{mo}^2)$$

For combined bending plus membrane action

$$U_{mo} = U_{bo} = U_o$$

$$p_e = \left(\frac{u_o}{0.52} + \frac{u_o{}^3}{0.213}\right)$$

$$U = 715\ p_e u_o = 715\left(\frac{u_o^{\ 2}}{0.52} + \frac{u_o^{\ 4}}{0.213}\right)$$

Hence

$$\frac{I_t^{\ 2} A}{2\ p_m h} = 715\left(\frac{u_o^{\ 2}}{0.52} + \frac{u_o^{\ 4}}{0.213}\right)$$

$$\frac{A}{2\ p_m h} = \frac{84 \times 84 \times 386}{2 \times 0.1 \times 0.75} = 1.82 \times 10^7$$

$$25400\ I_t^{\ 2} = 1.92\ u_o^{\ 2} + 4.7\ u_o^{\ 4}$$

$$160.5\ I_t = \sqrt{1.93\ u_o^{\ 2} + 4.7\ u_o^{\ 4}}$$

Limit f_m to $f_{yield}/2 = 10000$ psi

$$u_o = \left(\frac{10000}{3480}\right)^{1/2} = \underline{\underline{1.7''}} \qquad \underline{I_t = 0.041}$$

To limit I_t to ~0.041 lb.sec/in.^{2}, W_c must be limited to 2 pounds from Table 3-2. Therefore, the analysis indicates that a safe charge weight of 2 pounds of TNT equivalent could be exposed in the north room MSM.

APPENDIX C
QUALITY ASSURANCE PROGRAM FOR THE NUCLEAR TEST REACTOR

1. INTRODUCTION

A comprehensive quality assurance program is required for modification and maintenance of pertinent plant systems and equipment of the General Electric (GE) Nuclear Test Reactor (NTR). The objective and scope of the program is to maintain managerial and administrative controls over actions relative only to those scram systems and safety-related systems as identified in the Safety Analysis Report, NEDO-12727.

2. ORGANIZATION AND RESPONSIBILITIES

This section describes the organizational structure and functional responsibilities for the quality program (Figure 9.1)*.

2.1 NTR OPERATIONS

NTR Operations is responsible for operation of reactor and experiment systems in accordance with established Standard Operating Procedures. NTR operations is also responsible for those items in Sections 2.2 and 2.7.2.10, as required.

2.2 ENGINEERING

The NTR engineer(s) (reactor supervisor, as appropriate, specialist, etc.) or other personnel, as specified, is responsible for the performance of engineering on the NTR scram systems and safety-related systems. The engineer generates designs and design changes; prepares specifications, work instructions, and procedures; participates in design reviews; specifies which items in the scram systems and safety related systems require quality assurance and the level of quality assurance required; and assures adequate proof of component and systems operability and other related engineering functions as required.

The engineer also evaluates system and structural performance and effects solutions, as appropriate, where operation is found to be inadequate.

2.3 REACTOR ANALYST

The Reactor Analyst is an independent function within the Reactor Irradiations organization. The Reactor Analyst, as required, is responsible for reviewing procedures and implementing instructions to assure compliance with license conditions, technical specifications, standard operating procedures, and applicable regulations; performing audits at NTR to ensure compliance with procedures, specifications, and applicable regulations.

*Section 9, NEDO-12727.

2.4 NUCLEAR SAFETY AND QUALITY ASSURANCE (NS&QA)

Nuclear Safety and Quality Assurance is organizationally independent of the reactor operating functions and has full authority and responsibility to identify, evaluate, and recommend solutions to quality and safety-related problems. NS&QA performs an independent review of selected procedures; establishes overall safety and quality assurance programs, as required; acts as the primary interface between operating components and regulatory agencies; conducts audits, as required, to assure compliance with this plan, specifications, procedures, regulations, and company policy; and performs inspection services and conducts evaluations of nonconforming items, as required. The details of these responsibilities are defined in NS&QA Procedures.

2.5 DRAFTING

The drafting component is responsible for issuance of engineering definition documents, such as drawings and specifications, and changes to such documents.

2.6 PURCHASING

Purchasing performs activities related to procurement of materials and services required from outside vendors in accordance with procedures issued by the Purchasing Function.

2.7 TRANSPORTATION AND MATERIALS DISTRIBUTION

The Nuclear Energy Traffic Operation (NETO) is responsible for receiving, shipping, and on-site movement of materials. The routine handling of materials is in accordance with procedures issued by NETO. Special handling instructions may be specified in implementing instructions issued by the organization requesting the service.

2.8 INSTRUMENT AND ELECTRICAL MAINTENANCE

The Instrument and Electrical Maintenance components perform installation, calibration, repair, and maintenance services on electrical and instrumentation systems for the NTR. Records of work performed and calibration standards traceability, as required, are maintained.

