Resubmittal

of the

Technical Specifications for the
Nuclear Science Center Reactor
Facility License R-83

April 1982

The Technical Specifications for the NSCR have been revised to conform with ANS 15.1 "Standard for the Development of Technical Specifications for Research Reactors". The attached Technical Specifications are resubmitted as part of the license renewal package submitted July 1979.

The following information is provided to aid in the review of the Technical Specifications when comparing them with those submitted July 1979:

1.0 Definitions - Several new definitions have been added from ANS 15.1.

2.0 Safety Limit and Limiting Safety System Setting

No changes.

3.0 Limiting Conditions for Operation

Format of ANS 15.1 was used. Applicability, Objective and Basis were modified to establish the new format.

3.1.2 Pulse Mode Operation - Changed.

3.2.2 Reactor Safety Systems - Pool Level has been removed from Table I and Table I has a new title "Minimum Reactor Safety Circuits".

3.3 Confinement - New specification.

4.0 Surveillance Requirements

New format of ANS 15.1

4.2.4 Reactor Fuel Element - Changed.

5.0 Design Features

5.4 Radiation Monitoring System - Changed to read: "The radiation monitoring equipment listed in the following table will have these characteristics".

6.0 Administrative Controls

Changed to ANS 15.1 format.

6.1.2 Responsibility - Addition.

6.1.3 Staffing - Addition.

6.1.4 Selection and Training of Personnel - Addition

6.2.2 RSB Charter and Rules - Meeting frequency was added.

6.2.4 RSB Audit Function - Changes to areas of audits and their intervals.

6.2.5 Audit of ALARA Program - Addition.

TECHNICAL SPECIFICATIONS FOR THE
NUCLEAR SCIENCE CENTER REACTOR
FACILITY LICENSE NO. R-83

TABLE OF CONTENTS

TECHNICAL SPECIFICATIONS FOR THE
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Included in this document are the Technical Specifications and the "Bases" for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, are included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1.0 DEFINITIONS

1.1 Abnormal Occurrence - An "Abnormal Occurrence" is defined for the purposes of the reporting requirements of Section 208 of the Energy Reorganization Act of 1974 (P.L. 93-438) as an unscheduled incident or event which the Nuclear Regulatory Commission determines is significant from the standpoint of public health or safety.

1.2 ALARA - The ALARA program (As Low as Reasonably Achievable) is a program for maintaining occupational exposures to radiation and release of radioactive effluents to the environs as low as reasonably achievable.

1.3 Channel - A channel is the combination of sensors, lines, amplifiers, and output devices which are connected for the purpose of measuring the value of a parameter.

1.3.1 Channel Test - A channel test is the introduction of a signal into the channel for verification that it is operable.

1.3.2 Channel Calibration - A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.

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

1.4 Confinement - Confinement means a closure on the overall facility which controls the movement of air into it and out through a controlled path.

1.5 Core Lattice Position - The core lattice position is that region in the core (approximately 3" x 3") over a grid plug hole. It may be occupied by a fuel bundle, an experiment, or a reflector element.

1.6 Experiment - An operation, hardware, or target (excluding devices such as detectors, foils, etc.) which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within the pool, on or in a beamport or irradiation facility and which is not rigidly secured to a core or shield structure so as to be a part of their design.

1.7 Experimental Facilities - Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems and in-pool irradiation facilities.

1.8 Experiment Safety Systems - Experiment safety systems are those systems, including their associated input circuits, which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information which requires manual protective action to be initiated.

1.9 FLIP Core - A FLIP core is an arrangement of TRIGA-FLIP fuel in the reactor grid plate.

1.10 Fuel Bundle - A fuel bundle is a cluster of three or four fuel or non-fueled elements secured in a square array by a top handle and a bottom grid plate adaptor. Non-fuel elements shall be fabricated from stainless steel, aluminum, or graphite materials.

1.11 Fuel Element - A fuel element is a single TRIGA fuel rod of either standard or FLIP type.

1.12 Instrumented Element - An instrumented element is a special fuel element in which a sheathed chromel-alumel or equivalent thermocouple is embedded in the fuel near the horizontal center plane of the fuel element at a point approximately 0.3 inch from the center of the fuel body.

1.13 Limiting Safety System Setting - The limiting safety system setting is the setting for automatic protective devices related to those variables having significant safety functions.

1.14 Measuring Channel - A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output device which are connected for the purpose of measuring the value of a variable.

1.15 Measured Value - The Measured Value is the value of a parameter as it appears on the output of a channel.

1.16 Mixed Core - A mixed core is an arrangement of standard TRIGA fuel elements with at least 35 TRIGA-FLIP fuel elements located in a central contiguous region of the core.

1.17 Movable Experiment - A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.18 Operable - Operable means a component or system is capable of performing its intended function.

1.19 Operating - Operating means a component or system is performing its intended function.

1.20 Operational Core - An operational core may be a standard core, mixed core, or FLIP core for which the core parameters of shutdown margin, fuel temperature and power calibration have been determined. The maximum allowable pulse reactivity insertion will also be determined if the core is to be pulsed.

1.21 Pulse Mode - Pulse mode operation shall mean any operation of the reactor with the mode selector switch in the pulse position.

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

1.23 Reactor Console Secured - The reactor console is secured whenever all scrammable rods have been fully inserted and verified down and the console key has been removed from the console.

1.24 Reactor Operating - The reactor is operating whenever it is not secured or shutdown.

1.25 Reactor Safety Systems - Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action. In this standard, manual protective action is considered part of the reactor safety system.

1.26 Reactor Secured - A reactor is secured when:

a. It contains insufficient fissile material or moderator present in the reactor and adjacent experiments to attain criticality under optimum available conditions of moderation and reflection, or

b. The reactor console is secured, and

   1. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, and

   2. No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value of one dollar.

1.27 Reactor Shutdown - The reactor is shut down when the reactor, at ambient temperature and xenon-free condition and including the reactivity worth of all experiments, is subcritical by at least one dollar.

