9 AUXILIARY SYSTEMS

9.1 HEATING, VENTILATION, AND AIR CONDITIONING SYSTEMS

The control room and office spaces at the NTR share a central HVAC unit with other non-NTR workspaces in Building 105. The Setup Room is also provided HVAC by way of an independent central HVAC unit. There are no safety features or controls associated with these HVAC systems.

Potentially contaminated air and other gases are collected in the reactor cell ventilation system. Ducts draw air from the reactor cell, the ceiling of the south cell, and from the forward position (for neutron radiography of radioactive objects) in the Modular Stone Monument in the north room. Air is then passed through absolute filters before being discharged from the NTR stack. This ventilation system does not have heating or air conditioning capabilities.

The reactor cell ventilation system operates whenever the reactor is operated above 100 watts. For more details refer to Chapter 11.

9.2 HANDLING AND STORAGE OF REACTOR FUEL

The NTR core fuel assemblies were installed in 1957 and have operated satisfactorily since then. There are no unirradiated fuel assemblies or discs stored on site. Refer to Section 4.2.1 for more information about NTR fuel.

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. If it is necessary to remove more than one fuel assembly, special arrangements may be made to use a shielded transfer cask and storage facilities elsewhere on the Site. 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 (3.66 m) high by 5 feet (1.5 m) long by 4 feet (1.2 m) 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 3.5-in.-diameter and one 2-in.-diameter 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.

9.3 FIRE PROTECTION SYSTEMS AND PROGRAMS

The design of the building containing the reactor and the reactor itself makes maximum use of noncombustible structural material. A 500,000-gallon water storage tank on a hillside above the Site provides a gravity flow water supply to the Site, including the fire protection system. A fire alarm system initiates an alarm in building 105 when flow is initiated in the fire header. Designated personnel are trained to extinguish incipient fires.

Equipment, buildings, procedures, etc., are in accordance with company-wide standards, state and local regulations, and the recommendations of insurance agencies.

Six-inch 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 a sprinkler system located within the building. Fire hoses and nozzles are maintained in the Building 105 hallway and the southeast corner of the building. In addition to the fire protection water system, conventional portable fire extinguishers are located throughout the NTR facility and Building 105. The balance of fire protection is according to the VNC sitewide fire protection plan.

The site's radiological emergency plan includes information about offsite organizations that provide emergency support.

Flammable liquids and combustible materials are limited and controlled in all areas. These are regularly checked by the building safety inspections and by insurance carrier audits.

The reactor shall be shut down immediately if a fire occurs in the control room, the south cell or the adjacent hallway or laboratory rooms. In the event of a fire in another part of Building 105, reactor operation may continue if the fire is small and contained.

Due to the control of flammable materials, administrative controls, and facility design, a fire in the NTR would not result in more than a minimal release of radioactive material.

9.4 COMMUNICATION SYSTEMS

The NTR is a small, simple, compact facility. Most work is performed in the south cell which is entered from the control room. Coordination with experimenters is minimal and is accomplished with an intercom between the control room to the north room. Standard telephones in these areas may also be used.

Numerous means of communication available for normal operating, maintenance and emergency situations include: two-way radios (which are limited to areas where radio frequency sensitive pyrotechnic material is not present), a local public address system activated from the control room, [redacted], and a High-level Conference Circuit (HICON), that provides an open line to the security building.

9.5 POSSESSION AND USE OF BYPRODUCT, SOURCE, AND SPECIAL NUCLEAR MATERIAL

The NTR facility license, R-33, applies to Byproduct, Source, and Special Nuclear Material needed for operation of the reactor and its experimental programs. All Byproduct, Source, and Special Nuclear Material used in other laboratory areas of Building 105, and other locations at the VNC, is possessed and used in accordance with other licenses issued by the NRC or the State of California. Material received for irradiation at the NTR is according to license R-33. The R-33 license also authorizes the receipt, possession, and use of activated solids and byproduct materials used for instrument calibrations, startup sources and as may be produced from reactor operations. No changes to possession limits (type or quantity) are being requested coincident with this license renewal.

9.6 COMPRESSED AIR

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 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. A low-pressure switch provides an audible and visual alarm. 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. Conveniently located outlets are provided to supply service air. A loss of compressed air would cause the south cell

door and the south shutter to fail in the as-is position. This failure would have no safety significance.

9.7 RADIOGRAPHY VACUUM SYSTEM

Two vacuum pumps provide vacuum to the neutron radiography areas. One pump provides vacuum to the dark room, south cell, and control room. The second pump provides vacuum to the north room.

The neutron radiography vacuum film cassettes are connected to these vacuum sources so that the radiography film maintains intimate contact with the conversion screen in the cassette for a quality radiography image.

The vacuum system is not used for any reactor system and has no safety significance.

10 EXPERIMENTAL FACILIITIES AND UTILIZATION

10.1 SUMMARY DESCRIPTION

The NTR is used primarily for nondestructive testing of materials and for irradiating various types of materials. The experiment facilities at the NTR are defined as locations for experiments on or against the external surfaces of the main graphite pack and thermal column including the horizontal facility, vertical facility, fuel storage tank, Cable Held Retractable Irradiation System (CHRIS) facility, and the fuel loading chute. These facilities, according to types, are as follows.

10.1.1 Incore Facilities

The Horizontal facility is used for three different types of experiments depending upon its configuration. It can be configured with a source log and pinhole unit and used for neutron radiography (nondestructive testing) or filled with graphite and used for reactivity testing of materials (nondestructive testing), or if completely empty it is used for an irradiation facility.

10.1.2 In-reflector Facility

The vertical facility extends vertically from the top of the graphite pack, through the graphite reflector, tangential to the east side of the fuel container.

Another experiment location is the fuel loading chute which extends diagonally through the graphite pack from the fuel loading tank to the reactor core.

10.1.3 In-reflector and Automatic Transfer Facility

The CHRIS is a dry tube that allows access for a cable-held carrier to an experiment position during reactor operation. The experiment position is a horizontal tube in the graphite pack in line with and parallel to the horizontal facility approximately 6-inches above the top of the reactor core can. The CHRIS is discussed further in section 10.2.4.

10.1.4 Thermal Columns

There are three experimental areas that utilize the radiation coming from the external surfaces of the main graphite pack and thermal column. Two of them, the Top and East Face, are in the reactor cell on the top and to the east side of the main graphite pack. The third, the Thermal Column, is located on the south side of the reflector.

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.

Access to the Thermal Column is from the south cell. Sections of the biological shield may be removed to provide access to the graphite reflector or to permit use of the radiation beams.

10.2 EXPERIMENTAL FACILITIES

10.2.1 The Horizontal Facility

The horizontal facility (Figure 10-1) is a 5-inch-diameter hole traversing the horizontal axis of the reactor. It contains a removable sleeve which makes it a 3-inch-diameter hole for current use. A source log which is 30.5 inches long and 2.96 inches in diameter is centered with the core can in the horizontal facility. The source log is an aluminum pipe containing pieces of graphite, lead and plastic to make the neutron beam more uniform. The thermal column end is stepped to 3.125, 5.0, and 8.0 inches diameter. A pinhole is installed in the 3.125-inch-diameter area for proper focus of the neutron beam for neutron radiography.

Figure 10-1 NTR Neutron Radiographic Facilities (Side View)

The horizontal facility is accessible from either the south cell or reactor cell. The south cell is provided with an air-operated radiation shield shutter that is also used for neutron radiography.

An electrically operated shutter is used at the penetration to the Modular Stone Monument (MSM), in the north wall of the reactor cell. Both shutters have their own timers and can be controlled from either the reactor control console or at the entrances to the south cell or the north room.

The MSM (Figure 10-2) is a dual neutron radiography facility, located in the north room, designed to allow neutron radiography on unirradiated or irradiated objects. The design involves six concrete blocks that make up the shield and structural unit. A 12-inch-ID stainless steel pipe, capped off at the bottom, penetrates 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.

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

Figure 10-2 Modular Stone Monument Neutron Radiography Facility

Irradiated objects normally arrive at the NTR in large casks which are placed on the MSM, using the overhead crane. The objects can be then 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.

Irradiations may be performed anywhere in the horizontal facility or in the radiation beams streaming from the tube ends. The pinhole or the pinhole and source log may be removed to install the sample. 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, set up room 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 irradiation device behavior, the operator will take whatever immediate action is required to ensure the safety of personnel, the reactor, and the irradiation device. The operator 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 shutdown; 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 properly control exposure, temporary shielding and remote handling tools will be utilized. To minimize radiation exposure to the operators, irradiated samples are normally allowed to decay to the extent practical before handling them.

10.2.2 The Vertical Facility

The vertical facility is defined by a 4-inch^2 by 5-foot-long aluminum can, which extends vertically through the graphite reflector tangentially 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 of the top of the facility for connection to equipment in the reactor cell or through cell penetrations to or above the control room, set up room, or north room.

The precautions and procedures during irradiation are the same as those discussed 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 always required for sample unloading, 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.

10.2.3 Fuel 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 irradiation. 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 of 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 normal operating conditions.

10.2.4 CHRIS Facility

The CHRIS is a dry tube that allows access for a cable-held carrier to an experiment position during reactor operation. Entrance to the irradiation system is from the NTR North Room mezzanine. The tube runs through the NTR reactor cell wall, across the reactor cell, and connects to the experiment position.

The peak thermal neutron flux in the experiment position (approximately at the core centerline) is approximately 8×10^{11} n/cm^2-sec.

Samples to be irradiated are placed in a carrier. The carrier is a capped aluminum tube that slides through the irradiation system. One cap is removable for sample loading and unloading. A flexible cable is attached to the carrier to insert it into and retract it from the irradiation system.

At least 18 hours of decay is allowed after irradiation before handling the irradiated portion of the cable.

10.2.5 Top Face and East Face

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-foot by 4-foot removable sections on the top and east faces. The removable section for the east face contains a 20-inch by 18-inch 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 [redacted] plug in the shield. 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.

10.2.6 Thermal Column

The thermal column is a 4-ft cube of high purity graphite located against the south side of the reactor's 5-ft cube graphite reflector. The thermal column is traversed by the horizontal facility.

A 20-inch-wide by 20-inch-high by 4-foot-long centrally located section, made of 4-inch by 4-inch by 4-ft graphite 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-inch boral plate, [redacted]. 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 using the thermal column normally have a negligible effect on core reactivity and may be loaded or unloaded with the reactor operating or shutdown. Radiation exposure to the operator is usually of more concern during such activities than the effects of the irradiation device on the reactor.

10.3 EXPERIMENT REVIEW

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, approximately 60 years of experience in conducting experiments at the NTR, and sound engineering practice. Most of these limits are contained in the Technical Specifications. Adherence to the limits and restrictions below is mandatory and provides assurance that:

- There is no anticipated mode of experiment operation that will endanger the health or safety of the general public or plant personnel.
- No experiment will be performed that involves a technical specification change or that requires prior NRC approval according to 10 CFR 50.59.
- A proposed experiment type will be evaluated in detail and its execution controlled to reduce any radiation exposure to the public and plant personnel to the lowest practicable level.

10.3.1 General Experiment Requirements

10.3.1.1 A written description and analysis of the possible hazards involved for each type of experiment shall be evaluated and approved by the Area Manager or a designated alternate before the experiment may be conducted. Records of such evaluation and approval shall be maintained.

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

10.3.1.3 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.

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

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

10.3.2 Reactivity Limits

10.3.2.1 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 0.50$ and the operation is performed with the knowledge of the licensed operator at the console. All power operated, 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. All manually operated mechanisms for moving an object into the reactor graphite pack shall be done with the knowledge and consent of the reactor operator at the controls of the reactor.

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$.

10.3.2.2 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.

10.3.3 Explosive and Flammable Material Lists

Limits for explosives and flammable materials are developed in Chapter 13 and listed in Section 3.6.3 of the Technical Specifications.

11 RADIATION PROTECTION PROGRAM / WASTE MANAGEMENT

11.1 RADIATION PROTECTION

11.1.1 Radiation Sources

11.1.1.1 Airborne Radioactive Sources

The most significant source of airborne radionuclides during the normal operation of the NTR is the thermal neutron activation of the naturally occurring argon gas in the air and dissolved in the cooling water in and around the reactor core. A small contribution to the total airborne radionuclides can occur from the gaseous fission products emitted from the trace quantities of uranium which may have contaminated the aluminum skin of the reactor fuel elements during fabrication. The radioactive noble gas isotope Ar-41 is the predominant radionuclide emitted from the NTR reactor cell through the exhaust ventilation system.

It can be demonstrated using fission product generation and decay codes that the activity inventory of fission produced halogens and fission produced noble gasses are approximately equal at shutdown (zero decay time). Further it can be shown by the same method that the ratio of the total fission produced noble gasses is approximately 46 times the I-131 activity at shutdown. Therefore, if the measured I-131 release rate is known, and the measured total noble gas release rate is known, the quantity of the total noble gas release due to fission product noble gasses can be estimated. By this process, the fraction of the total noble gas release due to fission produced noble gas turns out to be a very small fraction of the total noble gas release, and therefore it is confirmed that most of the measured noble gas release during normal operation is due to Ar-41. For example:

- During 2018, which is typical of recent years of operation, the total measured release of I-131 from the NTR stack was calculated to be 6.2 mCi.
- The total measured release of noble gas during the same period was about 190 curies.
- A radioisotope inventory of the NTR core calculated with the Radioisotope Buildup and Decay, [1]RIBD, computer code showed that the ratio of fission produced total noble gas activity to the I-131 activity at shutdown was approximately 46:1.
- Applying the 46:1 ration to 2018 data, the total activity of fission produced noble gas released during 2018 was approximately 285.2 mCi (46 times 6.2 mCi), assuming equal fractions of I-131 and noble gasses were released from the core to the exhaust system.

[1] RIBD, Radio Isotope Buildup and Decay Code and Library, RSIC Computer Code Collection, CCC-137

- The difference of 190 curies minus 0.2852 curies is the activity of Ar-41 released during the year.

The other general groups of isotopes which are measured in the stack effluent consist of gross beta particulate and gross alpha particulate. Both releases are equivalently low like the I-131 releases. For example:

The total particulate activities released from the NTR stack during 2018 were measured to be:

Gross Beta Activity = 0.560 μCi

Gross Alpha Activity = 0.021 μCi

The committed effective dose equivalent due to exposure to these releases is demonstrated to be very low.

11.1.1.2 Liquid Radioactive Sources

The only liquid radioactive source at the NTR is the primary coolant. Primary contributors to the radioactivity of the primary coolant are N-16, produced during reactor operation, and activated sodium (Na-24) from the aluminum primary coolant piping, also produced during reactor operation. The primary coolant is sampled periodically to monitor fuel leakage into the primary system. The primary coolant system is vented to a holdup tank prior to startup. The amount vented is small enough that the water in the holdup tank evaporates and the tank does not fill. Dose rate measurements of the holdup tank indicate that no long-lived radioactive material accumulates in the tank. The only liquid radioactive waste generated is a result of sampling - approximately one liter for each sample. This waste is placed in tanks with other laboratory generated liquid radioactive waste and subsequently disposed of in accordance with approved site practices and procedures.

11.1.1.3 Solid Radioactive Sources

Solid Radioactive Sources at the NTR (Table 11-1 and Table 11-2) consist of: the reactor itself during operation; the reactor fuel; a Radium-Beryllium neutron source used for pre-startup instrument checks; an ion exchange demineralizer and filter system for the primary coolant system; spent power reactor fuel rod sections which are received from the hot cell facilities, neutrographed as part of a NDE process, and returned to the hot cell facilities; activated experiments and neutrographed parts and fixtures which are exposed to the neutron beam or irradiated by placement in the horizontal cavity; byproduct material in experiments; instrument check sources; and solid radioactive waste.

Table 11-1 STANDARD, CHECK, AND STARTUP SOURCES AT THE NTR

Source	Isotopic Content	Radiation Type / Energy (MeV)	Activity	Location	Form	Sealed/ Unsealed
1	Cl-36	β^- / 0.709	8 μCi	CAM Cs7A2	Solid	Sealed
2	Cs-137	β^- / 0.514 γ / 0.662	8 μCi	Stack Monitor, Setup Room	Solid	Sealed
3	Cl-36	β^- / 0.709	0.02 Ci	Stack Monitor, Setup Room	Solid	Sealed
4	Radium-Beryllium	Neutron*	0.2 Ci	Graphite Pack	Solid	Sealed
5	Sr-90	β^- / 0.546	0.767 mCi	Control Room	Solid	Sealed

* Neutrons from this source are characterized as 1.0E+06 n/sec.

Table 11-2 FISSILE AND FISSIONABLE MATERIAL AT THE NTR

		Original Enrichment 93.17% (wt.% U-235)		Current Enrichment 91.54% (wt.% U-235)		
		grams U	grams U-235	grams U	grams U-235	grams Pu
NTR Fuel	In-core	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

The quantity of solid waste generated by NTR activities is very small. A more detailed description is provided in Section 11.2.3.

Solid radioactive sources in the form of experiments vary depending of the nature of the material and the experiment to be performed. Other radioactive materials present in the NTR facility are limited by the R-33 license.

Typical radiation levels for occupied or accessible areas of the facilities under license R-33 are discussed in Section 11.1.5. The radiation shielding of the facility is also described in Section 4.3. Based on the last five years of dosimetry for personnel at the facility, the estimated maximum annual dose to a single worker is 862 mRem, and the average dose to all workers is 443 mRem. No single worker at NTR exceeded a total of 2.5 Rem over this period. This is well within the 10 CFR 20.1201 requirements for occupational exposure. Exposure to individual members of the public as a result of NTR operations is essentially zero due to the existence of a restricted area, an access-controlled area, and the distance to the site boundary.

11.1.2 Radiation Protection Program

11.1.2.1 Organization and Minimum Qualifications

The radiation protection program for the site, including the NTR, is the responsibility of the Regulatory Compliance (RC) organization. The Manager, Regulatory Compliance (RC), has overall responsibility for the RP program. The Radiation Safety Officer (RSO) is responsible as the site radiation safety function leader for the ongoing implementation of the program and reports directly to the Manager, RC. The Manager, RC, may alternatively serve as the RSO if applicable radiation protection experience requirements are met.

The Manager, RC, shall hold a bachelor's degree or equivalent and have two years of management experience in assignments involving regulatory activities or a high school diploma and five years supervisory or technical experience in a nuclear, manufacturing, or other technical field.

The RSO shall hold a bachelor's degree in an engineering or scientific subject or equivalent (based on a combination of education and experience) and have more than two years of experience in applied radiation protection. Alternatively, a professional certification in health physics (CHP) may be credited for two years of experience in radiation protection.

Technical personnel within the RC component shall have at least an associate degree or equivalent technical experience in the nuclear industry, and enough professional experience to provide authoritative and competent discharge of assigned responsibilities. Radiation monitoring technicians are trained and qualified in accordance with a comprehensive VNC certification program.

Additional support is provided by GEH subject matter experts in the regulations and standards, quality, and operations organizations.

The staffing level for the RC organization is dependent on the level of activity at the site. Staffing for the RC activities for the NTR includes those necessary to perform health physics monitoring and nuclear safety oversight.

11.1.2.2 RP Program Implementation

The RP program for the NTR is a subset of the broader VNC site-wide RP program and is implemented by procedures that ensure compliance with all applicable federal and California regulations. Procedures implement the use of Radiation Work Permits to ensure safe, authorized

work in restricted areas. RWPs provide information and instruction to the worker and prescribe necessary precautions and protective equipment when performing tasks in those areas.

The following is an inexhaustive list of program areas covered by implementing procedures:

- Organization and administration
- Training and qualifications
- Occupational dose and controls
- External exposure monitoring
- Internal exposure monitoring
- Records and reports
- ALARA program
- Radiological work management
- Radiological surveys
- RP instruments
- Radioactive material control
- Access control and radiological posting
- Respiratory protection
- Declared Pregnant women and prenatal radiation protection
- Radiation protection of members of the public
- Radwaste processing, packaging, shipping, and storage

The use of Change Authorizations and Engineering Releases for NTR activities provide documentation of changes and work and includes the determination whether the proposed change requires prior NRC approval pursuant to 10 CFR Part 50.59.

All procedures at VNC are administratively controlled to ensure procedures and changes that impact the radiation protection program are reviewed for adequacy, approved by authorized personnel, and distributed to the applicable staff.

11.1.2.3 RP Training

Radiation safety training is defined by procedure, managed by the RSO, and implemented by the responsible managers. Procedures describe radiation safety courses as well as how they are developed and maintained, delineate responsibility for ensure training is performed, and ensure training is documented.

Radiation Monitors are trained and certified in accordance with a comprehensive Health Physics program that covers all site operations, including those at the NTR. NTR personnel receive radiological safety training per the Reactor Operator Initial and Requalification programs. In addition, they receive annual radiation safety refresher training for site radiation workers and personnel assigned to site response teams.

11.1.2.4 RP Program Oversight

The Vallecitos Technological Safety Council (VTSC) is the review and audit committee associated with the activities of the site as a whole and the NTR (see Ch 12).

The radiation protection program is reviewed each year pursuant to 10 CFR 20.1101(c), in a report to the site manager. The VTSC reviews the report annually and determines the effectiveness of the program. The VTSC also receives and reviews incident investigation reports and countable event reports and uses all the information to implement program improvement and to ensure root causes are determined and effective corrective action is taken.

