NUREG–1135
Supplement No. 1

Safety Evaluation Report

related to the construction permit and operating license for the research reactor at The University of Texas

Docket No. 50–602

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

January 1992

ABSTRACT

The Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission (NRC) has prepared Supplement 1 to NUREG-1135, "Safety Evaluation Report Related to the Construction Permit and Operating License for the Research Reactor at the University of Texas" (SER) May 1985. The reactor facility is owned by The University of Texas at Austin (UT, the applicant) and is located at the university's Balcones Research Center in Austin, Texas. This supplement to the SER (SSER) describes the changes to the reactor facility design from the description in the SER. The SER and SSER together reflect the facility as built. The SSER also documents the reviews that the NRC has completed regarding the applicant's emergency plan, security plan, and technical specifications that were identified as open in the SER.

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1 INTRODUCTION

The Nuclear Regulatory Commission (NRC, staff) issued NUREG-1135, "Safety Evaluation Report related to the Construction Permit and Operating License for the Research Reactor at the University of Texas" (SER), in May 1985 regarding the application by the University of Texas at Austin (UT, applicant) to receive a construction permit to construct and a 20-year license to operate a research reactor at power levels up to 1100 kilowatts (thermal), and in the pulse mode, with reactivity insertions not to exceed 2.2 percent $\Delta k/k$.

The staff is issuing Supplement 1 to the SER (SSER) to provide detail on changes to the reactor facility design from the description in the SER. The SER and SSER reflect the facility as built. The design changes made during the construction of the facility do not affect the original conclusions by the staff in the SER that the facility can be constructed and operated without endangering the health and safety of the public. The SSER also provides the evaluation and conclusions of the staff regarding the facility emergency plan, security plan and technical specifications, which were identified in the SER as open items requiring additional information from the applicant to close.

UT possesses an NRC-licensed TRIGA Mark I research reactor located on the main campus of UT (Facility Operating License No. R-92, Docket 50-192). On April 29, 1988, operations ended at the TRIGA Mark I research reactor. On March 9, 1987, the Commission issued an Order authorizing UT to dismantle the Mark I research reactor. UT will dismantle the reactor after it transfers the fuel to the new reactor facility and makes the new facility operational. In August 1991, UT transferred the fuel to the new reactor under the UT Special Nuclear Materials license (SNM-180).

The staff performed the review of the construction of the facility and closed the open items upon reviewing additional information provided by the applicant and the results of the Commission's inspection program at the facility. This material is available for review at the Commission's Public Document Room at 2121 L Street, N.W., Washington, D.C. 20555. Material regarding the physical security plan is protected from public disclosure under 10 CFR 2.790(d)(1).

The staff assigned the same number to each of the following sections as was assigned to the corresponding SER section that is being updated, and provided the discussions to supplement and not to replace the material in the SER unless otherwise noted. The appendix contains errata to the SER.

This SSER was prepared by A. Adams, Jr., Project Manager, Division of Advanced Reactors and Special Projects, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission. Major contributors to the review include the project manager, J. Hmama of NRC, and R. Carter, R. Carpenter, and P. Napper of the Idaho National Engineering Laboratory (INEL) under contract to the NRC.

The applicant has reexamined the amount of contained uranium-235 required for operating the reactor and has amended the original request to increase the authorized amount from 5800 g to 5831 g. The increase included foils and reference standards used in connection with operation of the reactor.

1.1 Summary and Conclusions of Principal Safety Considerations

(5) The applicant submitted the final version of its Technical Specifications by letter of February 12, 1991. The applicant's Technical Specifications, which provide limits controlling the operation of the facility, provide a high degree of assurance that the facility will be operated safely and reliably.

(7) The applicant submitted its updated physical security plan by letter of August 13, 1990. The applicant's program for providing for the physical protection of the facility and its special nuclear material complies with the requirements of 10 CFR Part 73.

(8) On November 19, 1990, the applicant submitted by letter an updated version of the Operator Requalification Plan. The plan update was required to maintain compliance with 10 CFR Part 55 which had been revised since the original plan was submitted and reviewed. The applicant's procedures for training operators and the plan for operator requalification are acceptable. The plan gives reasonable assurance that the reactor facility will be operated with competence.

(9) The applicant submitted by letter of November 21, 1990, as supplemented on April 15, 1991, a revision of the quality assurance (QA) program that complies with the regulations (10 CFR 50.34) regarding the overall QA program for research reactors.

(10) The applicant submitted the final version of the Emergency Plan by letter of January 11, 1991 as supplemented on April 15, 1991. The applicant's Emergency Plan provides reasonable assurance that the applicant is prepared to assess and respond to emergency events.

1.2 Reactor Description

The reactor core typically contains 86 fuel elements. The operational and experimental requirements may require the applicant to vary the number of fuel elements. The elements are assembled in hexagonal rings in the reactor, not concentric rings as stated in the SER. Three of the reactor control rods have fuel followers. The applicant had planned to reuse the reactor bridge assembly from the original reactor on the main campus, instead, the applicant has installed a new reactor bridge.

1.5 Summary of Open Items

Additional information submitted by the applicant has enabled the staff to review all open items in the SER to find acceptable the proposed resolutions, and thus to close these items. These open items identified in the SER were (1) the emergency plan (Section 13.3), (2) the physical security plan (Section 13.7), and (3) the facility technical specifications (Section 15). The staff included the results of this review in the sections of this SSER corresponding to the sections of the SER.

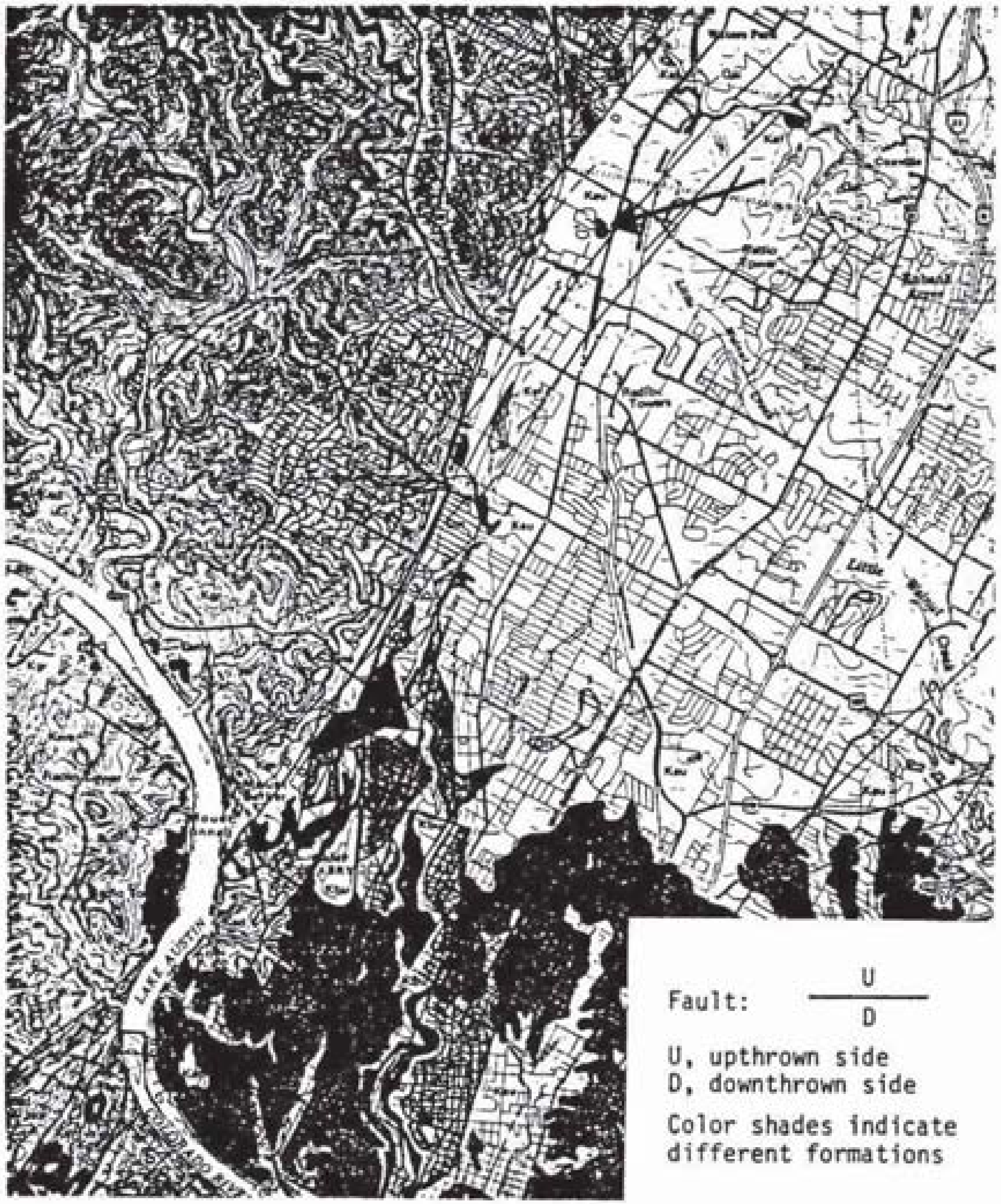
2 SITE CHARACTERISTICS

2.2 Demography

The 1990 census project estimated the population of Austin to be 465,000.

2.5 Geology

Replace Figure 2.3 with the revised figure. The revised figure shows the correct location of the Nuclear Engineering Teaching Laboratory (NETL).



Fault: $\frac{U}{D}$

U, upthrown side
D, downthrown side

Color shades indicate different formations

BALCONES FAULT ZONE

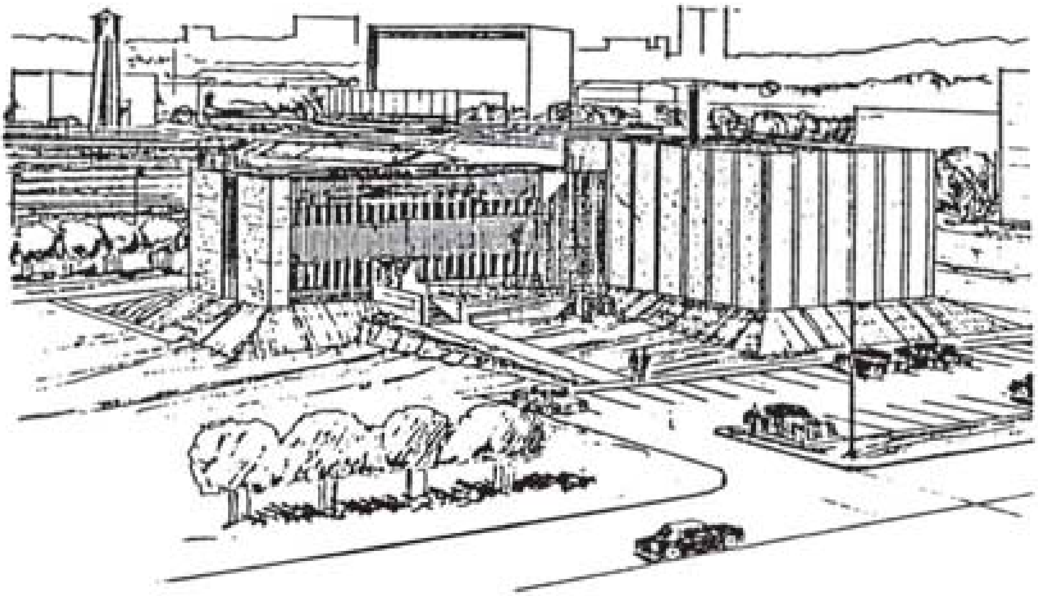
Figure 2.3

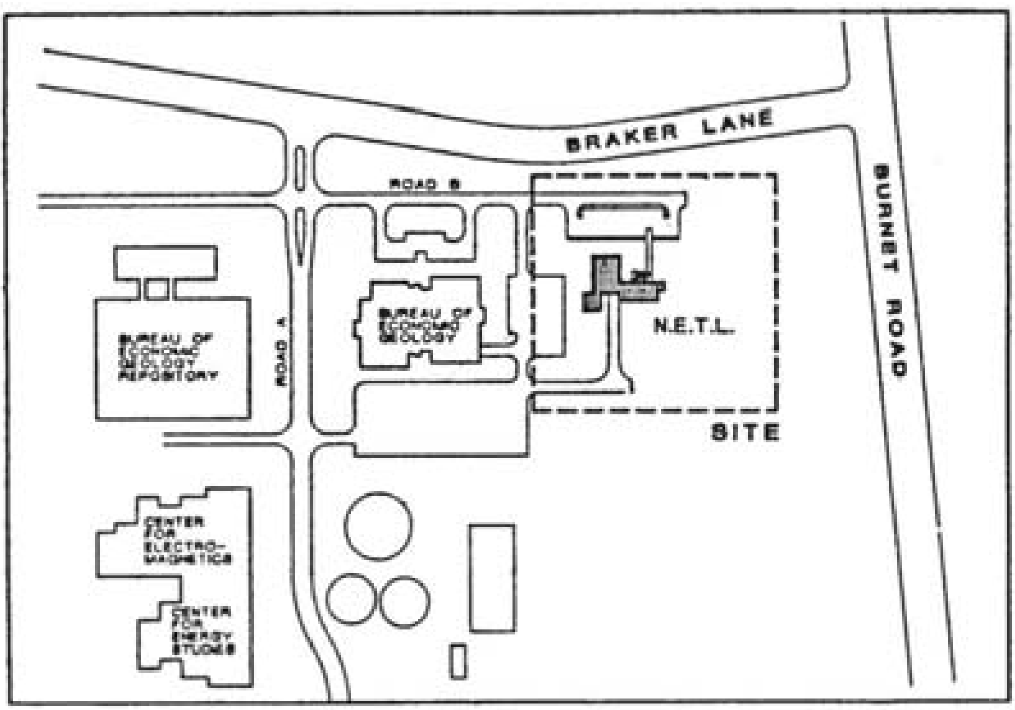
3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

Replace Figures 3.1 to 3.4 with new Figures 3.1 to 3.6. These figures reflect the as-built facility. The changes in the facility are primarily limited to the location and orientation of rooms. The staff concludes that these changes do not affect the reactor safety analysis.

3.3 Seismically Induced Reactor Damage

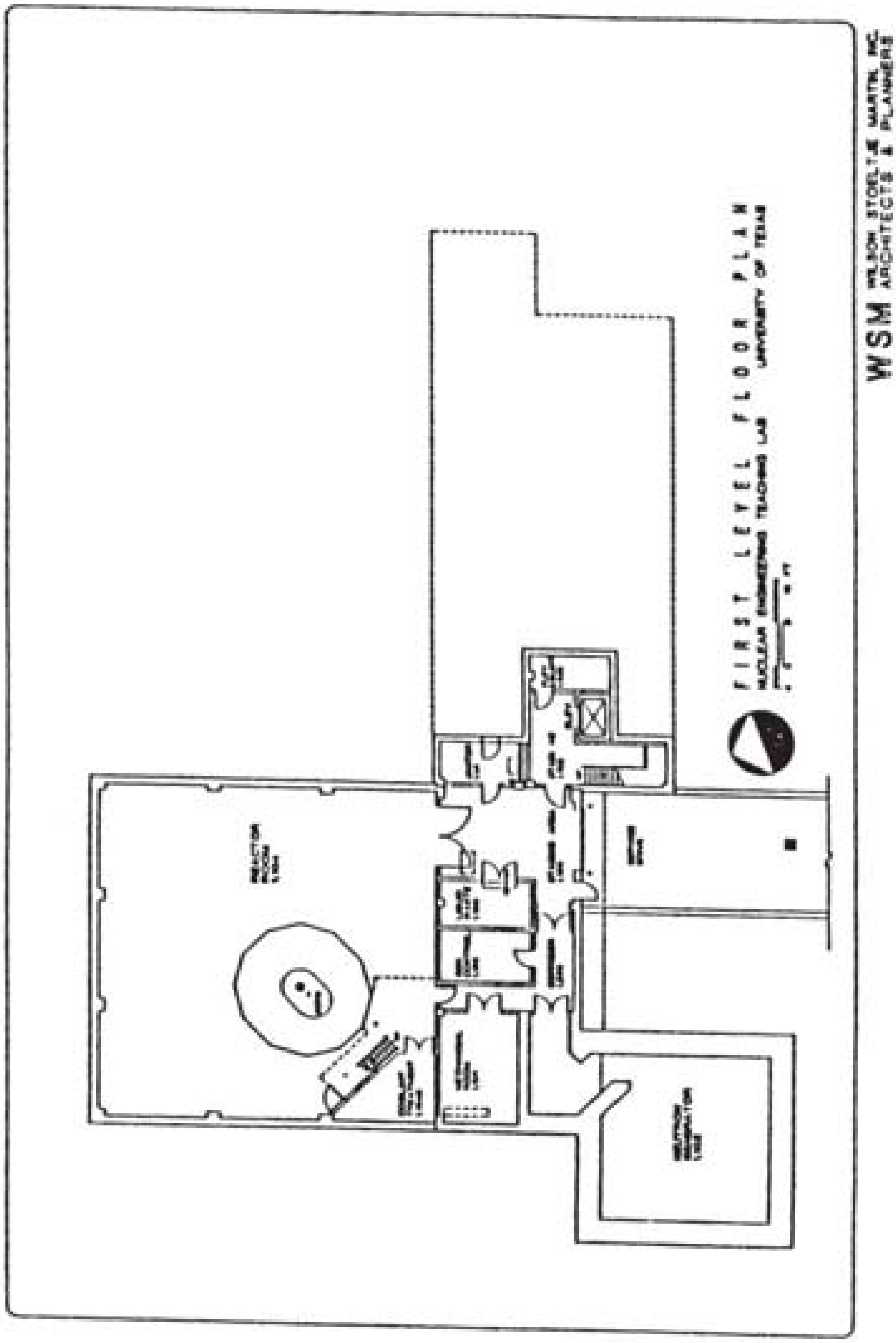
The UT reactor facility is in a 0 seismic zone where no damage from earthquakes is expected (see Section 2.6). The NETL building is designed with state-of-the-art engineering practices to the Uniform Building Code for seismic zone 0. The integrity of the building will not be affected by an earthquake of intensity VI (MMI). However, even if a rare severe earthquake damaged the building and the reactor, the staff concludes in the analyses in Section 14 that the health and safety of the public will not be adversely affected.