2.9 SHOP OPERATIONS

Shop Operations provides fabrication services as requested by the responsible engineer. This includes primarily machining and welding operations.

2.10 MECHANICAL MAINTENANCE

Facilities Maintenance and Reactor Irradiations Mechanical Maintenance components perform installation, repair, and maintenance services on mechanical systems for the NTR. Records of work performed and calibration standards traceability, as required, are maintained.

3. INSTRUCTIONS, PROCEDURES, AND SPECIFICATIONS

Organizations responsible for work and/or performing work within the scope of this program are responsible for establishment and maintenance of documented systems and procedures for the performance of that work, unless provided for by NTR Operations or determined by NTR operations to be not required. Any changes of these documents are approved by the same function that authorized their issuance and use, unless otherwise specified within the document, or by governing Standard Operating Procedures.

Planning and/or implementing documents shall:

1. Provide, when warranted, space for sign-off by the person who performs the work to show that he has followed the prescribed instructions.

2. Call out essential controls and hold points, as required, which provide an independent assessment that the work was performed as prescribed and that the results meet specifications.

3. Include, as necessary, special instructions for handling and transportation.

4. DESIGN CONTROL

4.1 DESIGN STANDARDS

The responsible engineer identifies in the design drawings and specifications, required codes and standards and practices that provide the basis for design methods, material evaluation and process controls.

4.2 DESIGN VERIFICATION

Design verification is required for new systems or significant changes to existing systems for NTR safety-related items. This is accomplished by independent reviews (normally, NS&QA review is sufficient), alternate calculations, or the execution of a test program. The verification is performed by individuals other than those who performed the original design. The normal method for documentation is the Change Authorization, which is discussed in Subsection 9.2 of the NTR Safety Analysis Report, NEDO-12727.

4.3 ENGINEERING CHANGE

Changes to engineering definition documents are implemented and recorded by means of the Engineering Change Notice (ECN). Field changes during installation, as determined by the responsible engineer, may be implemented by "redlining" the drawing or specification, provided the change is documented on an ECN and the change is evaluated by the same functions that approved the original prior to the operation of the component or system.

5. PROCUREMENT CONTROL

5.1 PROCUREMENT FLOW

Materials are ordered in accordance with the requirements of the engineering definition document, if applicable. Purchasing from outside vendors is performed by Purchasing in accordance with Purchasing procedures. Requests for Quotation (RFQs) and material to be purchased from outside vendors are documented on a Material Request form (MR). NS&QA reviews MRs prior to submittal to Purchasing as required both for procurement or RFQ. Receiving inspection instructions, if required, are included on the requests.

5.2 VENDOR SELECTION AND SURVEILLANCE

Purchasing is responsible for soliciting quotes, negotiation of contracts, and procurement. Vendor evaluation from a technical standpoint is performed by the responsible engineer. Vendor quality capability evaluation, if required, is performed by NS&QA. The quality of purchased materials is verified by supplier-furnished evidence, source inspection, receiving inspection, or a combination of these, as appropriate.

6. DOCUMENT CONTROL

Organizations performing work within the scope of this program generate documents such as Standard Operating Procedures, drawings, specifications, and work instr ctions. Procedures are established describing the document control system in each organization, as required. The document control system assures the proper review, approval, distribution, and control of documents and their revisions.

7. MATERIAL CONTROL

Procedures are established, as required, to control the identification, handling, storage, shipping, cleaning and preservation of safety related material and equipment. The system provides measures to ensure the use of correct materials, to maintain traceability of components, and to clearly identify discrepant materials.

Storage areas are provided, if necessary, to shelter material from natural elements, and to protect material in special environments. Materials held in storage are properly identified, adequately protected to preclude damage, and segregated to prevent the use of incorrect or defective parts.

8. PROCESS CONTROLS

When required by engineering specifications or planning documents, production processes are accomplished under controlled conditions in accordance with applicable codes, standards, specifications, or other engineering criteria using appropriately qualified personnel and procedures.

8.1 PROCESS QUALIFICATION

Qualification of a production process is achieved by performing the process under controlled conditions on samples and then analyzing the output to determine acceptability. When the process can be duplicated on a repetitive basis by holding essential variables constant, and meet the requirements, the process is considered qualified. Qualifications are performed to written instructions based upon engineering specifications and include essential variables.

8.2 PERSONNEL QUALIFICATION

All personnel performing work activities have capabilities commensurate with their assigned functions, a thorough understanding of the operation they perform, the necessary training or experience, and adequate information concerning application of pertinent quality provisions to their respective functions. Supervisors responsible for directing work activities are responsible for assuring that personnel under their direction meet these qualification requirements.

9. INSPECTION

9.1 INSPECTION PLANNING

Inspections are performed to documented and approved plans for each work operation where it is necessary to measure quality. Inspection plans, as required, are incorporated into the detailed work instructions of the performing components.