11.1.2.5 RP Program Records

Radiation program safety records are generated per procedures. Procedures direct record retention requirements and reviews of those records. Records are required to be stored in a safe location and be easily retrievable. Radiation program safety records are used to develop trends, inform management, develop ALARA actions, and reporting to regulatory agencies. Radiation safety records include, employee exposure records, visitor and contractor exposure records, training records, medical records, radiation monitoring records, RWP's, VTSC meeting minutes, and the site decommissioning file.

11.1.3 ALARA Program

The radiation program at NTR includes a commitment to maintain radiation exposure as low as reasonably achievable, ALARA. The VNC site manager, who has overall responsibility for the NTR license, and the Manager, RC, who has responsibility for radiation safety, are fully committed to the ALARA principle and oversee its effective implementation across the site. The Manager, NTR, is fully committed to the ALARA principle and ensures its proper procedural implementation specific to the NTR.

During the first quarter of each calendar year, the Manager, NTR, reviews the NTR ALARA program and prior year accomplishments and establish goals for the current year. All licensed NTR operators review each radiation exposure report. The Manager, NTR, also reviews radiation

exposure records for all NTR personnel. Any unusual exposure is discussed, and a probable cause determined. Individual workers, both NTR, and non-NTR personnel, are adequately trained for the job and periodically reminded of ALARA principles. ALARA is considered when facility changes are made, via the Change Authorization (CA) process, when new or changed experiments are reviewed, and when major maintenance is planned.

11.1.4 Radiation Monitoring and Surveying

Radiation monitoring and surveying include the following:

- Monitoring and Surveying routines performed by a Radiation Monitoring Technician (RMT)
- Special monitoring and surveying by RMT
- Fixed air sampling system
- Stack Monitoring system
- Continuous air monitor (CAM)
- Remote area monitors (RAMs)
- Smear surveys documented by NTR operations personnel
- Personal dosimeters
- Sampling and counting of Industrial Wastewater prior to release.

Radiation monitoring equipment at the NTR includes the following:

Table 11-3 RADIATION MONITORING EQUIPMENT AT THE NTR

Description	Location	Function
Portable survey instruments	Control Room North Room Setup Room Hallway outside Control Room	Dose rate (radiation fields); β, γ, n Contamination; α, β, γ
Continuous Air Monitor (CAM)	Control Room	Reactor Cell Air monitor
Hand and Foot Counter	Hallway near building exit	Contamination detector
Remote Area Monitor (RAM)*	**Readout:** Reactor panel, control room **Detectors:** North Room (adj to CHRIS) South Cell Reactor Cell Control Room North Room (MSM)	Radiation field detection alarm
Fixed Air Filters	Control Room South Cell Reactor Cell (2) North Room (3)	Airborne contamination detection
Stack Monitor System	Setup Room	Stack particulate and noble gas monitoring alarm
Personal Dosimetry	NTR Personnel	Exposure monitoring

*The RAM system is installed to ensure protection of personnel by ensuring exposures are maintained ALARA.

Records of monitoring and surveys conducted by RMTs are maintained, reviewed, and archived, and undergo independent review by RC personnel. Dosimetry and fixed air filter records are also maintained, reviewed and archived by RC personnel. Stack monitoring records and CAM records are maintained and archived by NTR personnel and reviewed by RC personnel.

Procedures are maintained to ensure the proper calibration of radiation protection instruments. Calibration of radiation protection instruments used at the NTR is required upon initial acquisition, after major maintenance, and at least annually. Radiation monitoring instruments are calibrated on-site by VNC Instrument Maintenance or by authorized offsite vendors. Calibration sources used for calibration are traceable to NIST standards. Radiation monitoring instruments are controlled, and timely calibration is assured by RC personnel.

Monitoring and Surveying routines by a VNC RMT for the NTR are directed by procedures. The frequency of routine surveys is based on potential for changes to affect the radiological situation.

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure control is achieved at the NTR by shielding, the ventilation system, security, entry control devices, an active ALARA program, the radiation protection program, environmental monitoring, equipment and materials, and through the VNC dosimetry program.

Shielding and typical radiation levels for occupied or accessible areas of the NTR facility are discussed in Section 4.3. Reactor cell ventilation is discussed in Section 3.5 and in Chapter 9.

The ALARA program, the radiation protection program, and the environmental monitoring program, are described in Sections 11.1.3, 11.1.2 and 11.1.7 respectively, of this Chapter.

Equipment and materials used in radiation exposure control consist primarily of protective clothing and respiratory protection equipment. Procedures provide for type and application of protective clothing in high radiation and/or radioactive materials areas.

The respiratory program ensures:

- proper respiratory equipment is available and assigned,
- work environment respiratory hazards are evaluated,
- air sampling and analysis is adequate to identify hazards,
- individual personnel exposures are evaluated,
- respiratory protective equipment is properly selected and assigned in accordance with 10 CFR Part 20,
- minimum qualification requirements are set forth that include initial and periodic training, refitting, and medical clearance requirements.

Radiation exposure to NTR personnel varies somewhat from year to year, however, the expected average annual exposure, based on historical records, is 443 mRem per person. Non-NTR personnel, working in the same building, again based on historical exposure records, are expected to receive < 100 mRem per year per person, due to radiation associated with the NTR. Personnel providing service for the NTR, are expected to have a total annual exposure, from all site sources of < 1.0 Rem, with an estimated < 30% of that exposure attributable to NTR sources of exposure.

Because it is unlikely that any individual would receive, in one year, and intake in excess of 10 percent of on ALI(s) of Table 1, Columns 1 and 2 of Appendix B to 10 CFR 20, committed effective dose equivalent (CEDE) is not typically added to external dose for determination of total effective dose equivalent (TEDE) at VNC. The philosophy at VNC and the NTR is to use procedures and other controls to limit the intake of radioactive materials during normal work in controlled areas. Air activity is controlled using ventilation systems and contamination control. Intakes of radioactive materials is limited using respiratory protection devices, control of access and limitation on the time of exposure when other controls are impractical. Whole body counts are done routinely, based on the Manager, NTR's input, to confirm the lack of intake.

Exposure limits for occupational workers are pursuant to 10 CFR 20.1201 requirements and more restrictive administrative limits are established by procedures. Exposure to the Embryo/Fetus of declared pregnant workers is in accordance with 10 CFR 20.1208, 20.1502 and 10% of 20.1201 requirements as informed by Regulatory Guide 8.13, Instruction Concerning Prenatal Radiation Exposure.

Procedures describe the exposure and dosimetry requirements for NTR Operations personnel. Exposure control limits are given, over the full range of operations, including normal operations, emergency conditions and planned special exposures. Administrative dose action levels are established in accordance with the site RP Program and require management approval to exceed.

Personal dosimetry used at the NTR includes beta-gamma dosimeters, neutron albedo dosimeters, and electronic dosimeters (EDs). These are processed and read routinely or as necessary to estimate exposure between routine badge processing. Special use dosimeters such as TLD finger rings are issued for extremity exposure and high dose rate exposure according to the site RP program. Dosimetry records are kept indefinitely by RC for beta-gamma badge, neutron badge, and TLD finger ring exposures.

11.1.6 Contamination Control

Contamination control at the NTR is accomplished through some of the elements of the Radiation Protection Program discussed in the previous sections. These elements include the ALARA program, routine surveys and monitoring, the dosimetry program including evaluation and testing for internal depositions, the use of anti-contamination clothing, training programs for staff and visitors, and survey records. In addition, access control, area posting, and the use of RWP's are integral parts of contamination control at the NTR.

11.1.7 Environmental Monitoring

The primary purpose of the environmental surveillance program is to obtain information essential to assessing and controlling the exposure of the neighboring population to industrial chemicals, radiation and/or radioactive materials. Secondary objectives include identifying the sources of specific contaminants that might be released, predicting trends in pollutant levels, and improving public relations by showing that the operations at VNC are not adversely affecting the health and safety of the public and surrounding areas. This program is responsive to several site licenses, permits, and procedures.

At VNC the overall environmental program is separated into two distinct categories: (1) effluent, monitoring and (2) environmental surveillance. Effluent monitoring for the NTR is limited to the ventilation stack. Sampling and analysis of water in the site retention basins is performed prior to release as a function of environmental surveillance and to coincide with required state industrial effluent sampling. This program provides measurements of the amount of radioactivity that is released to the environment in gaseous effluents and validates that licensed byproduct material does not make its way to the environment through the discharging of industrial wastewater.

Environmental surveillance covers all measurements and observations made of the environment on and adjacent to the site. This includes environmental air samplers and TLD stations; the sampling of water, vegetation, soil and stream bottom sediment; and the resulting radiological and chemical analysis. This program provides assurance that there are no deleterious impacts on the environment from operations conducted at the site.

A complete description of the current VNC environmental program is contained in the VNC Environmental Monitoring Manual. A method for determining action levels is established in implementing procedures and ensures specific actions are undertaken to determine the possible sources of the activity before regulatory limits are reached.

The specialist assigned to environmental protection is responsible to assure the requirements of the environmental protection program are met within the time frames established. This includes:

- sample collection (method and frequency) and analysis (technique and sensitivity);
- preparing required summary reports;
- assuring the proper installation, operation, and maintenance of environmental monitoring equipment.

The specialist assigned to environmental protection has been granted the necessary authority by management to meet these responsibilities.

Regulatory Compliance (RC) is responsible for reviewing the environmental protection program for adequacy and for recommending changes as necessary. Further, RC prescribes equipment in support of the environmental protection program and shall review periodically the activities of the specialist assigned to environmental protection.

11.2 RADIOACTIVE WASTE MANAGEMENT

11.2.1 Radioactive Waste Management Program

Radioactive waste management at VNC addresses a broad range of activities across the site that are inclusive of the NTR. Radioactive Waste at the VNC is any material in which radioactivity is distributed or the surfaces of which are contaminated with radioactive material to levels that prevent release for unrestricted use or which is potentially contaminated and cannot be shown to be less than these levels and which has no further functional or monetary value to the user or owner.

The radioactive waste management program at VNC is implemented by station procedures and sets forth management policy for the handling of low-level radioactive waste (radwaste) materials generated at VNC, including the NTR, establishes a radwaste handling program designed to meet the objectives of this policy, and defines the responsibilities for carrying out the program. Responsibilities for program implementation for the NTR are appropriately divided among the Area Manager, MLO; RC; the Manager NTR; and those individuals performing radwaste activities. Procedures describe the criteria, method and responsibilities to be used at the NTR for collection, interim storage, identification, characterization, and transfer of low-level radioactive waste to the site inspection/packaging area.

VNC activities involved in the processing, packaging, transfer, receiving, interim storage, and shipment of radwaste comply with applicable federal and state regulations.

Low-level radioactive wastes undergo volume reduction as appropriate and are then packaged and ultimately shipped to licensed commercial radwaste disposal sites.

Records and checklists associated with radwaste activities are maintained by the Area Manager, MLO. Such records consist of on-site transfers of radwaste to the MLO, shipment to off-site

disposal facilities, packaging checklists, and personnel training records. These records are reviewed as part of the overall periodic radwaste program review.

11.2.2 Radioactive Liquid Waste Management

The only liquid radioactive waste generated is as a result of the annual sampling, approximately one liter. This waste is placed in tanks with other laboratory generated liquid radioactive waste and subsequently disposed of in accordance with approved site practices and procedures. No liquid radioactive waste is released directly to the unrestricted environment.

Contaminated wastewaters created from NTR operation are processed in the site waste evaporator. Evaporator bottoms are then processed and shipped as radioactive solid wastes. Industrial wastewater from the NTR single pass, non-contact, secondary cooling water heat exchanger is tested for radiological constituents as well as other potentially polluting constituents in accordance with a National Pollutant Discharge Elimination System (NPDES) permit prior to release to the environment.

11.2.3 Radioactive Solid Waste Management

Low level dry active wastes, consisting of contaminated paper and plastic, filters, and resins, are infrequently produced and are also processed and shipped as a fractional component of the VNC solid waste stream. The quantity of solid waste generated by NTR activities is very small; estimated to be one to three cubic feet annually with the radioactive content measured in millicuries. Radwaste reduction techniques, include planning, decontamination, use of reusable vs. disposable materials, unpackaging (debulking) of supplies and equipment prior to transfer into a radioactive materials area (RMA), and dedication of appropriate tools and equipment to RMA's for reuse as needed. The site also uses commercial facilities, where cost effective, to perform radwaste volume reduction and recycling of contaminated materials, e.g., metal melt technology.

Solid radioactive waste is characterized, handled, packaged, surveyed and shipped in accordance with all applicable DOT and NRC regulations. Shipments of solid radwaste are intermittent at VNC and annual Curie makeup of those shipments in recent years have been dominated by unique shipments of byproducts of sealed source manufacturing and materials from the SAFESTOR reactor facilities.

11.2.4 Radioactive Gaseous Waste Management

Release of routine gaseous effluents is dominated by Ar-41, which is generated by neutron activation of Ar-40 in air. Airborne radioactive waste exiting through the NTR stack is monitored as radioactive effluent and is well within the Technical Specification and 10 CFR 20 requirements. Monitoring and alarms associated with the NTR stack have been discussed in this Chapters 7 and Section 3.5 of this SAR.

The annual average dilution factor from the NTR stack to the site boundary based on historical meteorological data, 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 not be greater than 1/20,000 of the concentration at the stack when averaged over one year.

The reactor cell and stack ventilation system were originally required to mitigate an analyzed fueled experiment failure of a type that has not been performed at the NTR for more than 30 years. Nevertheless, the reactor cell will contain any radioactive release while it is exhausted through the ventilation system and out the stack.

Table 11-4 STACK RELEASE ACTION LEVELS

Stack	Nominal Flow Rate, cfm	Noble Gas Ci/wk µCi/cc	Halogen Ci/wk µCi/cc	Alpha Ci/wk µCi/cc	Beta Ci/wk µCi/cc
105, NTR	1.80E+03	9 *9.5E-05	1.74E+02 1.9E-06	8.69E+00 9.5E-11	8.69+E02 9.5E-09

* The NTR noble gas concentration limit during non-operating time, i.e., when the reactor is shut down and the cell can be open, is set at 2E-6 µCi/cc.

11.2.5 Bases for the Stack Action Levels

The stack release action levels are defined as the release rates for each radionuclide group (noble gas, I-131, beta particulate, or alpha particulate) at which action should be taken to reduce the release rate. These action levels ensure doses to members of the public due to airborne releases are at or below the 10 CFR 20.1101(d) limit of 10 mRem per year. The method for establishing these action limits is described below.

10 CFR 20, Appendix B, Table 2, Column 1 gives airborne radioactive material concentration limits for releases to the general environment. Inhalation of a single radioisotope limited by the stochastic annual limits on intake (ALIs) at that concentration continuously over the course of a year would produce a total effective dose equivalent of 50 millirem while inhalation of a radioisotope limited by submersion dose (mostly noble gases) at that concentration continuously

over the course of a year would produce a total effective dose equivalent of 100 millirem. Therefore, the release rates from the NTR stack is controlled to a level which will not exceed 20% of the 10 CFR 20 effluent concentrations of isotopes limited by the stochastic ALIs and 10% of the 10 CFR 20 effluent concentrations of isotopes limited by submersion dose.

Annual average release rates are converted to boundary concentrations by a dilution-dispersion factor. Dilution-dispersion factors are calculated from the measured meteorological conditions for a year's period (or more). Consideration also is given to concurrent releases from the other stacks on site, and the release limit is further reduced to account for multiple releases.

The action level for the noble gas releases from the NTR stack is selected as the rate which would give an annual average concentration of Ar-41 at the site boundary of 10% of the effluent concentration limit (ECL), further divided by a factor of two for other stack releases. Ar-41 has been shown to be the predominant noble gas in the stack effluent (Climent, 1969). Fission produced noble gases are a minor fraction unless fuel material is exposed to the effluent air. Ar-41 is produced by the neutron irradiation of the air passing through the reactor.

The action levels for all other isotope groups are conservatively selected at 10% of the concentration limit for the restrictive, credible isotopes of each of the isotope groups: 1-131, unidentified beta radionuclide and Np-237. These, too, are reduced further by a factor of two for other stack releases. The release limits are specified as release rates (μCi/sec); this makes the limit independent of the stack flow rate. A limit expressed as a concentration (μCi/ml) is dependent on the stack flow rate. However, radioactive concentrations are commonly used in measuring and reporting effluent releases.

Stack flow rates fluctuate, sometimes by design and sometimes randomly. For example, the NTR flow depends on the position of the cell door. The flow in all filtered systems varies as the dust loading on filters increases and as containment systems are changed. The 1,800-cfm average stack flow rate for Building 105 is used for limiting concentrations and calculating measured releases.

The applicable effluent concentration limit values from Appendix B, Table 2, Column 1 of 10 CFR 20 are given below:

Release Category	Limiting Isotope	10 CFR 20 Effluent Concentration Limit, µCi/ml
Noble Gas*	Ar-41	1.00E-08
Halogen	I-131	2.00E-10
Alpha Particulate	Np-237	1.00E-14**
Beta-Gamma Particulate	***	1.00E-12

The dilution-dispersion (χ/Q) factor and reduction factor to account for releases from "other stacks on site" are given below:

Stack Location	χ/Q, sec/ml	"Other stack" reduction factor
Building 105, NTR	3.48E-11	2

* The NTR noble gas inventory available to the boundary has been found to be primarily Ar-41, which is an activation product of air. Fission products would be of concern in the event of fuel failure, an abnormal condition.

** There are several isotopes with more restrictive limits, but they can be shown to be insignificant fractions of the typical mix of alpha emitters found at VNC.

*** Unidentified isotopes, where several natural, transuranic, and other rare elements are known to be absent. These are mainly alpha emitters which would be accounted for in the alpha analysis.

The annual average dilution-dispersion factor for the NTR, and the other stacks at VNC, was calculated from valid hourly records of measured meteorological conditions for a two-year period in 1976 and 1977. The sector average χ /Q factors were conservatively computer calculated for each of 16 sectors (22.5 degrees each) using:

- Scaled distances from a site layout map to determine the distances from the reactor to the center of the sector at the site boundary.
- A building cross-section of 281 square meters, for wake effects.
- A ground level release elevation.
- No credit taken for plume depletion.

The single maximum calculated annual average χ /Q value of 3.48E-11 sec/ml was selected from the 16 sector average values. This value, which happens to occur in the east-southeast sector at 622 meters from the stack, is used to determine the NTR stack release limits.

The ECL release rate, i.e., the continuous release rate which would produce an annual average boundary concentration equivalent to the ECL, would be calculated by division of the ECL by the χ/Q value. The Action Level release rates are calculated by reducing the ECL release rates by a factor of 10 for noble gas (a factor of five and another factor of two for "Other Stacks", and a

factor of 20 for the other isotope groups. These release rate limits, in units of microcuries per second, are shown below.

Isotope Group	Action Limit Release Rates µCi/sec
Noble Gas	2.87E+01
Halogen	2.87E-01
Alpha	1.44E-05
Beta	1.44E-03

These conservative release rate limits are converted and presented as action levels based on cumulative weekly releases in Table 11-4 and the Technical Specification weekly release rate limits of Table 3-3 in Chapter 14, Technical Specifications. A normal maximum operating time for the NTR typically would not exceed 30 hours in a week. Therefore, this partial operating time is used to calculate the operating stack effluent concentration limits.

12 CONDUCT OF OPERATIONS

12.1 ORGANIZATION

12.1.1 Structure

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

The NTR facility organization and interrelationships are shown in Figure 12-1. 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 is organized so that decisions are communicated via the proper levels 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, VNC Operations has overall responsibility for the reactor license. Reporting directly to the Manager, VNC Operations, is the Manager, NTR, who is the Area Manager. The Manager, NTR 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 Area Manager, is the Regulatory Compliance organization (RC). Within this organization are specialists in nuclear safety, health physics, licensing, safeguards, security, and criticality. This organization also contains or is supported by health physics monitoring personnel and quality control personnel.

Also available to the Area Manager are many other highly specialized technical individuals on and off the VNC in the GEH organization.

12.1.2 Responsibility

The responsibilities of selected NTR facility positions are as follows:

12.1.2.1 Manager, VNC Operations

The Manager, VNC Operations, has the overall responsibility for the NTR facility's license.

12.1.2.2 Manager, Nuclear Test Reactor (NTR)

The Manager, NTR, reports to the Manager, VNC Operations. The Manager, NTR is the Area Manager and has the overall responsibility for the safe, reliable, and efficient operation of the NTR.