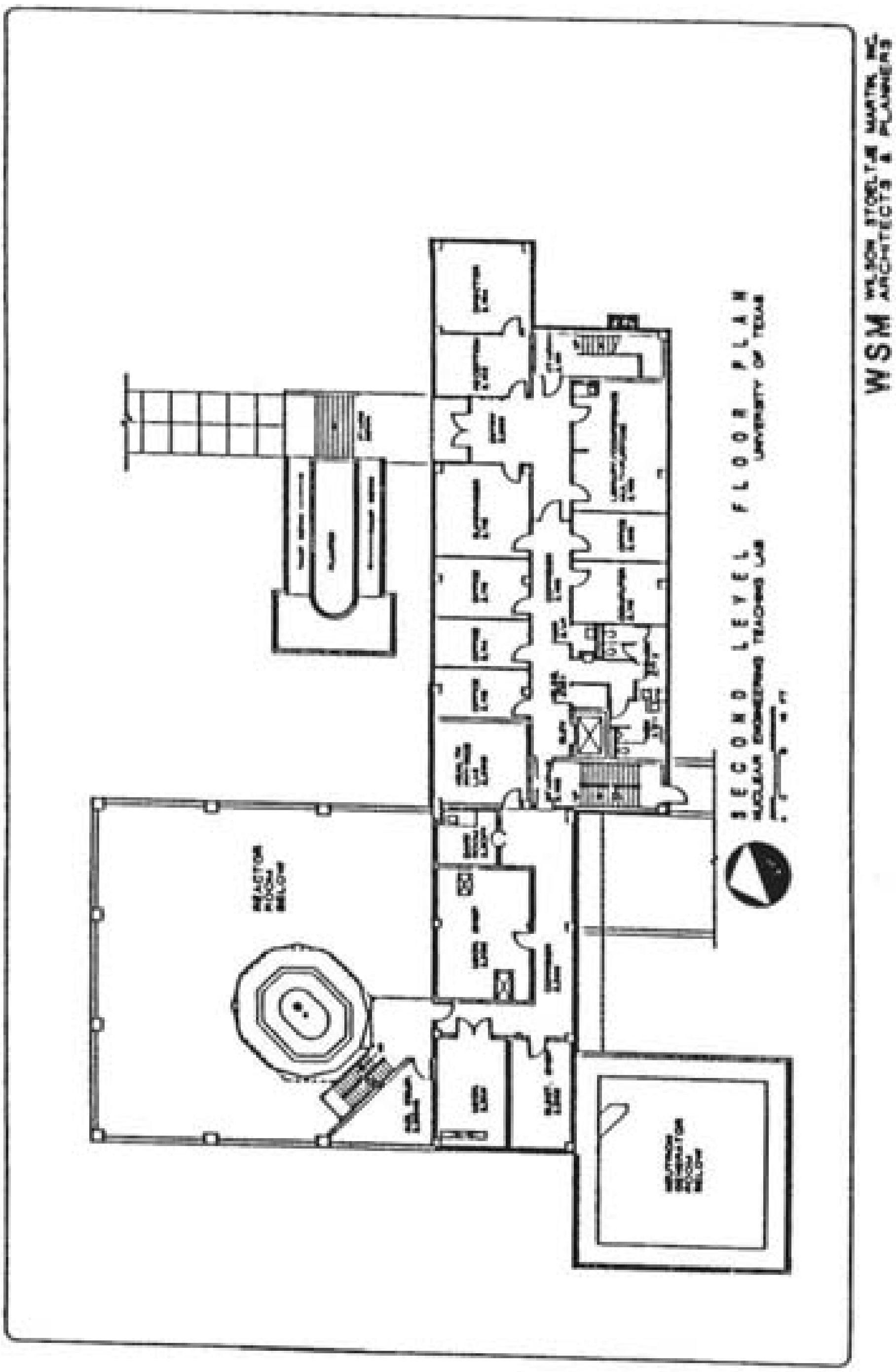
NETL SITE PLAN FOR
BALCONES RESEARCH CENTER

Figure 3.1



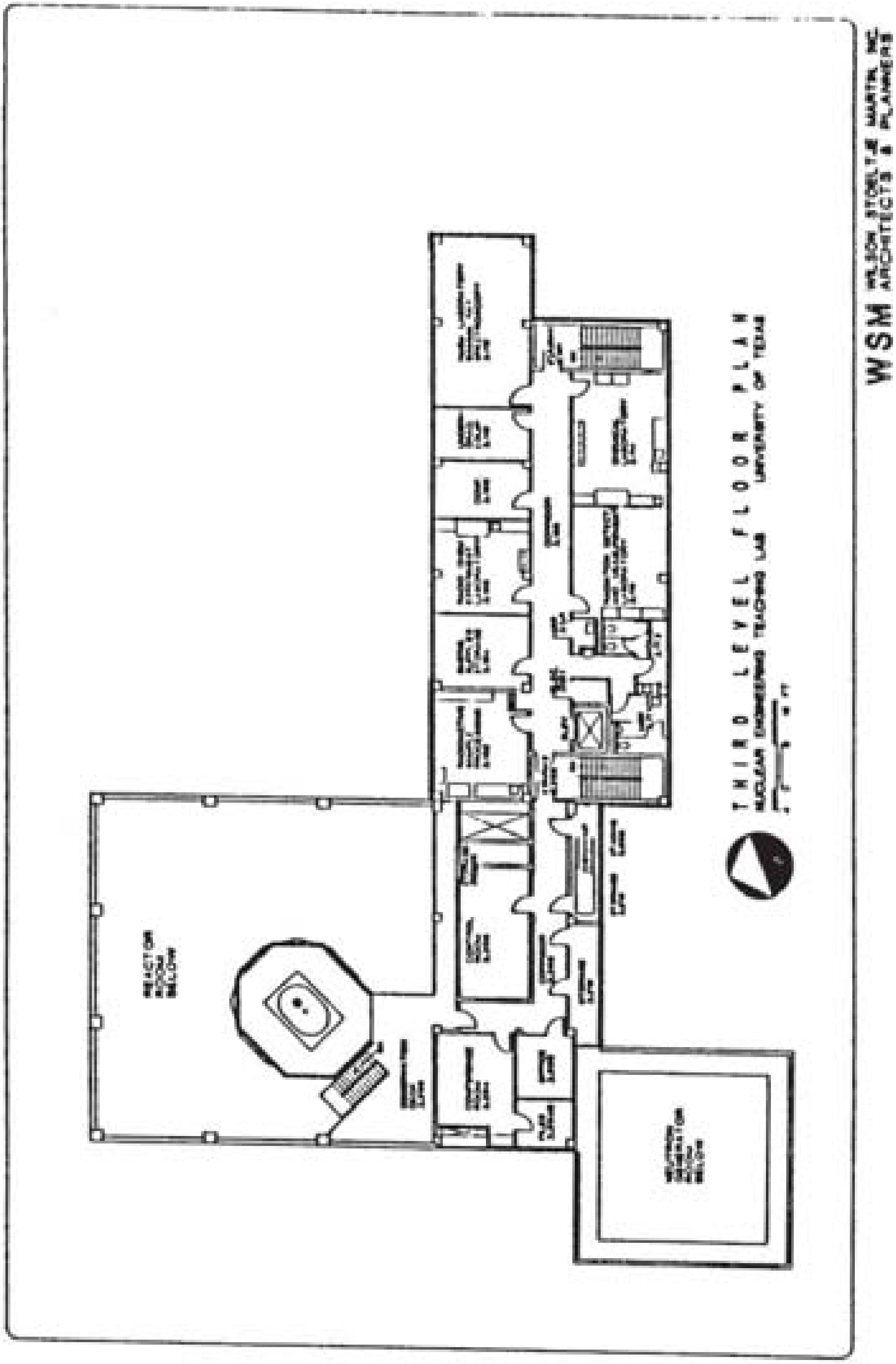
FIRST LEVEL FLOOR PLAN

Figure 3.2



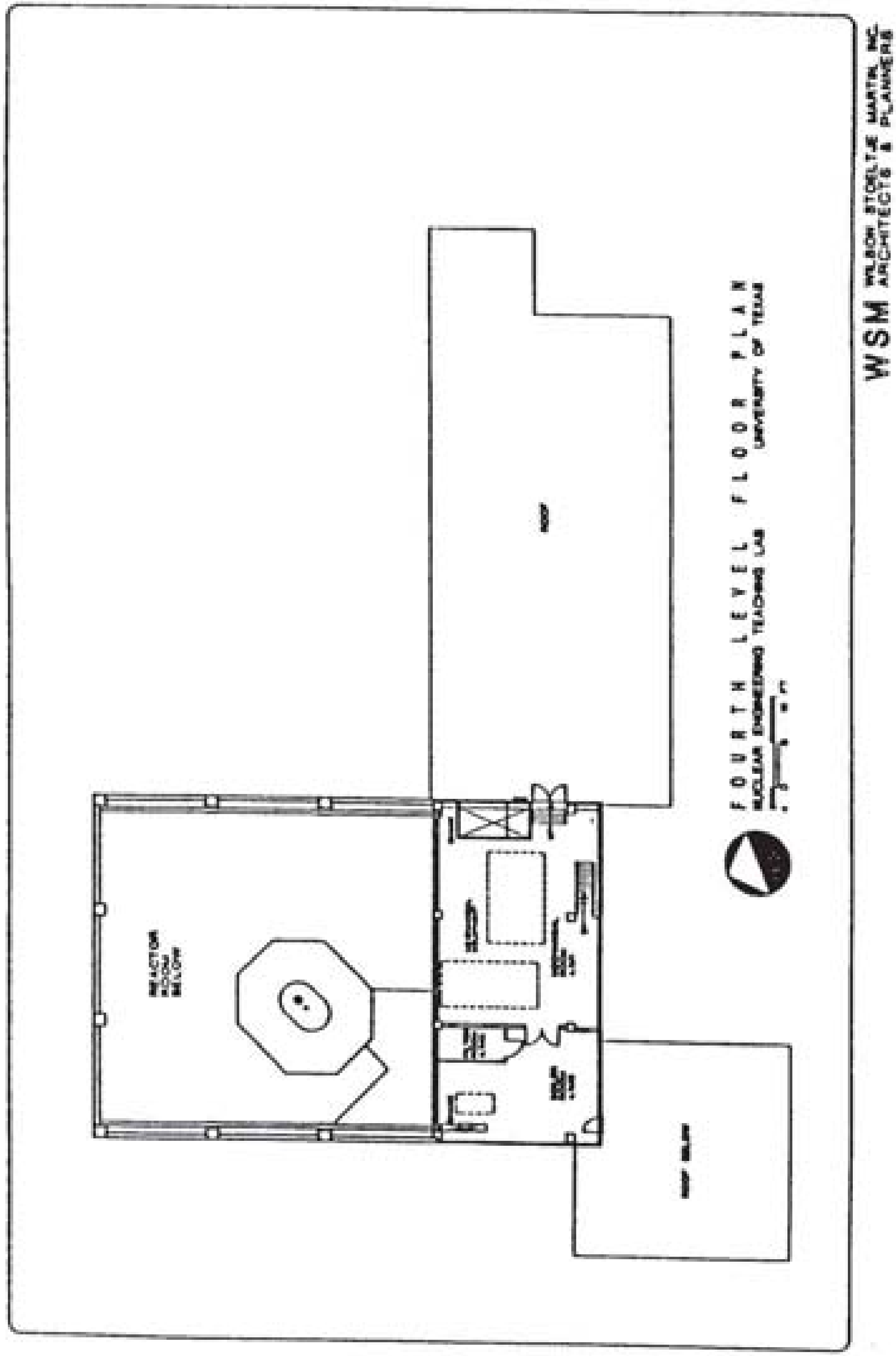
SECOND LEVEL FLOOR PLAN

Figure 3.3



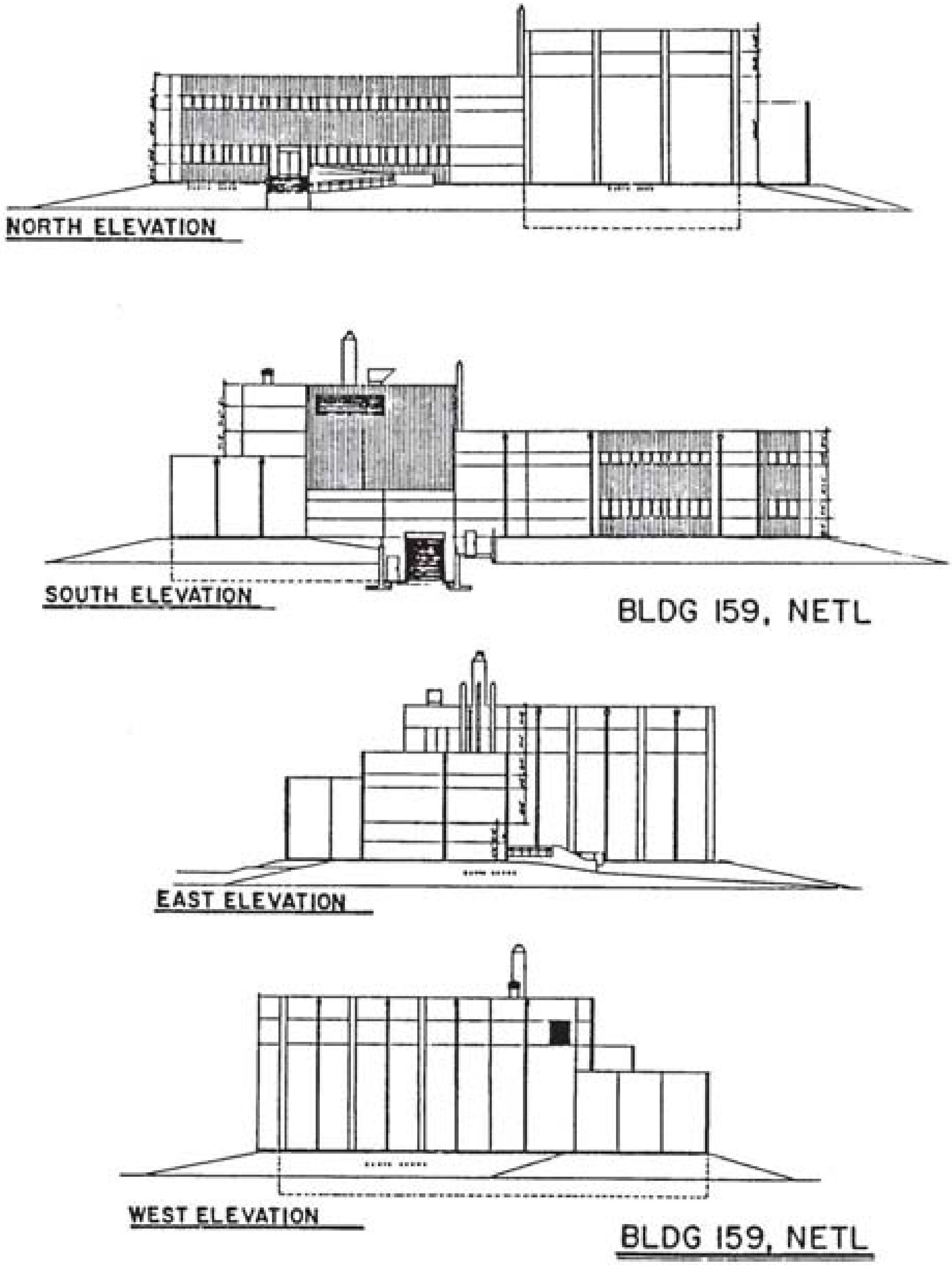
THIRD LEVEL FLOOR PLAN

Figure 3.4



FOURTH LEVEL FLOOR PLAN

Figure 3.5



ELEVATION PLANS

Figure 3.6

4 REACTOR

Replace the existing Figure 4.1 of a typical TRIGA Mark II reactor with the new Figure 4.1 which shows the UT TRIGA Mark II reactor.