1.28 Reportable Occurrence - A reportable occurrence is any of the following which occurs during reactor operation:

a. Operation with actual safety-system settings for required systems less conservative than the limiting safety-system settings specified in technical specifications 2.2.

b. Operation in violation of limiting conditions for operation established in the technical specifications.

c. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns. (Note: Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

d. An unanticipated or uncontrolled change in reactivity greater than one dollar.

e. Abnormal and significant degradation in reactor fuel, or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks) where applicable which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

f. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

1.29 Rod-Control - A control rod is a device fabricated from neutron absorbing material or fuel which is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod may be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

1.30 Rod-Regulating - The regulating rod is a low-worth control rod used primarily to maintain an intended power level that need not have scram capability and may have a fueled follower. Its position may be varied manually or by the servo-controller.

1.31 Rod Shim-Safety - A shim-safety rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled follower section.

1.32 Rod-Transient - The transient rod is a control rod with scram capabilities that is capable of providing rapid reactivity insertion to produce a pulse.

1.33 Safety Channel - A safety channel is a measuring channel in the reactor safety system.

1.34 Safety Limit - Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity.

1.35 Scram Time - Scram time is the time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control rods reaches its fully inserted position.

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

1.37 Shall, Should and May - The "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation. In order to conform to this standard, the user shall conform to its requirements but not necessarily to its recommendations.

1.38 Shutdown Margin - Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition, if the most reactive rod is stuck in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.39 Standard Core - A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate.

1.40 Steady State Mode - Steady state mode operation shall mean operation of the reactor with the mode selector switch in the steady state position.

1.41 True Value - The true value is the actual value of a parameter.

1.42 Unscheduled Shutdown - An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, not to include shutdowns which occur during testing or check-out operations.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1 SAFETY LIMIT-FUEL ELEMENT TEMPERATURE

Applicability

This specification applies to the temperature of the reactor fuel.

Objective

The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element cladding will result.

Specifications

a. The temperature in a TRIGA-FLIP fuel element shall not exceed 2100°F (1150°C) under any conditions of operation.

b. The temperature in a standard TRIGA fuel element shall not exceed 1830°F (1000°C) under any conditions of operation.

Bases

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured. A loss in the integrity of the fuel element cladding could arise from a build-up of excessive pressure between the fuel-moderator and the cladding if the fuel temperature exceeds the safety limit. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The safety limit for the TRIGA-FLIP fuel element is based on data which indicate that the stress in the cladding due to the hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided the temperature of the fuel does not exceed 2100°F (1150°C) and the fuel cladding is water cooled.

The safety limit for the standard TRIGA fuel is based on data, including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided that the temperature of the fuel does not exceed 1830°F (1000°C) and the fuel cladding is water cooled.

2.2 LIMITING SAFETY SYSTEM SETTING

Applicability

This specification applies to the scram setting which prevents the safety limit from being reached.

Objective

The objective is to prevent the safety limits from being reached.

Specification

The limiting safety system setting shall be 525°C (975°F) as measured in an instrumented fuel element. The instrumented element shall be located adjacent to the central bundle with the exception of the corner positions.

Basis

The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. A setting of 525°C provides a margin of 625°C for FLIP type fuel elements and a margin of 475°C for standard TRIGA fuel elements. A part of this margin is used to account for the difference between the maximum and measured temperatures resulting from the actual location of the thermocouple. If the thermocouple element were located in the hottest position in the core, the difference between the true and measured temperatures would be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. However, this position is normally not available due to the location of the transient rod. The location of the instrumented element is therefore restricted to the positions closest to the central element. Calculations indicate that, for this case, the true temperature at the hottest location in the core will differ from the measured temperature by no more than 40%. Thus, when the temperature in the thermocouple element reaches the trip setting of 525°C, the true temperature at the hottest location in a standard core would be no greater than 632°C and 690°C in a mixed core, providing a safety margin of at least 368°C for standard fuel elements and 460°C for FLIP type elements. These margins are ample to account

for the remaining uncertainty in the accuracy of the fuel temperature measurement channel and any overshoot in reactor power resulting from a reactor transient during steady state mode operation.

In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting the "tail" of the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

3.0 Limiting Conditions for Operation

3.1 Reactor Core Parameters

3.1.1 Steady State Operation

Applicability

This specification applies to the energy generated in the reactor during steady state operation.

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded during steady state operation.

Specifications

The reactor power level shall not exceed 1.3 megawatts under any condition of operation. The normal steady state operating power level of the reactor shall be 1.0 megawatts. However, for purposes of testing and calibration, the reactor may be operated at higher power levels not to exceed 1.3 megawatts during the testing period.

Basis

Thermal and hydraulic calculations indicate that TRIGA fuel may be safely operated up to power levels of at least 2.0 megawatts with natural convection cooling.

3.1.2 Pulse Mode Operation

Applicability

This specification applies to the peak temperature generated in the fuel as the result of a pulse insertion of reactivity.

Objective

The objective is to assure that repetitive pulsing will not induce damage to the reactor fuel.

Specification

The reactivity to be inserted for pulse operation shall not exceed that amount which will produce a peak fuel temperature of 830°C. In the pulse mode the pulse rod will be limited by

mechanical means so that the reactivity insertion will not inadvertently exceed the maximum value.

Basis

TRIGA Fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 for FLIP fuel and 1.65 for Standard. This yields delta phase zirconium hydride which has a high creep strength and undergoes no phase changes at temperatures over 1000°C. However, after extensive steady state operation at 1 Mw the hydrogen will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions. When the fuel is pulsed the instantaneous temperature distribution is such that the highest values occur at the surface of the element and the lowest values occur at the center. The higher temperatures in the outer regions occur in fuel with a hydrogen to zirconium ratio that has now substantially increased above the nominal value. This produces hydrogen gas pressures considerably in excess of that expected for $ZrH_{1.6}$. If the pulse insertion is such that the temperature of the fuel exceeds 874°C, then the pressure will be sufficient to cause expansion of microscopic holes in the fuel that grows with each pulse. The pulsing limit of 830°C is obtained by examining the equilibrium hydrogen pressure of zirconium hydride as a function of temperature. The decrease in temperature from 874°C to 830°C reduces hydrogen pressure by a factor of two, which is an acceptable safety factor. This phenomenon does not alter the safety limit since the total hydrogen in a fuel element does not change. Thus, the pressure exerted on the clad will not be significantly affected by the distribution of hydrogen within the element.