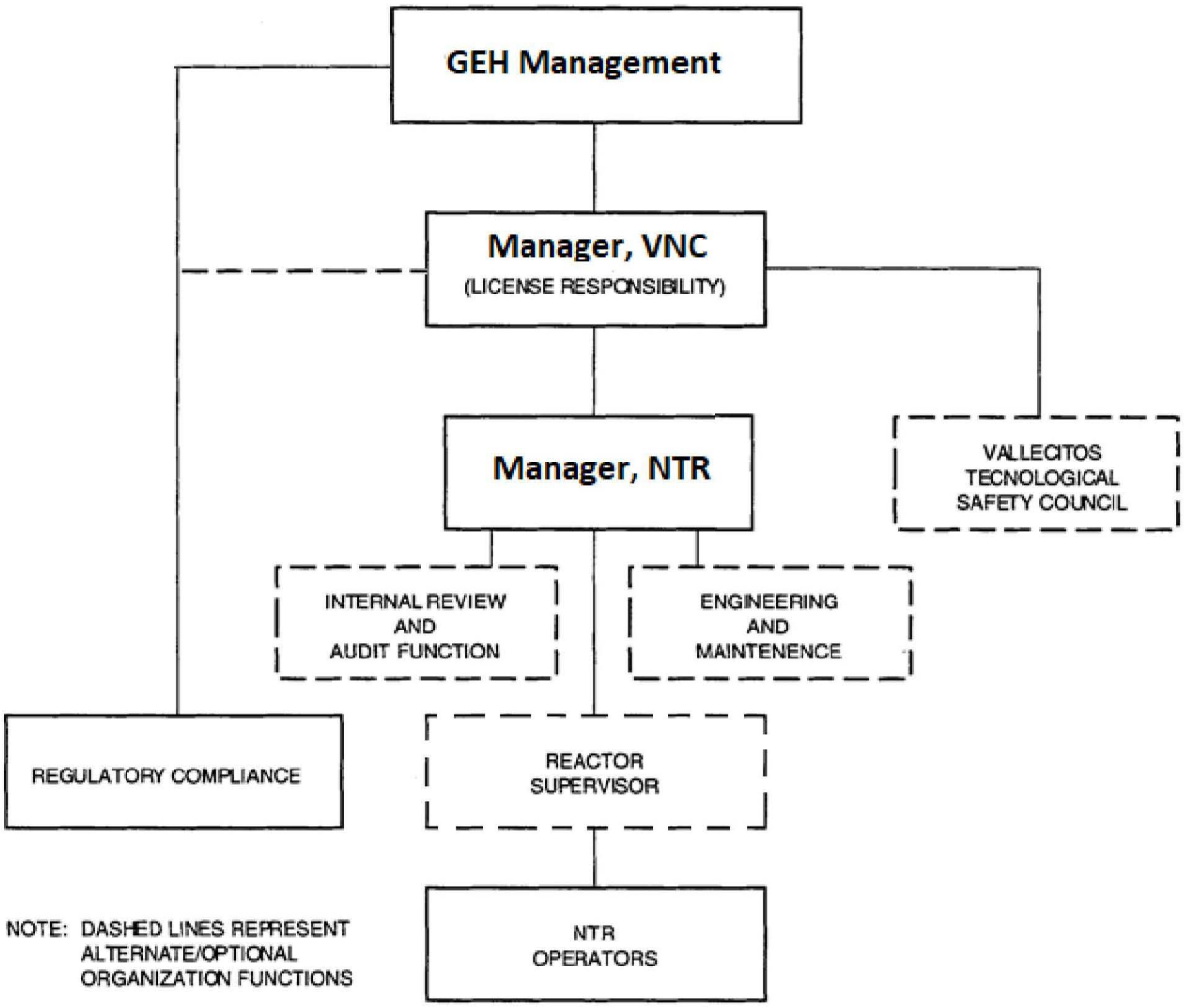

Figure 12-1 NTR Organization

The Manager, NTR is responsible for maintaining a competent staff and an effective organization structure. All changes to the facility or facility procedures and all new tests and experiments require the approval of the Manager, NTR, or designated alternate. The Manager, NTR is responsible for an adequate safety review and may utilize resources of other GEH personnel (or outside consultants) not on the NTR staff. When the positions exist, the Manager, NTR, directs the activities of the reactor supervisor, and the NTR engineer(s). The Manager, NTR 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 requalification of operating personnel, and the safety of assigned and visitor personnel.

12.1.2.3 Reactor Supervisor

The Reactor Supervisor (shift supervisors, Operations Supervisor, etc.) reports to the Manager, NTR and supervises the NTR operation in accordance with written operating procedures, administers planned work, and handles emergencies at the facility. The Reactor Supervisor 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. In the absence of a separate Reactor Supervisor, the Manager, NTR, has these responsibilities.

12.1.2.4 NTR Engineer (engineering and maintenance personnel)

The NTR Engineer (specialist, etc.) may report to the Manager, NTR and usually provides detailed direction and guidance in the installation and operation of experiment facilities and programs and overall plant maintenance, repairs, and modifications within the framework of operating procedures. In the absence of a separate Engineer, the Manager, NTR, has these responsibilities.

12.1.2.5 NTR Operators

The NTR operators consist of trainees, 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 Manager, NTR, or 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.

12.1.3 Minimum Staffing

12.1.3.1 Reactor Not Secured

- A licensed Reactor Operator or Senior Reactor Operator shall be in the control room with access to the nuclear console.
- A second person present at the site who is familiar with the VNC Radiological Emergency Plan and Emergency Procedures relevant to the NTR and is capable of carrying out facility written procedures.

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

12.1.3.2 Reactor Operating

- A Licensed Reactor Operator or Senior Reactor Operator shall be present at the controls at all times during the operation of the facility.

12.1.3.3 During Evolutions that Affect Reactivity

- Apparatus and mechanisms other than controls, the operation of which may affect the reactivity or power level of the reactor, shall be manipulated only with the knowledge and consent of a Licensed Reactor Operator or Senior Reactor Operator present at the controls.

12.1.3.4 A Senior Reactor Operator shall be present at the NTR facility during:

- Recovery from an unplanned or unscheduled shutdown.
- Reactor fuel loading or reactor fuel relocation.
- During any experiment or facility changes with a reactivity worth greater than one dollar.
- Manual poison sheet changes.

12.1.4 Selection and Training of Personnel

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:

12.1.4.1 Manager, NTR

At the time of appointment to the active position, the Manager, NTR, (Area Manager) shall have at least 6 years of nuclear experience. Additionally, the Manager, NTR shall have a baccalaureate or higher degree in an engineering or scientific field. Equivalent education or experience may be substituted for a degree. The degree may fulfill up to 4 of the 6 years of nuclear experience required on a one-for-one basis. The Manager, NTR shall have or immediately pursue steps for obtaining an NTR Senior Reactor Operator license.

12.1.4.2 Reactor Supervisor

At the time of appointment to the active position, the Reactor Supervisor, when utilized, shall have at least 3 years of nuclear-related experience. A maximum of 2 years equivalent full-time academic training may be substituted for 2 of the 3 years of nuclear-related experience required. The individual should have a high-school diploma or have successfully completed a General Education Development (GED) test. The individual shall have an NTR Senior Reactor Operator license or shall immediately pursue steps for obtaining an NTR Senior Reactor Operator license.

12.1.4.3 NTR Engineer

At the time of appointment to the active position, the NTR Engineer shall have at least 1 year of nuclear-related experience. Additionally, the NTR Engineer shall have a baccalaureate or higher degree. Equivalent education and experience may be substituted for a degree.

12.1.4.4 NTR Operators

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.

Minimum qualifications as determined by the Manager, NTR are:

a. Senior Reactor Operator
   - An NTR Senior Operator's license.
   - A sufficient level of experience in NTR reactor operations, experiment setup and operation, and a high level of leadership.

b. Reactor Operator
   - A high school diploma or equivalent, with a high degree of mechanical dexterity.
   - NTR Operator's license.
   - Sufficient training or experience in related nuclear fields.

c. Trainee
   - A high school diploma or equivalent.

12.1.4.5 Initial Operator Training

a. Initial training for the Manager, NTR, and the reactor operators shall be sufficient for the individuals to obtain a Reactor Operator or Senior Reactor Operator license issued by the NRC. Topics shall include the following:

- Fundamentals of reactor theory and operation,
- Facility design and operating characteristics,
- Instrumentation and control,
- Procedures and Technical Specifications,
- Radioactive material handling and exposure control,
- Code of Federal Regulations,
- Emergency response,
- Security.

12.1.4.6 Operator Requalification Program

Licensed operators participate in a comprehensive Operator Requalification Program.

The 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 55.59. The NTR Requalification Plan approved by the NRC, is described in and is administratively controlled by the NTR procedures.

12.1.5 Radiation Safety

Radiation protection is discussed in Chapter 11. Radiation protection functions at NTR are performed by NTR operators and radiation protection staff reporting to the Manager, Regulatory Compliance (RC). RC involvement ensures the radiation safety function is performed independently from the operations organization. All individuals performing a radiation safety function have stop work authority and the authority to raise a concern.

The Manager, RC, raises concerns to the Manager, VNC Operations, the Manager, NTR, or by using the corrective action program.

12.2 REVIEW AND AUDIT ACTIVITIES

An effective independent review and audit process at NTR assures the following:

- Operations comply with the facility license, the Code of Federal Regulations, and established procedures;
- The operating organization discharges its responsibilities consistent with good safety practices; and

- The records accurately and adequately reflect actual operation.

12.2.1 Composition and Qualifications

Review and audit of the NTR is routinely conducted by members of the RC organization which provides a diverse resource of expertise in engineering and health physics. VNC's optional Vallecitos Technological Safety Council (VTSC) may be convened to provide an independent reviewing body. The VTSC is maintained via charter and is composed of personnel from a broad range of activities and technical disciplines. Membership is by appointment from the Manager, VNC Operations. The VTSC is responsible to the Manager, VNC Operations, and is independent of both the Regulatory Compliance and NTR organizations. Both, the RC and the VTSC are supported by audit participation of offsite GEH organizations.

12.2.2 Charter

The RC organization assists operating groups in developing methods of implementing the regulations, licenses, permits and solutions to safety issues and performs independent reviews and audits. Reviews are performed as documents are submitted. Audits are performed quarterly. Comments and recommendations are made to the Area Manager. Disputes are resolved by the Manager, VNC Operations.

The optional VTSC is a review body, independent of all operating organizations, that provides expert advice and counsel, but it is not responsible for conducting routine inspections. The VTSC reports its deliberations and recommendations to the Manager, VNC Operations. Responses to VTSC safety- or compliance-related recommendations should be in writing, addressed to the Chairman, VTSC. The VTSC maintains records of its safety- or compliance-related recommendations and follow-on actions.

The VTSC meets quarterly unless there is no business to conduct. The council may meet as frequently as necessary. A quorum is 50% or more of its members, but VTSC recommendations are based on a majority vote of members present.

12.2.3 Review and Audit Functions

The Regulatory Compliance organization is responsible for reviewing the following:

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

- Proposed types of experiments, facility modifications, and facility procedures; as described in this document;
- Proposed changes to the facility operating license, including Technical Specifications and revised bases;
- Violation of the Federal Regulations, Technical Specifications, facility license requirements, and internal procedures having safety significance;
- Unusual or abnormal occurrences which are reportable to the NRC, as required by the Federal Regulations or Technical Specifications;
- Significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety; and
- Periodic audit of facility operation, maintenance, and administration, to include:
  - The conformance of facility operation to federal regulations, Technical, Specifications, and facility license requirements.
  - 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.
  - The facility emergency procedures, security plan, requalification programs, and their implementing procedures.

The optional VTSC evaluates the overall effectiveness and relevance of safety studies and RC review activities.

The VTSC has the authority to review:

- reportable incident investigations
- applicability determinations and screenings performed pursuant to 10 CFR 50.59;
- operating standards, experiments, and receipt, possession, separation, use, processing, and transfer of radioactive material; and
- processes, operations and procedures which involve toxic, flammable, etc., materials.

Additionally, the VTSC has the authority to:

- Consider and provide advice, as requested by management, on problems of nuclear safety, criticality control, and industrial safety.
- Investigate problems of nuclear safety, criticality control, and industrial safety.
- Evaluate the overall effectiveness and relevance of safety studies and RC review activities as they collectively influence the safety conditions.

- Review and make recommendations on "special topics" as requested by the nuclear or industrial safety function or operations management.
- Review any other matter which it conceives to be of safety importance.

12.3 PROCEDURES

12.3.1 Procedure Process

The facility license, Technical Specifications, and Code of Federal Regulations establish the bounds within which the reactor must be operated. The VNC maintains a series of procedures for the NTR that are exclusive from those used for the broader operation of the site.

Site-specific procedures are reviewed by the applicable Area Manager and accepted by the Manager, VNC Operations or designated alternate.

Procedures exclusive to the implementation of administrative and operational requirements of the NTR Licensing basis and their revisions are approved by the Manager, NTR or his designated alternate.

NTR and site-specific procedures address:

- Normal startup, operation, and shutdown of the reactor and all related systems and components involving nuclear safety of the facility;
- Defueling, refueling, and fuel transfer operations when required;
- Preventive or corrective maintenance which could have an effect on the safety of the reactor;
- Off-normal conditions relative to reactor safety for which an alarm is received;
- Response to abnormal reactivity changes;
- Surveillance, testing, and calibrations required by the Technical Specifications;
- Operator requalification;
- Emergency conditions involving potential or actual release of radioactive materials;
- Radiation protection consistent with 10 CFR 20 requirements;
- Review and approval of changes to all required procedures;
- Facility Change Authorizations
- Security and Emergency Planning; and

12.3.2 Procedure Change Process

12.3.2.1 Major revisions to the NTR-specific procedures undergo independent review by the RC function.

12.3.2.2 Minor changes do not require independent review.

12.3.2.3 Temporary changes to an NTR operating procedure that are to be effective for 6 months or less are made according to the VNC / GEH temporary instruction change process. If a temporary change is to remain in effect longer than six months, a procedure revision shall be performed. A procedure revision is performed for a temporary change that is to remain in effect longer than six months.

12.3.2.4 An SRO may authorize deviations from NTR-specific procedures during emergencies to prevent injury to personnel or damage to the facility. An SRO shall document the required emergency action in the logbook and notify the Manager, NTR.

12.3.2.5 Up-to-date copies of the NTR procedures are maintained and available to all personnel at the facility.

12.3.3 Engineering Release (ER)

An Engineering Release (ER) is issued, as required, to request work, distribute information, document actions, and otherwise ensure the safe and efficient operation of the NTR. An ER cannot contradict an effective procedure (A temporary instruction may be necessary.).

The ERs are written by NTR personnel and reviewed and approved in accordance with the procedure covering ERs. Independent review in accordance with Section 12.2 is required for ERs affecting those activities listed in Section 12.3.1.

12.3.4 Change Authorization (CA)

A Change Authorization (CA) is required for changes to the facility and changes to this document. The CA provides the documented description and a record of safety evaluations required by 10 CFR 50.59, and the review and approval of the change. A CA is required for changes, activities, or projects that are judged to involve significant safety considerations or a potential Technical Specification violation or that may require prior NRC approval. A Change Authorization is also required for new types of experiments or changes to types of experiments.

Change Authorizations involving experiments (experiment type approval, as discussed in Chapter 10) require the following as a minimum:

a) All new types of experiments which could be postulated to affect reactivity or 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.

b) Changes to approved experiments shall receive appropriate review and approval.

c) Approved experiments are implemented in accordance with written procedures.

Change Authorizations are administratively controlled by a procedure. The Change Authorization is reviewed independently by Regulatory Compliance to determine that the following criteria are satisfied:

- The proposed change can be made without prior NRC approval (10 CFR 50.59).
- The change does not violate any license requirement or federal regulations.
- 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 procedure.

RC provides an independent review and concurrence of all Change Authorizations. RC may request evidence that the specified criteria are satisfied.

The Manager, NTR, or his designated alternate has the responsibility of approving Change Authorizations.

12.4 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 until the cause is determined and required corrective action is completed.

Should the true value of the reactor thermal power exceed 190 kW, the reactor shall be shut down and secured immediately and notification made to the Manager, NTR; Manager,

Regulatory Compliance; and the Manager, VNC Operations. The NRC is notified according to the Technical Specifications Section 6.6.2.

The reactor shall remain shut down and secured until reactor operation is authorized by the Manager, VNC Operations.

If operation, maintenance, testing, or inspection reveals an unusual or unexpected result or a situation which is potentially reportable, the individual noting the occurrence shall notify the Manager, NTR immediately. If the reactor is in operation, the condition or situation shall be immediately returned to normal or the reactor shut down. The Manager, NTR shall notify RC personnel. If the event is determined to be reportable, the NRC shall be notified according to the Technical Specifications Section 6.6.2.

12.5 REPORTS

Reports shall be submitted to the NRC as required by the applicable portions of Title 10 CFR, Parts 20, 40, 70, 71 and 73.

12.5.1 Special Reports

Special Reports of unplanned events at the NTR as well as planned major facility and administrative changes are addressed in Chapter 14 (Technical Specifications). Technical Specifications also describes the content of the annual report that is used to meet the reporting requirements of 10 CFR 50.59.

12.5.2 Annual Operating Reports

Contents of the annual operating report are submitted to the NRC each year and include the following:

a. A narrative summary of reactor operating experience including the hours the reactor was critical and total energy produced.

b. The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.

c. Tabulation of major preventive and corrective maintenance operations having safety significance.

d. A summary report in accordance with 10 CFR 50.59.

e. A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the owner-operator as determined at or before the point of such release or discharge.

f. Summarized results of environmental surveys performed outside the facility.

g. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed or recommended.

12.6 RECORDS

NTR records are maintained and retention periods specified according to Chapter 14 (Technical Specifications). Sitewide at the VNC, records are maintained in accordance with the applicable Federal Regulations such as 10 CFR 20.2103, 10 CFR 30.51, 10 CFR 40.61, 10 CFR 50.71, 10 CFR 55, 10 CFR 70.51, 10 CFR 73.70 and 10 CFR 74.19.

Records, in the form of logs, data sheets, recorder charts, and computer disks are maintained in file cabinets, binders, archive boxes, or electronic format.

12.7 EMERGENCY PLANNING

The Radiological Emergency Plan is informed by ANSI/ANS-15.16, Emergency Planning for Research Reactors, and provides the framework of the site's response to emergency situations: to identify, communicate, respond to, and minimize the consequences of an emergency.

The objective of the Emergency Plan is to provide a basis for action, to identify personnel and material resources, and to designate areas of responsibility for coping with any emergency at the VNC that could impact public health and safety. This plan identifies both on-site and off-site support organizations that are required to be contacted for specialized assistance depending upon the nature of the emergency. It is the basis for detailed implementing procedures, which provide staff with the flexibility to cope with a wide range of emergency situations without requiring frequent revisions to the plan.

Provisions for reviewing, modifying, and approving the emergency implementation procedures are defined in the plan to assure that adequate measures to protect the staff and general public are available.

The emergency plan is reviewed biennially and revised as necessary.

12.8 SECURITY PLANNING

Since its inception, VNC has operated under a controlled-access security plan. The perimeter of the Site is posted "No Trespassing." The VNC Site Developed Area includes an access road from SR 84 with a vehicle access gate and personnel turnstile at the access point to the permanent controlled access area. This controlled access area is the portion the VNC Site Developed Area that is within the vehicle access gate.

Several security measures are in effect at NTR to prevent unauthorized access to the facility and theft of materials.

Written procedures are available for incidents [redacted]. The plan requires periodic testing of systems, recordkeeping, and reports to the NRC. Authorized individuals may refer to the current Security Plan for more details.

12.9 QUALITY ASSURANCE

Design and construction of new and modification of existing structures, systems and components (SSCs) that are important to safety are subject to a comprehensive quality assurance program. The objective of the program is to maintain an assurance of quality of the scram systems (Table 7-1) and safety-related items (Table 7-2) of the NTR.

12.9.1 Organization and Responsibilities

This section describes the organizational structure and functional responsibilities for the quality program (Figure 12-1).

12.9.1.1 NTR Operations

NTR Operations is responsible for operation of reactor and experiment systems in accordance with established procedures. NTR operations is also responsible for those items in Section 12.9.2 as required.

12.9.1.2 Engineering

The NTR engineer (or the individual performing this function) is responsible for:

a) The performance of engineering on the NTR scram systems and safety-related items,
b) The generation of designs and design changes,
c) The preparation of specifications, work instructions, and procedures,
d) Participation in design reviews,

e) Specifying which items in the scram systems and which safety-related items require quality assurance and the level of quality assurance required,
f) Ensuring adequate proof of component and systems operability,
g) Evaluating system and structural performance and effecting solutions, as appropriate, where operation is found to be inadequate, and
h) Performing related engineering functions as required.

12.9.1.3 Regulatory Compliance

The roles of Regulatory Compliance and the optional VTSC are discussed in Sections 12.1 through 12.3 of this Chapter.

12.9.1.4 Purchasing

Purchasing performs activities related to procurement of materials and services required from outside vendors in accordance with procedures. Parts that are purchased for use in safety systems or safety-related items, from a vendor or original manufacturer having a 10 CFR 50, Appendix B Quality Assurance program, will have directions included in the Change Authorization for notifying the seller in the event the part fails or is found to be deficient according to purchasing documents.

12.9.1.5 Transportation and Materials Distribution

The VNC Shipping and Receiving organization is responsible for receiving, shipping, and on-site movement of materials. The routine handling of materials is in accordance with procedures.

12.9.1.6 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.

12.9.1.7 Fabrication

Fabrication may be performed on- or off- site as requested by the responsible engineer.

12.9.1.8 Mechanical Maintenance

Facilities Maintenance components perform installation, repair, and maintenance services on mechanical systems not performed by the NTR staff. Records of work performed and calibration standards traceability, as required, are maintained.

12.9.2 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:

- Provide, when warranted, space for sign-off by the person who performs the work to show that he has followed the prescribed instructions.
- 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.
- Include, as necessary, special instructions for handling and transportation.

12.9.3 Design Control

12.9.3.1 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.

12.9.3.2 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, RC review is adequate), 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 Section 12.3.4.