4.1 Reactor Facility

The reactor bridge assembly from the original reactor on the main campus will not be reused. A new reactor bridge has been installed on the new reactor.

4.2 Reactor Core

The reactor core will consist of a lattice of approximately 86 fuel elements, which will vary in number according to the operational and experimental needs. The final design of the core has fuel elements in hexagonal rings, not concentric circular rings. The active (fueled) region of the reactor core forms a hexagon about 17.5 inches (44 cm) in diameter and about 15 inches (38 cm) high instead of the right circular cylinder discussed in the SER.

4.2.1 Reflector Assembly, Grid Plates, and Core Support Structure

In the final design of the reactor, the safety plate is fastened to the core support assembly instead of being welded to the radial graphite reflector. Replace Figure 4.4 which shows details of the design of the reactor and reflector.

Table 4.1 Principle Design Parameters

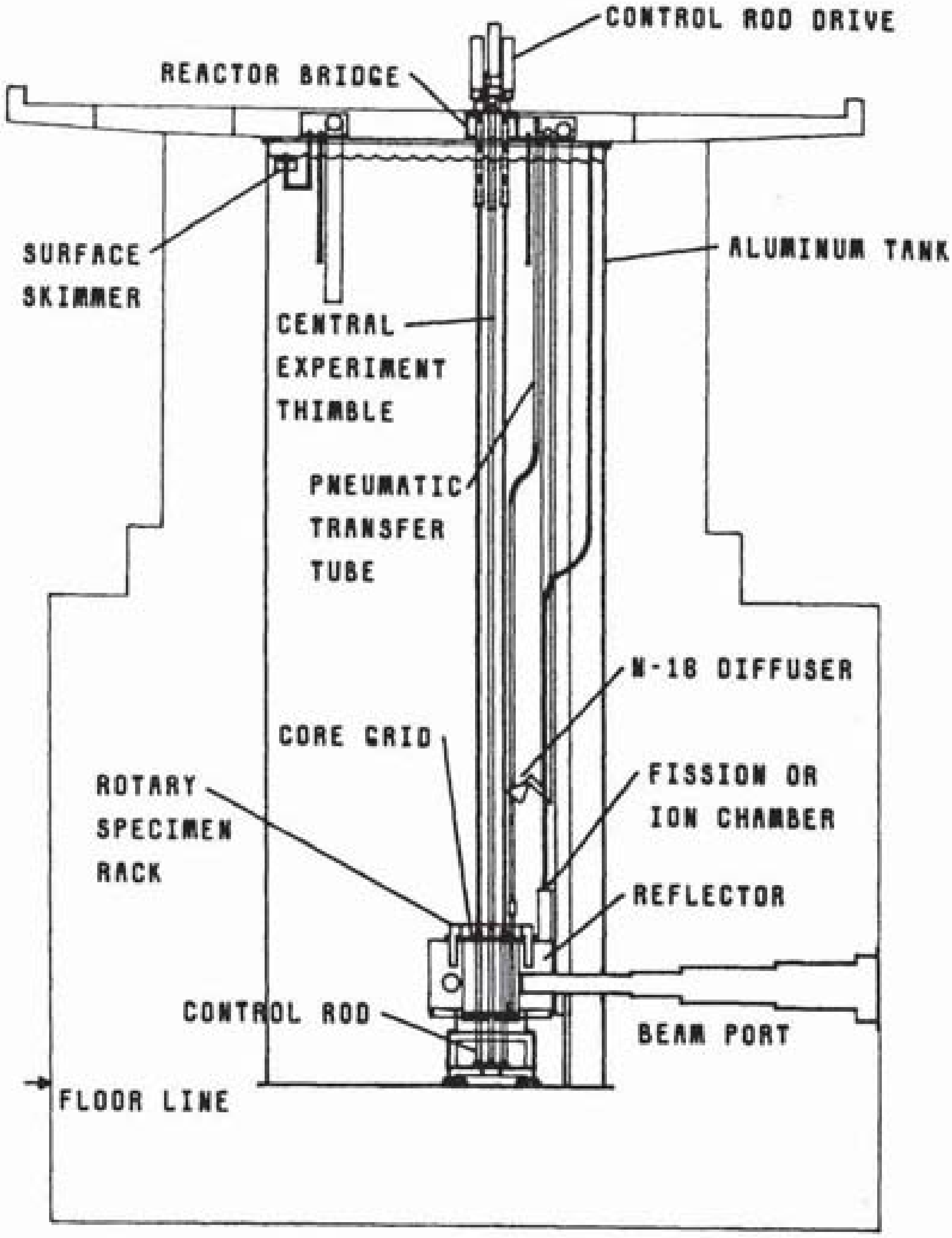
Three changes are made to Table 4.1. The UT TRIGA has two rods designated as shim rods: shim 1 and shim 2. Replace "shim" with "shim 1" and "safety" with "shim 2." This is a change in designation only. The function of the control rods has not changed. Remove $\Delta k/k$ to correct an error in the units for beta effective.

4.2.2 Fuel Elements

Replace Figure 4.3 which shows the core arrangement for the UT TRIGA reactor. The transient control rod is in location C-1, the shims rods are in locations D-6 and D-14, and the regulating rod is in location C-7.

4.2.4 Control Rods

The UT TRIGA as built has two rods designed as shim rods, shim 1 and shim 2. Shim 2 takes the place of the safety rod. This is a change in designation only. The function of the control rods has not changed.



TRIGA MARK II REACTOR

Figure 4.1

4.3 Reactor Tank and Biological Shield

The reactor tank as constructed has a depth of 26.7 feet (8.1 m) and a capacity of approximately 11,000 gallons (41,700 l). Normal system inventory is 10,500 gallons (39,750 l). This represents a small increase in depth and capacity over the original design. The description of the outside of the reactor tank previously discussed only bituminous tar and paper coating. However, it is now also coated with epoxy paint. Adding the epoxy paint will provide better corrosion protection than was provided by the bituminous tar and paper coating alone.

A cobalt-60 source having a maximum source strength of 10,000 Curies (Ci) may be located in the reactor pool. The source is located at least 10 feet (3.04 m) below the water level. The original SER referred to a source strength of 9,000 Curies, which was the original source strength, not the maximum strength allowed by the license.

The core will be shielded horizontally by a minimum of 7.97 feet (2.43 m) of medium density concrete (150 lb/ft^3 (2.88 g/cm^3)). The shield design radiological exposure constraint of 1 mrem per hour for the most accessible areas of the shield is not changed. Add Figure 4.5 to show the as-built tank and shield structure.

4.4 Reactor Instrumentation

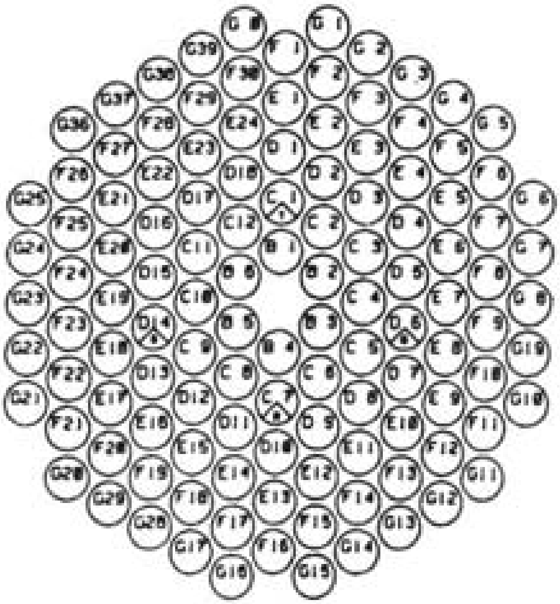
Replace Section 4.4 in its entirety with the following paragraph.

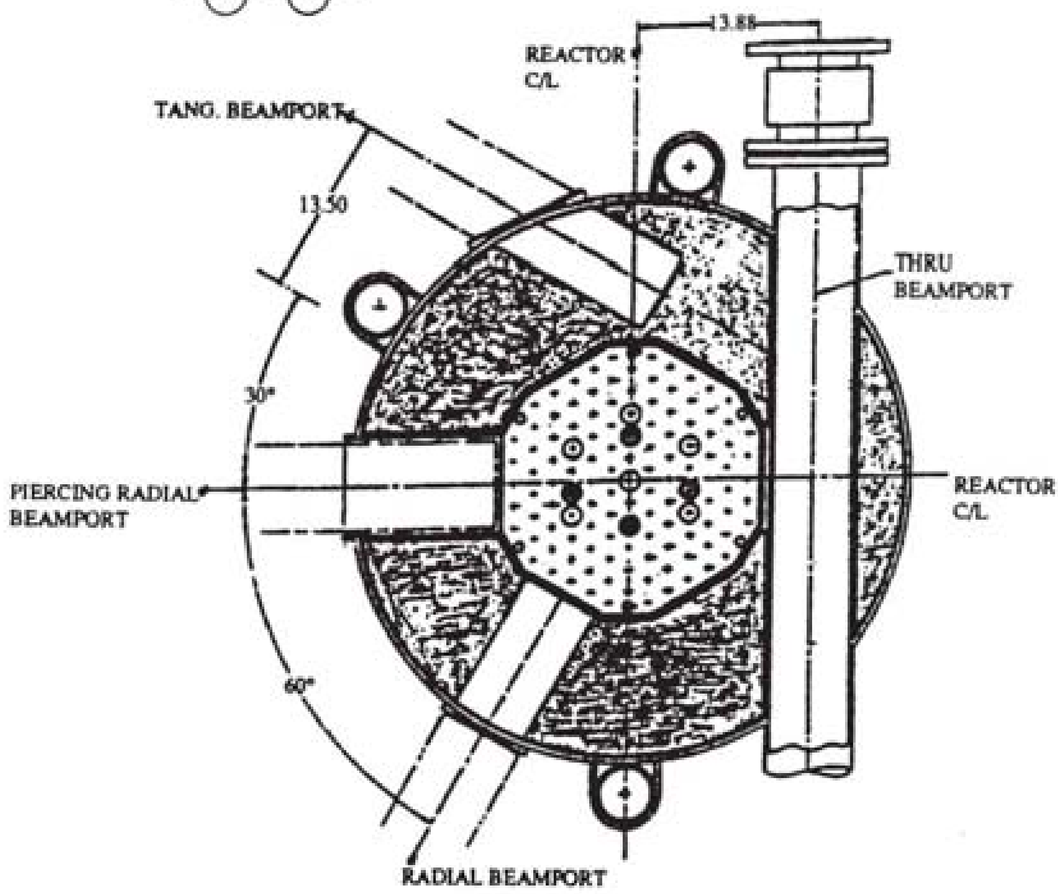
> The reactor instrumentation will use a multifunction computer to process the input from a low-noise fission chamber and from two ionization chambers. One of the ionization chambers will be used during pulsing mode to measure peak power and energy release. Section 7 provides a detailed description of the reactor instrumentation.

4.5 Dynamic Design Evaluation

4.5.1 Excess Reactivity and Shutdown Margin

The Technical Specifications require that the reactor shall not be operated unless the shutdown margin provided by the control rods is greater than 0.2 percent $\Delta k/k$ with the reactor in the reference core condition, the most reactive control rod fully withdrawn, and all moveable experiments in their most reactive state. The reference core condition is when the core is at ambient temperature and the reactivity worth of xenon is negligible. The original SER did not discuss the state of the core and discussed the non-secured experiment of highest worth. The new definition of shutdown margin is more conservative than that in the original SER because it accounts for core conditions that can change with time and reduce the amount of reactor shutdown and because it accounts for a class of experiments that can affect shutdown. The staff finds this change to be acceptable.