In practice the pulsing limit of 830°C will be translated to a reactivity insertion limit for each specific core. The peaking factors from the thermocouple element to the hottest spot in the core must be calculated for each core configuration that is to be used. Temperature would then be measured for small pulse insertions. The pulse insertions would be increased by small increments allowing an extrapolation of peak temperatures, thereby establishing the maximum allowed pulse insertion for a given core.

3.1.3 Shutdown Margin

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods and experiments. They apply for all modes of operation.

Objective

The objective is to assure that the reactor can be shutdown at all times and to assure that the fuel temperature safety limit will not be exceeded.

Specifications

The reactor shall not be operated unless the shutdown margin provided by control rods is greater than 0.25 dollar with:

(a) The highest worth non-secured experiment in its most reactive state,

(b) The highest control rod and the regulating rod (if not scrammable) fully withdrawn, and

(c) The reactor in the cold condition without xenon.

Basis

The value of the shutdown margin assures that the reactor can be shut down from any operating condition even if the highest worth control rod should remain in the fully withdrawn position. If the regulating rod is not scrammable, its worth is not used in determining the shutdown reactivity.

3.1.4 Core Configuration Limitation

Applicability

This specification applies to mixed cores of FLIP and standard types of fuel and to full FLIP cores.

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded due to power peaking effects in mixed cores and FLIP cores.

Specifications

(a) The TRIGA core assembly may be standard, FLIP, or a combination thereof (mixed core) provided that any FLIP fuel be comprised of at least thirty-five (35) fuel elements, located in a contiguous, central region.

(b) The reactor shall not be taken critical with a core lattice position vacant except for positions on the periphery of the core assembly. Water holes in the inner fuel region shall be limited to single rod positions. Vacant core positions shall contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core.

(c) The instrumented element shall be located adjacent to the central bundle with the exception of the corner positions (Reference: 2.2 Limiting Safety System Setting).

Bases

(a) In mixed cores, it is necessary to specify the minimum number of FLIP elements and arrange them in a contiguous, central region of the core to control flux peaking and power generation values in individual elements.

(b) Vacant core positions containing experiments or an experimental facility will prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core or a single rod position to prevent power peaking in regions of high power density.

(c) Reference: 2.2 Limiting Safety System Setting.

3.2 Reactor Control and Safety Systems

3.2.1 Reactor Control Systems

Applicability

This specification applies to the information which must be available to the reactor operator during reactor operation.

Objective

The objective is to require that sufficient information is available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the measuring channels listed in the following table are operable.

Measuring Channel	Min. No. Operable	Effective Mode S.S.	Effective Mode Pulse
Fuel Element Temperature	1	X	X
Linear Power Level	1	X	
Log Power Level	1	X	
Integrated Pulse Power	1		X

Bases

Fuel temperature displayed at the control console gives continuous information on this parameter which has a specified safety limit. The power level monitors assure that the reactor power level is adequately monitored for both steady state and pulsing modes of operation. The specifications on reactor power level indication are included in this section since the power level is related to the fuel temperature.

3.2.2 Reactor Safety Systems

Applicability

This specification applies to the reactor safety system circuits.

Objective

The objective is to specify the minimum number of reactor safety system channels that must be operable for safe operation.

Specification

The reactor shall not be operated unless the safety circuits described in Table 1 are operable.

TABLE 1

Minimum Reactor Safety Circuits

Safety Channel	Number Operable	Function	Effective Mode S.S.	Effective Mode Pulse
Fuel Element Temperature	1	SCRAM @ LSSS	X	X
Hi Power Level	2	SCRAM @ 125%	X	
Console Scram Button	1	SCRAM	X	X
Hi Power Level Detector Power Supply	2	SCRAM on loss of supply voltage	X	
Preset Timer	1	Transient rod scram 15 seconds or less after pulse		X
Log Power	1	Prevent withdrawal of shim-safeties at $<4 \times 10^{-3}$ watts	X	
Log Power	1	Prevent pulsing above 1 kW		X
Transient Rod Position	1	Prevent application of air unless fully inserted	X	
Shim-safeties & Regulating Rod Position	1	Prevent withdrawal		X

Bases

The fuel temperature and power level scrams provide protection to assure that the reactor can be shutdown before the safety limit on the fuel element temperature will be exceeded. The manual scram allows the operator to shut down the system if

an unsafe or abnormal condition occurs. In the event of failure of the power supply for a safety chamber, operation of the reactor without adequate instrumentation is prevented. The preset timer insures that the reactor power level will reduce to a low level after pulsing.

The interlock to prevent startup of the reactor at power levels less than 4×10^{-3} watts which corresponds to approximately 2 cps assures that sufficient neutrons are available for proper startup.

The interlock to prevent the initiation of a pulse above 1 kW is to assure that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing the reactor in the steady state mode. The interlock to prevent withdrawal of the shim-safeties or regulating rod in the pulse mode is to prevent the reactor from being pulsed while on a positive period.

3.2.3 Scram Time

Applicability

This specification applies to the time required for the scrammable control rods to be fully inserted from the instant that a safety channel variable reaches the Safety System Setting.

Objective

The objective is to achieve prompt shutdown of the reactor to prevent fuel damage.

Specification

The scram time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds.

Basis

This specification assures that the reactor will be promptly shutdown when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to assure the safety of the reactor.