12.9.3.3 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.

12.9.4 Procurement Control

12.9.4.1 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). RC reviews MRs prior to submittal to Purchasing as required both for procurement and RFQs. Receiving inspection instructions, if required, are included on the requests. Receiving inspections are performed by the NTR personnel according to procedure to verify by objective evidence such features as proper configuration, identification, and cleanliness, and to determine any shipping damage, fraud, or counterfeit.

12.9.4.2 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 RC. The quality of purchased materials is verified by supplier-furnished evidence, source inspection, receiving inspection, or a combination of these, as appropriate.

12.9.5 Document Control

Organizations performing work within the scope of this program generate documents such as procedures, drawings, specifications, and work instructions. Procedures are established describing the document control system. The document control system assures the proper review, approval, distribution, and control of documents and their revisions.

12.9.6 Material Control

Procedures are established, as required, to control the identification, handling, storage, shipping, cleaning and preservation of safety-related items. 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.

12.9.7 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.

12.9.7.1 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.

12.9.7.2 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.

12.9.8 Inspection

12.9.8.1 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.

12.9.8.2 Inspection Requirements

Inspections are performed, as required, to written instructions and the inspection results are documented. NTR staff inspects raw materials, fabricated parts, assembly, and installation to the specifications provided. For purchased material, the receiving member of NTR staff identifies and matches quantities received with the purchase order and performs the receipt inspection.

12.9.8.3 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.

12.9.9 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.

12.9.10 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.

12.9.11 Nonconformances

12.9.11.1 Nonconforming 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 ensure their proper disposition. Nonconforming conditions shall be evaluated for further reporting to appropriate regulatory agencies.

12.9.11.2 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. Technical justification for acceptability of a nonconforming item dispositioned “repair” or “use as-is” shall be documented in a non-conformance report and is subject to design control measures commensurate with those applied to the original design.

12.9.12 Corrective Action

Documentation of agreed-upon corrective action for conditions adverse to quality are governed by established procedures. In the case of a condition adverse to quality or a significant condition

adverse to quality, the cause of the condition shall be investigated, and corrective action taken according to the GEH corrective action program.

12.9.13 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.

12.9.14 Records

Records are retained in accordance with the requirements of Section 12.6.

12.9.15 Audits

RC 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.

Procedures include a program for the performance of radiation safety audits. Audits/reviews to verify conformance of an item or activity to requirements, shall be planned, documented, and performed. Audits/reviews should include radiation safety, procurement, construction, modification, maintenance, and experiment fabrication, and should ensure effective implementation of the program is assessed, deficiencies identified, and corrective actions have been taken. Audits are scheduled to cover audited programs and activities over a two-year period. Reviews of several NTR Technical Specifications are performed four times per year. The reviews are conducted by RC staff who are independent of the NTR organization and who have experience or training commensurate with the scope, complexity, or special nature of the activities to be reviewed. The written reports resulting from the reviews are sent to the Area Manager and/or Manager, RC, as appropriate. Follow-up action is performed by the reviewer and tracked to completion, to assure corrective action is accomplished.

12.10 OPERATOR TRAINING AND REQUALIFICATION

All licensed operators participate in a comprehensive Operator Requalification Program. The 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 55.59. Refer to the "Requalification Program for the General Electric Nuclear Test Reactor" for details.

12.11 ENVIRONMENTAL REPORTS

Operation of the NTR has had minimal effect on the environment. Refer to "Vallecitos Nuclear Center Environmental Report 2020," NEDO-12623.

12.12 REFERENCES

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13 ACCIDENT ANALYISIS

13.1 ACCIDENT-INITIATING EVENTS AND SCENARIOS

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 Section 13.2.

Events categorized as anticipated operational occurrences are discussed in Section 13.3. 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:

- Loss of electric power
- Loss of secondary cooling
- Loss of facility air supply
- Inadvertent core inlet temperature change
- Fuel handling errors.

Unacceptable consequences for anticipated operational occurrences are:

- Release of radioactive materials to the environs that result in exceeding the limits of 10 CFR 20.1301/20.1302 for members of the public;
- Radiation exposure of any person in excess of 10 CFR 20.1201 limits for occupational workers or 10 CFR 20.1301 limits for members of the public; and
- Violation of an established safety limit.

Events categorized as accidents are discussed in Section 13.4. 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 burners. The postulated accidents analyzed in this chapter are:

- Uncontrolled reactivity increases
- Loss of primary coolant flow (pump shaft seizure)
- Rod withdrawal
- Loss of primary coolant.

Unacceptable consequences for postulated accidents are:

- Radioactive material release to an extent that exceeds the guideline values of 10 CFR 20.1201 for occupational workers or 10 CFR 20.1301 for members of the public.
- Violation of a safety limit.

Section 13.5 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 Section 13.7.

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 Section 13.4.6. 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 Section 13.6 to demonstrate

the capability of the facility, even though such a release is not possible under the 0.76$ reactivity limit.

13.2 TRANSIENT MODEL

The reactor dynamics were simulated with a computer model. TRACG computer code is used to simulate the thermal-hydraulics response of the reactor with kinetics feedback. The TRACG code (Reference 30) is the GEH proprietary version of the Transient Reactor Analysis Code (TRAC), a system code widely used for boiling water reactor safety analyses. The code capabilities include 3D and point kinetics models, a multi-dimensional, two-fluid model for the reactor thermal hydraulics, and an implicit integration scheme for numerical calculations. The code was reviewed and approved as part of the application methodologies for Anticipated Operational Occurrence (AOO), stability, Anticipated Transient Without Scram (ATWS), and Loss of Coolant Accident (LOCA) analyses. Its applicability range extend from full operating pressure of boiling water reactors to atmospheric pressures (Reference 33).

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 (Reference 33).

Figure 13-1, 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. 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 13-1 shows three initial fuel temperatures for (a) average channel representing average power, flow, and temperature conditions, (b) a channel with less than average power (90%) but much less than average coolant flow (50%), and (c) a channel with highest power (130%) and highest flow (153% of the average). 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. In TRACG, the values of fuel surface temperatures during nucleate boiling conditions are calculated with Chen correlation.

(3) In the transient shown in Figure 13-1, 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 450,000 Btu/h-ft^2. 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 13-1, 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 Sections 13.4.1 through 13.4.4.

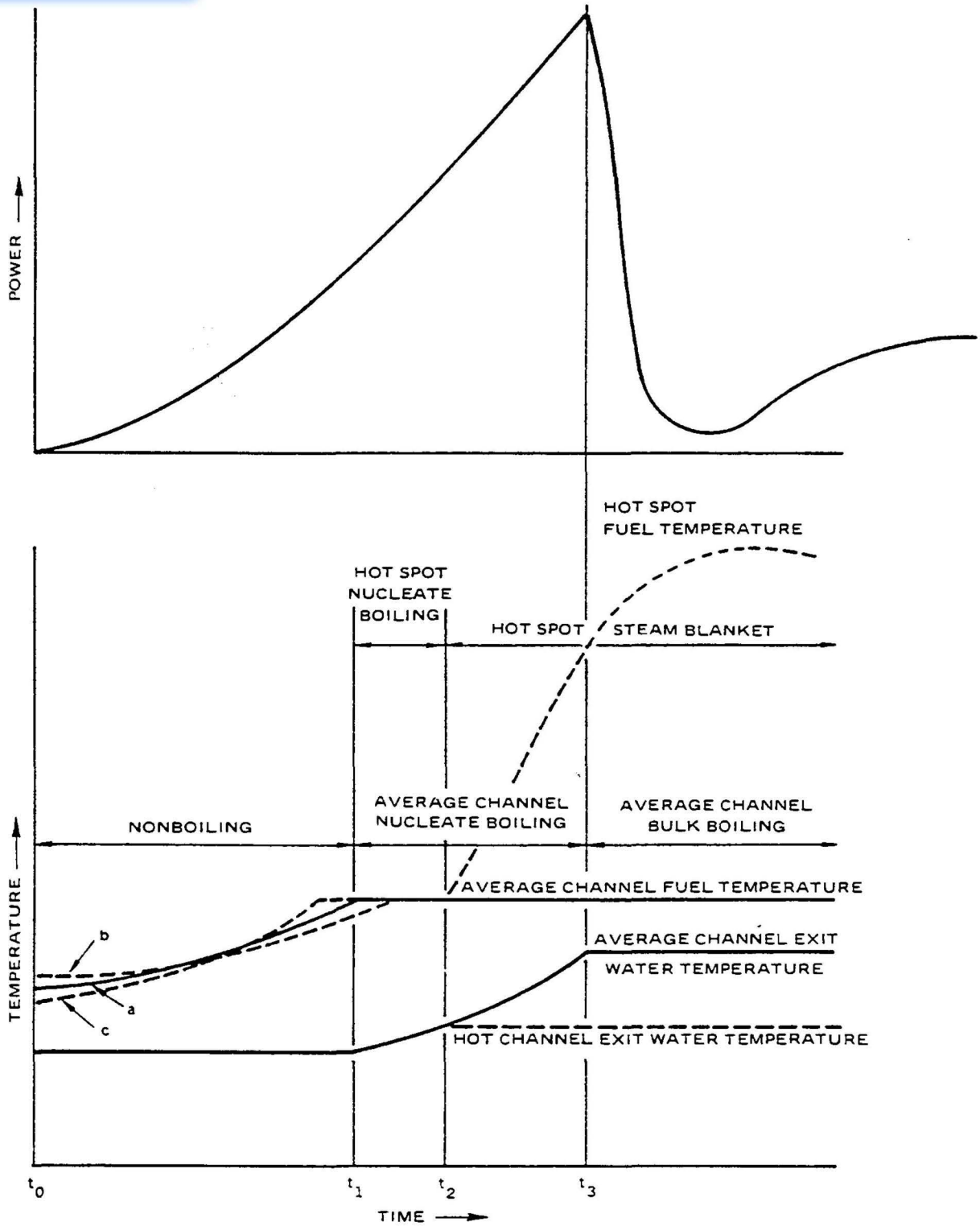


Figure 13-1 NTR Transient

13.3 ANTICIPATED OPERATIONAL OCCURRENCES

13.3.1 Loss of Electrical Power

[redacted]

13.3.2 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.

13.3.3 Loss of Secondary Coolant

Secondary coolant flows by gravity through the tube side of the primary heat exchanger, as described in Section 15.3. Loss of secondary coolant or loss-of-coolant flow will cause primary reactor coolant temperature to gradually increase. This will cause power to increase until core temperature exceeds 124 °F at which time the negative coolant temperature coefficient will turn power over and power will decrease. If reactor power level increases to a level that produces an appreciable heating rate, the reactor will scram from high primary coolant temperature. If reactor power is not high enough to produce a high-reactor power scram or result in a heating rate that could trigger a high primary coolant temperature scram, the loss of secondary coolant will soon be evident to the operator by:

- The slightest changes in temperature, which cause an observable reactivity effect.
- The temperature monitor system readout at the console.
- The secondary flow control in the control room.

13.3.4 Inadvertent Core Inlet Temperature Change

If the primary pump were inadvertently started, the effect would be to decrease the reactor inlet temperature. This and any other event that causes a decrease in inlet temperature while the average reactor coolant temperature is below 124 °F will cause reactor power to drop. There is no safety or radiological concern related to an inadvertent pump start.

Other events can be postulated that would increase reactor coolant temperature. 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. There is currently no source of energy to produce an increase in primary coolant temperature. However, the system design includes a 5-kW heater that was removed from the system years ago that could be reinstalled if needed. The amount of positive reactivity which could be added from this heater 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.

The worst possible case would be a coolant heatup to 124°F from reduced temperature conditions. A case is performed with initiation of a 5-kW heater without scram. The temperature and power characteristics are shown in Figure 13-2. The power continues to rise

until power increases sufficiently to raise the core coolant temperature above 124°F and reduce the net reactivity. Power and temperature stabilize at a higher power and temperature.

For such a slow transient, a high-power scram would clearly stop the excursion without fuel damage. For even higher heat additions, 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.

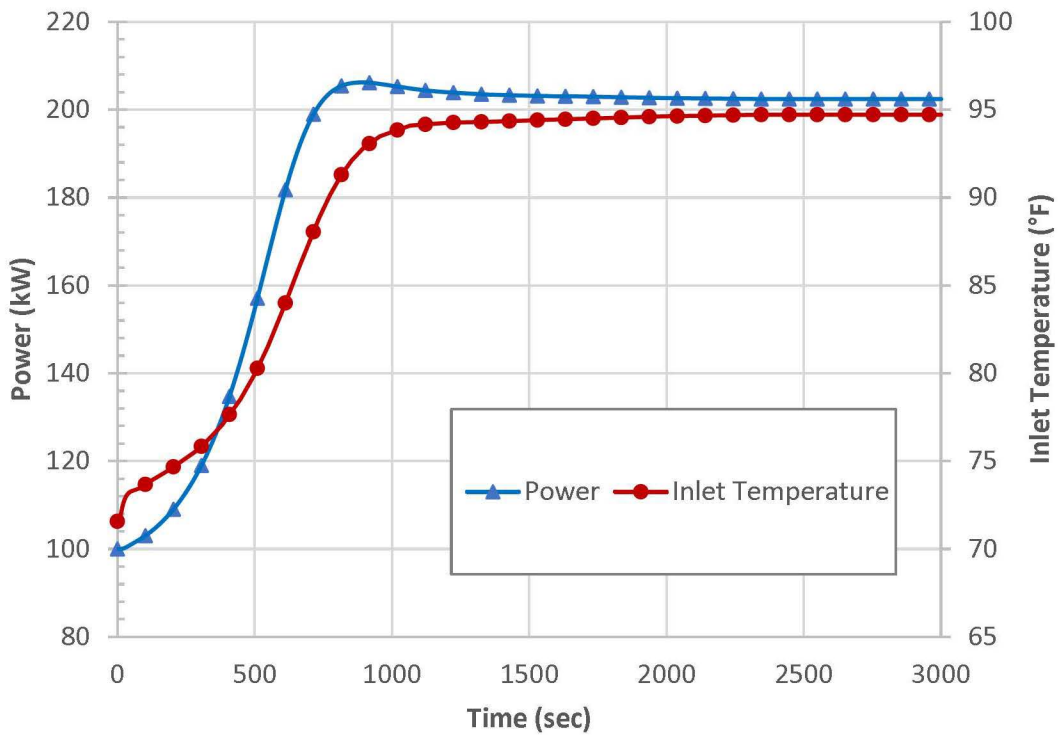


Figure 13-2 Initiation of 5 kW Heater from Reduced Coolant Temperature (65°F)

13.3.5 Fuel Handling Errors

Fuel handling equipment and procedures are discussed in Section 9.2. It should be reemphasized here that refueling for reactivity increase is not necessary, and fuel handling is very rare. The most recent fuel handling occurrence was in support of core container replacement in 1976. 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 Chapter 10.

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:

- Reactor design, fuel handling equipment, and administrative controls are such that not more than two elements can be handled at one time.
- All fuel movement must be performed in accordance with written procedures.
- The cell high-gamma-level alarm system will be in operation.
- 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$.
- Any movement of source and special nuclear material within the NTR facility must have the approval of the licensed operator on duty.
- Any storage arrangement used will be analyzed to ensure a subcritical configuration.

13.4 POSTULATED ACCIDENTS

The transient model used to simulate the reactor dynamics is presented in Section 13.2, except that analysis in 13.4.1 and 13.4.2 do not use the TRACG code.

13.4.1 Idealized Step Reactivity Insertions – with Scram

Transients resulting from step reactivity insertions up to 1.4$ 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 13-3. 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 13-4 and the sequence of events for this transient is as 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 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.

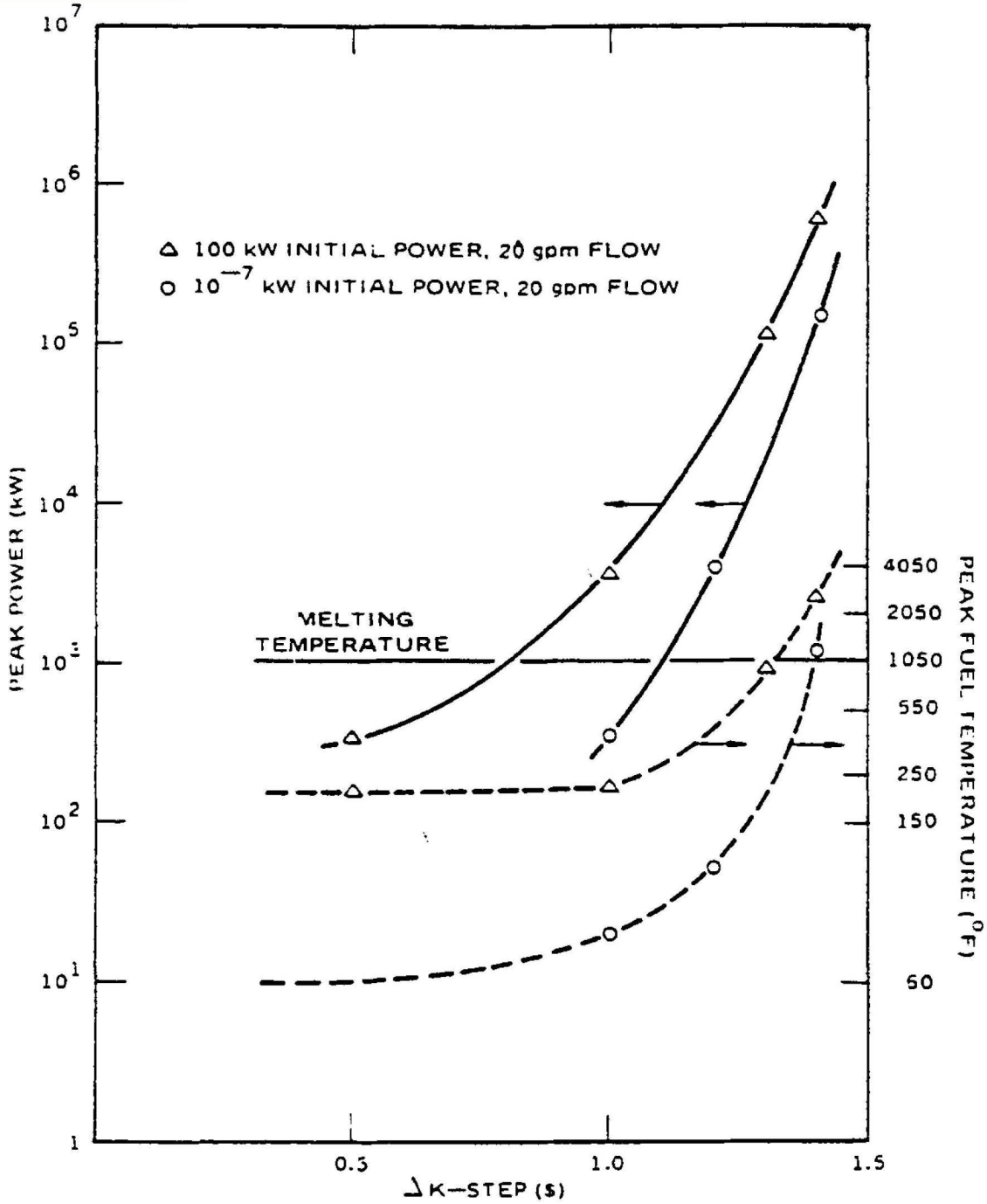
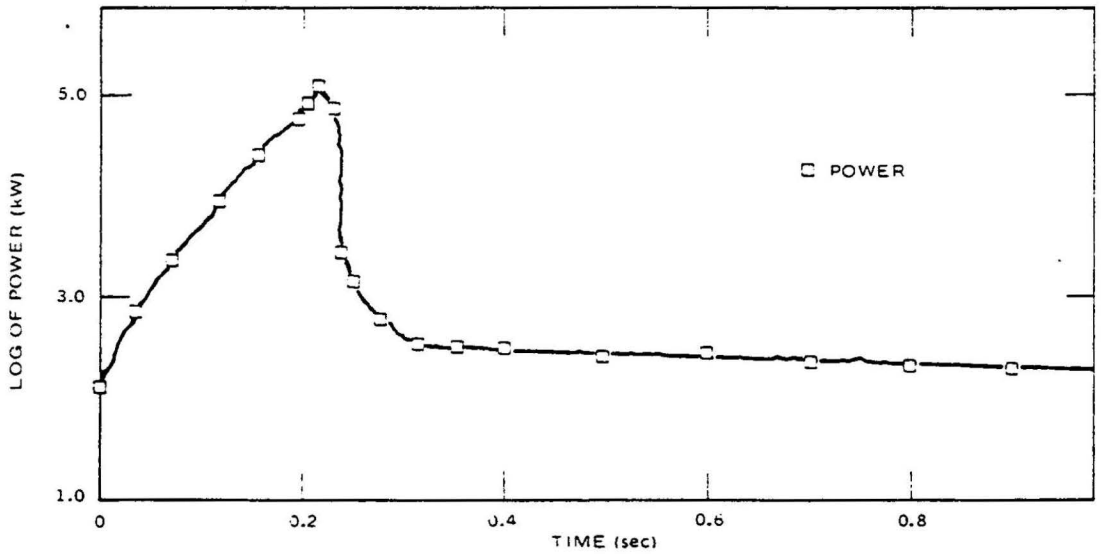
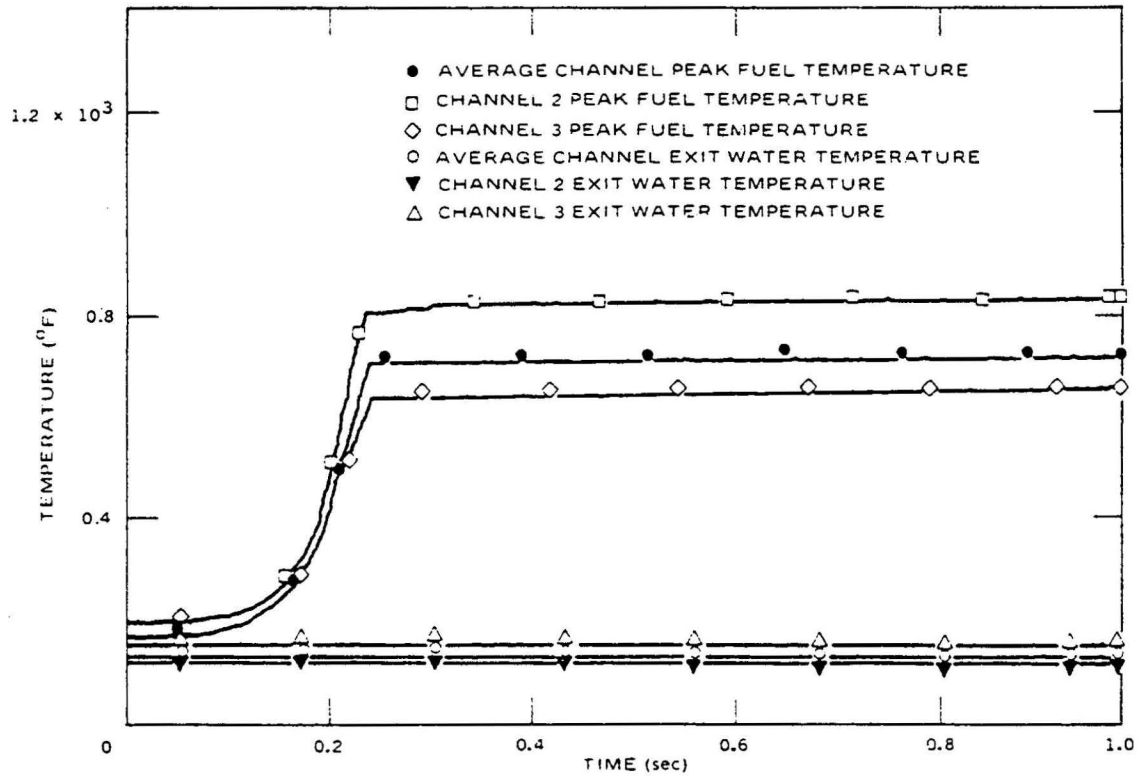