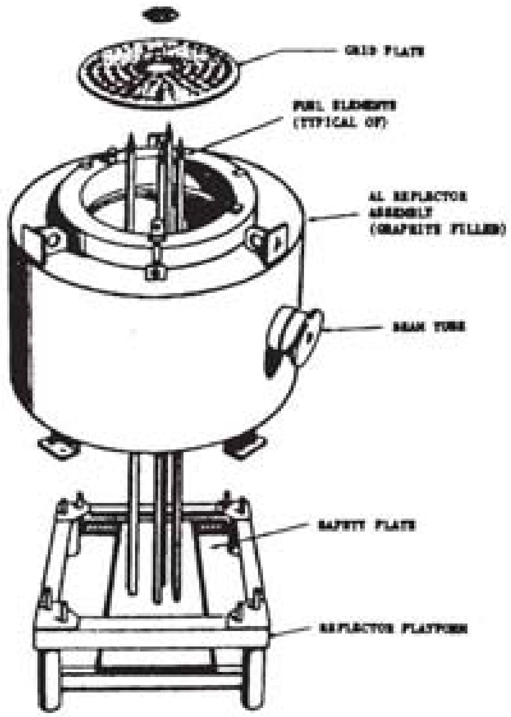


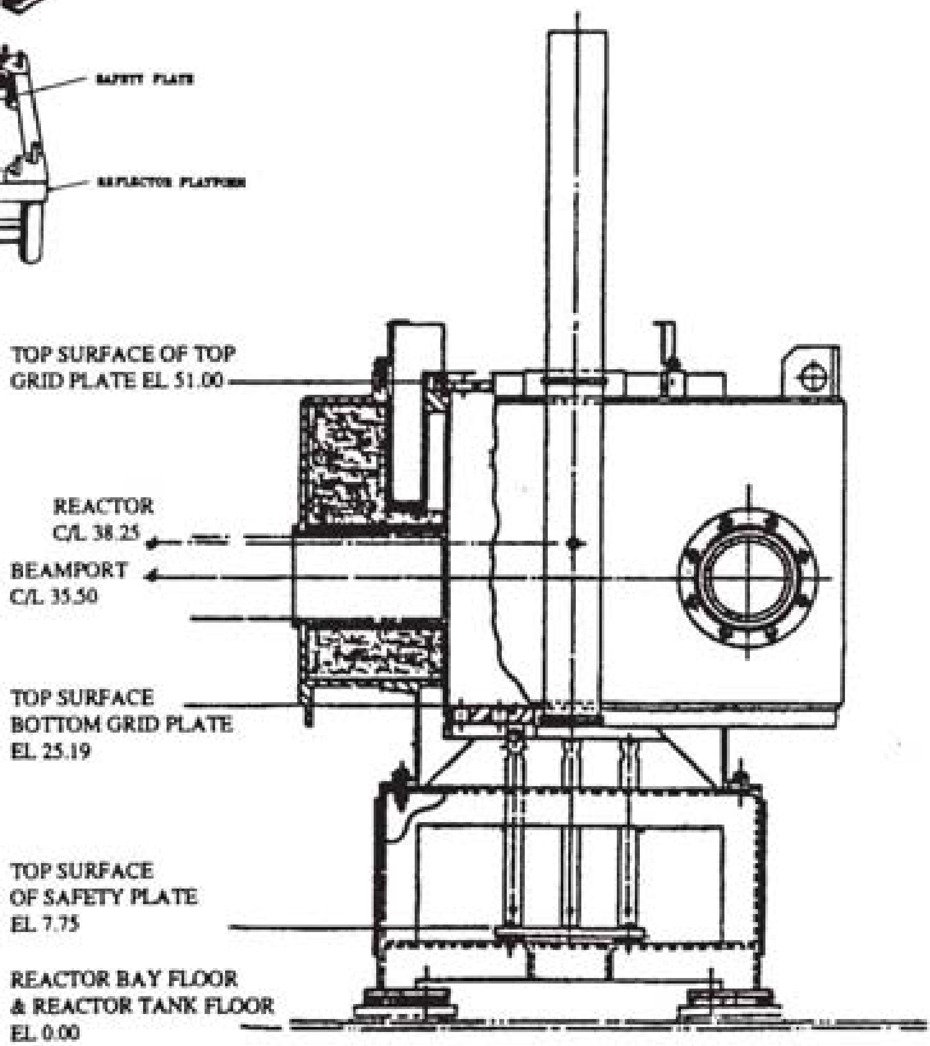


TRIGA reactor - cross section through beamports

CORE ARRANGEMENT

Figure 4.3

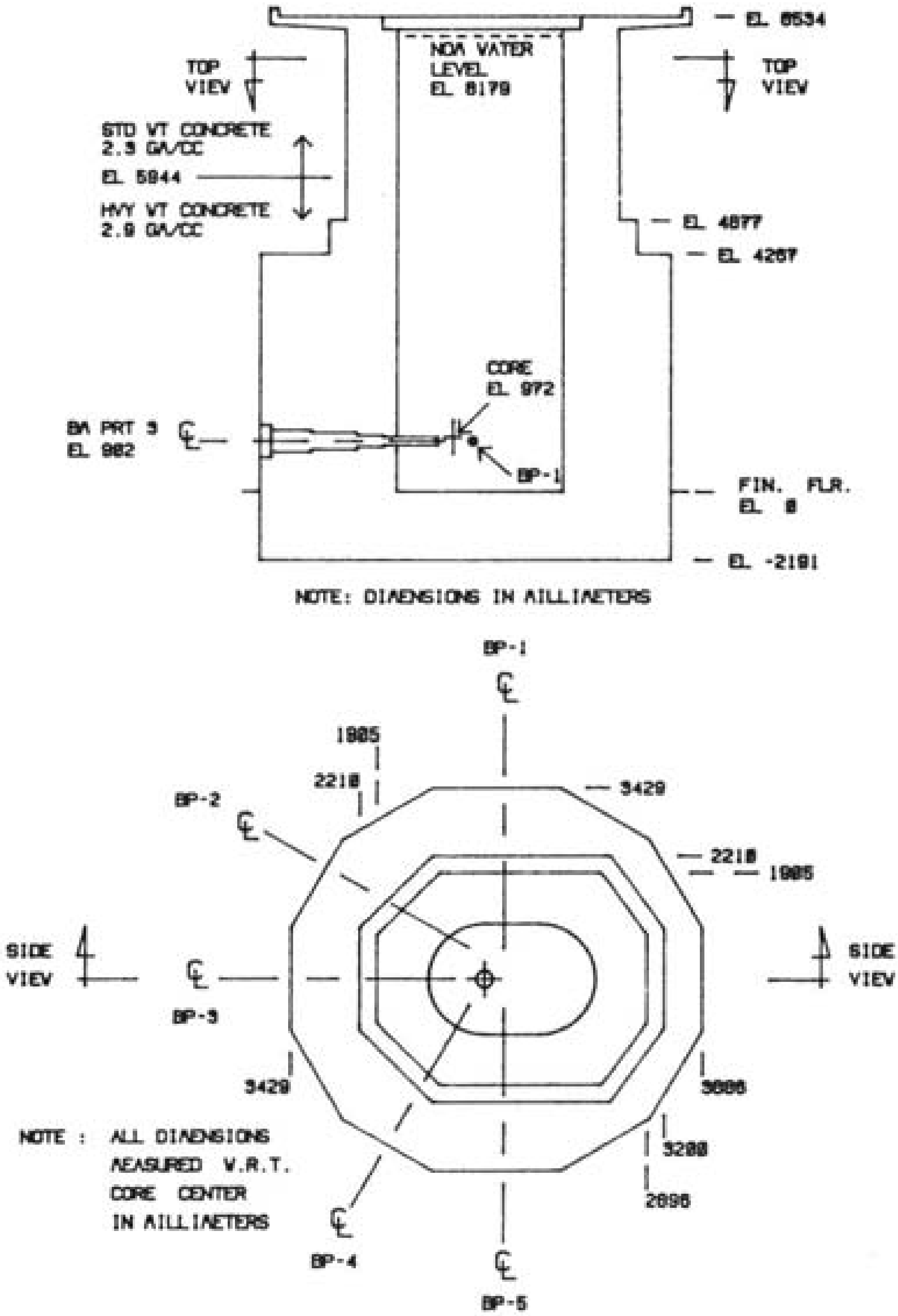




TRIGA reactor - elevation view

REACTOR, REFLECTOR, AND SHIELDING

Figure 4.4



REACTOR SHIELD STRUCTURE

Figure 4.5

The UT TRIGA as built has two control rods designated as shim rods, shim 1 and shim 2. Shim 2 takes the place of the safety rod. This is a change in designation only. The function of the control rods has not changed. The total worth of the control rods remains the same but the redesignation of the control rods causes the worth of the individual rods to be different than described in the SER. The control rod worths are 2.1 percent $\Delta k/k$ (3.0$) for the transient rod, 2.6 percent $\Delta k/k$ (3.7$) for the regulating rod, 2.0 percent $\Delta k/k$ (2.9$) for shim 1 and 2.0 percent $\Delta k/k$ (2.9$) for shim 2. This is a change in control rod designation that the staff finds acceptable.

4.6 Functional Design of Reactivity Control System

The UT TRIGA as built has two rods designated as shim rods, shim 1 and shim 2. Shim 2 takes the place of the safety rod. This is a change in designation only. The nuclear function of the control rods has not changed.

Each control rod drive system will be energized from the data acquisition and control system. This statement is more accurate than the original SER which stated the following: "The control rods will be energized by the control console."

4.6.1 Control Rod Drive Assembly

The information in this Section applies only to the two shim rods. Each of the shim rod drive assemblies consists of a nonsynchronous, single phase electric motor connected to a rack and pinion drive system. The regulating rod drive has been changed to a linear drive actuated by an electric stepping-motor. Section 4.6.1.1 provides a detailed discussion of the regulating rod.

The drive motors for the two shim rods will insert or withdraw the shim rods at an approximate rate of 18 inches per minute (0.75 cm/s). This is an increase over the original approximate rate of 11.5 inches per minute (0.5 cm/s). The staff concludes that this change in rate does not affect the safe operation of the reactor because the TRIGA reactor is designed to pulse and because Section 14.2 of the SER concludes that insertions of reactivity at a rapid rate will not damage the reactor.

The original SER states that a helipot generates the position indications for the shim, safety, and regulating rods. In the final reactor design, a helipot is connected to the pinion to generate the position indication for the two shim rods.

4.6.1.1 Regulating Rod Drive Assembly

The regulating rod can be used as a manual control rod by the reactor operator or can be used in automatic mode to bring and maintain reactor power at a preset demand level by the reactor control system. The rod drive mechanism for the regulating rod is an electric stepping-motor-actulated linear drive equipped with a magnetic coupler. The stepping motor drives a pinion gear that is engaged with a rack. The regulating rod uses a 10-turn potentiometer to generate position indication which is displayed on the control console. The rest of the system is similar to those for the other control rods. The maximum design rate at which the motorized system will insert or withdraw the control rod is approximately 33 inches per minute (1.4 cm/s). The rod speed has been measured during construction tests at 27 inches per minute (1.1 cm/s). The staff concludes that this rate of insertion and withdrawal does not affect the safe operation of the reactor because the TRIGA reactor is designed to pulse and because Section 14.2 of the SER concludes that insertions of reactivity at a rapid rate will not damage the reactor.

Stepping motors operate on phase-switched direct-current (dc) power. The circuitry supplying power to the motor has been chosen to optimize motor torque to the usual drive speeds of the TRIGA control rods. Licensees have only recently began using stepping motors for TRIGA control rods. Thus, little operating experience is available to provide data for this type of control rod. However, the total reactivity worth of the regulating rod is less than the maximum inadvertent insertion evaluated in Section 14 of the SER. Therefore, the staff concludes that a reactivity addition caused by a malfunction of the stepping motor is within the envelope of that evaluation and that the use of the stepping motor is acceptable.

4.6.2 Transient Rod Drive Assembly

The original SER indicated that the control logic of the new reactor control system would prevent the transient rod from being actuated if the reactivity insertion value of the rod was greater than a predetermined amount. However, this feature of the new control system design was not realized. In the final design of the reactor control system, administrative control will be used to restrict the travel of the transient rod to limit the reactivity insertion of the pulse to less than the license limit of 2.2 percent $\Delta k/k$ (3.14\$). Administrative control is the common method of controlling the worth of the transient rod insertion in TRIGA reactors and is acceptable to the staff.

The withdrawal speed of the transient rod is approximately 28 inches per minute (1.19 cm/s). Replace "safety rod" with "shim 2" in this section.

4.6.3 Scram-Logic Circuitry and Interlocks

The core instrumentation consists of a low-noise fission chamber and two ionization chambers. The safety rod has been redesignated as a shim rod. The SER listed a number of events that will cause the reactor to shut down automatically. The reactor will also scram upon a loss of electrical power to the control console and if the software does not update the timers to monitor computer status in the watchdog circuits for each computer.

4.6.4 Assessment

The reactor control system consists of one digital NM-1000 safety channel and the NP-1000 and NPP-1000 analog safety channels instead of two NM-1000 digital channels as discussed in the original SER.

4.8 Conclusions

The staff concludes that the reactor has been built in substantial agreement with the accepted design. The staff found that the minor deviations during construction from the originally reviewed design will not decrease the safety margins and will likely increase reliability and utility. Therefore, the staff concludes that there continues to be reasonable assurance that the as-built construction of the UT TRIGA reactor will pose no significant radiological risk to the health and safety of the public.

5 REACTOR COOLANT AND ASSOCIATED SYSTEMS

Revise Figure 5.1 and add Figure 5.2 to reflect the as-built reactor coolant and purification systems.

5.1 Cooling System

The SER discusses two cooling system suction intakes. The final reactor design includes only one suction intake for the reactor cooling system. The primary coolant purification system discussed in Section 5.2 provides the suction for water surface skimming.

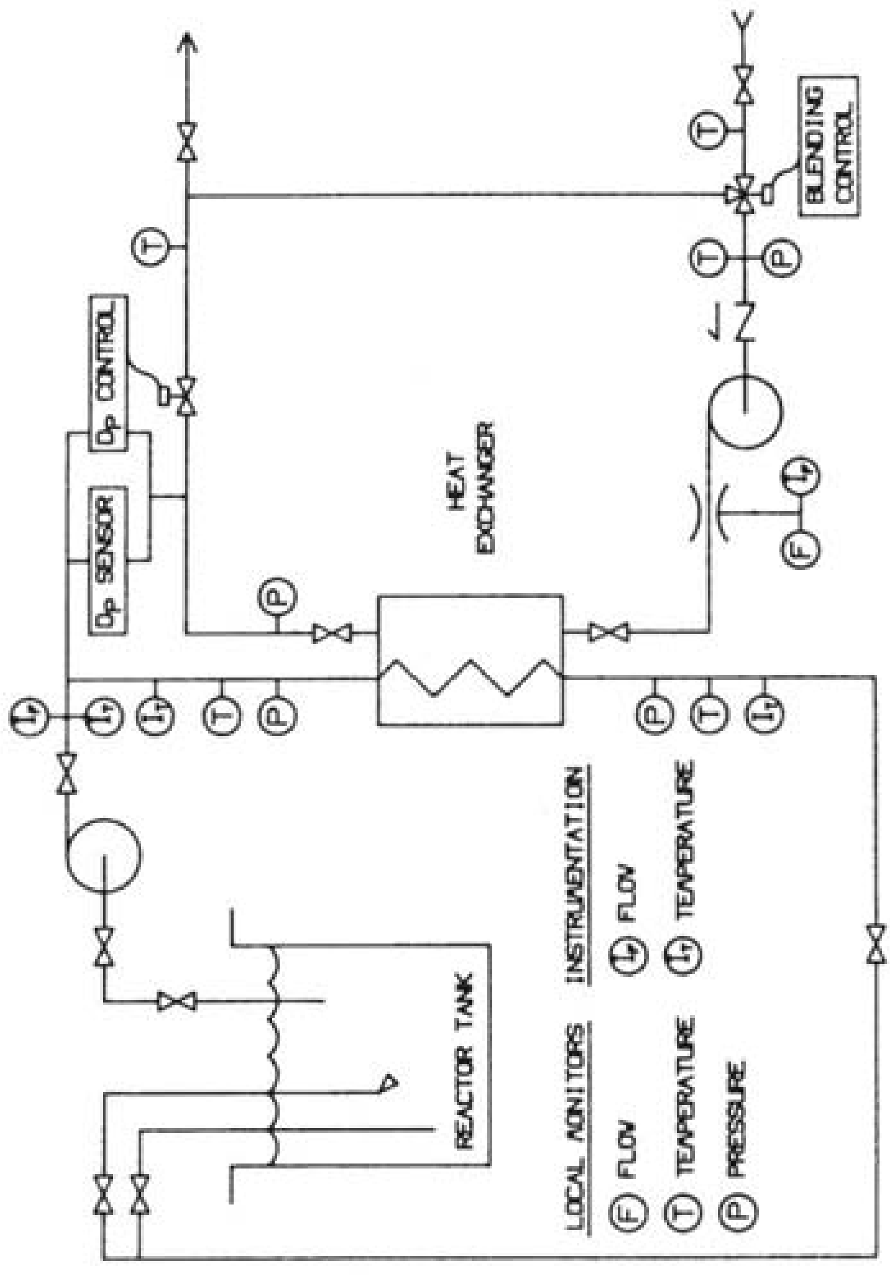
5.2 Primary Coolant Purification System

Replace the first paragraph of this section of the original SER with the following paragraph.

> A purification loop will be incorporated separate from the cooling system. Suction of water from the reactor pool for purification is provided by two inlets which extend no more than 6.5 feet (2 m) below the top of the reactor tank. Valves are used to select suction from either a surface skimmer or a subsurface inlet. The purification skid will be located at about the same vertical location as the heat exchanger. The purification loop pump will circulate continuously approximately 10 gallons per minute (0.6 l/s) of pool water to remove suspended particulates and soluble ions from the water coolant. Treated water is returned to the pool through a subsurface discharge pipe.
>
> Valves isolate the suction or return lines and system components.

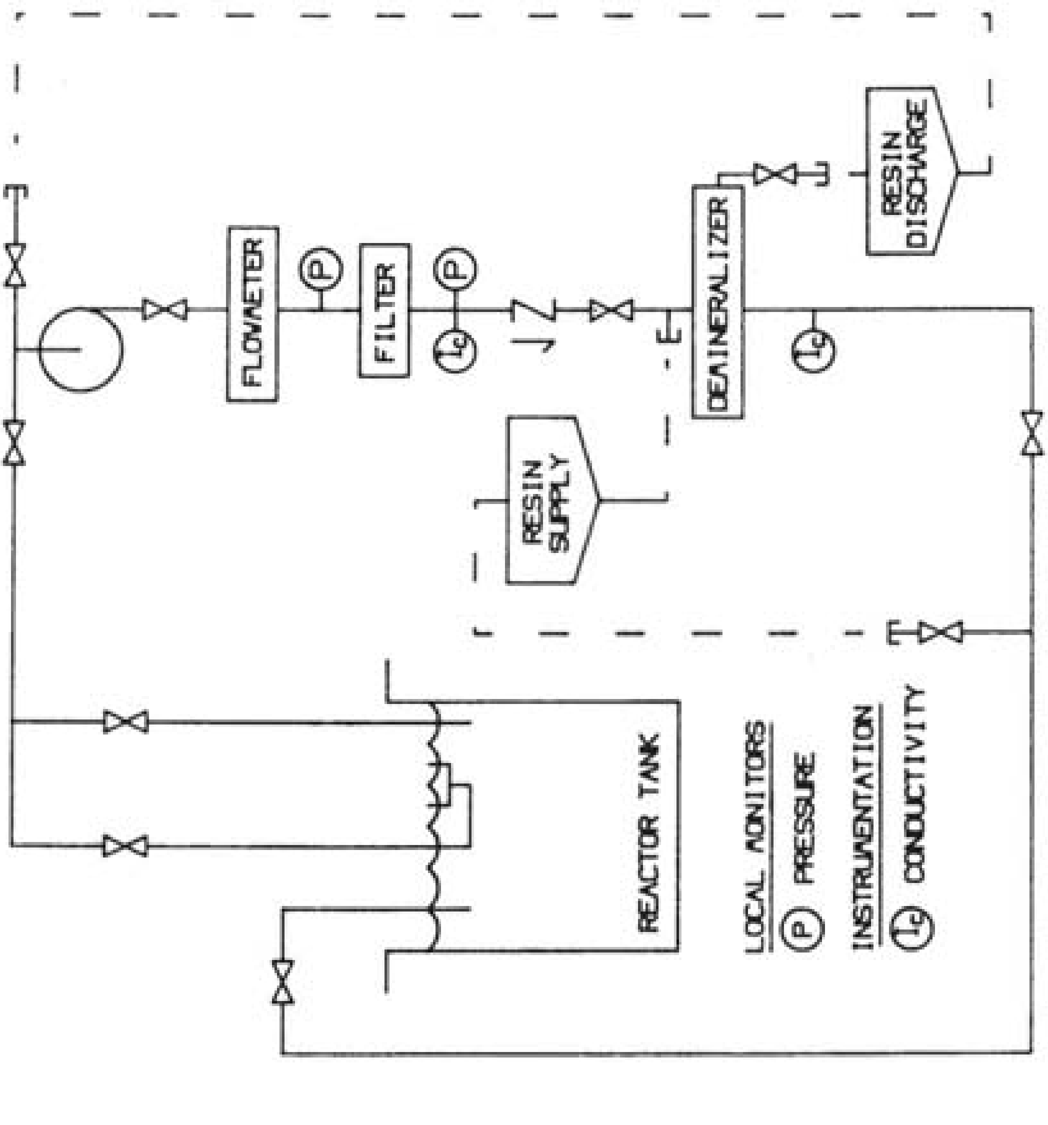
5.3 Primary Coolant Makeup System

A check valve in the piping and quick disconnect fittings eliminates the possibility of primary water entering the city water system. The quick disconnect fittings ensure that the two systems are isolated except when water is being added, and the check valve eliminates the possibility of backflow during water addition.