3.3 Confinement

3.3.1 Operations That Require Confinement

Applicability

This specification applies to confinement requirements during operation of the reactor and the handling of radioactive materials.

Objective

To maintain normal or emergency air flow into and out of the reactor building during operations that produce or could potentially produce airborne radioactivity.

Specification

Confinement of the reactor building will be required during the following operations.*

(a) Reactor operating

(b) Handling of radioactive materials with the potential for airborne release.

*For periods of time for maintenance to the central exhaust fan, entry doors to the reactor building will remain closed except for momentary opening for personnel entry or exit.

Basis

(a) This basis applies during the conduct of those activities defined as reactor operations. Argon-41 is produced during operation of the reactor in experimental facilities and in the reactor pool; thus, air control within the building and the exhaust system is necessary to maintain proper airborne radiation levels in the reactor building and release levels in the exhaust stack. Other radioactivity releases to the reactor building must be

considered during reactor operation, such as fission product release from a leaking fuel element or a release from fixed experiments in or near the core.

(b) The handling of radioactive materials can result in the accidental or controlled release of airborne radioactivity to the reactor building environment or direct release to the building exhaust system. In these cases the control of air into and out of the reactor building is necessary.

3.3.2 Equipment to Achieve Confinement

Applicability

This specification applies to the equipment and controls needed to provide confinement of the reactor building.

Objective

The objective is to assure that a minimum of equipment is in operation to achieve confinement as specified in 3.3.1 and that the control panel for this equipment is available for normal and emergency situations.

Specification

(a) The minimum equipment required to be in operation to achieve confinement of the reactor building shall be the central exhaust fan.*

(b) The controls for operation of the ventilation system during normal and emergency conditions shall be located in the reception room.

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*For periods of time for maintenance to the central exhaust fan, entry doors to the reactor building will remain closed to assure closure except for momentary opening for personnel entry or exit.

Bases

(a) Operation of the central exhaust fan will achieve confinement of the reactor building during normal and emergency conditions when the controls for air input are set such

that the central exhaust fan capacity remains greater than the amount of air being delivered to the reactor building. The exhaust fan has sufficient capacity to handle extra air intake to the building during momentary opening of entry doors.

(b) Controls for the ventilation system provide for the manual selection of air input to the reactor building and the automatic or manual selection of air removal. The two systems work together to maintain a small negative pressure in the reactor building. These controls are located in the reception room for accessibility during emergency conditions.

3.4 Ventilation System

Applicability

This specification applies to the operation of the facility ventilation system.

Objective

The objective is to assure that the ventilation system is in operation to mitigate the consequences of the possible release of radioactive materials resulting from reactor operation.

Specification

The reactor shall not be operated unless the facility ventilation system is operable, except for periods of time necessary to permit repair of the system. In the event of a substantial release of airborne radioactivity, the ventilation system will be secured automatically by signals from an exhaust air radiation monitor.

Basis

During normal operation of the ventilation system, the concentration of Argon-41 in unrestricted areas is below MPC (SAR, Section IX). In the event of a substantial release of airborne radioactivity, the ventilation system will be secured automatically. Therefore, operation of the reactor with the ventilation system shutdown for short periods of time to make repairs insures the same degree of control of release of radioactive materials. Moreover, radiation

monitors within the building independent of those in the ventilation system will give warning of high levels of radiation that might occur during operation with the ventilation system secured.

3.5 Radiation Monitoring Systems and Effluents

3.5.1 Radiation Monitoring

Applicability

This specification applies to the radiation monitoring information which must be available to the reactor operator during reactor operation.

Objective

The objective is to assure that sufficient radiation monitoring information is available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the radiation monitoring channels listed in the following table are operable.

Radiation Monitoring Channels*	Function	Number
Area Radiation Monitor	Monitor radiation levels within the reactor room	1
Continuous Air Radiation Monitor	"	1
Exhaust Gas Radiation Monitor	Monitor radiation levels in the exhaust air stack	1
Exhaust Particulate Radiation Monitor	"	1

*For periods of time for maintenance to the radiation monitoring channels, the intent of this specification will be satisfied if they are replaced with portable gamma sensitive instruments having their own alarms or which shall be kept under visual observation.

Bases

The radiation monitors provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings.

3.5.2 Argon-41 Discharge Limit

Applicability

This specification applies to the concentration of Argon-41 that may be discharged from the TRIGA reactor facility.

Objective

To insure that the health and safety of the public is not endangered by the discharge of Argon-41 from the TRIGA reactor facility.

Specification

The concentration of Argon-41 in the effluent gas from the facility as diluted by atmospheric air in the lee of the facility due to the turbulent wake effect shall not exceed 4.8×10^{-8} μCi/ml averaged over one year.

Basis

The maximum allowable concentration of Argon-41 in air in unrestricted areas as specified in Appendix B, Table II of 10 CFR 20 is 4.0×10^{-8} μCi/ml. Section IX of the S. A. R. for the NSCR substantiates a 5.0×10^{-3} atmospheric dilution factor for a 2.0 mph wind speed. This dilution factor represents the conditions at the site building for a wind speed of 2 mph, which occurs less than 10% of the time on an annual basis.

3.6 Limitations on Experiments

3.6.1 Reactivity Limits

Applicability

This specification applies to the reactivity limits on experiments installed in the reactor and its experimental facilities.

Objective

The objective is to assure control of the reactor during the handling of experiments adjacent to or in the reactor core.

Specifications

The reactor shall not be operated unless the following conditions governing experiments exist.

(a) Non-secured experiments shall have reactivity worths less than one dollar.

(b) The reactivity worth of any single experiment shall be less than two dollars.

Bases

(a) This specification is intended to provide assurance that the worth of a single unfastened experiment will be limited to a value such that the safety limit will not be exceeded if the positive worth of the experiment were to be suddenly inserted.

(b) The maximum worth of a single experiment is limited so that its removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit. Since experiments of such worth must be fastened in place, its removal from the reactor operating at full power would result in a relatively slow power increase such that the reactor protective systems would act to prevent high power levels from being attained.