Figure 13-3 Δk-Steps with 150-kW Scram



a. Log of Power



b. Temperature

Figure 13-4 1.3$ Step from 100-kW with Scram

13.4.2 Idealized Finite Ramp Reactivity Insertions – with Scram

Large reactivity insertions over short periods of time were studied for finite reactivity ramps. The study used a simplified point kinetics model with a lumped neutron precursor group with reactivity feedback. The feedback effects included, in addition to the scram, moderator temperature and voiding. Reactivity insertions of 2$ and 4$, with durations from 0.2 to 0.6 second, were analyzed. The results for initial powers of 100 kW with the overpower scram occurring at 150 kW are given in Figure 13-5. For the 4$ 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. Figure 13-6 and Figure 13-7 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 13-5** 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.

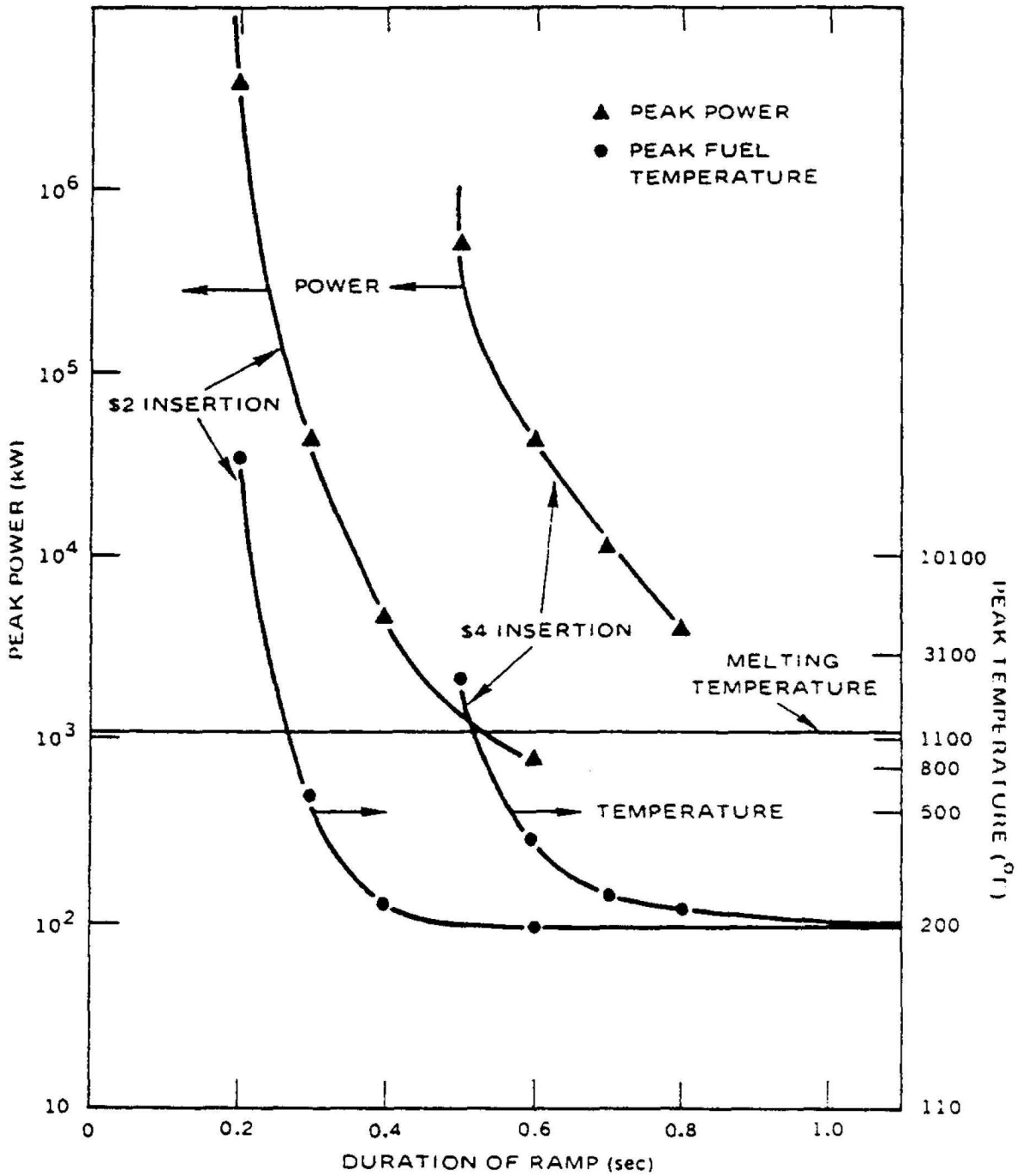
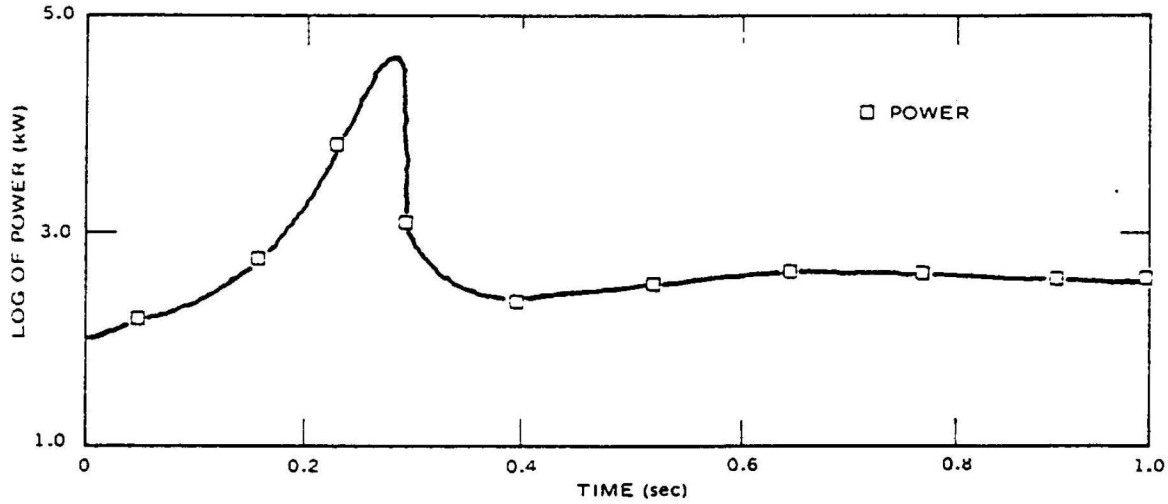
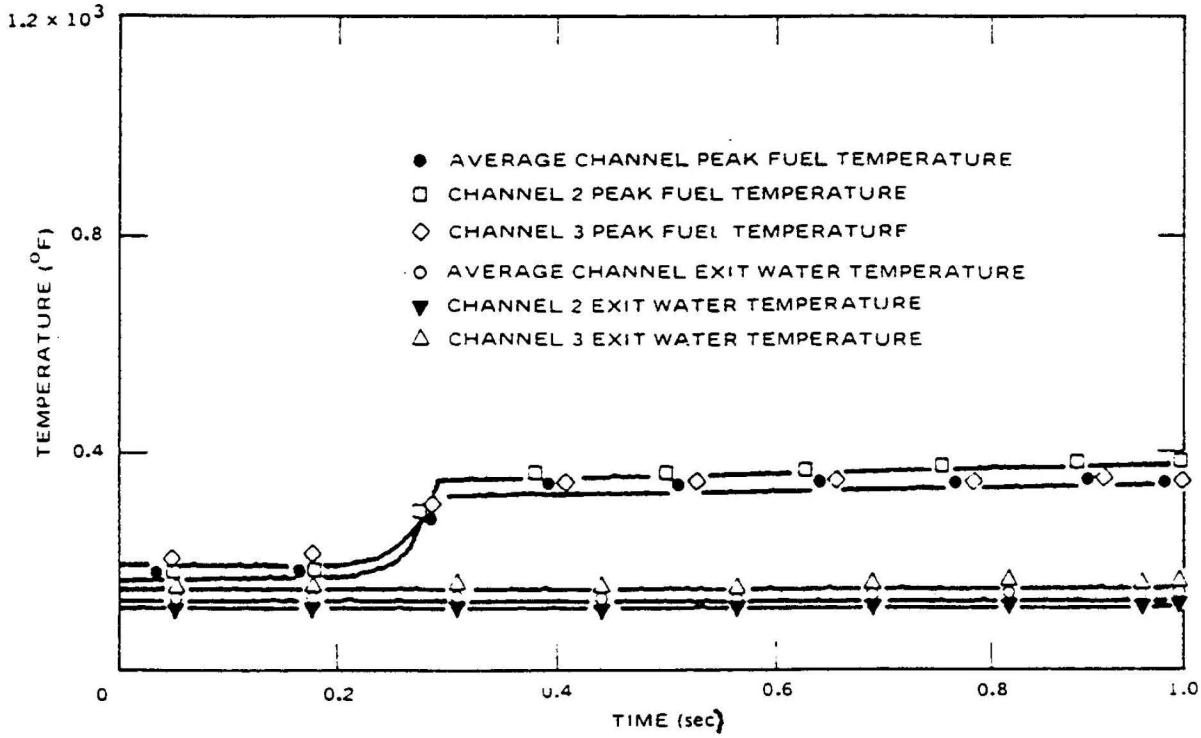


Figure 13-5 100-kW Finite Ramp Insertion with High-Flux Scram

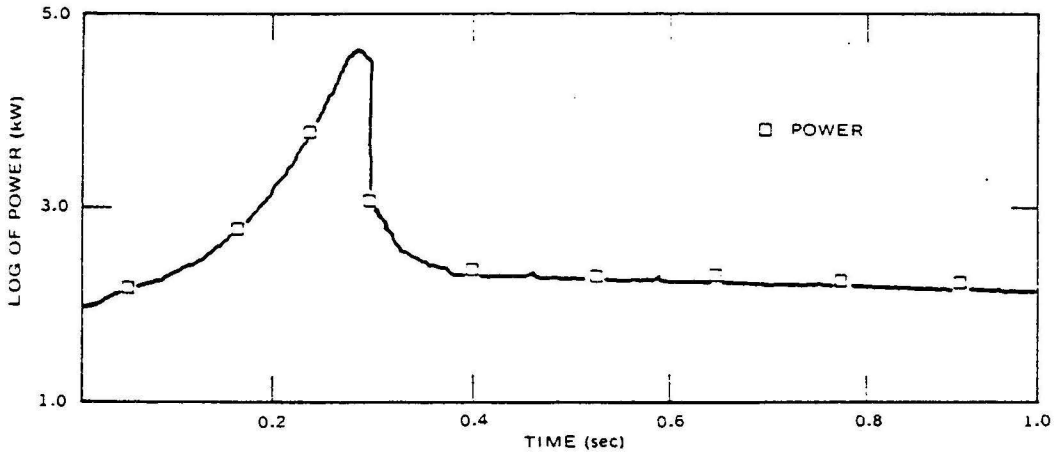


a. Log of Power

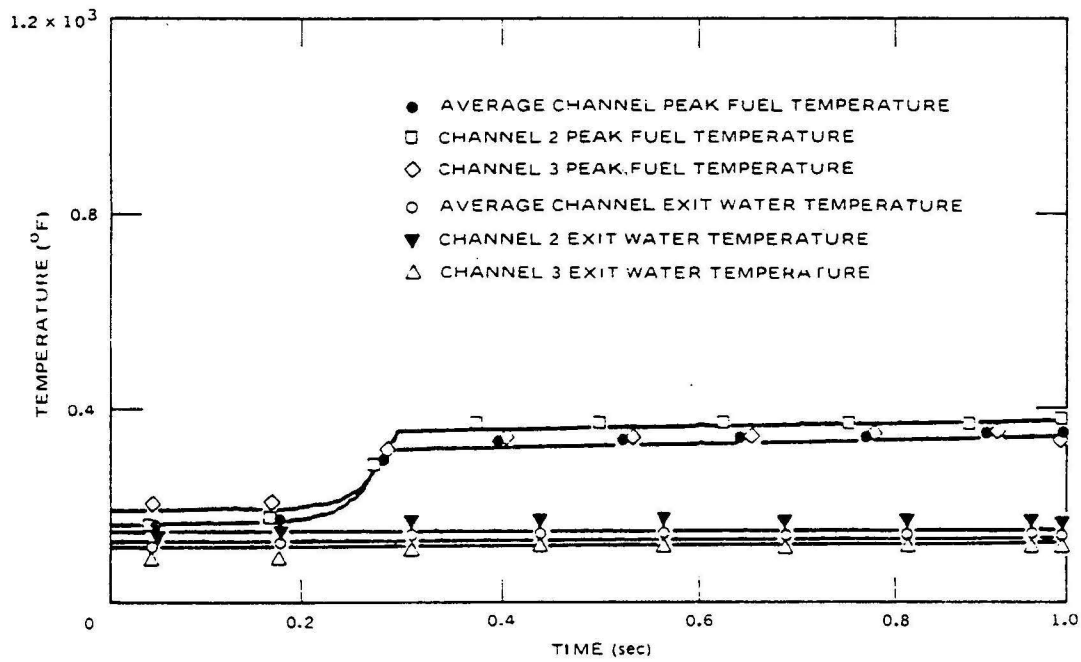


b. Temperature

Figure 13-6 4$ Ramp in 0.6 Second from 100 kW with Scram



a. Log of Power



b. Temperature

Figure 13-7 2$ Ramp in 0.3 Sec from 100 kW with Scram

13.4.3 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 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 sizable step reactivity insertions. The results of a 0.76$ step reactivity insertion are shown in Figure 13-8. Power peaked at 4 x 10^3 kW and PCT at 373 °F. The PCT and peak power characteristics versus magnitude of the reactivity step are shown in Figure 13-9.

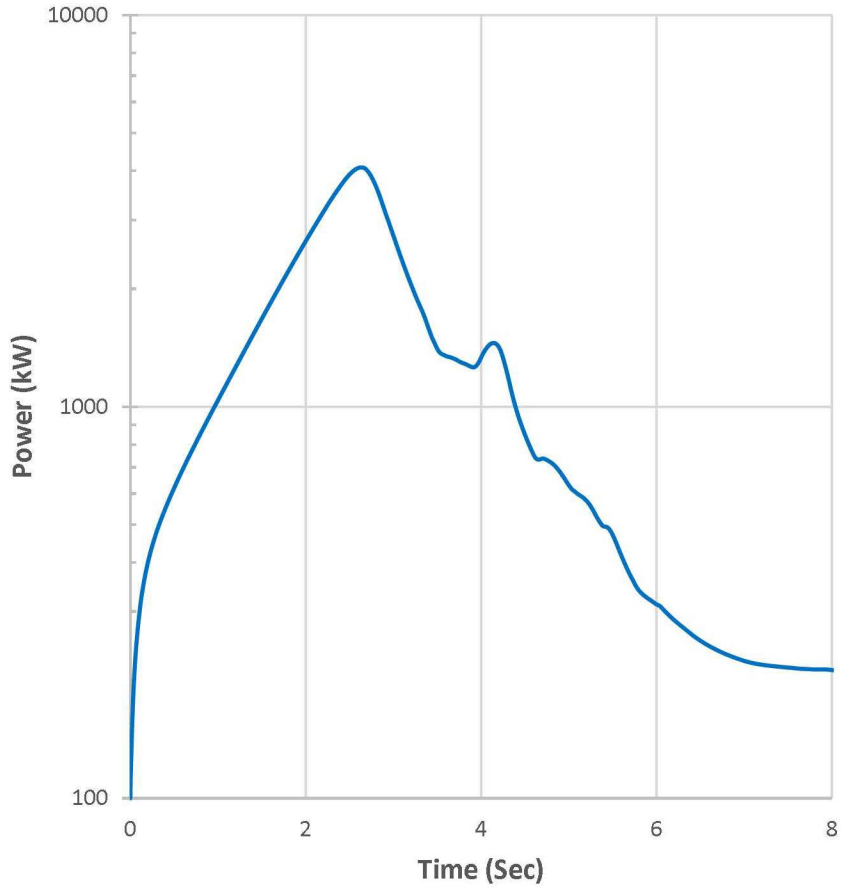


Figure 13-8 0.76$ Step from 100 kW – No Scram

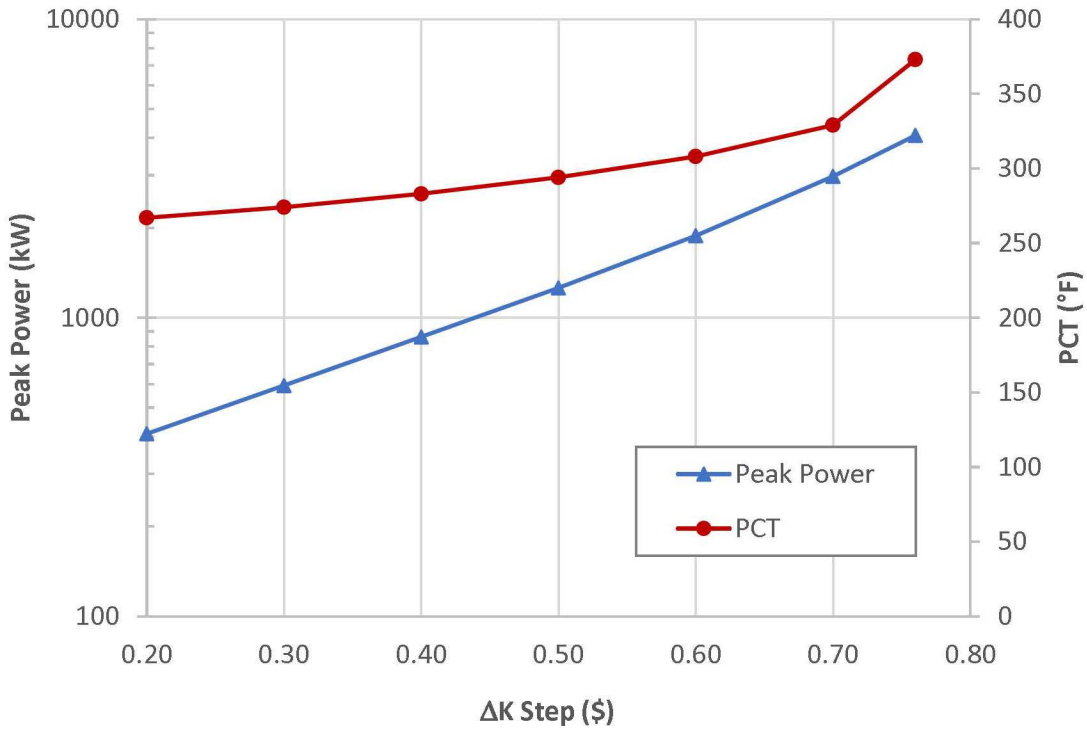


Figure 13-9 Δk Steps from 100 kW – No Scram

To determine the effects of positive reactivity additions from less than full power and temperature, additional transients were run with an initial power level of 1 x 10^{-7} kW. Inlet water temperatures ranged from 55 to 90°F and the initial positive reactivity step of 0.76$. Results show that the positive reactivity feedback from the temperature coefficient, while not as important for the full-power cases because the feedback is very small, it is more important for the zero-power cases because coolant temperatures are lower.