COOLANT SYSTEM LAYOUT

Figure 5.1



PURIFICATION SYSTEM LAYOUT

Figure 5.2

6 ENGINEERED SAFETY FEATURES

6.1 Reactor Room

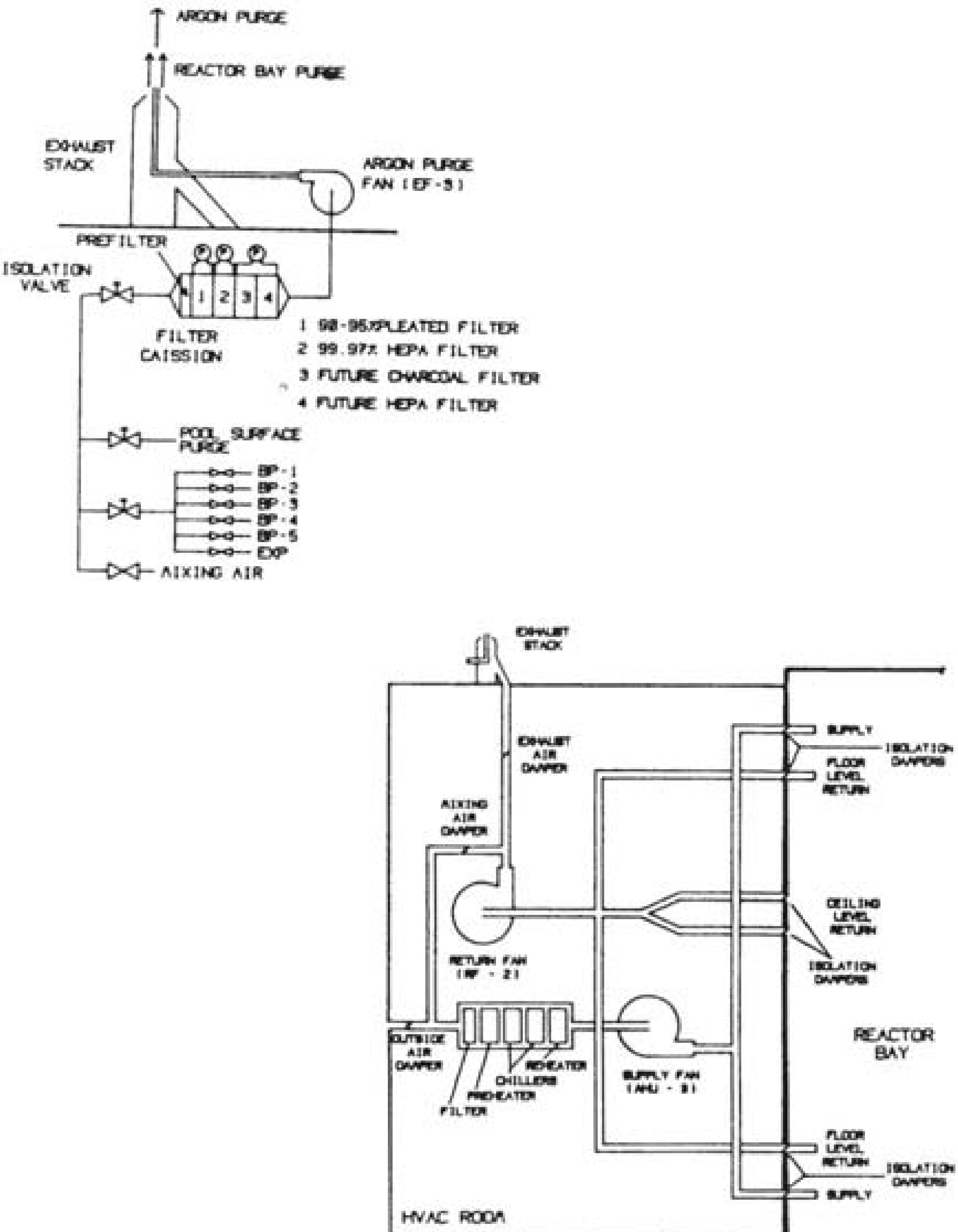
The reactor room is designed to withstand a negative pressure of 0.06 inches (0.15 cm) of water below the ambient atmospheric pressure and to normally operate at a negative pressure of 0.04 inches (0.10 cm) of water, not 2 inches as stated in the original SER.

6.2 Ventilation System

The main ventilation system does not ventilate through the high efficiency particulate (HEPA) filter system as stated in the original SER. The system exhausts to a roof stack at least 60 feet (18 m) above ground level. The main ventilation system maintains a negative pressure in the reactor room in relation to the outside and areas adjacent to the reactor room. The ventilation system has two modes of operation: (1) recirculation of the reactor room air and (2) a high volume mode that has no recirculation and completely exchanges the air in the reactor room more than twice each hour.

The air purge exhausts air from areas of argon-41 production such as the beam tubes, sample transfer systems, and the pool surface. The system exhausts this air to the roof stack through a prefilter and a HEPA filter. The design includes provisions to add charcoal filters if experimental needs dictate. The applicant can sample the air in this system using sample ports. This system can also be manually isolated.

Figure 6.1 provides details of the ventilation system.



SCHEMATIC OF VENTILATION SYSTEMS

Figure 6.1

7 CONTROL AND INSTRUMENTATION SYSTEMS

When the original SER was written in 1985, General Atomics (GA) had not yet completed the design of its new instrumentation and control (I&C) system. However, the design was advanced to the point that the NRC staff could conclude that although the system had not been previously used at NRC licensed non-power reactors, the design was acceptable and the system would be adequately tested and evaluated before it would be operated at the UT facility. Additional information provided by GA and NRC non-power reactor licensees and additional evaluation performed by the NRC staff since 1985 have not changed our conclusion.

GA has completed the design, and the staff has evaluated the hardware and software. The staff concluded that GA I&C systems are acceptable at a number of other TRIGA reactor facilities. As of November 1991, the staff approved license amendments and technical specifications changes to install the GA I&C system at the GA Mark I reactor, the Armed Forces Radiobiology Research Institute (AFRRI) reactor, the Dow Chemical Company reactor, and the United States Geological Survey (USGS) reactor. In addition, the NM-1000 digital power channel has been installed at the Veterans Administration research reactor.

An updated logic diagram for the I&C system is shown in Figure 7.1 which replaces Figure 7.1 in the original SER.

7.2 Control System

7.2.1 Control Console

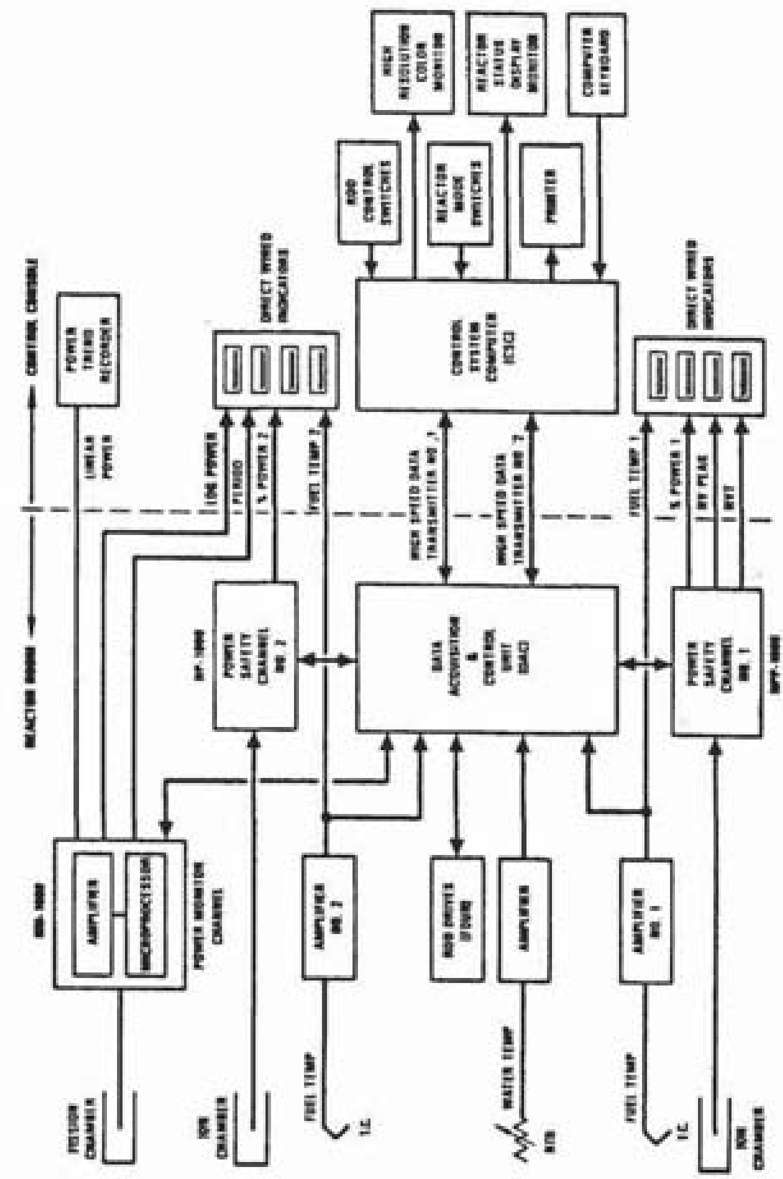
Figure 7.2 shows the final design of the control console and replaces Figure 7.2 in the original SER. The final I&C system does not have the two independent instrumentation computers discussed in the original SER. Figure 7.3 shows the final layout of the control panel which replaces the conceptual layout shown in Figure 7.3 in the original SER.

The console includes reactor control panels, a control system computer (CSC), two graphic CRT monitors, a keyboard interface, disk drives, and a printer.

The mode control panel contains buttons for selecting the reactor mode. The operator sets the power demand from this panel to be used in automatic (servo) power control. This panel also contains a selector switch for scram tests and contains buttons for control system instrument power and the prestart check.

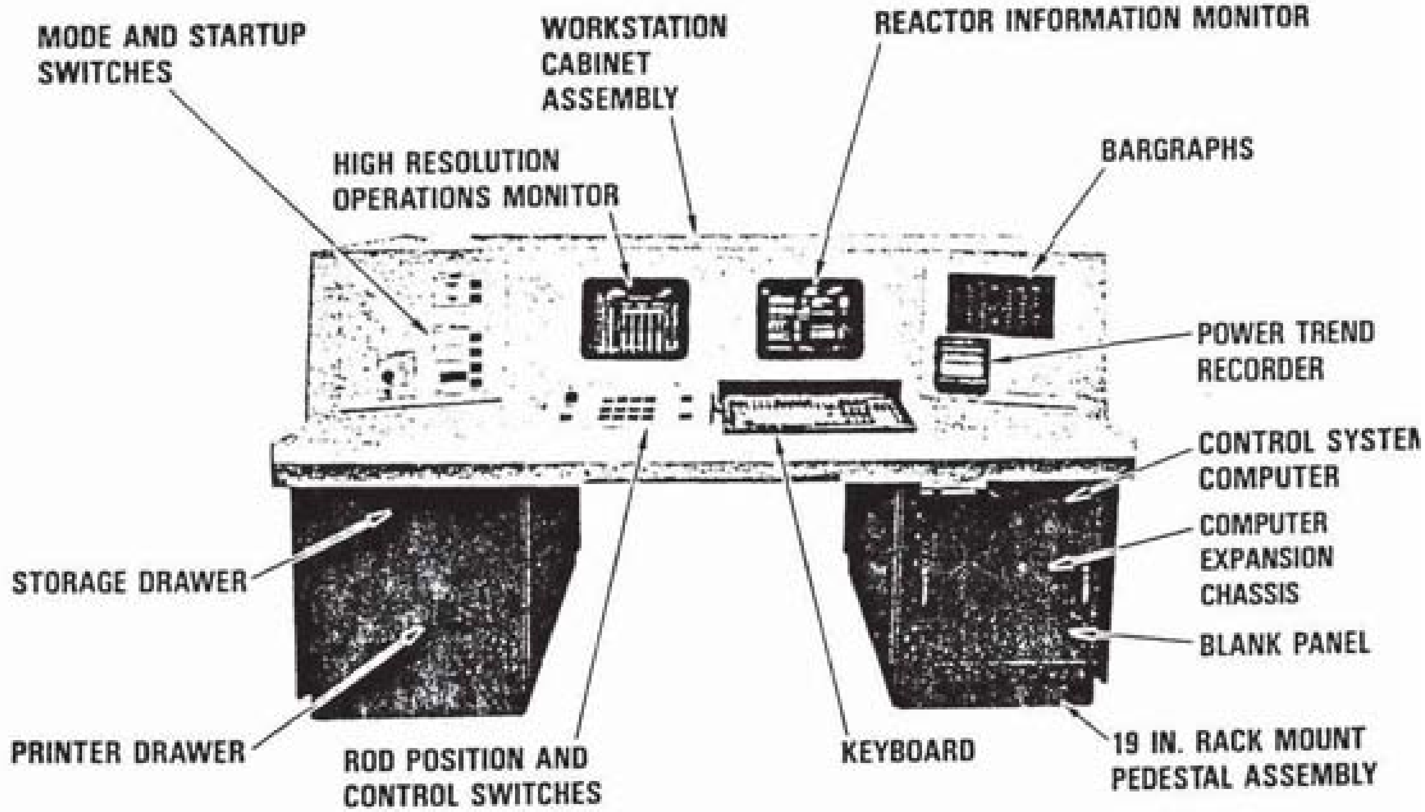
The analog display panel displays important information in a bar graph format about reactor status. This provides an additional display of important parameters to the CRT.

The CSC displays on the CRT information such as power level, control rod and drive position, and fuel temperatures. The operator also has access to status windows that display plant status information.



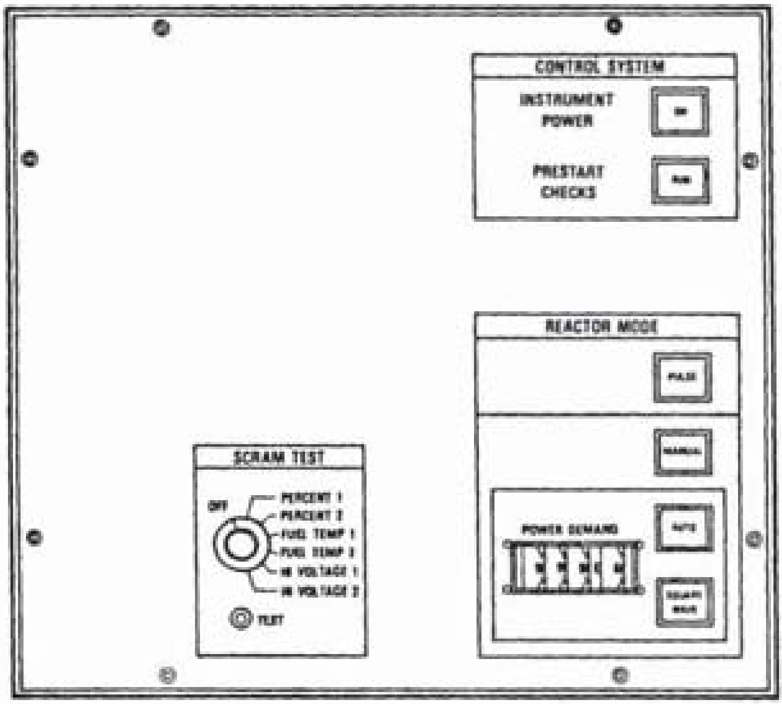
CONTROL SYSTEM BLOCK DIAGRAM

Figure 7.1

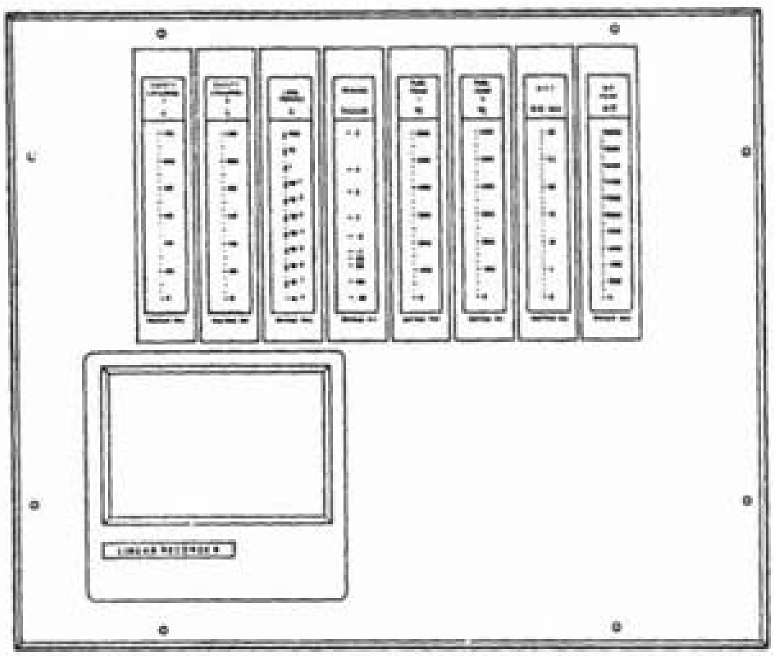


LAYOUT OF THE REACTOR CONTROL CONSOLE

Figure 7.2



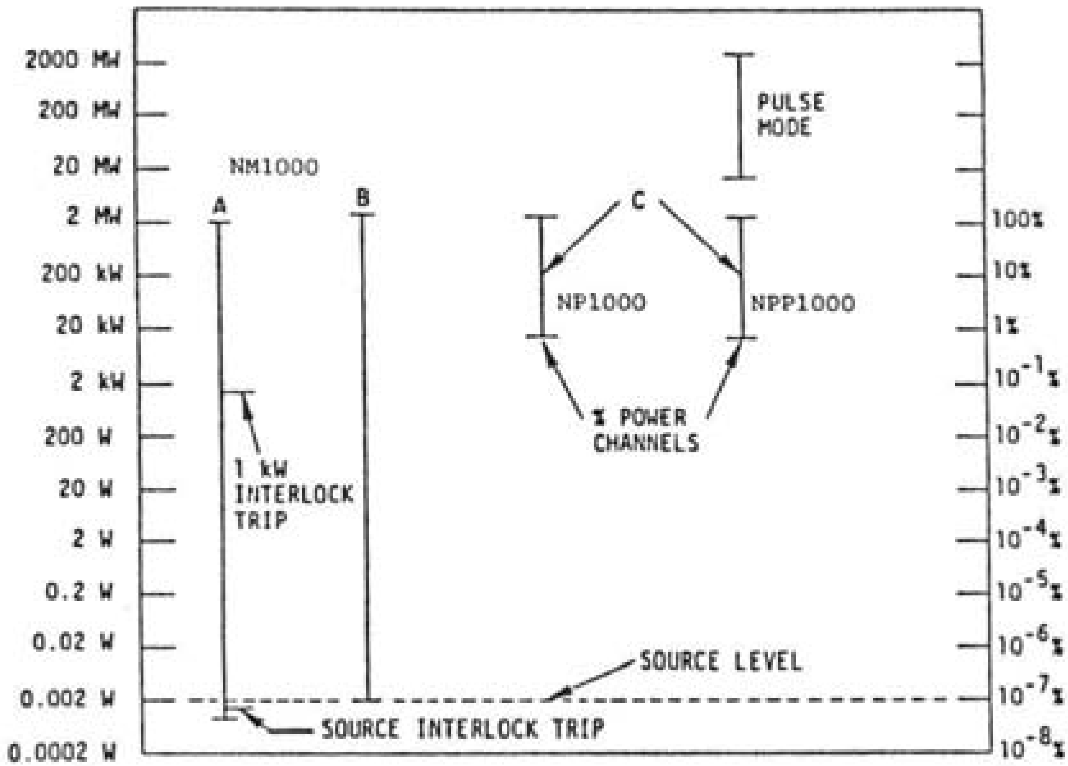
Mode Control Panel



Analog Display Panel

CONSOLE CONTROL PANELS

Figure 7.3



A = Wide Range Log Channel, NM1000
B = Wide Range Linear Channel, NM1000
C = Manual, Automatic, and Squarewave Modes, NP1000

NEUTRON CHANNEL OPERATING RANGES

Figure 7.4

The rod control panel contains the key switch for rod magnet power, the buttons for manual rod control, annunciators, and the scram switch.