3.6.2 Material Limitations

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactivity by limiting material quantity and radioactive material inventory of the experiment.

Specifications

(a) Explosive materials in quantities greater than 5 pounds shall not be allowed within the reactor building. Irradiation of explosive materials shall be restricted as follows:

(1) Explosive materials in quantities greater than 25 milligrams shall not be irradiated in the reactor pool. Explosive materials in quantities less than 25 milligrams may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the container.

(2) Explosive materials in quantities greater than 25 milligrams shall be restricted from the reactor pool, the upper research level, demineralizer room, cooling equipment room and the interior of the pool containment structure.

(3) Explosive materials in quantities greater than 5 pounds shall not be irradiated in experimental facilities.

(4) Cumulative exposures for explosive materials in quantities greater than 25 milligrams shall not exceed 10^{12} n/cm^2 for neutrons or 25 roentgen for gamma exposures.

(b) Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies.

Bases

(a) This specification is intended to prevent damage to the reactor or reactor safety systems resulting from failure of an experiment involving explosive materials.

(1) This specification is intended to prevent damage to the reactor core and safety related reactor components located within the reactor pool in the event of failure of an experiment involving the irradiation of explosive materials. Limited quantities of less than 25 milligrams and proper containment of such experiment provide the required safety for in-pool irradiation.

(2) This specification is intended to prevent damage to vital equipment by restricting the quantity and location of explosive materials within the reactor building. Explosives in quantities exceeding 25 milligrams are restricted from areas containing the reactor bridge, reactor console, pool water coolant and purification systems and reactor safety related equipment.

(3) The failure of an experiment involving the irradiation of up to 5 lbs of explosive material in an experimental facility located external to the reactor pool structure will not result in damage to the reactor or the reactor pool containment structure.

(4) This specification is intended to prevent any increase in the sensitivity of explosive materials due to radiation damage during exposures.

(b) The 1.5 curie limitation on iodine 131 through 135 assures that in the event of failure of a fueled experiment leading to total release of the iodine, the exposure dose at the exclusion area boundary will be less than that allowed by 10 CFR Part 20 for an unrestricted area.

3.6.3 Failure and Malfunctions

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

(a) Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, or (3) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limit of Appendix B of 10 CFR Part 20.

(b) In calculations pursuant to (a) above, the following assumptions shall be used:

(1) If the effluent from an experimental facility exhausts through a holdup tank which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.

(2) If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3-micron particles, at least 10% of these vapors can escape.

(3) For materials whose boiling point is above 130°F and where vapors formed by boiling this material can escape only through an undistributed column of water above the core, at least 10% of these vapors can escape.

(c) If a capsule fails and releases material which could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Director or his designated alternate and determined to be satisfactory before operation of the reactor is resumed.

Bases

(a) This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary of the NSC.

(b) These assumptions are used to evaluate the potential airborne radioactivity release due to an experiment failure.

(c) Operation of the reactor with reactor fuel or structure damage is prohibited to avoid release of fission products. Potential damage to reactor fuel or structure must be brought to the attention of the Director (NSC) or his designated alternate for review to assure safe operation of the reactor.

4.0 SURVEILLANCE REQUIREMENTS

4.1 GENERAL

Applicability

This specification applies to the surveillance requirements of any system related to reactor safety.

Objective

The objective is to verify the proper operation of any system related t reactor safety.

Specifications

Any additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the rod drive mechanism, or the reactor safety system shall be ma:le and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications approved by the Reactor Safety Board. A system shall not be considered operable until after it is successfully tested.

Basis

This specification relates to changes in reactor systems which could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications, then it can be assumed that they meet the presently accepted operating criteria.

4.2 REACTOR CORE PARAMETERS

4.2.1 Steady State Operation

Applicability

This specification applies to the surveillance requirement of the power level monitoring channels.

Objective

The objective is to verify that the maximum power level of the reactor meets the license requirements.

Specification

A Channel Calibration shall be made of the power level monitoring channels by the calorimetric method annually but at intervals not to exceed 14 months.

Basis

The power level channel calibration will assure that the reactor will be operated at the proper power levels.

4.2.2 Pulse Mode Operation

Applicability

This specification applies to the surveillance requirements for operation of the reactor in the pulse mode.

Objective

The objective is to verify that operation of the reactor in the pulse mode is proper and safe and to determine if any significant changes in fuel characteristics have occurred.

Specification

The reactor shall be pulsed semiannually to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value or the reactor shall not be pulsed until such comparative pulse measurements are performed.

Basis

The reactor is pulsed at suitable intervals to make a comparison with previous similar pulses and to determine if changes in fuel or core characteristics are taking place.

4.2.3 Shutdown Margin

Applicability

This specification applies to the surveillance requirement of control rod calibrations and shutdown margin.

Objective

The objective is to verify that the requirements for shutdown margins are met for operational cores.

Specification

The reactivity worth of each control rod and the shutdown margin shall be determined annually but at intervals not to exceed 14 months.

Basis

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide an accurate means for determining the reactivity worths of experiments inserted in the core. Past experience with TRIGA reactors gives assurance that measurement of the reactivity worth on an annual basis is adequate to insure no significant changes in the shutdown margin.

4.2.4 Reactor Fuel Elements

Applicability

This specification applies to the surveillance requirements for the fuel elements.

Objective

The objective is to verify the continuing integrity of the fuel element cladding and to ensure that no fuel damage has occured.

Specification

At least four fuel elements, which occupy the highest specific power density positions in the core shall be inspected visually for damage or deterioration and measured for length and bend annually, not to exceed 15 months. If any element is found to be damaged, the entire core will be inspected. The reactor shall not be operated with damaged fuel. A fuel element shall be considered damaged and must be removed from the core if:

a. In measuring the transverse bend, the bend exceeds 0.125 inch over the length of the cladding.

b. In measuring the elongation, its length exceeds its original length by 0.125 inch, or

c. A clad defect exists as indicated by release of fission products.