Reactor power and peak fuel temperature versus time for a 0.76$ step insertion from 1 x 10^{-7} kW and 55°F inlet water temperature is given in Figure 13-10. As can be seen from the results, limiting the step reactivity to 0.76$ or less ensures that there are no mechanisms available which will cause fuel damage.

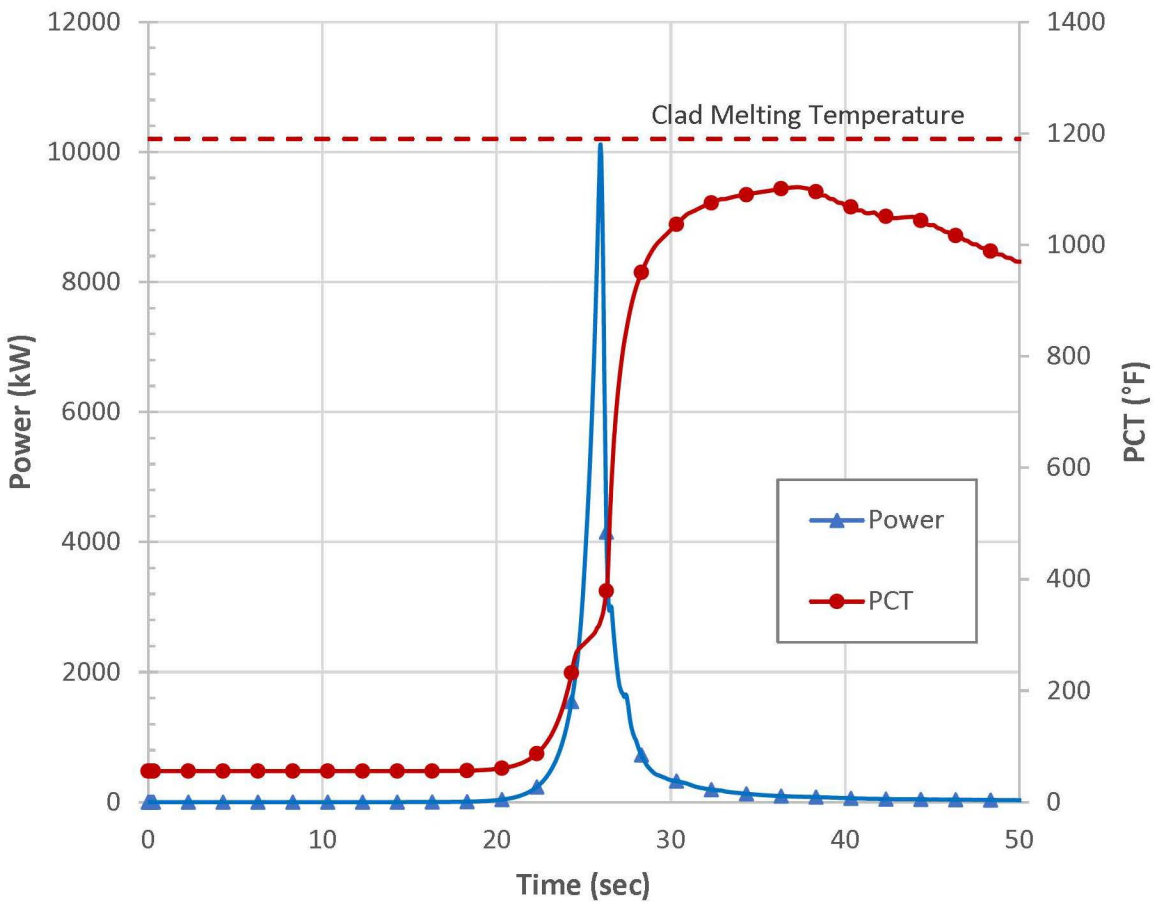


Figure 13-10 Reactor Power and Hot Spot Fuel Temperature Versus Time, 0.76$ Step from Source Level, 55°F Coolant Inlet Temperature – No Scram

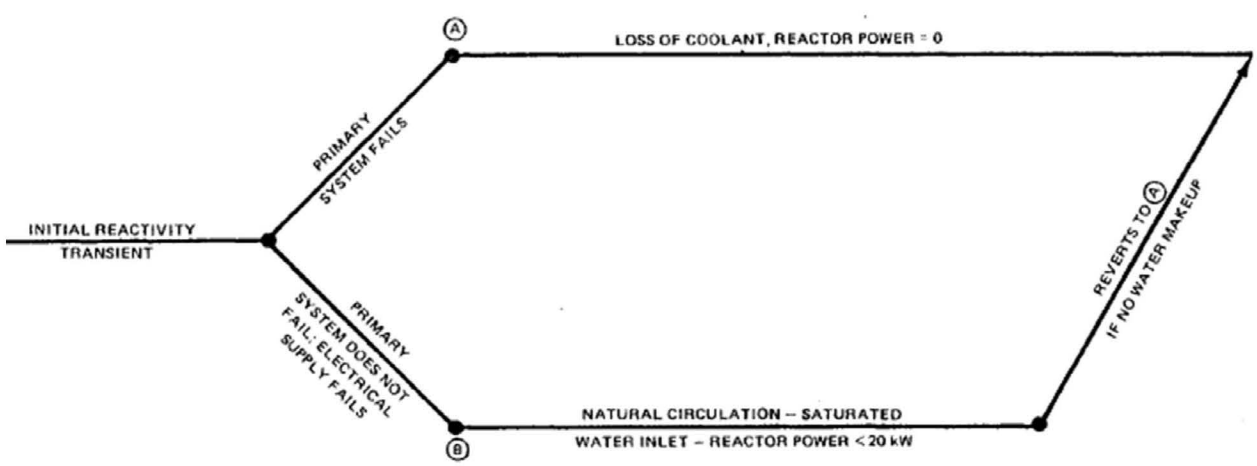


Figure 13-11 Possible Reactor States Following the Postulated Seismic Event

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 13-11 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 Section 13.4.6 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 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 power reduction by voiding the reactor core (Figure 13-11, State A).

We may also assume loss of electric power because: a) it is 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 fate of the secondary system becomes a moot question, and we need consider only the possibility of the primary system 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 low power and low flow. Since there is no secondary cooling, reactor power must be dissipated by the heat loss from the uninsulated reactor primary piping and by evaporation or boiloff of the

primary coolant. 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. The loss-of-coolant by boiling will be a less severe event than the loss-of-coolant event described in Subsection 13.4.6 for two reasons.

1. The reactor power is lower 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.
2. As the maximum fuel temperature for a loss-of-coolant occurring at a power level of 100 kW is less than the fuel melt temperature, there will be no fission product release from this accident.

13.4.4 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 Section 13.4.3 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 ≤0.76$) does not melt fuel. Therefore, the transient which would be caused by the withdrawal of all the rods can be accommodated.

13.4.5 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 106.2°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 reactivity coefficient (Equation 4-2), this final coolant temperature level is 151°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 temperature has been limited to a value well below the melting point, as a result of local surface boiling.

Maximum fuel temperature during the transient is 193°F at 54 seconds and then continues to decrease.

13.4.6 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:

- Primary system ruptures at some point below the core entrance so that gross removal of core coolant supply occurs.
- 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 13-12 as a fraction of the initial power. A power-peaking factor of 1.30 was assumed, which includes both the normal axial peaking and severe azimuthal skewing. The calculation was performed using the TRACG computer program. The nodal structure for the problem is adapted from earlier calculations using a version of the Transient Heat Transfer program (Reference 21) and is shown in Figure 13-13. Axial heat-transfer was neglected.

The peak fuel temperature and the volume-averaged graphite temperature are shown in Figure 13-14 and Figure 13-14 as a function of time after coolant loss. The fuel temperature reaches a maximum of about 626°F about 30 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 800°F at about 20 minutes. 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.

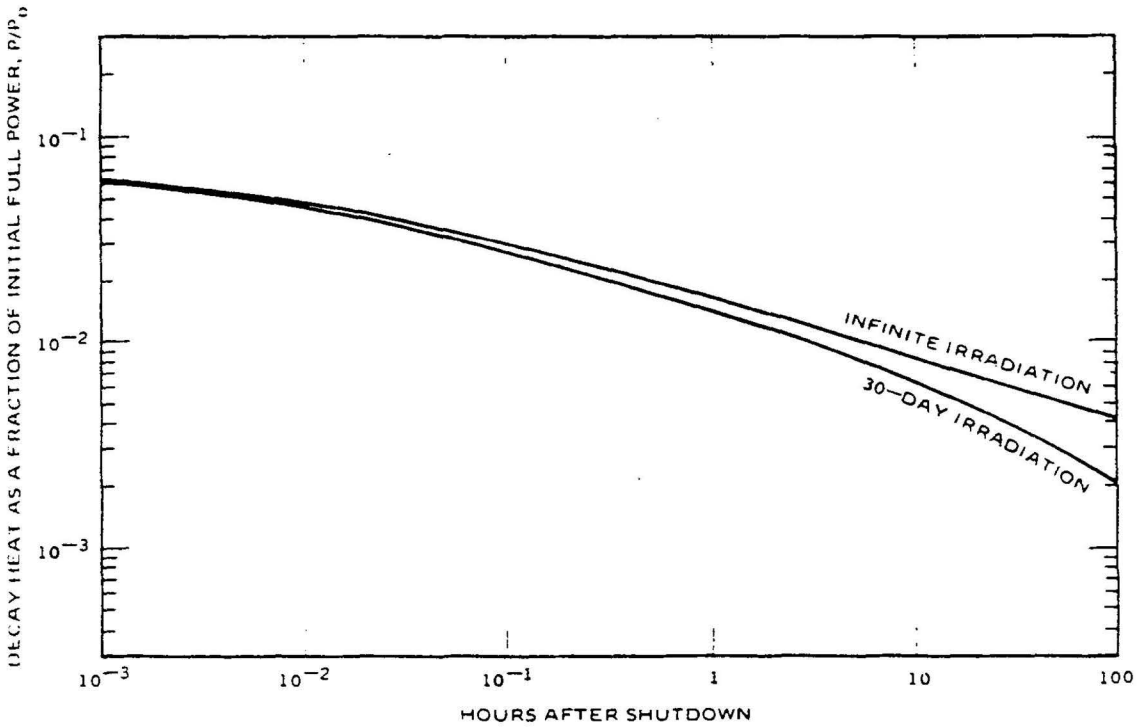


Figure 13-12 Decay Heat Rate

[redacted]

Figure 13-13 Node Structure Adapted for TRACG Analysis

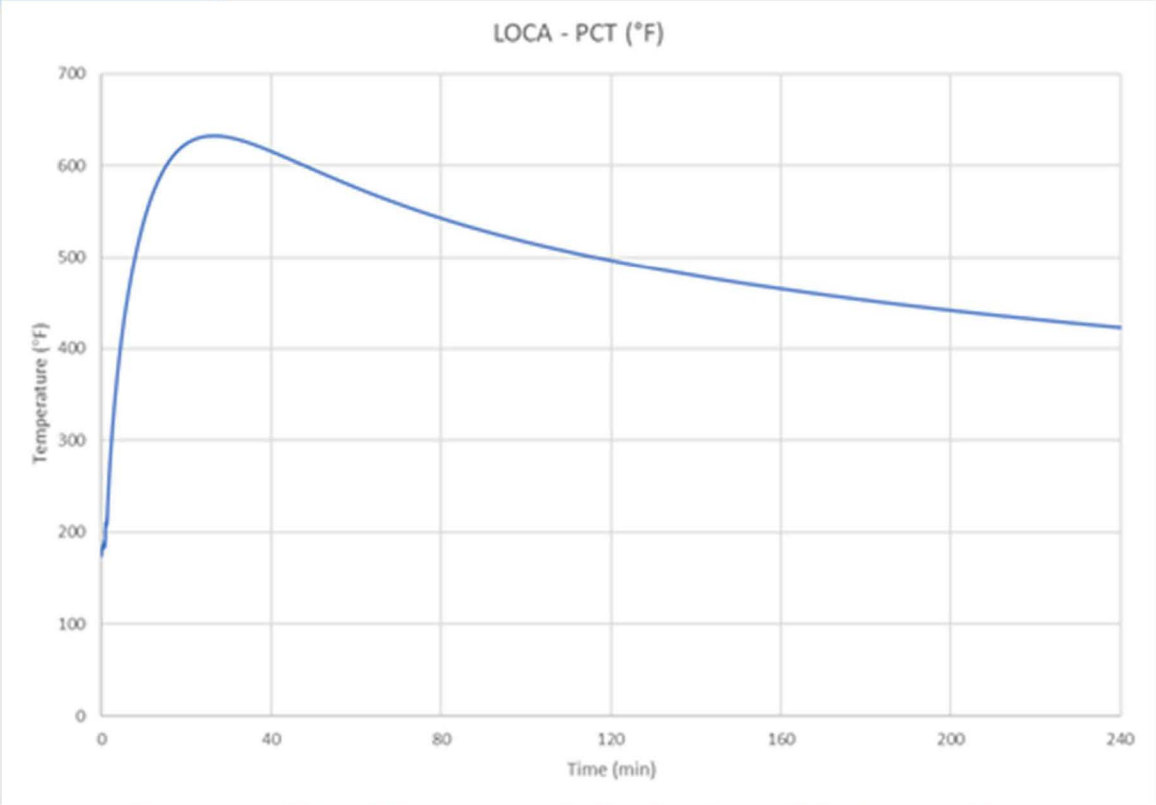


Figure 13-14 Fuel Temperature Following Loss of Coolant Accident

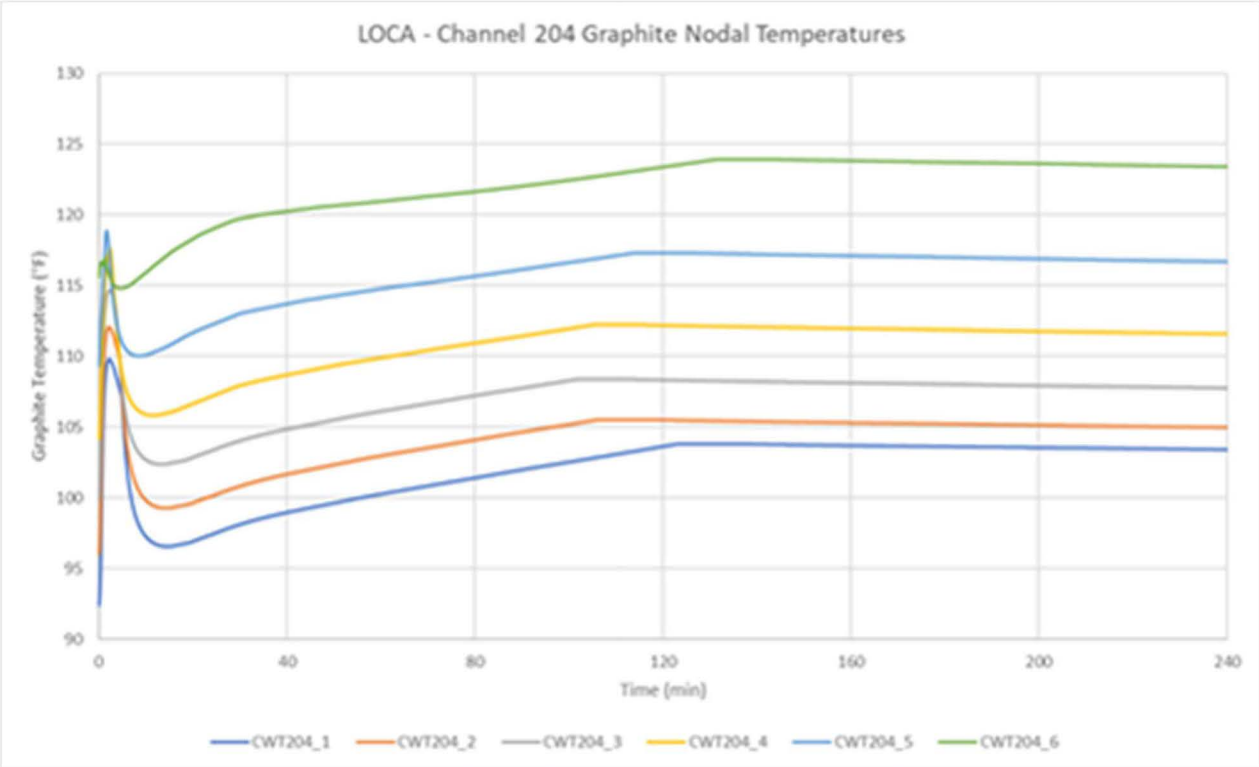


Figure 13-15 Graphite Heatup Following Loss of Coolant Accident

13.5 EXPERIMENT SAFETY ANALYSIS

13.5.1 Introduction

Descriptions of the NTR experiment safety programs and associated facilities, equipment, and procedures are discussed in Chapter 10. 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 Section 13.5.2. 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 13.5.3.

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.

13.5.2 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:

- Any interaction of an experiment with the reactor system that has potential for causing fission product release from the fuel.
- Any interaction that could adversely affect an engineered safety feature or control system feature designed to protect the public from fission product release.
- Inherent features of an experiment that could create beams, radiation fields, or unconfined radioactive materials.
- Potentially adverse interaction with concurrent experimental and operational activities.

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

- The physical conditions of the design and conduct of the experiment.
- The content of the material.
- The administrative controls employed to evaluate, authorize, and carry out the experiment.

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

13.5.2.1 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:

- Its potential reactivity worth.
- The rate of change of reactivity of unsecured experiments and movable experiments.

13.5.2.2 Thermal and Hydraulic Effects

An experiment will be evaluated to ensure its thermal limits are not exceeded 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.

13.5.2.3 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.

13.5.2.4 Material Content of Experiments

Certain kinds of materials which may be used in experiments possess properties which could have significant safety 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:

- Radioactive material
- Trace elements and impurities
- High cross section materials
- Highly reactive chemicals (explosives)

- Corrosive chemicals
- Radiation sensitive materials
- Flammable material
- Toxic material
- Cryogenic liquids
- Unknown materials

13.5.2.5 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 Area 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 Chapter 12.

Administrative controls applicable to all experiments are listed below:

1. An NTR Licensed Senior Reactor Operator must provide written approval of every experiment and must ensure that each person executing the experiment is knowledgeable in those procedures required to ensure safe conduct of the experiment.
2. Detailed written procedures must be provided for the use, or operation of, each experiment facility and each experiment type.
3. 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.
4. 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.

13.5.3 Consequences of Accidental Explosions

The facilities, equipment, and procedures used for experiment programs that involve explosive material are described in Chapter 10. 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.

13.5.3.1 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 n/cm^2-sec 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 uranium 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.

13.5.3.2 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 showed 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.

13.5.3.3 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 Section 13.4.3.

Therefore, the consequences would be less severe than those analyzed, which assumed 0.76$ step insertion both with and without scram.

13.5.4 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, approximately 60 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 that has not undergone a 10 CFR 50.59 review.

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.

13.5.4.1 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 Area 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.

13.5.4.2 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 power operated, 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. All manually operated mechanisms for moving an object into the reactor graphite pack shall be done with the knowledge and consent of the reactor operator at the controls of the reactor.

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.

13.5.4.3 Explosive and Flammable Material Lists

1. a) The maximum amounts of explosives (detonating and deflagrating, DOT Hazard Class/Divisions 1.1, 1.2, 1.3 and 1.4) permitted in the NTR facilities are as follows:

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

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

   iii. Set-up room: W = 25 pounds

   b) The maximum amounts of explosives allowed in the North room MSM (inclusive in the limit of 1.a.ii above) are as follows:

   1. for DOT Hazard Class Divisions 1.1, 1.2, and 1.3 (detonating): W = 2 pounds
   2. for DOT Hazard Class Division 1.4 (deflagrating): W = 4 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. With the exception of communication equipment utilizing low-energy electromagnetic waves in radiofrequencies, such as mobile phones and two-way radios used by reactor personnel, unshielded high-frequency generating equipment shall not be operated within 50 feet of any explosive device.

4. No explosive device shall be placed in a radiation field greater than 1×10^4 roentgens or consisting of greater than 3×10^{12} n/cm^2 thermal neutrons.

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.

13.5.4.4 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, 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.

13.6 EXPERIMENT DESIGN BASIS ACCIDENT

13.6.1 Introduction

The material quantity limits, 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 (Reference 35). This analysis specifically addresses the limits for a singly clad U-235 powder fueled capsules and shows the capability of the facility and site to accommodate a radioactive material release with no credit taken for filtration of the release by the NTR stack filter system.