9.2 INSPECTION REQUIREMENTS

Inspections are performed, as required, to written instructions and the inspection results are documented. When requested, Quality Assurance inspects raw materials, fabricated parts, assembly, and installation to the specifications provided by the requesting organization. For purchased material, the receiving clerk identifies and matches quantities received with the purchase order and then notifies the requesting organization that the material has arrived. The requesting organization is then responsible for making arrangements for receiving inspection, as required.

9.3 HOLD POINTS - APPROVALS

Hold points are stages in the planned activity beyond which work cannot proceed until the preceding work has been evaluated and approved. Hold points are determined by specific job requirements. Hold points and approval requirements for each organization are specified, as required, in the appropriate work instruction or procedure.

10. TEST CONTROL

The responsible engineer identifies the need for development testing and/or for establishing test criteria for items not proven in design standard, mathematical analyses, or in state-of-the-art practices. Tests are aimed toward evaluation of performance capability under various conditions required by the design. Tests are conducted in accordance with written procedures; the test results are documented and evaluated to assure that the test requirements have been satisfied.

11. CONTROL OF MEASURING AND TEST EQUIPMENT

Each component which performs work is responsible for the inventory, identification, and calibration of gages and instruments used for measuring quality parameters as required or as specified by the requesting engineer. Inspection gages and instruments are calibrated, as required, with traceability to certified standards. If no certified national standards exist, the basis for calibration is documented.

12. NONCONFORMANCES

12.1 NONÇONFORMING MATERIAL PROCEDURES

Procedures will be provided, as required for the control of materials or parts as specified by the responsible engineer, which do not conform to requirements, in order to prevent their inadvertent use.

12.2 DISPOSITION OF SCRAP MATERIALS

Disposition of nonconforming materials shall be accomplished after a review by responsible personnel or groups and will consist of acceptance, repair, rework, or rejection.

13. CORRECTIVE ACTION

Documentation of agreed-upon corrective action for conditions adverse to quality are governed by established NS&QA procedures. The procedure assures that corrective action commitments are implemented on a systematic and timely basis.

14. EXPERIMENTAL EQUIPMENT

This program provides, as applicable, controls over the fabrication and installation of experimental equipment to the extent that these relate to reactor safety.

15. RECORDS

Records are retained in accordance with the requirements of NEDO-12727, the Nuclear Test Reactor Safety Analysis Report, Subsection 9.7.

16. AUDITS

NS&QA conducts audits in accordance with established procedures to verify compliance with the various elements of this Quality Assurance program. Audits are conducted on a scheduled or random unscheduled basis, or both, as appropriate.

Appendix D
BASIC RELATIONS USED IN NTR HEAT TRANSFER ANALYSIS

1. LAMINAR HEAT-TRANSFER COEFFICIENT

The Sieder-Tate correlation for laminar heat transfer inside tubes (and channels) was used to compute fuel-surface-to-coolant heat-transfer coefficients.[1] To demonstrate that the flow is laminar at all points in the core, the Reynolds number is calculated:

$$N_{RE} = \frac{\overline{W} D_H}{\mu A_F}$$

where:

$\overline{W}$ = maximum rate in a single channel, lb/h = 1.557 x average flow-rate;
μ = dynamic viscosity of the coolant, lb/ft-h;
D_H = equivalent hydraulic diameter of channel, ft; and
A_F = channel cross-sectional area for flow, ft^2.

For rated operation at 20-gpm total recirculation flow, the average Reynolds number is

$$N_{RE} = 597.$$

The flow in all regions is laminar, because the Reynolds number is much less than 2600, the number at which transition from laminar to turbulent flow occurs.

The Sieder-Tate correlation is

$$\frac{h D_H}{K} = 1.86 \, N_{RE}^{1/3} \, N_{PR}^{1/3} \left[\frac{D_H}{L}\right]^{1/3},$$

where:

h = film heat-transfer coefficient, Btu/h-ft^2-°F;

D_H = channel hydraulic diameter, ft;

N_{RE} = Reynolds number;

N_{PR} = Prandtl number; and

L = flow length of channel, ft (average chord length of disk).

The heat-transfer coefficient computed from this relation is an average value over the disk but is used at all axial positions in the channel. All thermal properties are evaluated at the bulk coolant temperature.

2. FILM BOILING HEAT-TRANSFER COEFFICIENT

In those instances in which the surface heat flux locally rises above the burnout heat flux, it is necessary to recalculate the heat-transfer coefficient,[2] since burnout is the point at which local surface boiling changes to film boiling. The film boiling coefficient is computed from

$$h = 0.724 \left[\frac{\lambda \; g k_v{}^3 \; \rho_v (\rho_\ell \rho_v)}{D_m \mu_v \Delta T} \right]^{1/4} ,$$

where:

- λ = heat of vaporization, Btu/lb;
- g = gravitational constant, ft/h^2;
- K_V = thermal conductivity of vapor, Btu/h-ft-°F;
- ρ_v = density of saturated vapor, lb/ft^3;
- ρ_ℓ = density of saturated liquid, lb/ft^3;
- D_H = hydraulic diameter, ft;
- μ = dynamic viscosity of saturated vapor, lb/h-ft;
- T = temperature difference between surface and coolant, °F.