Basis

The frequency of inspection and measurement schedule is based on the parameters most likely to affect the fuel cladding of a pulsing reactor operated at moderate pulsing levels and utilizing fuel elements whose characteristics are well known. Experience has shown that temperature is the major contributor to fuel damages. By examining the 4 "lead" elements, the remainder of the fuel does not require inspection due to operation at substantially lower temperatures.

The limit of transverse bend has been shown to result in no difficulty in disassembling fuel bundles. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel element bowing that could result in touching has occurred without deleterious effects. The elongation limit has been specified to assure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to assure adequate coolant flow.

4.3 REACTOR CONTROL AND SAFETY SYSTEMS

4.3.1 Reactor Control Systems

Applicability

These specifications apply to the surveillance requirements for reactor control systems.

Objective

The objective is to verify the condition and operability of system components affecting safe and proper control of the reactor.

Specifications

a. The control rods shall be visually inspected for deterioration at intervals not to exceed 2 years.

b. The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated as necessary semiannually at intervals not to exceed 8 months.

Basis

The visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation of the reactor. Inspection and maintenance of the transient rod drive assembly reduces the probability of failure of the system due to moisture-induced corrosion of the pulse cylinder and piston rod assembly.

4.3.2 Reactor Safety Systems

Applicability

These specifications apply to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems.

Objective

The objective is to verify the performance and operability of those systems and components which are directly related to reactor safety.

Specifications

a. A Channel Test of each of the reactor safety system channels for the intended mode of operation shall be performed prior to each day's operation or prior to each operation extending more than one day, except for the pool level channel which shall be tested weekly.

b. Whenever a reactor scram caused by high fuel element temperature occurs, an evaluation shall be conducted to determine whether the fuel element temperature safety limit was exceeded.

c. A calibration of the temperature measuring channels shall be performed semiannually but at intervals not to exceed 8 months.

d. A Channel Check of the fuel element temperature measuring channel shall be made daily whenever the reactor is operated by recording a measured value of a meaningful temperature indication.

Basis

Channel tests will assure that the safety system channels are operable on a daily basis or prior to an extended run. Operational experience with the TRIGA system gives assurance that the thermocouple measurements of fuel element temperatures have been sufficiently reliable to assure accurate indication of this parameter.

4.3.3 Scram Time

Applicability

This specification applies to the surveillance of control rod scram times.

Objective

The objective is to verify that all scrammable control rods meet the scram time requirement.

Specification

The scram time shall be measured annually but at intervals not to exceed 14 months.

Basis

Measurement of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control rods to perform properly.

4.4 EQUIPMENT TO ACHIEVE CONFINEMENT - VENTILATION SYSTEM

Applicability

This specification applies to the building confinement ventilation system.

Objective

The objective is to assure the proper operation of the ventilation system in controlling releases of radioactive material to the uncontrolled environment.

Specification

It shall be verified weekly that the ventilation system is operable.

Basis

Experience accumulated over several years of operation has demonstrated that the tests of the ventilation system on a weekly basis are sufficient to assure the proper operation of the system and control of the release of radioactive material.

4.5 RADIATION MONITORING SYSTEMS AND EFFLUENTS

Applicability

This specification applies to the surveillance requirements for the area radiation monitoring equipment and the continuous air monitoring system.

Objective

The objective is to assure that the radiation monitoring equipment is operating and to verify the appropriate alarm settings.

Specification

The area radiation monitoring system and the continuous air monitoring system shall be calibrated annually but at intervals not to exceed 14 months and shall be verified to be operable at weekly intervals.

Basis

Experience has shown that weekly verification of area radiation and air monitoring system set points in conjunction with annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span.

4.6 EXPERIMENTS

Applicability

This specification applies to the surveillance requirements for experiments installed in the reactor and its experimental facilities and for irradiations performed in the irradiation facilities.

Objective

The objective is to prevent the conduct of experiments or irradiations which may damage the reactor or release excessive amounts of radioactive materials as a result of failure.

Specifications

a. A new experiment shall not be installed in the reactor or its experimental facilities until a hazards analysis has been performed and reviewed for compliance with the Limitations on Experiments, Section 3.9 by the Reactor Safety Board. Minor modifications to a reviewed and approved experiment may be made at the discretion of the senior reactor operator responsible for the operation provided that the hazards associated with the modifications have been reviewed and a determination made and documented that the modifications do not create a significantly different, a new, or a greater safety risk than the original approved experiment.

b. The performance of an experiment classified as an approved experiment shall not be performed until it has been reviewed for compliance by a licensed senior operator and a person qualified in health physics.

c. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.

Basis

It has been demonstrated over a number of years of experience that experiments and irradiations reviewed by the Reactor Staff and the Reactor Safety Board as appropriate can be conducted without endangering the safety of the reactor or exceeding the limits in the Technical Specifications.

5.0 DESIGN FEATURES

5.1 REACTOR FUEL

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

a. TRIGA-FLIP Fuel

The individual unirradiated FLIP fuel elements shall have the following characteristics:

(1) Uranium content: maximum of 9 Wt-% enriched to nominal 70% Uranium 235.

(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms.

(3) Natural erbium content (homogeneously distributed): nominal 1.5 Wt-%.

(4) Cladding: 304 stainless steel, nominal 0.020 inch thick.

(5) Identification: Top pieces of FLIP elements will have characteristic markings to allow visual identification of FLIP elements employed in mixed cores.

b. Standard TRIGA fuel

The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

(1) Uranium content: maximum of 9.0 Wt-% enriched to a nominal 20% Uranium 235.

(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.7 H atoms to 1.0 Zr atoms.

(3) Cladding: 304 stainless steel, nominal 0.020 inch thick.