13.6.2 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 all radioactive material into the immediate environment of the experiment or to the reactor 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 50 mg of U-235 powder in a singly encapsulated container.
    a. Dose consequences for doubly cladded or pellet forms of U-235 equal to or less than 50 mg are bounded by the results of this analysis.
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 established conditions would presumedly include the ventilation system being in operation; however, ventilation (filtration) is conservatively not credited in the analysis.
3. The release fractions of U-235 fuel and fission products to the environment are assigned as follows:

    Release from capsule to reactor cell: Powder / Pellet (%)

- U-235 100
- Noble Gas 100
- Iodine 100
- All Remaining Fission Products 100

Release from reactor cell to the environment:

- U-235 100
- Noble Gas 100
- Iodine 100
- All Remaining Fission Products 100

4. Dose Limits, total effective dose equivalent (TEDE):

2-hour Fence-Post Dose to Member of Public	0.1 rem, TEDE
Operator Dose, during 5-Minute Evacuation	0.5 rem, TEDE

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 (specifically, the reactor cell) exposure will result from the submersion in and inhalation of the isotopes released to the reactor cell for a period of 5 minutes during evacuation. The bases for this postulated exposure are as follows:

   a. It is assumed that a complete release of the experiment capsule contents to the restricted area will occur uniformly 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.

   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 alarm system and evacuation procedures.

   e. Assuming the operator is in the reactor cell where the accident occurs for the duration of the evacuation ensures the operator dose for this event bounds the

operator doses that would be received at all other locations inside or adjacent to the NTR for a 5-minute evacuation.

13.6.3 Calculation Method

ORIGEN2 Version 2.1 is used to calculate the fission products created during operation of the U-235 capsule, and the RADTRAD computer code (version 3.10) is used to calculate the TEDE dose resulting from exposure to the released radioactive materials. Decay and progeny have been accounted for in the RADTRAD dose consequence model. The inputs required for the evaluation of doses resulting from exposure to the released isotopes from a U-235 experiment in the NTR are:

1) Capsule operating power = 60 watts (for 50 mg U-235)
2) Capsule operating time = 1 day
3) Decay time after shutdown = 0 minutes (not credited)
4) ORIGEN2 cross-section library = BWRUE.lib (based on ENDF/B-V)
5) Meteorology type is Pasquill Type F with a wind speed of 1 m/sec (for the boundary dose)
6) Effective release height is at ground level
7) The breathing rate of the exposed subject is 350 cc/sec
8) The distance from the nearest boundary point to the NTR stack is 510 meters
9) Release to environment = 2 hours
10) Environmental release flow rate = 1000 cfm
11) Reactor cell airspace volume = 10,500 ft^3
12) Dose conversion factors from Federal Guidance Reports No. 11 and 12.
13) Inventory of nuclides released in the reactor cell:

Table 13-1 NTR EXPERIMENT DBA ISOTOPIC RELEASE TO REACTOR CELL

Nuclide	Curies	Nuclide	Curies	Nuclide	Curies	Nuclide	Curies
Kr-85	1.78E-05	Zr-97	1.84E+00	Te-131m	7.71E-02	La-140	3.11E-02
Kr-85m	6.20E-01	Nb-95	3.35E-04	Te-132	4.11E-01	La-141	2.84E+00
Kr-87	1.29E+00	Mo-99	6.71E-01	I-131	1.03E-01	La-142	2.86E+00
Kr-88	1.82E+00	Tc-99m	3.98E-01	I-132	3.69E-01	Ce-141	4.76E-02
Rb-86	1.50E-06	Ru-103	2.76E-02	I-133	1.82E+00	Ce-143	1.16E+00
Sr-89	3.23E-02	Ru-105	5.00E-01	I-134	3.83E+00	Ce-144	6.63E-03
Sr-90	1.87E-04	Ru-106	3.93E-04	I-135	2.90E+00	Pr-143	3.16E-02
Sr-91	2.43E+00	Rh-105	1.48E-01	Xe-133	1.25E-01	Nd-147	6.86E-02
Sr-92	3.00E+00	Sb-127	1.21E-02	Xe-135	7.67E-01	Np-239	1.74E-12
Y-90	4.99E-05	Sb-129	3.35E-01	Cs-134	2.05E-08	Pu-238	2.43E-14
Y-91	1.78E-02	Te-127	5.44E-03	Cs-136	1.84E-04	Pu-239	4.04E-20
Y-92	2.92E+00	Te-127m	5.37E-06	Cs-137	1.94E-04	Pu-240	1.62E-22
Y-93	2.60E+00	Te-129	2.98E-01	Ba-139	3.14E+00	Pu-241	3.39E-31
Zr-95	3.43E-02	Te-129m	8.08E-04	Ba-140	1.65E-01	U-235	1.08E-07

The source term for this evaluation is calculated by modeling the irradiation of a capsule containing 50 mg of U-235 at a capsule operating power level of 60 W for a period of 1 day simulated in an ORIGEN2 model. The model results include activities and masses for several hundred isotopes (>700 nuclides), however, many of these nuclides are not important contributors to dose consequences or are not present in sufficient quantity to impact the dose consequences of the NTR experiment DBA.

Thus, this analysis considers the 60 dose important isotopes used for offsite dose consequence evaluations for commercial nuclear power plants that use U-235 enriched fuel. This set of isotopes in commonly referred to as the Alternative Source Term (AST) nuclide set. There are 55 of the AST isotopes present in the ORIGEN2 NTR experiment DBA results. Along with these 55 dose important isotopes, the remaining unburned U-235 is added to the release source term. The resulting source term comprised of 56 isotopes is shown in Table 13-1.

While these isotopes are known to be the main dose contributors for releases involving irradiated U-235, there may be a contribution from the remaining isotopes not being modeled. To account for this contribution a safety factor of 20% is added to the final dose consequences.

13.6.4 Results

The results generated by the RADTRAD model shown in Table 13-2 confirm the material quantity and operating limits established for the NTR experiment DBA results in dose

consequences at the unrestricted area boundary and the restricted area that are below the dose limits. Dose consequences for doubly cladded or pellet forms of U-235 equal to or less than 50 mg are bounded by the results of this analysis.

Table 13-2 NTR EXPERIMENT DESIGN BASIS ACCIDENT DOSES

Exposure Location and Duration	Dose Consequences, TEDE	Dose Limits, TEDE
Unrestricted Area Boundary with 2-hour Exposure	66.2 mrem (0.662 mSv)	100 mrem (1 mSv)
Restricted Area (NTR Reactor Cell) with 5-Minute Exposure	470.0 mrem (4.70 mSv)	500 mrem (5 mSv)

13.7 REACTOR SAFETY LIMITS

13.7.1 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.

13.7.2 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

$$\text{DNBR} = \frac{DNB\ surface\ heat\ flux}{operating\ surface\ heat\ flux}$$

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 Section 13.4 reach a DNB heat flux with the core coolant significantly subcooled.

13.7.3 Analysis for Development of Safety Limits

13.7.3.1 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 Macbeth, (Reference 25), developed an empirical correlation of experimental data which presents the critical heat flux (Reference 25) 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

$$\text{DNB} = \frac{(\text{H}_{\text{fg}})(\text{G} \times 10^{-6})^{0.5}}{(135)}(1 - \text{X}_{\text{max}})$$

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 = $\frac{mass\ of\ vapor}{mass\ of\ vapor\ and\ mass\ of\ liquid}$

The critical heat flux is calculated for the hottest location of the hottest channel.

13.7.3.2 Transient Critical Heat Flux Relationship

The DNB correlation used to evaluate the reactor safety under transient conditions must be applicable to subcooled boiling. Macbeth developed the following empirical correlation for the DNB critical heat flux under subcooled, low-pressure, low-flow conditions:

$$\mathrm{DNB} = (0.247)(\mathrm{H_{fg}})(\frac{\rho_v}{\rho_\ell})^{0.024}(\frac{\mathrm{DG}}{\mathrm{L}}) + (0.00213)(\rho_\ell)^{1/2}(\mathrm{H_{fg}})^{1/6}(10^6)^{1/3}(\frac{\mathrm{DG}}{\mathrm{L}})^{2/3}\Delta\mathrm{Hi}$$

where

H_{fg} = latent heat of vaporization (Btu/lb)

ρ_v = density of the vapor (lb/ft^3)

ρ_ℓ = 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 Section 13.4, the hottest surface location within the NTR core will attain DNB above a heat flux of 600,000 Btu/h-ft^2.

The postulated accidents were originally analyzed using a DNB heat flux of 450,000 Btu/h-ft^2).

13.7.3.3 Thermal Hydraulic Computer Model

The computer model CORLOOP (Reference 27) 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 13-16; the circulation loop is illustrated in Figure 13-17. 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 (Reference 28).

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.

In the current updated safety analysis report, the analyses of select accidents and postulated transients are reevaluated using TRACG code. TRACG is a two-phase, non-equilibrium code capable of simulating the thermal-hydraulics response of the reactor with kinetics feedback. The TRACG code (Reference 30) is the GEH proprietary version of the Transient Reactor Analysis Code (TRAC), a system code widely used for boiling water reactor safety analyses. The code capabilities include 3D and point kinetics models, a multi-dimensional, two-fluid model for the reactor thermal hydraulics, and an implicit integration scheme for numerical calculations. The code was reviewed and approved as part of the application methodologies for Anticipated Operational Occurrence (AOO), stability, Anticipated Transient Without Scram (ATWS), and Loss of Coolant Accident (LOCA) analyses. Its applicability range extend from full operating pressure of boiling water reactors to atmospheric pressures (Reference 31)

13.7.4 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 13-18 shows, as one would expect, that the departure from nucleate boiling ratio approaches unity as the reactor power increases. Figure 13-19 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 Figure 13-20 and Figure 13-21. 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.

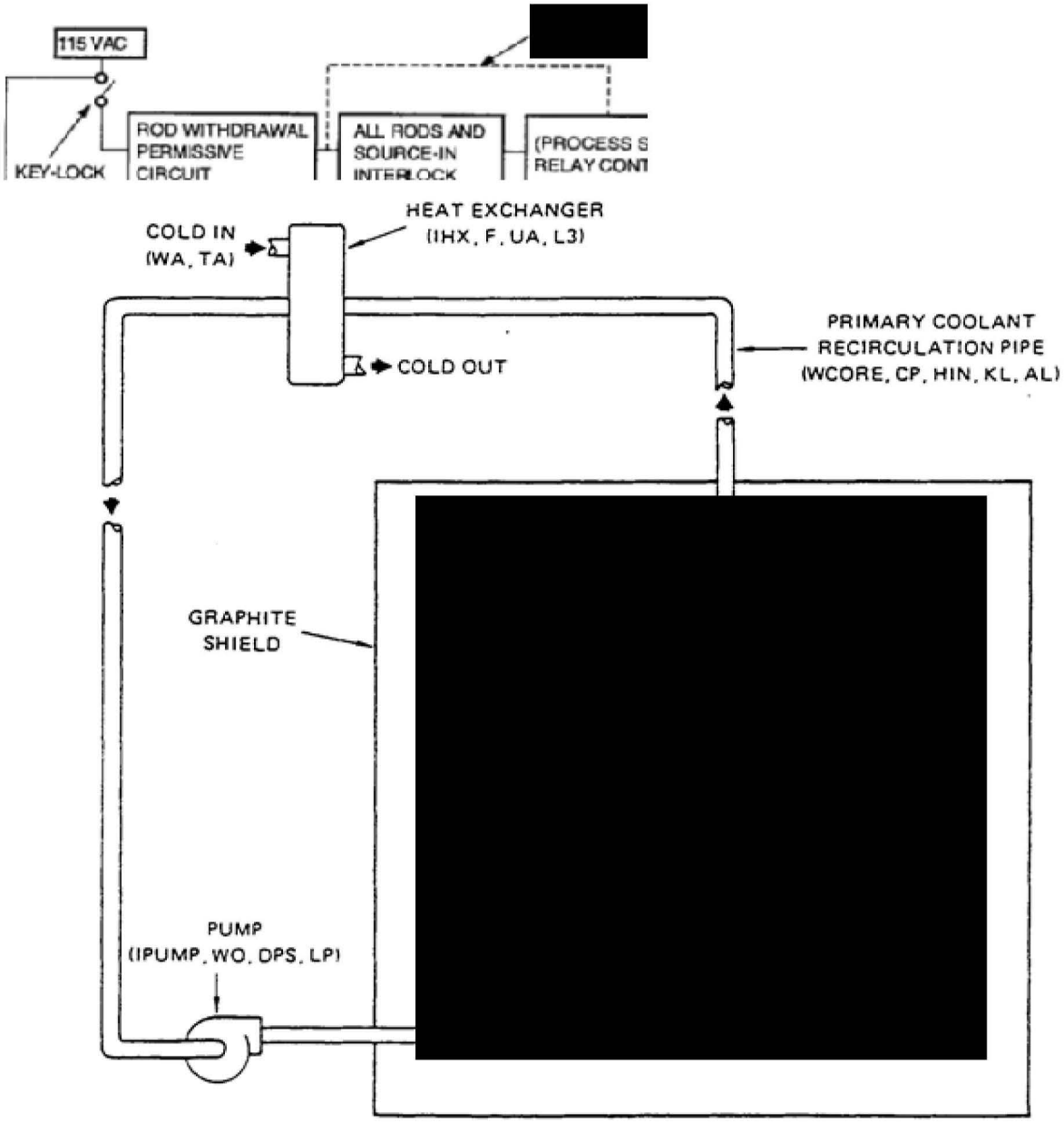


Figure 13-16 Multi-Channel Core Model of NTR (CORLOOP)

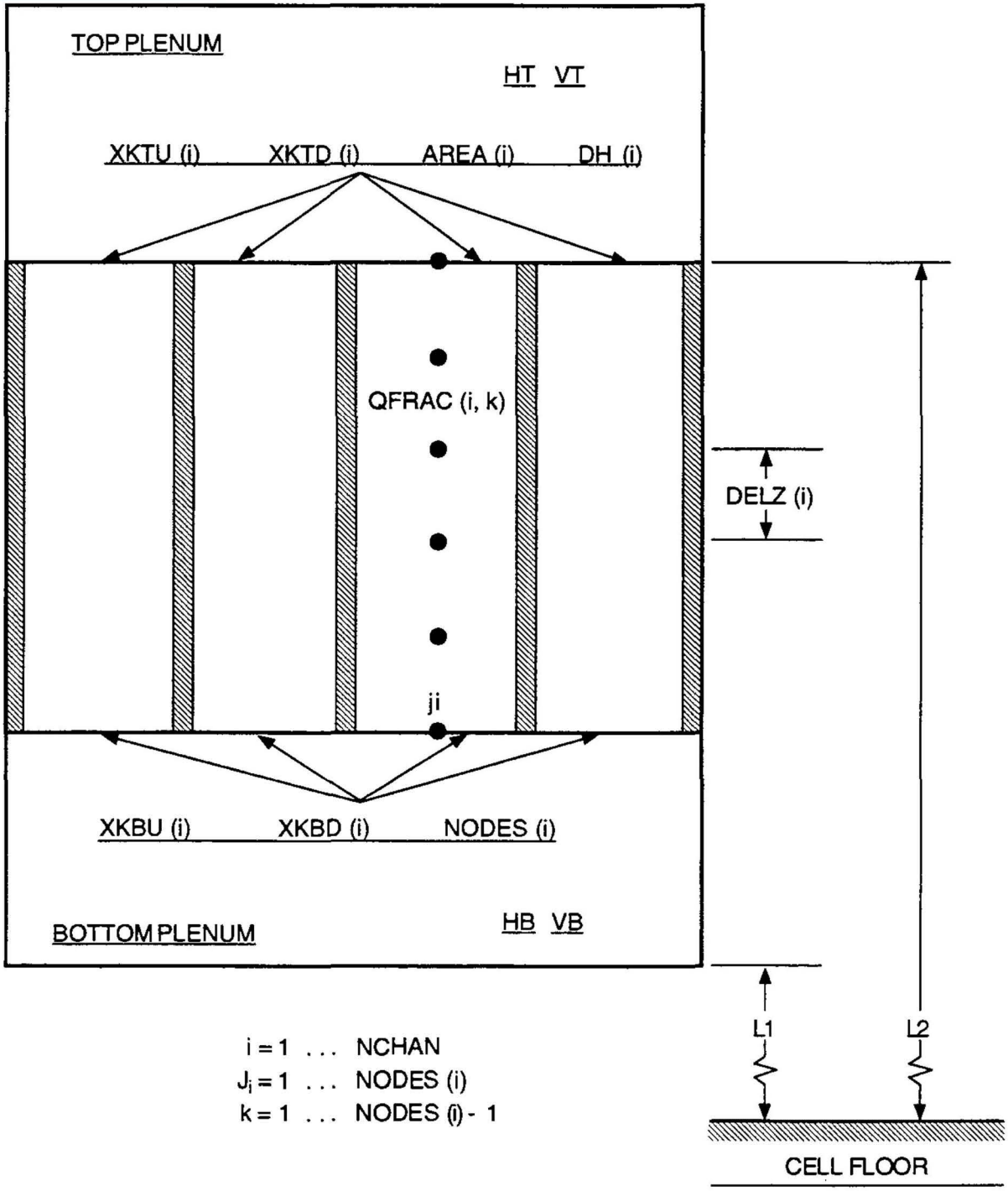


Figure 13-17 Schematic Diagram of the NTR Circulation Loop Model (CORLOOP)