3. JENS-LOTTES SURFACE BOILING CORRELATION

The fuel disk surface temperature that will exist in the presence of local surface film boiling was predicted from the Jens-Lottes[3] correlation:

$$T_{(Surf)} = T_{(Sat)} + \frac{1.9\ (q/A)^{1/4}}{e^{P/900}},$$

where:

T_{Sat} = coolant saturation temperature, °F;

(q/A) = surface heat flux, Btu/h-ft^2; and

P = system pressure, psia.

When the surface temperature locally rises to the value given by the above expression from fuel internal heat generation and laminar flow heat transfer, local surface boiling will start and thereafter, the surface temperature will be held at that value.

4. BURNOUT HEAT FLUX CORRELATIONS

The burnout correlation that was used to compute burnout heat fluxes[4] is:

$$(q/A)_{crit} = 7000\ (T_{(Sat)} - T_L)\ V^{0.5},$$

where:

T_{Sat} = coolant saturation temperature, °F;

T_L = local coolant temperature, °F; and

V = coolant velocity, ft/sec.

This correlation was proposed to predict low pressure burnout for subcooled water. Its range of application is: pressure, 1 to 11 atm; velocity, 1 to 40 ft/sec; subcooling, 20 to 200 Btu/lb.

The preceding correlation is applied locally and the burnout heat flux varies from channel to channel and along the flow direction in each channel.

The channel velocities encountered in the NTR core are actually smaller than the recommended lower limit of application of the above correlation. However, the margin of conservatism can be examined by referring to existing data correlations on pool boiling. Pool boiling is boiling with no forced flow of the coolant, which would occur if the velocity were dropped to zero. The NTR maximum velocity is only 0.2 ft/sec; therefore, pool boiling data should serve as an indicator of a limit on the burnout heat fluxes.

A recent pool boiling correlation which shows good agreement with available data at all pressures is that of Zuber and Tribus.[5] The correlation is:

$$(q/A)_{crit} = \frac{\pi}{24} L\rho_v \gamma + \frac{2k}{\sqrt{\pi\ \alpha\ \tau}}\ (T_s - T_L)$$

with

$$\pi = \frac{\sqrt{2\pi^3}}{3\gamma} \left[\frac{\sigma}{g\ (\rho_\ell \quad \rho_v)}\right]^{1/2}$$

and

$$\gamma = \left[\frac{\sigma g(\rho_\ell - \rho_v)}{\rho_v^2}\right]^{1/4}$$

where:

g = gravitational constant,
k = thermal conductivity of liquid,
L = latent heat of vaporization,
α = thermal diffusivity of liquid,
σ = surface tension, and
ρ = density.

For a pressure of 14.7 psia (approximately the pressure level of the NTR), the following values of the burnout heat flux have been calculated from the expression:

T_{liquid} (°F)	$(q/A)_{crit}$ ($Btu/f\text{-}ft^2$)
100	1.14×10^6
150	0.78×10^6
200	0.42×10^6
212	0.32×10^6

As indicated by these values when compared to the values computed from the first burnout correlation presented above, the use of the first correlation is conservative.

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AUTHOR	SUBJECT	TIE NUMBER
Irradiation Processing Operations Staff	730, 980	80NED029
		DATE: April 1981
TITLE: Nuclear Test Reactor Safety Analysis Report		GE CLASS: I
		GOVERNMENT CLASS: -
REPRODUCIBLE COPY FILED AT TECHNICAL SUPPORT SERVICES, R&UO, SAN JOSE, CALIFORNIA 95125 (Mail Code 211)		NUMBER OF PAGES: 329

SUMMARY

The General Electric Nuclear Test Reactor is described and a summary of the facility safety evaluation is presented. The description includes the General Electric Nuclear Test Reactor history; the Vallecitos Nuclear Center Site and area characteristics: a detailed facility description; descriptions of Irradiation Facilities, instrumentation and control systems; an facility administration, including the Quality Assurance programs and shielding around the facility. The safety evaluation contains a summary of the analyses performed and the consequences of normal and off-normal conditions, and postulated reactor accident conditions.

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DOCUMENT NUMBER NEDO-12727

INFORMATION PREPARED FOR Nuclear Engineering Division

SECTION Irradiation Processing Operation

BUILDING AND ROOM NUMBER 401-202 MAIL CODE V18

NED-914 (6/77)