Bases

a. A maximum uranium content of 9 Wt-% in a TRIGA-FLIP element is about 6% greater than the design value of 8.5 Wt-%. Such an increase in loading would result in an increase in power density of about 2%. Similarly, a minimum erbium content of 1.1% in an element is about 30% less than the design value. This variation would result in an increase in power

density of only about 6%. An increase in local power density of 6% reduces the safety margin by at most ten percent. The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad.

When standard and FLIP fuel elements are used in mixed cores, visual identification of types of elements is necessary to verify correct fuel loadings. The accidental rotation of fuel bundles containing standard and FLIP elements can be detected by visual inspection. Should this occur, however, studies of a single FLIP element accidently rotated into a standard fuel region indicate an insubstantial increase in power generation in the FLIP element.

b. A maximum uranium content of 9 Wt-% in a standard TRIGA element is about 6% greater than the design value of 8.5 Wt.%. Such an increase in loading would result in an increase in power density of less than 6%. An increase in local power density of 6% reduces the safety margin by at most 10%. The maximum hydrogen-to-zirconium ratio of 1.8 will produce a maximum pressure within the clad during an accident well below the rupture strength of the clad.

5.2 REACTOR CORE

Applicability

This specification applies to the configuration of fuel and and in-core experiments.

Objective

The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications

a. The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

b. The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water.

Bases

a. Standard TRIGA cores have been in use for years and their characteristics are well documented. FLIP cores have been operated at General Atomics and the Puerto Rico Nuclear Center and their operational characteristics are available. General Atomics has also performed a series of experiments using standard and FLIP fuel in mixed cores. In addition, studies performed at Texas A&M for a variety of mixed core arrangements and operational experience with mixed cores indicate that such loadings would safely satisfy all operational requirements.

b. The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

5.3 CONTROL RODS

Applicability

This specification applies to the control rods used in the reactor core.

Objective

The objective is to assure that the control rods are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specification

a. The shim-safety control rods shall have scram capability and contain borated graphite, B_4C powder or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding. These rods may incorporate fueled followers which have the same characteristics as the fuel region in which they are used.

b. The regulating control rod need not have scram capability and shall be a stainless rod or contain the materials as specified for shim-safety control rods. This rod may incorporate a fueled follower.

c. The transient control rod shall have scram capability and contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The transient rod shall have an adjustable upper limit to allow a variation of reactivity insertions. This rod may incorporate an aluminum or air follower.

Bases

The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, B_4C powder or boron and its compounds. Since the regulating rod normally is a low worth rod, its function could be satisfied by using a solid stainless steel rod. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to insure mechanical stability during movement and to isolate the poison from the pool water environment. Control rods that are fuel followed provide additional reactivity to the core and increase the worth of the control rod. The use of fueled followers in the FLIP region has the additional advantage of reducing flux peaking in the water filled regions vacated by the withdrawal of the control rods. Scram capabilities are provided for rapid insertion of the control rods which is the primary safety feature of the reactor. The transient control rod is designed for a reactor pulse. The nuclear behavior of the air or aluminum follower which may be incorporated into the transient rod is similar to a void. A voided follower may be required in certain core loadings to reduce flux peaking values.

5.4 RADIATION MONITORING SYSTEM

Applicability

This specification describes the functions and essential components of the area radiation monitoring equipment and the system for continuously monitoring airborne radioactivity.

Objective

The objective is to describe the radiation monitoring equipment that is available to the operator to assure safe operation of the reactor.

Specification

The radiation monitoring equipment listed in the following table will have these characteristics.

Radiation Monitoring Channel and Function

Area Radiation Monitor (gamma sensitive instruments)
Function - Monitor radiation fields in key locations, alarm and readout at control console and readout in reception room.

Continuous Air Radiation Monitor (beta - gamma sensitive detector with air collection capability)
Function - Monitor concentration of radioactive particulate activity in building, alarm and readout at control console, and readout in reception room.

Gas and Particulate Stack Radiation Monitors (gamma and beta-gamma sensitive detectors with air collection calability)

Function - Monitor concentration of radioactive particulate activity and radioactive gases in building exhaust, alarm and readout at control console, and readout in reception room.

Basis

The radiation monitoring system is intended to provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings.

5.5 FUEL STORAGE

Applicability

This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective

The objective is to assure that fuel which is being stored will not become critical and will not reach an unsafe temperature.

Specifications

a. All fuel elements shall be stored in a geometrical array where the k-effective is less tnan 0.8 for all conditions of moderation.

b. Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel element or fueled device temperature will not exceed design values.

Basis

The limits imposed by Specifications 5.5.a and 5.5.b are conservative and assure safe storage.

5.6 REACTOR BUILDING AND VENTILATION SYSTEM

Applicability

This specification applies to the building which houses the reactor.

Objective

The objective is to assure that provisions are made to restrict the amount of release of radioactivity into the environment.

Specifications

a. The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall be 180,000 cubic feet.

b. The reactor building shall be equipped with a ventilation system designed to filter and exhaust air or other gases from the reactor building and release them from a stack at a minimum of 85 feet from ground level.

c. Emergency shutdown controls for the ventilation system shall be located in the reception room and the system shall be designed to shut down in the event of a substantial release of fission products.

Bases

The facility is designed such that the ventilation system will normally maintain a negative pressure with respect to the atmosphere so that there will be no uncontrolled leakage to the environment. The free air volume within the reactor building is confined when there is an emergency shutdown of the ventilation system. Controls for startup, emergency filtering, and normal operation of the ventilation system are located in the reception room. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the reception room with a minimum of exposure to operating personnel

5.7 REACTOR POOL WATER SYSTEMS

Applicability

This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications

a. The reactor core shall be cooled by natural convective water flow.

b. The pool water inlet and outlet pipe to the demineralizer shall not extend more than 15 feet below the top of the reactor pool when fuel is in the core.

c. Diffuser and skimmer pumps shall be located no more than 15 feet below the top of the reactor pool.

d. Pool water inlet and outlet pipe to the heat exchanger shall have emergency covers within the reactor pool for manual shut off in case of pool water loss due to external pipe system failure.

e. A pool level alarm shall indicate loss of coolant if the pool level drops approximately 10% below operating level.