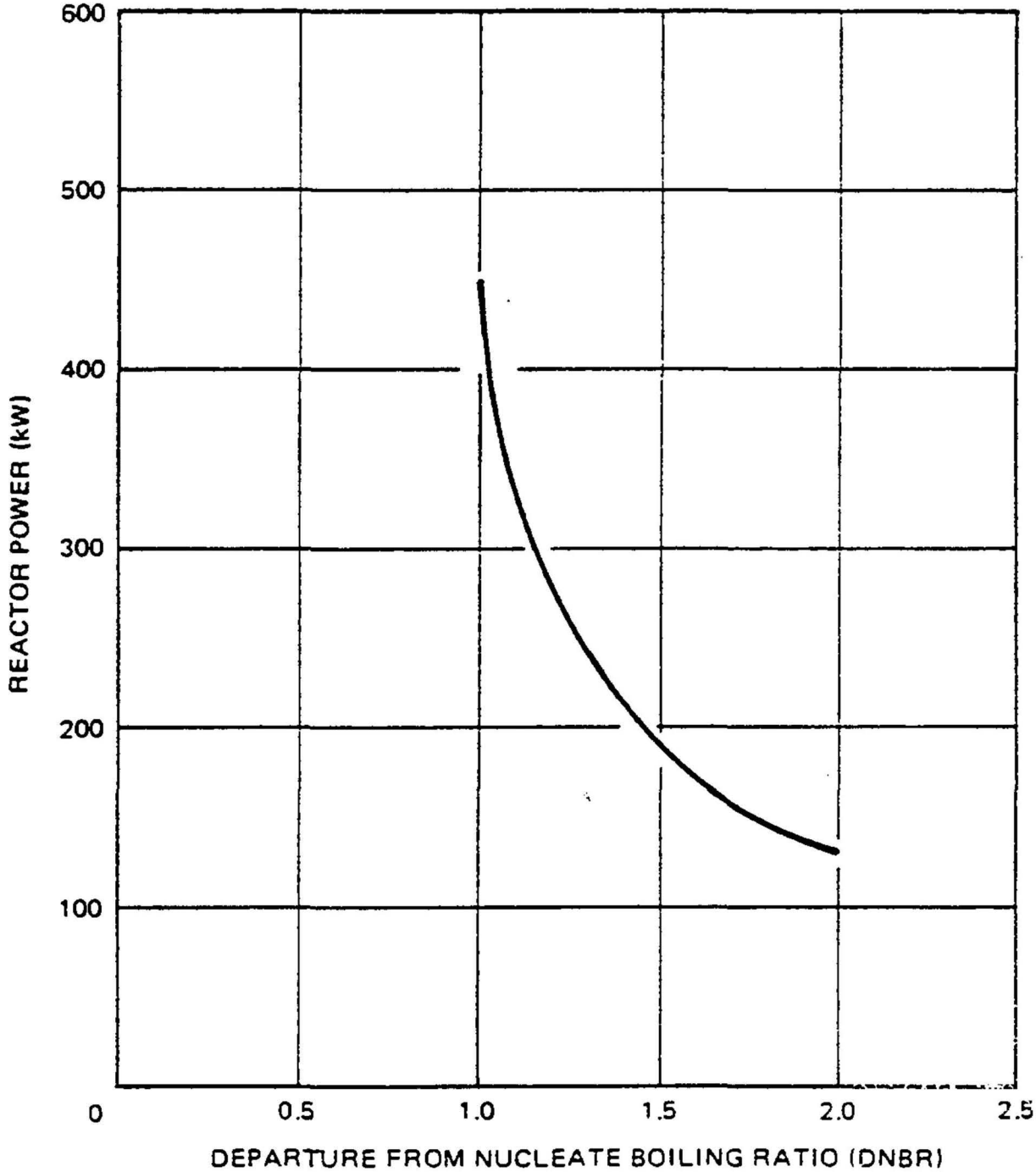


Figure 13-18 Reactor Power Versus DNBR = Depart from Nucleate Boiling Ratio

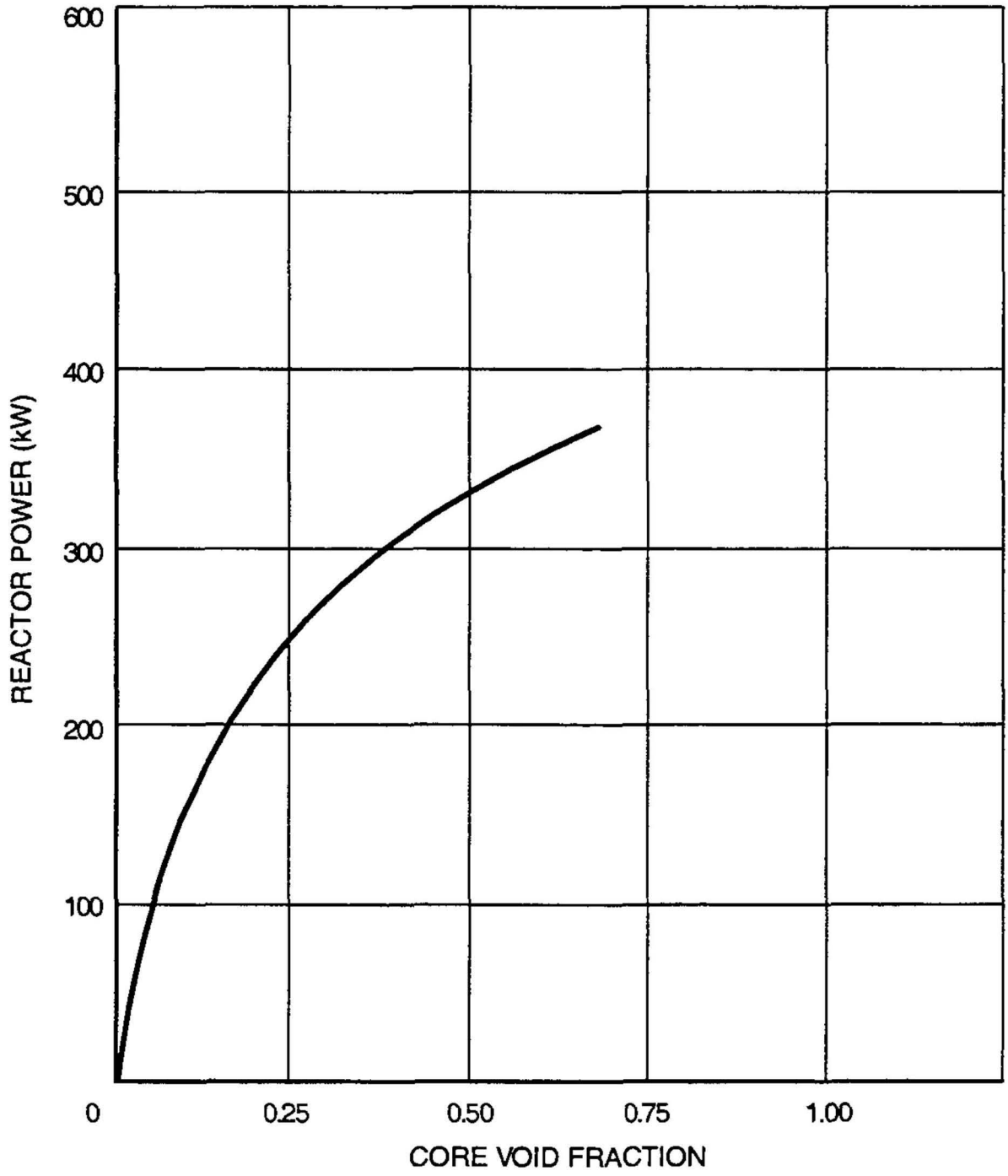


Figure 13-19 Reactor Power Versus Core Void Fraction

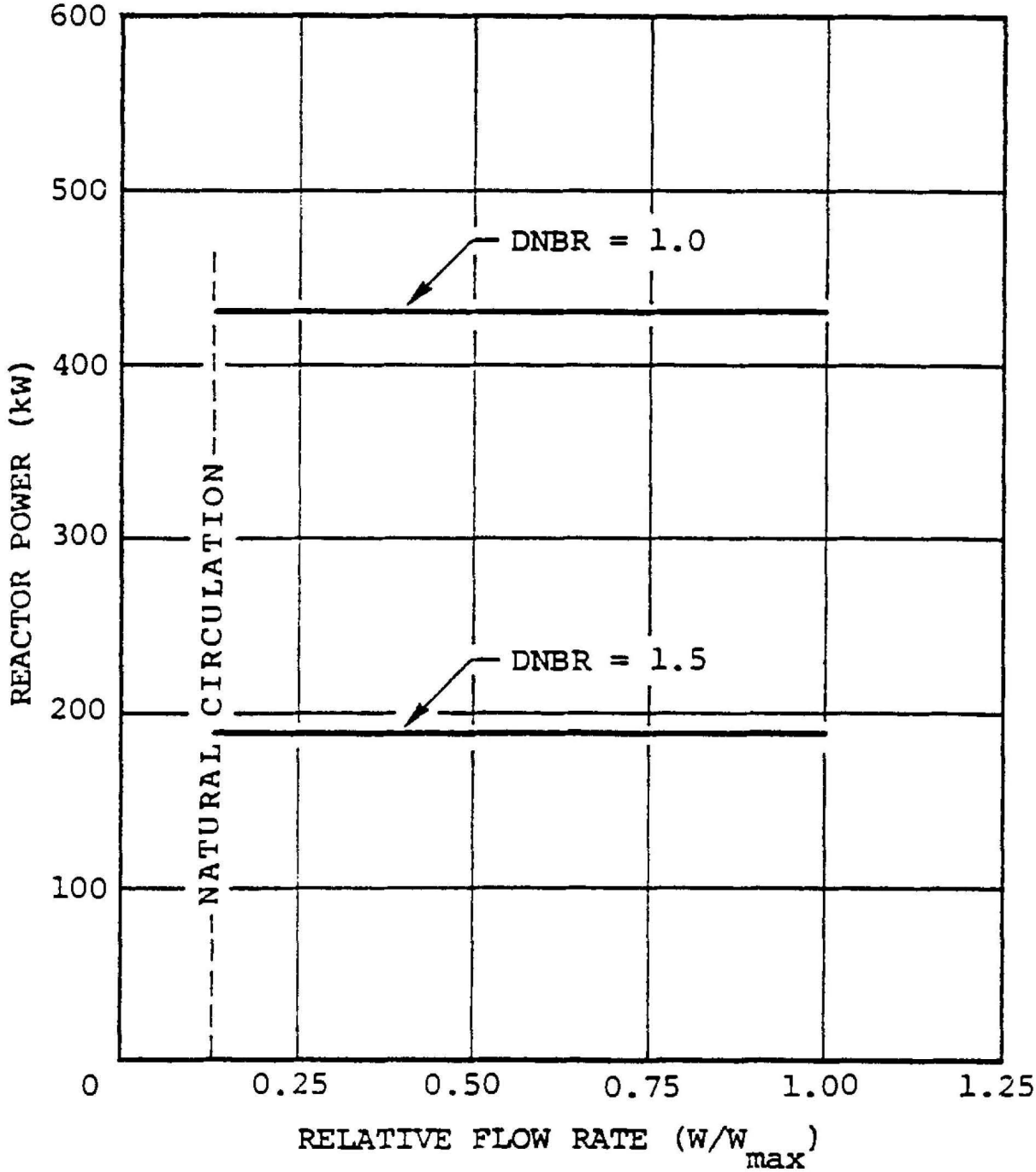


Figure 13-20 Reactor Power Versus Relative Flow Rate

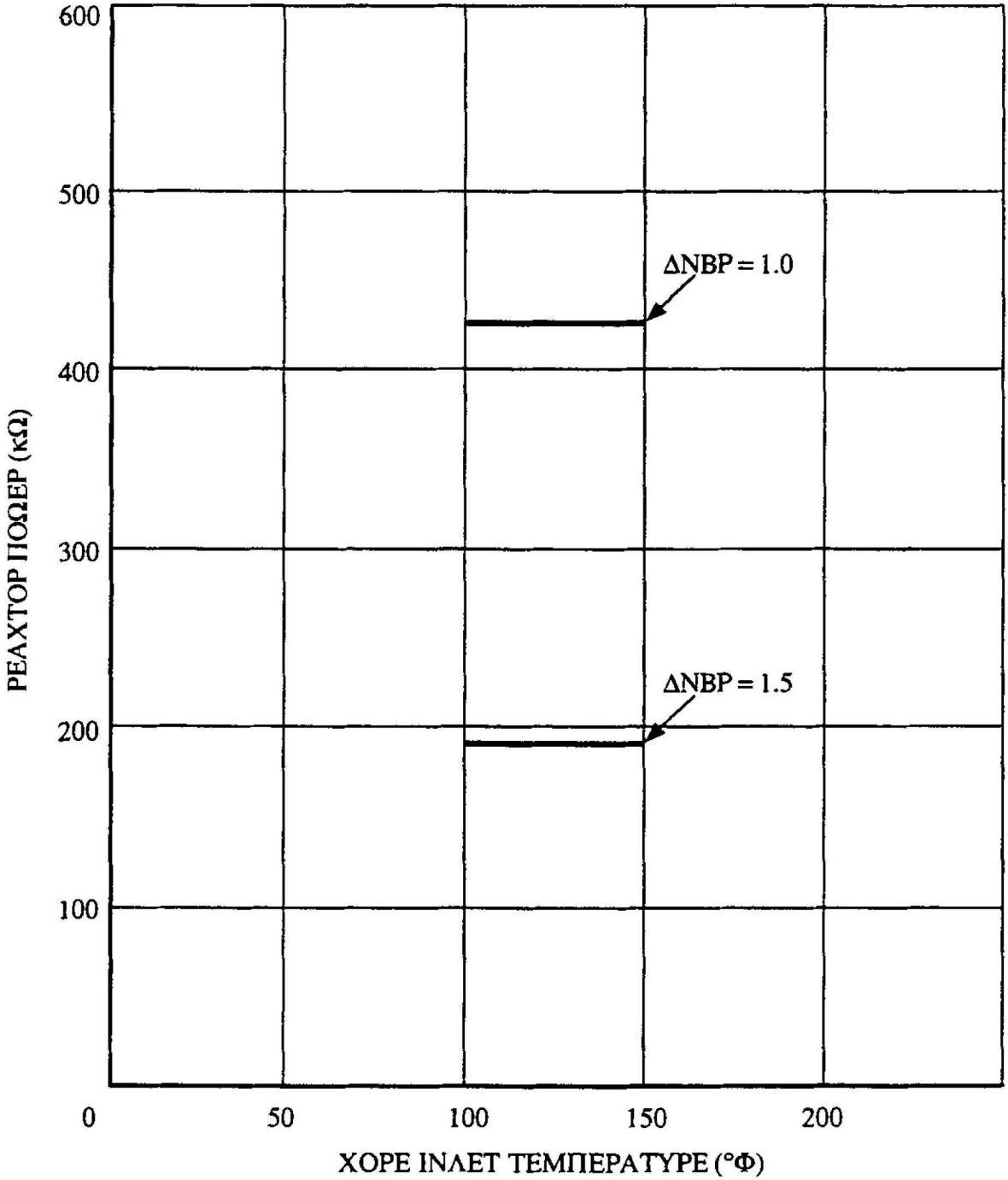


Figure 13-21 Reactor Power Versus Core Inlet Temperature

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 13-20, 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 Figure 13-20 and Figure 13-22.

The curves presented in Figure 13-20 and Figure 13-22 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 base transient analysis presented in Section 13.4, which required a reactor scram, were originally 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 reevaluated using the more realistic thermal-hydraulics tool predicting higher critical heat flux, confirming the adequacy of the thermal limits set for operation.

13.7.5 Instrument Uncertainties

The instrument uncertainties are presented in Table 13-3 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.

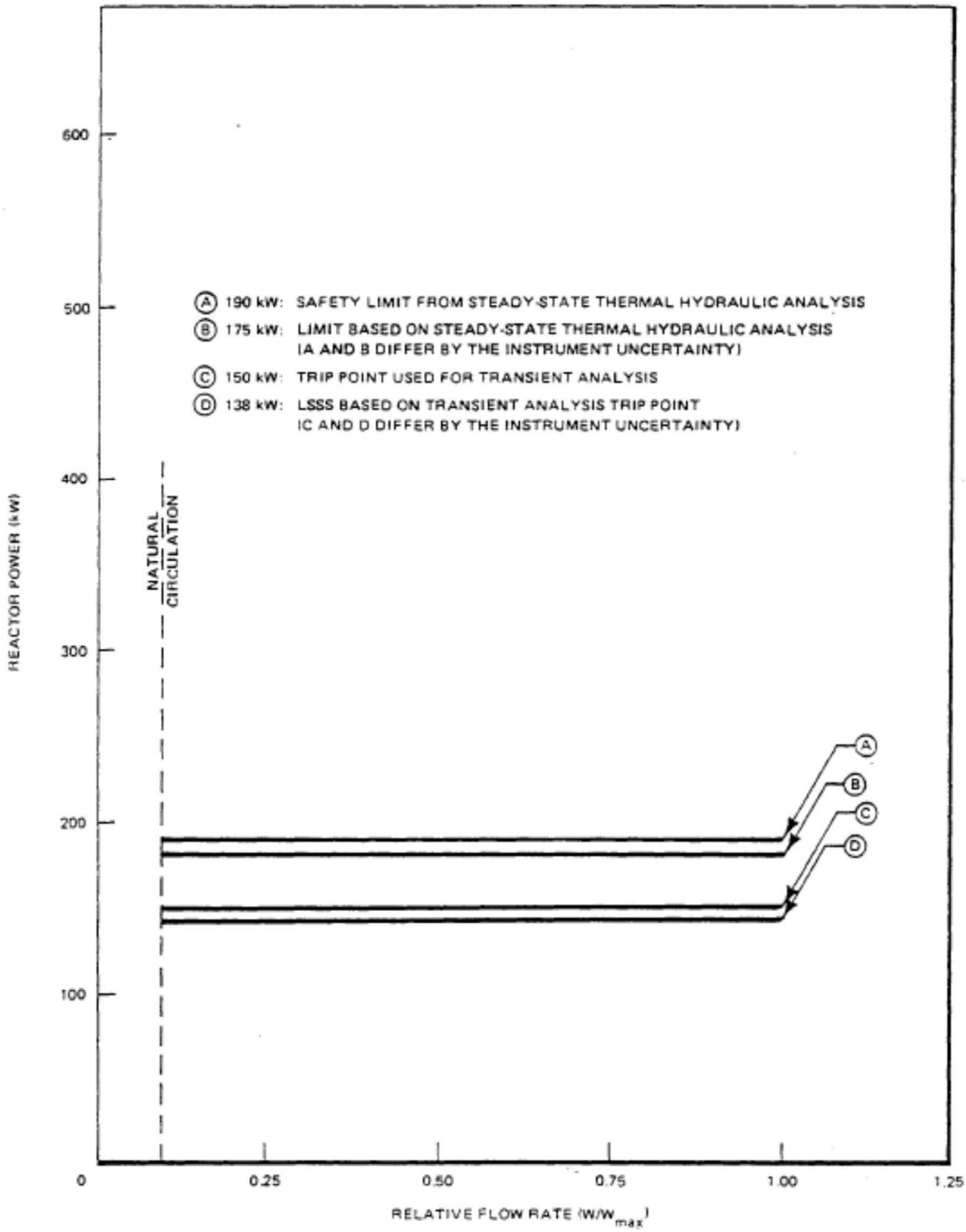


Figure 13-22 LSSS and Safety Limit for Reactor Power in Terms of Relative Core Flow Rate

Table 13-3 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%

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 13-22. The limiting safety system setting for the reactor power is 125 kW over the entire range of core flow conditions, including natural circulation. The value of 125 kW is a conservative setpoint well below 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 13-22 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 13-22 shows the scram trip point used in the transient analysis. Curve D, the LSSS curve, represents Curve C adjusted to account for instrument uncertainties.

14 TECHNICAL SPECIFICATIONS

Technical Specifications have been developed for the NTR which follow the format of the 2013 revision to American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1. Operation of the reactor within the limits of the Technical Specifications will not result in offsite radiation exposure in excess of 10 CFR 20.1201, 20.1301, and 20.1101(d) limits. Operation within the Technical Specifications also limits the likelihood and consequences of malfunctions and assures the health and safety of the on-site personnel and the public, and protection of the environment.

15 FINANCIAL QUALIFICATIONS

15.1 FINANCIAL ABILITY TO CONSTRUCT A NON-POWER REACTOR

Not Applicable

15.2 FINANCIAL ABILITY TO OPERATE A NON-POWER REACTOR

The actual costs and estimates of costs to operate the NTR for the first five years of the renewal period are considered by GE Hitachi Nuclear Energy (GEH) to be proprietary. However, based on approximately 60 years of experience in operating the NTR at VNC, costs of NTR operation are well known and understood. GEH has significant assets and is capable of assuming total operating costs for the NTR for the duration of the license renewal period.

Current and anticipated sources of funding for the NTR, would come from the sales of services to various customers in the areas of neutron radiography, irradiation of test and research materials, and reactivity testing. The sales of these services cover all of the NTR operating costs, excluding certain landlord type costs, which are required for operations of the site as a whole, and the expense of which is not directly attributed to the NTR. These landlord costs include cleaning, landscape maintenance, utilities, facilities maintenance, and security.

15.3 FINANCIAL ABILITY TO DECOMMISSION THE FACILITY

GEH provides appropriate financial assurance instruments to demonstrate that sufficient funds will be available when needed for required decommissioning activities. In 2018, GEH began using a payment surety bond for a prescribed amount based on a site-specific NTR facility decommissioning cost estimate. The surety bond amount is reviewed annually, and a revised bond is submitted to NRC as necessary.

16 OPERATING EXPERIENCE

All reactor components of the NTR have been used exclusively in the NTR since initial criticality on November 15, 1957, with even the uranium-clad, uranium-aluminum alloy reactor fuel being manufactured on site exclusively for use in the NTR. The reactor was originally 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 irradiation or exponential experiments, and (3) provide a sensitive device for measuring reactivity. Since initial criticality, the NTR has performed innumerable experiments, sample irradiations (including rocks from the moon) reactivity measurements, sensor calibrations and uranium enrichment analyses. The NTR is now used predominantly for performing neutron radiography. The first neutron radiograph in the NTR was performed On August 30, 1966.

In order to accommodate all possible types of experiments conceived by GE, the original bounding for the facility was established using an extremely conservative fueled experiment involving the rupture of an irradiated plutonium capsule. Design of the facility followed this experiment by incorporating robust Engineered Safety Features that include confinement and ventilation systems that more closely resemble those of a large power reactor rather than those typical of a Class 2 research reactor.

The NTR is an easy to operate and easy to maintain facility. It is a low temperature, low pressure, low heat reactor so components are not unduly stressed. The primary system is constructed of aluminum and stainless-steel components and the primary coolant system is maintained at a high purity, so corrosion is not an accelerated concern. The reactor is also very accessible so that control rod and safety rod drives may be inspected and maintained regularly. These inspections and tests have demonstrated that the NTR can be operated safely and that components with degraded performance may be detected and replaced.

16.1 REACTOR FUEL

The fuel used in the NTR has an excellent performance history and does not release large amounts of fission gas without melting. Corrosion of the aluminum in water is minimized when the pH of the water is 6.5.

The reactor primary coolant pH is maintained between 4.8 and 8.7 by maintaining the water purity below 10 µS/cm. High specific conductivity can be tolerated for short time durations during unusual circumstances.

The effect of aging of fuel was evaluated in 2020 and both the condition of fuel and the adequacy of monitoring fuel aging were determined to be adequate to support ongoing operation. Fuel cladding was concluded to be half of its original thickness after more than 60 years of operation and, while at least one fuel disc has a cladding breach, monitoring of Sr-91 and Sr-92 indicates the amount of fuel in the coolant has not been increasing at a significant rate over the last decade.

Chronic fuel degradation is monitored by primary coolant water conductivity and periodic sampling for Sr-91 and Sr-92. Significant degradation of the fuel cladding would be indicated in increased dose rates and escaped fission gasses by local radiation monitors and the stack effluent monitor.

16.2 SAFETY RODS

The safety rods have an exceptional performance history. The rods are accessible for inspection and testing. The tests include scram times, low-current magnet separation, rod withdrawal time, limit switches and interlocks, residual magnetism of the electromagnets, sliding friction, and spring force. Any component of the Safety Rods may be replaced if required.

16.3 CONTROL RODS

The Control Rods have also had an exceptional performance history. The rods are accessible for inspection and testing. The tests include limit switches and interlocks, automatic insertion, rate of withdrawal, and position indication. Any component of the Control Rods may be replaced if required.

16.4 RADIATION AREA MONITORS (RAM)

The RAMs are accessible for periodic inspection, calibration, and testing. In March 2020, the entire radiation monitoring system was replaced with a digital system to increase reliability and eliminate installed check sources.

16.5 CONCLUSION

The NTR is a simple, compact, accessible reactor. This is evidenced by the replacement of the primary; core can which occurred in 1976. At that time, the control rods, safety rods, startup sources and reactor fuel were removed. Most of the graphite blocks were relocated and the aluminum core can and fuel reel assembly were replaced with new units and the reactor was reassembled. The extensive surveillance, testing and calibration program has resulted in a facility with an outstanding record of safe operation. Degradation of performance is evident when it occurs, and components replaced easily; assuring the NTR will continue to be a safe facility.