7.2.2.1 Manual Mode

The "contact/on" (C/O) buttons have been renamed "magnet" (shim and regulating rods) or "AIR" (transient rod) push buttons. Their function remains to interrupt current to the shim or regulating rod magnets or to the transient rod air supply solenoid valve. The new Figure 7.5 shows the rod control panel.

7.2.2.2 Automatic Mode

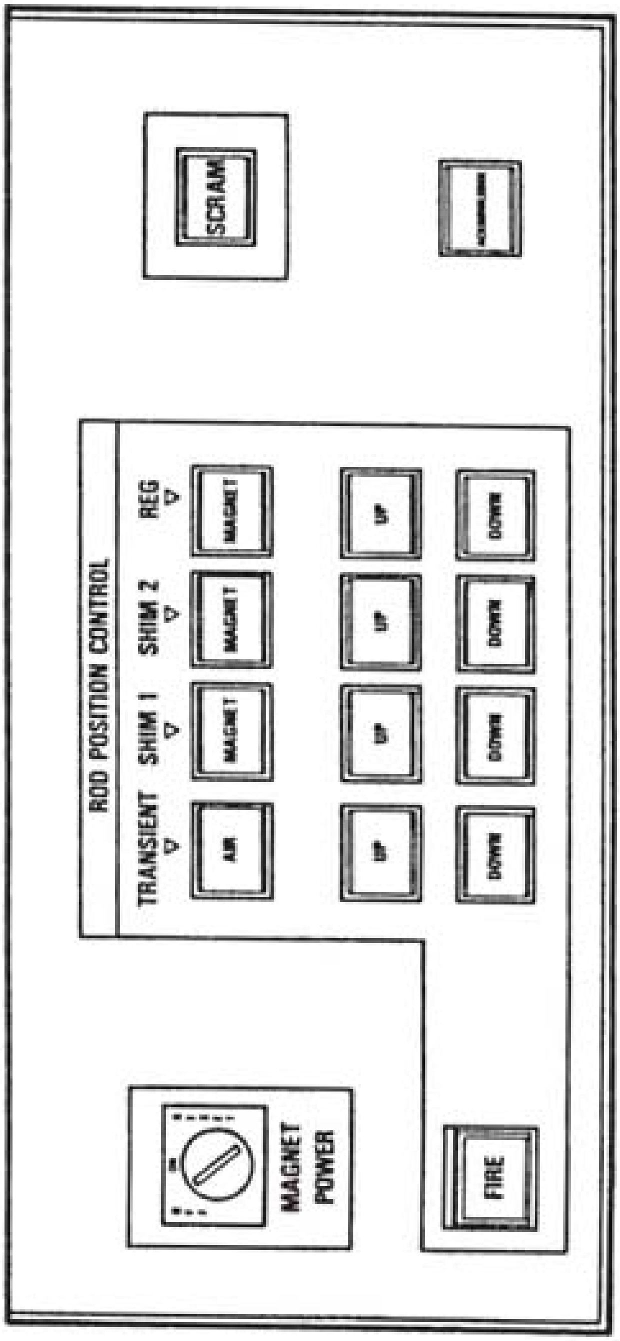
The applicant will not use the option discussed in the SER to operate in the automatic mode using control rods other than the regulating rod. The regulating rod is the only rod used in automatic mode.

7.3 <u>Instrumentation Systems</u>

7.3.1 Nuclear Instrumentation

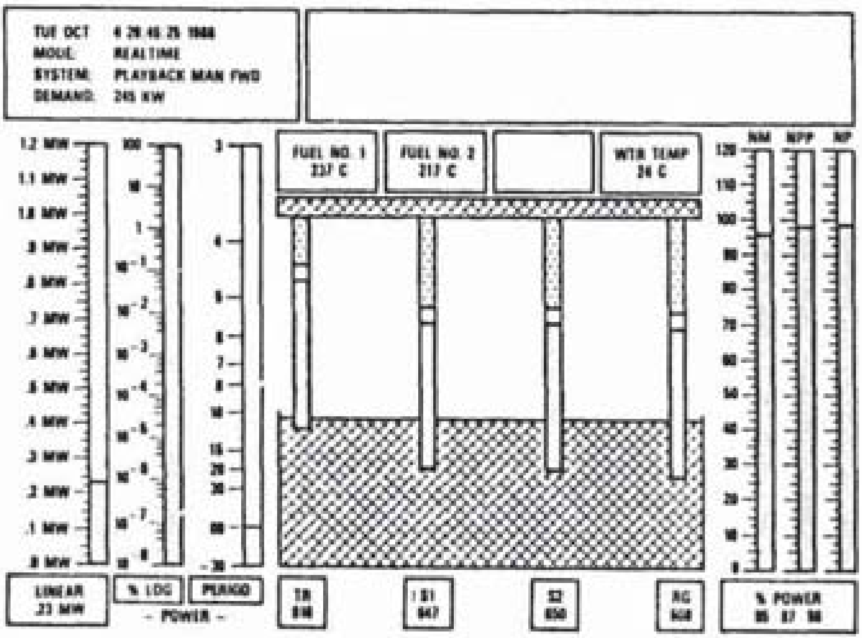
Replace Section 7.3.1 with the following:

> The nuclear instrumentation will use a multifunction computer processing the input of a low-noise fission chamber and two analog ionization channels to measure linear power level. This instrumentation will use a gamma-sensitive chamber to measure peak power and energy release during the pulsing mode.
>
> The nuclear instrumentation computer will provide (1) multirange linear power indication, (2) wide-range log power indication from source range to 150 percent of full power, (3) a separate output to the linear percent-power safety channel with power level scram, and (4) the adjustable power level scram function. The computer will receive an input signal from the fission chamber and convert it into 10 linear power ranges, which provide a more precise indication than the log channel. The computer will switch between ranges automatically. The computer will also provide a period indication and information to the adjustable period scram channel. The computer will test the instrumentation automatically to ensure that the instruments can operate at high power ranges while the reactor is operating in the low ranges and can operate at the low ranges when the reactor is at high power.
>
> The fission chamber that provides input signals to the computer is of a similar design to those previously used in the original UT TRIGA facility except that additional shielding has been used to improve the signal-to-noise ratio providing a usable signal from source range to maximum power. The new Figure 7.4 shows the operating ranges of the neutron channel.
>
> A low sensitivity ionization chamber will provide signals to a microprocessor that provides output to the control console and CRT when the reactor is in the pulsing mode.



ROD CONTROL PANEL

Figure 7.5



High-resolution graphic display

TEXAS STATUS WINDOW

Linear Power	1.0e+0
Σ Power #1	0
Σ Power #2	0
Fuel Temp #1	25
Fuel Temp #2	26
Shim #1 Position	0
Shim #2 Position	0
REG Position	50
Transient Position	600
Min Source Interlock	OK
Power >1, KW Interlock	OK
Pool Level Lo/Hi	OK
Primary Coolant Flow	0 gpm
Secondary Coolant Flow	0 gpm
HX Pool Water Inlet Temp	24.6°C
HX Pool Water Outlet Temp	26.3°C
Pressure Difference HX	OK
Pool Temperature	23.4°C
DEMIN Conductivity	2.02μmho

Date	
Time	
Reactor Mode	SCRAM
Current Pulse Number	1
Control Room Arm	1.0e-1 mR
Pool Surface Arm	2.0e-1 mR
Area 1 Arm	2.0e-1 mR
Area 2-3 Arm	2.0e-1 mR
Area 4-5 Arm	2.0e-1 mR
Portable Arm	1.0e-1 mR
Particulate Monitor	200 cpm
Stack AR-41 Monitor	20 cpm
Rx Bay Doors	OK
Rx Bay Neg Air Pressure	OK
Beam Port 1	OK
Beam Port 2	OK
Beam Port 3	OK
Beam Port 4	OK
Beam Port 5	OK

Color text display

VIDEO DISPLAY DATA

Figure 7.6

7.3.2 Nonnuclear Instrumentation

The reactor fuel thermocouples embedded in the reactor fuel provide signals that are displayed in analog bar graphs and are displayed in the status window (Figure 7.6). These signals and displays are in addition to the CRT display discussed in the original SER. The pool water temperature is displayed on the control console both on the high-resolution graphic display and on the status window. The reactor pool outlet and inlet temperature can be displayed on the status window. The I&C system does not include in the control console the temperature meter for the bulk pool water as discussed in the original SER.

In the SER, Table 7.1 listed the reactor safety system channels and Table 7.2 listed some control console alarm settings. The following are the replacements for Tables 7.1 and 7.2.

Table 7.1 Minimum reactor safety system channels

Safety Channel	Function	Set Point
Manual scram	Scram	Scram on demand
Fuel temperature	Scram	$\leq$ 550 °C
Power level (2 required)	Scram	$\leq$ 1.1 Mw
Pulse power	Scram	$\leq$ 2000 Mw
High voltage (2 required)	Scram	Loss of voltage
Magnet current	Scram	Loss of current
Watchdog (2 required)	Scram	loss of timer reset
Minimum period	Scram	As desired (not required by license)
External safety switch	Scram	As required

Table 7.2 Console alarm settings

Instrument Channel	Alarm Setting
Pool water level	24.6 ft (7.8 m) above grid plate
ΔP between primary and secondary coolant systems	5 psi (34.5 KPa)
Pool water temperature	113 °F (45 °C)

7.4 Evaluation of Instrument and Control System

7.4.1 Hardware and Systems Assessment

The staff evaluated the new control console to determine if it had vulnerabilities that might compromise its ability to present accurate information to the operator and to provide scram signals when required. The staff did not assess the reliability of the nonsafety-related controls. Issues investigated included single failure, environmental qualification, seismic qualification, power supplies, electromagnetic interference (EMI), failure modes and effects, reliability, error detection, and independence.

The primary review criteria for instrument and control systems for research reactors are presented in ANSI/ANS 15.15 (1978) "Criteria for the Reactor Safety Systems of Research Reactors." The staff performed this evaluation also using criteria that apply to current nuclear power plants. However, as discussed in Section 14 of the SER, the TRIGA design has an inherent reactivity insertion safety feature and generates minimal decay heat, thus reducing the probability of fuel damage to a minimal amount. The staff has concluded that these power plant criteria are guidelines and need not be strictly followed.

7.4.1.1 Environmental and Seismic Qualification

The new control system is installed in the control room and the reactor room. The staff considers the reactor room to be a mild environment when compared to power plant requirements. Therefore, the entire system can be considered to be in a mild environment. The system has been constructed in standard commercial enclosures suitable for a mild environment. The testing and operations have not revealed any problems regarding temperature or humidity. The new system should not be unduly susceptible to temperature or humidity and is therefore acceptable to the staff.

Although the NRC has not promulgated requirements for the seismic qualification testing of research reactor control equipment, the staff evaluated the equipment to determine general ruggedness. The equipment is mounted in a commercial quality fashion which should prevent the components from moving significantly within the console and racks. In this TRIGA reactor, an inadvertent scram does not present a significant challenge to reactor safety systems because a scram consists of the removal of current to the control rod magnets allowing the control rods to drop into the core by gravity. No other equipment is required to maintain the reactor in a safe shutdown condition. The primary concern remaining would be that the chatter of relay contacts could prevent a scram when required. The safety system scram circuits for this system are designed to scram on failure (which includes contact chatter). Therefore, the staff concludes that the system is acceptable.

7.4.1.2 Electromagnetic Interference (EMI)

The staff evaluated the new equipment to determine if common mode EMI could disable more than one system at a time. The design characteristics of the TRIGA reactor do not allow an inadvertent scram to present a significant challenge to safety systems, although it might hinder operations such as by disrupting an experiment.

The TRIGA uses industrial isolators, which prevent conducted EMI from being transmitted between the control and safety mechanisms. The neutron flux signal cables are shielded to reduce the effect of radiated EMI. Previous experience with similar equipment provided by several different vendors at other facilities has indicated that if EMI causes any perturbance in the system, it will most likely cause a scram, which is not a safety concern. Therefore, the staff concludes that EMI should not prevent a scram when required and that the design is acceptable.

7.4.1.3 Power Supplies

The power supplies for the system are buffered to reduce the effect of minor fluctuations in the line power. The scram circuits for the new system are designed to scram when power is lost to them. The NP-1000 and NPP-1000 are analog devices and will respond to power fluctuations similar to the existing analog equipment. The digital NM-1000 nuclear power channel uses a random access memory (RAM) with alternate dc battery power to store constant data during a loss of power. The NM-1000 has self-diagnostic circuits and also has a watchdog timer circuit which places the NM-1000 in a tripped condition and scrams the reactor if power fluctuations prevent the software from operating properly. The NM-1000 Software Functional Specification and Software Verification Program (March 1989) describes the tests performed on the NM-1000 to verify that the system returns to proper operation after the power is restored. The staff finds this acceptable.

7.4.1.4 Failure Modes and Effects

The applicant performed a scram circuit safety analysis to identify the various ways in which the reactor safety system could fail. These include the following:

(1) Physical system failure (wire breaks, shorts, ground fault circuits)
(2) Limiting safety system setting failure (failure to detect)
(3) System operable failure (loss of monitoring)
(4) Computer/manual control failure (automatic and manual scram)

The applicant performed this analysis using fault trees to predict a failure to scram for various failure modes. The applicant concluded that a failure of all safety systems and therefore failure to scram was extremely unlikely. The applicant evaluated all failures attributable to the unique failure modes of the software of the NM-1000. The staff has reviewed the applicant's analysis of the failure modes and effects of the new system and finds this acceptable.

7.4.1.5 Independence, Redundancy, and Diversity

The staff reviewed the data link between the safety channels and the nonsafety systems. The safety channels provide hard-wired scram inputs and are also wired directly to independent indicators on the control console. The operators receive information from both the analog NP-1000 and NPP-1000 power monitors and the digital NM-1000 monitor. The information is displayed on both direct wired bar graphs and on a graphic CRT. The safety channels also provide inputs to the non class 1E data acquisition computer (DAC) through isolators. The isolators used have not been tested for the maximum credible faults that the staff requires for isolators used in power plants. However, the manufacturer has tested them to standard commercial criteria. The staff concludes that the use of isolators tested to standard commercial criteria is acceptable for the UT TRIGA reactor. The DAC is then connected through redundant high speed serial data trunks to the non class 1E control system computer (CSC) which interfaces with the operator by controls, a keyboard, and CRT displays. The CSC would not meet the independence requirements of a power plant because the CSC does not interface with the safety channels. However, the staff concluded that this interface was not necessary for the current application at UT.

The scram circuit has a fail safe design using automatic and manual contacts which open to remove power to the control rod magnets. Redundant fuel temperature inputs are provided to the scram circuit at the UT facility. Redundant power level inputs (NP-1000, NPP-1000) to the scram circuit are also provided.

The analog and digital neutron monitors and the watchdog scram function provide additional diversity and redundancy to the scram system. The system as installed meets most of the requirements of IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," and IEEE-379-1977, "Application of the Single-Failure Criteria to Nuclear Power Generating Station Class 1E Systems."

The staff has concluded that the UT control system design maintains an acceptable level of independence, redundancy, and diversity for the UT TRIGA reactor.

7.4.1.6 Testing and Operating History

Both GA and AFRRI have extensively tested the new system and made a significant number of changes to the design during the testing and initial operation of the new system. The staff has reviewed the problems discovered during testing of the system and concluded that the resolutions appear acceptable. The staff concludes that the installation of equipment having readily available spare parts improves operability and safety. The new self-diagnostic feature allows continuous online testing and reduces the possibility of undetected failures.

7.4.2 Software Assessment

7.4.2.1 Criteria

The staff requires an approved verification and validation (V&V) plan for software that performs a safety function or provides information to the operator. At UT, the NM-1000 provides inputs to the scram circuit and to the rod withdrawal prevent interlock system block function. The staff reviewed GA's program for developing the NM-1000 software to determine if the V&V plan was acceptable. The staff compared the GA V&V plan to Regulatory Guide 1.152, "Criteria for Programmable Digital Computer Software in Safety-Related Systems at Nuclear Power Plants," which endorses ANSI.IEEE 7-4.3.2 1982, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations." The staff has concluded that this standard is appropriate for use in reviewing research reactor software.

7.4.2.2 Verification and Validation Plan

The staff audited the V&V documentation provided by GA. The NM-1000 at the UT TRIGA is wired directly into the scram circuit, and therefore requires highly reliable software to perform its safety function when required. To assess the NM-1000 software developed by GA, the staff assessed the methodology and procedures used to develop the software by reviewing the V&V documentation through the development process.

Verification and validation are two separate but related activities performed throughout the development of software. Verification is the process by which to determine if the requirements of one phase of the development cycle have been consistently, correctly, and completely transferred to the next phase of the cycle (that is, to determine if the requirements have been fulfilled).

Validation is the testing of the final product to ensure that performance conforms to the requirements of the initial specification. The need for V&V arose because software is very complex, and prone to human errors of omission, commission, and interpretation. V&V provides for an independent verifier to work in parallel with, but independent of, the development team to ensure that human errors do not hinder the production of safety software that is reliable and testable.

In executing V&V, certain principles have proven over time to be very effective in software programs:

- Well defined systems requirements expressed in well written documents
- Development methodology to guide the production of software
- Comprehensive testing procedures
- Independence of the V&V team from the development organization

These principles comprise the foundation from which to apply the applicable criteria for software evaluations of Class 1E safety systems. These principals were used by the staff as guidance in the following review areas.

7.4.2.3 Independence

The independence of the verifier is a key ingredient in an effective verification process. Sorrento Electronics developed the original software for the NM-1000. After GA obtained the rights to market the NM-1000 for research reactors, it used a software consultant to modify the software. After many changes had been made, GA hired another contractor. Each contractor provided an additional level of independent review for the original design. Although the requirements imply a concurrent review, the staff finds that the verification has been sufficiently independent and is therefore acceptable for research reactors.

7.4.2.4 Validation Testing

The validation testing must be done by a team that did not help design or implement the software product. GA used the neutron monitoring system acceptance test procedure as part of the validation testing. The staff also reviewed substantial additional validation testing performed at the AFRRI facility. The staff did note a functional description of unknown date which included samples of the computer code. Though the developers knew the specific functions which the NM-1000 was to perform, these functions had never been documented which allows possibilities for omission when preparing test procedures. Upon request from the staff, GA provided functional specification E117-1001 "NM-1000 Software Functional Specification," (March 1989) which lists in detail the functions performed by the NM-1000. This specification included a system of cross reference by which the vendor verified that each specific functional requirement had been tested. The staff finds that this testing and verification is acceptable.

7.4.2.5 Discrepancy Resolution

Each V&V program should include a process by which to identify, record, correct, and resolve discrepancies uncovered during development. The resolution of a discrepancy must be reflected in all applicable documents, including the source code, the software design specification, the software requirements, and the original systems specification. The staff reviewed discrepancies and other comments provided to GA by the Console Owners Group and found that the process and resolution were documented and appeared adequate. When discrepancies prompted GA to modify the code, GA added to the code notation a description of the changes and the corresponding rationale. The staff finds that GA used acceptable methods to resolve discrepancies.

7.4.2.6 Design Approach

The primary software specification provides the foundation for sound development and effective V&V. The individual requirements in the specification for any software system describe the manner in which the software is to behave in any circumstance. The specification must be reliable and testable. A reliable specification exhibits the following characteristics:

- Correct - Each requirement of the safety function has been stated correctly.
- Complete - All of the requirements for the safety function are included.
- Consistent - The requirements are complementary and do not contradict each other.
- Feasible - The requirements can be satisfied with available technology.
- Maintainable - The requirements will be satisfied for the lifetime of the equipment.
- Accurate - The requirements include the acceptable bounds of operation.

The staff reviewed the design approach with GA. The early development is not well documented because the product was sold to GA without all of the supporting information. Though the staff finds that the design approach for the NM-1000 since inception has been erratic, the staff finds acceptable the recent developmental work and the design approach, because it appears to be better organized and controlled.

7.4.2.7 Software Evaluation

The software development plan for the NM-1000 indicates that GA developed the software for a very specific design goal and that the designers knew the application and the basic requirements for the hardware and software. However, GA did not develop a plan to specify the individual steps in the design project. To verify that each design requirement had been tested, GA developed the NM-1000 software verification program E117-1002 "NM-1000 Software Verification Program" (March 1989). The staff also reviewed working copies of the NM-1000 design input, which demonstrated that the design team clearly understands the functional requirements. The staff concludes that the software should perform its intended safety function as required.

7.4.2.8 Operator Task Analysis

In reviewing the documents, the staff found that GA had not provided a formal task analysis to support the design of the operator interface. After the equipment and software were substantially designed, the functional requirements and working level descriptions did include the operator task requirements. The staff concluded that, through the V&V process, GA had specified the requirements and incorporated them in the design. Therefore, the task analysis is acceptable.

7.5 Conclusion

The staff concludes that the hardware design of the new GA console is acceptable for use in the UT TRIGA reactor. The software design in the CSC, DAC, and NM-1000 is acceptable because it will not prevent the safety functions of the direct wired scram circuit from performing.

8 ELECTRIC POWER SYSTEM

8.1 Electrical Power System and Emergency Power

Although no emergency power is required to safely shut the reactor down and maintain it in a safe condition, emergency power is provided for lighting for access to exits and entrances to the reactor area and building.

9 AUXILIARY SYSTEMS

9.2 Fire Protection System

The facility design includes passive fire protection elements such as fire-retardant materials and architectural features such as a fire wall between the reactor building and the academic wing.

The academic wing of the laboratory has an automatic sprinkler system with heat sensitive discharge nozzles, detectors for heat and smoke, and dampers in the ventilation system. The reactor building, except the reactor bay, has smoke and heat detectors. The ventilation system for the reactor bay has smoke detectors.

9.6 Fuel Handling and Storage

The storage racks in the pool are generally located below 8 feet (2.4 m) of water for shielding. The racks each hold six fuel elements in a linear array. The six fuel storage pits adjacent to the reactor pool are 10 inches (25.4 cm) in diameter and 15 feet (4.57 m) deep. Each pit can hold up to 19 elements and has provisions for fuel shielding and water circulation.

10 EXPERIMENTAL PROGRAMS

10.1.2 Pneumatic Transfer System

Although compressed air can be used to move the sample containers in the pneumatic transfer system, carbon dioxide or nitrogen will be routinely used to minimize the production of Ar-41. The pneumatic transfer system is exhausted to the auxiliary air exhaust system to minimize the accumulation of Ar-41 in the reactor room.

10.1.3 Rotary Specimen Rack

Figure 10.1 details the design of the rotary specimen rack.

10.2 Special Experimental Facilities

A 10,000 Ci cobalt-60 source will be located in the reactor pool. The source consists of 156 pencil-size elements that are clad with an inner cladding of aluminum and an outer cladding of stainless steel. The applicant will sample the pool water every 2 months for the presence of cobalt-60, which could indicate a source leak. The applicant has determined the level of pool water cobalt activity at which leaking sources are removed from service and isolated. Cobalt activity in the water would be controlled by the pool water demineralizer, which would remove cobalt from the water.

The source will be separated from the reactor core by at least 1.6 feet (0.5 m) of water. This will prevent the source from affecting the reactivity of the reactor and will ensure that the source is not activated by the reactor. The cobalt irradiator will be under at least 10 feet (3 m) of pool water. This will maintain dose rates under 1.0 mrem per hour outside of the reactor shield and 0.01 mrem per hour at the pool surface.

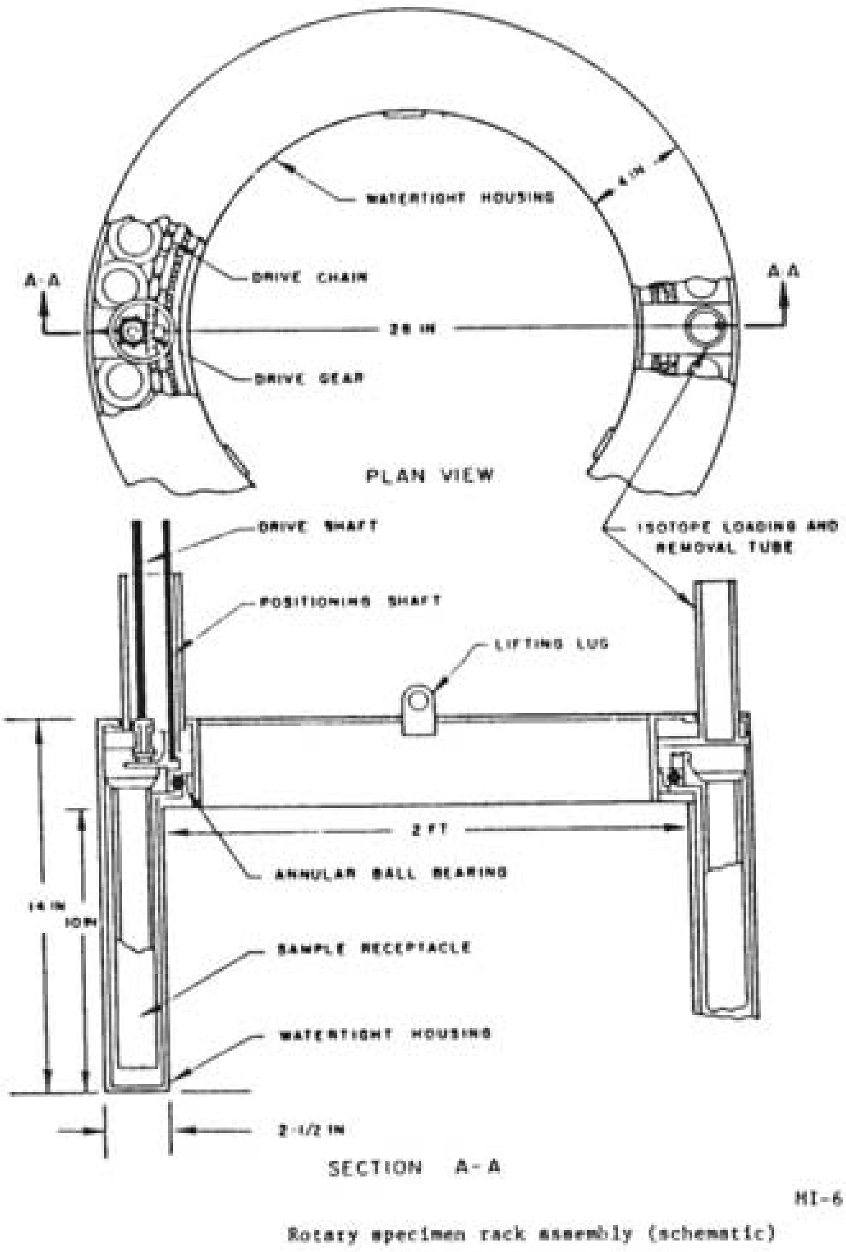
Experiments conducted with the irradiator will be subject to the Technical Specification limitations that apply to reactor experiments.

10.3 Beam Tube Facilities

The five beam tubes are 6 inches (15.2 cm) in diameter. Three of the tubes are located tangentially in relation to the reactor core and two are placed radially. Figure 10.2 provides details of the placement of the beam tubes.

10.4 Experimental Review

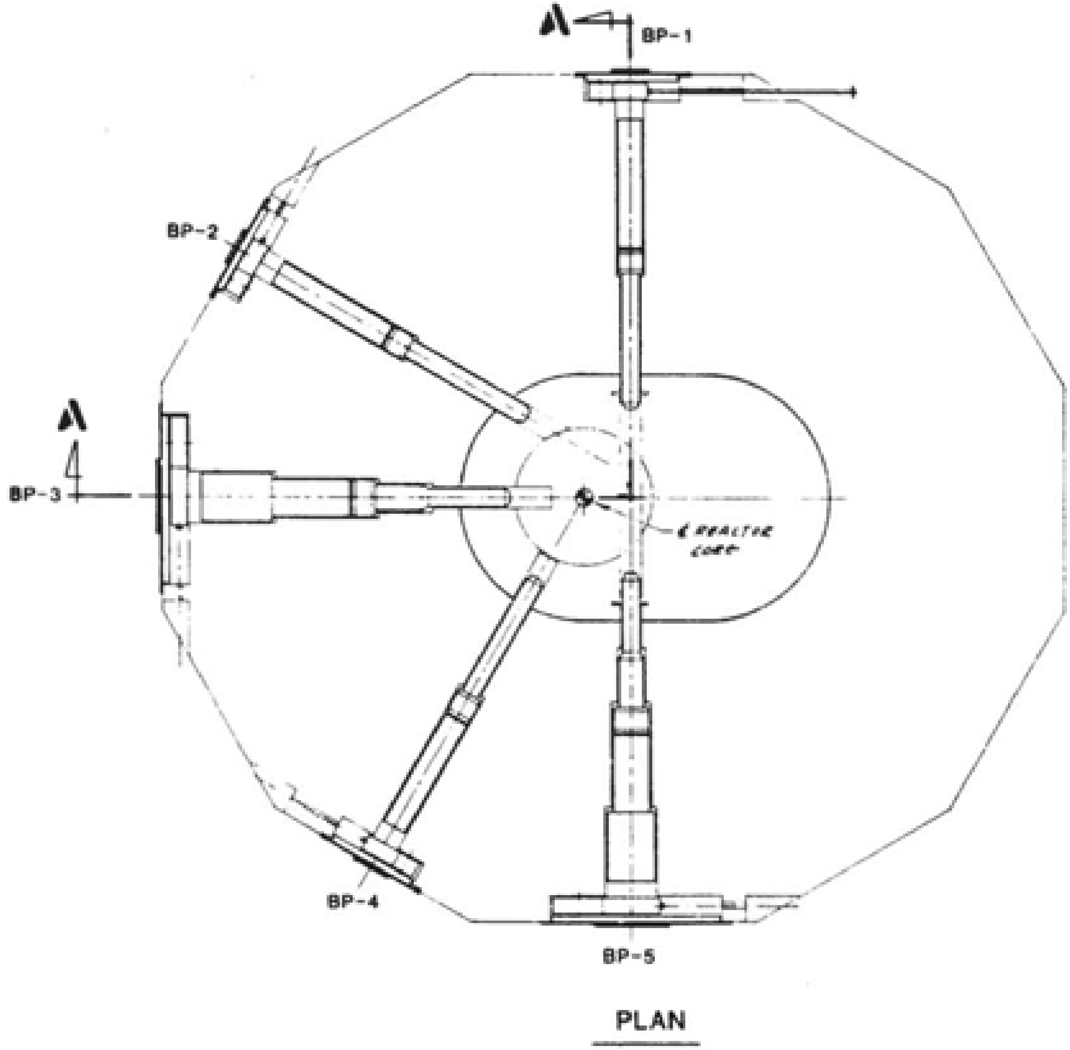
In reviewing each experiment to be conducted in the experimental facilities, the applicant will verify that the experiment conforms to the requirements in the Technical Specifications. The Technical Specifications include requirements concerning reactivity limitations, material encapsulation, irradiation of explosive material, fueled experiments, and experiments that could create airborne radioactivity.



Rotary specimen rack assembly (schematic)

ROTARY SPECIMEN FACILITY

Figure 10.1



BEAM TUBE CONFIGURATION

Figure 10.2

The UT will perform a safety analysis before conducting any experiments using the proposed three-element reactor core facility and six-element reactor core facility, if the experiment requires that holes be created in the reactor core to insert experiments by removing the specified number of fuel elements.

11 RADIOACTIVE WASTE MANAGEMENT

11.2.3 Airborne Waste

The calculations in this section are based on using the reactor 40 hours a week. The applicant believes that the dose rates produced from these calculations are very conservative and has committed to an effluent monitoring program to ensure that the actual doses are significantly less than those discussed in the SER.

12 RADIATION PROTECTION PROGRAM

12.2 Health Physics Program

12.2.4 Training

The observations of written examination and performance discussed in this section of the original SER apply to NRC-licensed reactor operators and senior reactor operators and are part of their requalification program.

12.3 Radiation Sources

12.3.1 Reactor

Access to the reactor bay will be controlled with a mechanical lock or with a security card system.

12.3.2 Extraneous Sources

The cobalt-60 irradiator will have a maximum capacity of 10,000 Ci.

12.4 Routine Monitoring

12.4.1 Fixed-Position Monitors

Fixed position area gamma radiation monitors will be located in six areas. The Technical Specifications require that monitors near the top of the reactor pool and two additional area radiation monitors be operating when the reactor is operating.

The facility also includes a continuous argon-41 air monitor located in the reactor control room area that can, by aligning valves, sample air from the reactor bay, the reactor pool access area, or the experimental systems manifold. Under abnormal conditions, this monitor can be used to obtain a particulate sample.

12.4.2 Experimental Support

A staff health physicist reviews all proposed procedures for methods of minimizing personnel exposures. The Radiation Safety Office does not review every proposed procedure as stated in the original SER.

12.5 Occupational Radiation Exposures

The applicant will provide self-reading dosimeters to visitors that could receive greater than 25 percent of the allowable dose limits.

12.8 Potential Dose Assessments

The maximum annual exposure of 100 mrem per year in the unrestricted area immediately outside the facility is above the level of natural background radiation.

13 CONDUCT OF OPERATIONS

13.1 Overall Organization

Figure 13.1 is an updated chart of the organization of the UT.

13.2 Training

On November 19, 1990, the applicant submitted by letter an updated version of the Operator Requalification Plan. The plan update was required for the applicant to maintain compliance with 10 CFR Part 55, which had been revised since the original plan was submitted and reviewed. The staff finds acceptable the applicant's procedures for training operators and the plan for operator requalification. The plan and procedures give reasonable assurance that the reactor facility will be operated with competence.

13.3 Emergency Planning

The applicant submitted its final version of the emergency plan by letter of January 11, 1991, as supplemented on April 15, 1991. The applicant's emergency plan provides reasonable assurance that the applicant is prepared to assess and respond to emergency events.

The staff reviewed the applicant's documents that address the two open items concerning the emergency plan, emergency procedures and guidance documents in support of maintaining emergency preparedness, and agreement letters with offsite support groups to the emergency plan. The staff concludes that the emergency procedures and guidance documents and offsite support group agreement letters are acceptable and that the open items are closed.

13.4 Reactor Startup Plan

The startup plan will not be appended to the Technical Specifications for the facility. The staff reviewed and found acceptable the plan as described in the SAR. The Technical Specifications still require the applicant to submit a report of startup testing to the NRC.

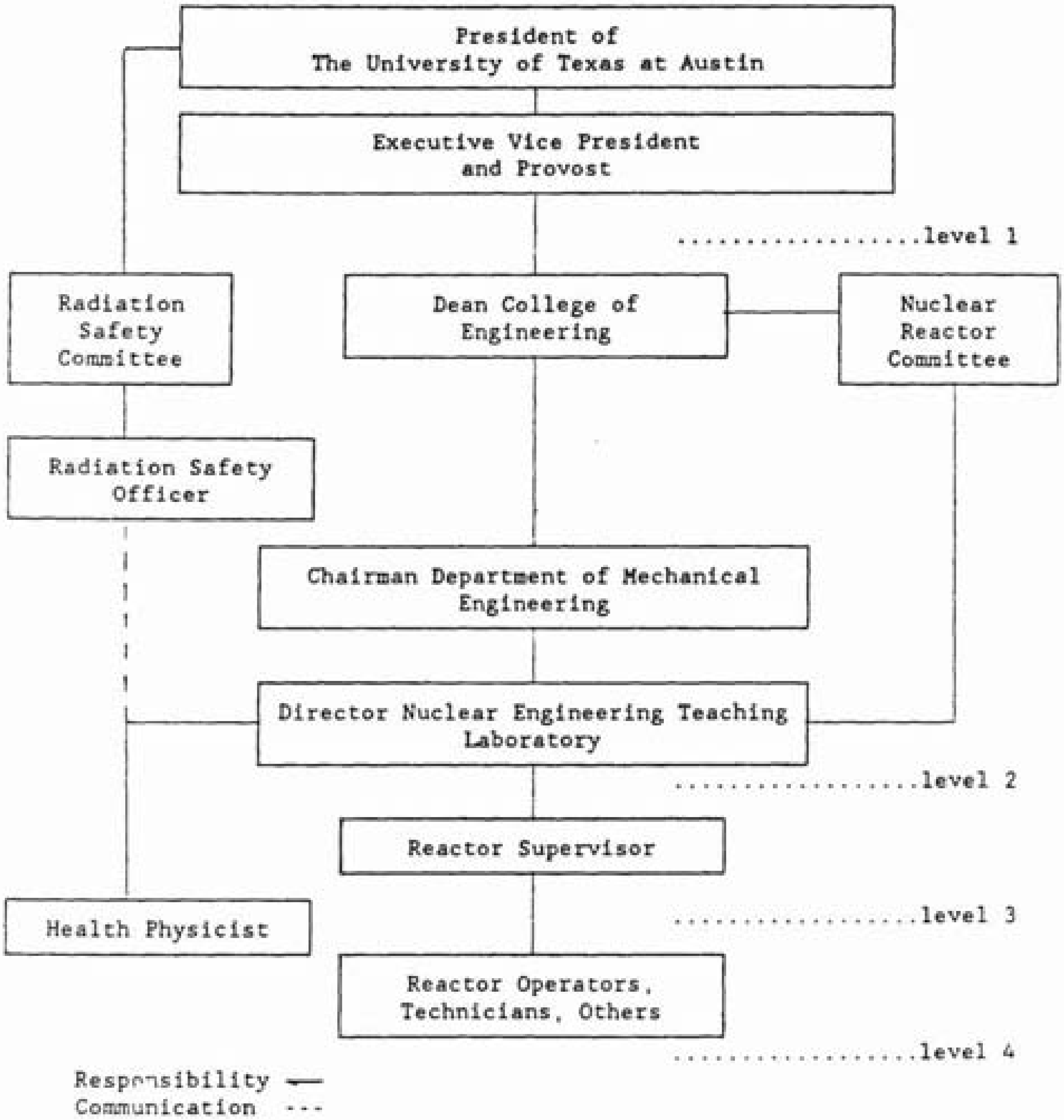
13.5 Operational Review and Audits

The Nuclear Reactor Committee has the responsibility to review the following:

- Determinations that proposed changes in equipment, systems, tests, experiments, or procedures do not involve an unreviewed safety question;

- All new procedures and major changes to procedure

- Proposed changes in reactor facility equipment or systems having safety significance



The Organization of the University of Texas at Austin
Figure 13.1

- All new experiments or classes of experiments that could affect reactivity or result in releases of radioactivity
- Changes in the Technical Specifications or license
- Violations of the license
- Operating abnormalities or violations of procedure having safety significance
- Reportable occurrences
- Audit reports

The Technical Specifications require that the results of audits preformed by the committee be reported to the Director and full committee within 3 months of the performance of the audit. The original SER stated that the results would be reported directly to the President of the University of Texas. The committee reports to the Dean of the College of Engineering who can elevate issues to the University President if the Nuclear Reactor Committee believes that elevation is necessary.

The staff finds acceptable this change in the process for reporting audits.

13.6 Quality Assurance Plan

The applicant submitted by letter of November 21, 1990, as supplemented on April 15, 1991, a revision of the quality assurance (QA) program which complies with the regulations (10 CFR 50.34) regarding the overall QA program for research reactors.

13.7 Physical Security Plan

By letter of August 13, 1990, the applicant submitted its updated physical security plan (PSP). The applicant's program for providing for the physical protection of the facility and its special nuclear material comply with the requirements of 10 CFR Part 73.

To satisfy the requirements of 10 CFR 73.67(f)(1), storage and use of special nuclear material of low strategic significance, the University has established permanent controlled access areas (CAA's) which are clearly demarcated, access to which is controlled and which affords isolation of the material or persons within them. Demarcation of the CAA's are provided through the use of normal construction type material. Access control to the CAA's is the responsibility of the facility director or supervisor. Control of access to the CAA's is established and implemented through the use of a validated access roster, a key and lock system, and escort system.

To satisfy the requirements of 10 CFR 73.67(f)(2), monitoring controlled access areas to detect unauthorized penetrations or activities, the University uses an intrusion alarm system and procedures for detecting unauthorized penetrations into the CAA's or unauthorized activities within the CAA's. To ensure the operability of the intrusion alarm system, functional and system operation tests are conducted at periodic intervals. Intrusion detection procedures are administrative in nature and are established for facility personnel and University security personnel.

To satisfy the requirements of 10 CFR 73.67(f)(3), response to unauthorized penetrations or activities, the University has designated the Chief of the University's Police Department as being responsible for security responses to the reactor facility. Back-up law enforcement is available from the City of Austin Policy Department, the Travis County Sheriff's Department, and the Texas Department of Public Safety.

To satisfy the requirements of 10 CFR 73.67(f)(4), procedures for dealing with threats and thefts of special nuclear material, the University has established and is maintaining procedures for response to specific events related to security of special nuclear material of low strategic significance. The response procedures detail the responsibilities and duties of the facility management and of the security organization.

All open items identified in the evaluation of the PSP submitted by the applicant on December 17, 1984, have been addressed by the updated PSP. The staff concludes that the PSP is acceptable because it meets the requirements of 10 CFR Part 73.

13.8 Review of Operational History

During the construction of the facility, inspectors from the NRC Region IV office conducted inspections to monitor construction progress, witness important construction events, and verify that the facility was constructed in accordance with the provisions of the construction permit. In a memorandum of July 31, 1990, from Robert D. Martin, Regional Administrator of Region IV, to Thomas E. Murley, Director of the Office of Nuclear Reactor Regulation, the regional staff determined

> that construction and preoperational testing of the University of Texas at Austin TRIGA MARK II Research Reactor have been completed in accordance with the FSAR, other docketed commitments, and regulatory requirements. We find that the facility is operationally ready with the exception of three open items.

The three open items involved the argon-41 monitor, the HEPA filter, and beam port shield plugs. In NRC Inspection Report No. 50-602/91-01, July 25, 1991, the staff confirmed that the applicant had addressed adequately the three open items and that thus these items were considered closed.

15 TECHNICAL SPECIFICATIONS

The staff has reviewed the final version of the applicant's proposed Technical Specifications of December 1990. These Technical Specifications define certain features, characteristics, and conditions governing the operation of the UT TRIGA reactor and are explicitly included in the operating license as Appendix A. The staff has reviewed the format and contents of the Technical Specifications using as a guide ANSI/ANS 15.1-1990, "Standard for the Development of Technical Specifications for Research Reactors."

The staff finds the Technical Specifications acceptable and concludes that normal plant operation within the limits of the Technical Specifications will not result in offsite radiation exposures in excess of 10 CFR Part 20 guidelines. Furthermore, the limiting conditions for operational and surveillance requirements will limit the likelihood of malfunctions and mitigate the consequences to the public of abnormal or accident events. The staff considers the open item concerning technical specifications to be closed.

16 FINANCIAL QUALIFICATIONS

The University of Texas is a State institute. The UT submitted a letter of September 24, 1991, in accordance with 10 CFR 50.75 certifying that decommissioning funds will be requested through appropriate state channels and will be obtained sufficiently before decommissioning to prevent a delay of required activities. The staff finds the decommissioning funding plans of UT to be acceptable and in accordance with the regulations.

17 OTHER LICENSE CONSIDERATIONS

17.1 Previous Use of Reactor Components

The reactor bridge assembly from the original UT TRIGA reactor was not transferred to the new reactor. A new reactor bridge was constructed. The three control rod drive mechanisms from the original reactor have been reconditioned by GA and have been tested and accepted by UT.

19 REFERENCES

Letter, Marvin M. Mendonca, NRC, to Dr. Keith A. Asmussen, General Atomics, "Issuance of Amendment No. 29 to Facility Operating License No. R-38 - General Atomics," October 4, 1990.

Bauer, T., and Goff, D., "Scram Circuit Analysis for The University of Texas TRIGA Reactor," The University of Texas at Austin, July 19, 1988.

APPENDIX

ERRATA TO THE SAFETY EVALUATION REPORT
REGARDING THE CONSTRUCTION PERMIT AND OPERATING LICENSE
FOR THE RESEARCH REACTOR AT THE UNIVERSITY OF TEXAS (DOCKET 50-602)

SER Section	Page	Change
1	1-1	Line 6, change "1" to "10"
2.1	2-1	Line 5, change "0.094 km^2" to "0.94 km^2"
2.2	2-1	Line 5, change "12 per 1076 ft^2 (100 m^2)" to "0.2 per 10758 ft^2 (1000 m^2)"
2.2	2-1	Line 6, change "1.2 to 2.0" to "0.2 to 0.3"
2.2	2-1	Line 12, change "to about" to "by"
2.2	2-1	Line 14, change "1000" to "500"
2.3	2-1	Line 6, change "1148 ft (350 m)" to "2000 ft (610 m)"
2.4	2-2	Line 18, change "333 ft^2 (30.8 m^2)" to "111,000 ft^2 (10,300 m^2)"
4.2.3	4-4	Line 1, change "americian" to "americium"
4.5	4-7	Line 16, change "^{235}U" to "^{238}U"
4.5.3	4-10	Line 2, change "$U\text{-}Zrh_x$" to "$U\text{-}ZrH_x$"
4.6.2	4-11	Line 17, change "magnet" to "cylinder"
4.6.2	4-11	Line 22, change "(3.00$)" to "(3.14$)"
4.7	4-13	Line 3 and Line 10, change "Reactor Operation Committee" to "Nuclear Reactor Committee"
7.2.2.1	7-5	Line 10, change "UP" to "up"
7.2.2.1	7-5	Line 16, change "AUTOMATIC" to "AUTO"
7.2.2.1	7-5	Line 17, change "DOWN" to "down"
7.2.2.1	7-5	Line 18, change "AUTOMATIC MODE" to "automatic mode"
8.2	8-1	Lines 4 to 11, remove these lines
10	10-1	Line 3, change "specialized" to "standard"
10.4	10-2	Line 2, change "Reactor Operation" to "Nuclear Reactor"
10.5	10-2	Line 2, Change "Radiation Safety" to "Nuclear Reactor"
11	11-1	Line 8, remove "reactor-related"
11.2.3	11-2	Line 14, change "2.1 x 10^{-3} Ci/ml" to "2.1 x 10^{-8} uCi/ml"
12	12-1	Line 1, change "radiation protection" to "radiation protection program"
12.2.1	12-1	Line 1, change "Reactor Operation Committee" to "Nuclear Reactor Committee"
12.2.2	12-1	Line 7, change "Reactor Operation Committee" to "Nuclear Reactor Committee"
12.3.2	12-2	Line 9, change "Reactor Operation Committee" to "Nuclear Reactor Committee"
12.6.1	12-3	Line 4, change "ice" to "air"

12.6.1	12-3	Line 7, change "2.2 x 10^{-5} uCi/mL" to "2.2 x 10^{-8} uCi/ml"
13.1.3	13-1	Title, line 1, and line 5, change "Reactor Operation Committee" to "Nuclear Reactor Committee"
13.5	13-3	Line 1, change "Reactor Operation Committee" to "Nuclear Reactor Committee"
13.5	13-3	Lines 1, 2, and 4, change "ROC" to "NRC"
14.1	14-2	Paragraph 5, line 15, change "will be" to "will lead to"
14.2.1	14-4	Line 15, change "insecured" to "unsecured"
14.2.2	14-4	Line 5, change "hotest" to "hottest"
14.6	14-7	Line 15, change "radical" to "radial"

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
NUREG-1135
Supplement No. 1

2. TITLE AND SUBTITLE
Safety Evaluation Report Related to the Construction Permit and Operating License for the Research Reactor at The University of Texas

Docket No. 50-602

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	1992

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)
Division of Advanced Reactors and Special Projects
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

same as 8. above

10. SUPPLEMENTARY NOTES
Docket No. 50-602

11. ABSTRACT (200 words or less)

The Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission (NRC) has prepared Supplement 1 to NUREG-1135, "Safety Evaluation Report Related to the Construction Permit and Operating License for the Research Reactor at the University of Texas" (SER) May 1985. The reactor facility is owned by The University of Texas at Austin (UT, the applicant) and is located at the University's Balcones Research Center in Austin, Texas. This supplement to the SER (SSER) describes the changes to the reactor facility design from the description in the SER. The SER and SSER together reflect the facility as built. The SSER also documents the reviews that the NRC has completed regarding the applicant's emergency plan, security plan, and technical specifications that were identified as open in the SER.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)
Safety Evaluation Report (SER)
construction permit
operating license
research reactor
The University of Texas
Supplement to the Safety Evaluation Report (SSER)

13. AVAILABILITY STATEMENT
Unlimited

14. SECURITY CLASSIFICATION
(This Page)
Unclassified
(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)