Bases

a. This specification is based on thermal and hydraulic calculations which show that the TRIGA-FLIP core can operate in a safe manner at power levels up to 2,700 kW with natural convection flow of the coolant water. A comparison of operation of the TRIGA-FLIP and standard TRIGA Mark III has shown them to be safe for the above power level. Thermal and hydraulic characteristics of mixed cores are essentially the same as that for TRIGA-FLIP and standard cores.

b. In the event of accidental siphoning of pool water through inlet and outlet pipes of the demineralizer system, the pool water level will drop no more than 15 feet from the top of the pool.

c. In the event of pipe failure and siphoning of pool water through the skimmer and diffuser water systems, the pool water level will drop no more than 15 feet from the top of the pool.

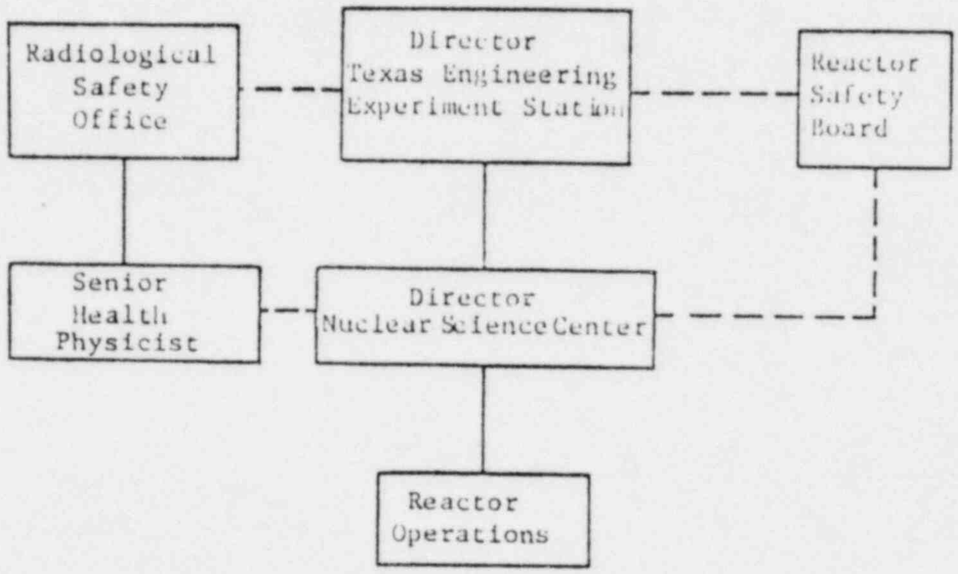
d. Inlet and outlet coolant lines to the pool heat exchanger terminate at the bottom of the pool. In the event of pipe failure, these lines must be manually sealed from within the reactor pool. Covers for these lines will be stored in the reactor pool. Time required to uncover the reactor core due to failure of a single pool coolant pipe system is 17 minutes.

e. Loss of coolant alarm after 10% loss requires corrective action. This alarm is observed in the reactor control room and the reception room.

6.0 Administrative Controls

6.1 Organization

6.1.1 Structure - The organization for the management and safe operation of the Nuclear Science Center Reactor shall be as shown in the following chart:



6.1.2 Responsibility - Responsibility for the safe operation of the reactor facility shall be in accordance with the chain of command established in the organizational chart. Individuals at the various levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license and technical specifications. Individual responsibilities at the various levels are as follows:

(a) Director, Texas Engineering Experiment Station - As licensee has ultimate responsibility for the safe and proper operation of the facility.

(b) Director, Nuclear Science Center - Has overall responsibility for insuring nuclear safety and providing administration of the Nuclear Science Center.

(c) Reactor Operations - Responsible for the day-to-day operations of the reactor.

(d) Radiological Safety Office - Responsible for providing "on-site" advice, technical assistance, and review in all areas related to occupational and radiological safety.

(e) Reactor Safety Board - Responsible for providing an independent review and audit of the safety aspects of reactor facility operations and for advising the directors in these areas.

6.1.3 Staffing

(a) The minimum staffing when the reactor is not secured shall be as follows:

(1) At least two individuals will be present at the facility complex and will consist of a licensed senior reactor operator and either a licensed reactor operator or operator trainee.

(2) A licensed reactor operator or senior reactor operator will be in the control room.

(3) The Director (NSC) or his designated alternate will be readily available for emergencies (i.e., capable of getting to the reactor facility within a reasonable time).

(4) At least one member of the health physics support group will be readily available to provide advice and technical assistance in the area of radiation protection.

(b) A list of reactor facility personnel by name and telephone number shall be readily available for use in the control room. The list shall include:

(1) Administrative personnel

(2) Radiation safety personnel

(3) Other operations personnel

(c) The following designated individuals shall direct the events listed:

(1) The Director (NSC) or his designated alternate shall direct any loading of fuel or control rods within the reactor core region.

(2) The Director (NSC) or his designated alternate shall direct any loading of an in-core experiment with a reactivity worth greater than one dollar.

(3) The senior reactor operator on duty shall direct the recovery from an unplanned or unscheduled shutdown other than a safety limit violation.

6.1.4 Selection and Training of Personnel - The selection and training of operations personnel shall be in accordance with the following:

(a) Responsibility - The Director (NSC) or his designated alternate is responsible for the training and requalification of the facility reactor operators and senior reactor operators.

(b) Requalification Program

(1) Purpose - To insure that all operating personnel maintain proficiency at a level equal to or greater than that required for initial licensing.

(2) Scope - Scheduled lectures, written examinations, and evaluated console manipulations will be used to insure operator proficiency is maintained.

6.2 Review and Audit

6.2.1 Reactor Safety Board - A Reactor Safety Board (RSB) of at least three (3) members knowledgeable in fields which relate to nuclear safety shall review, evaluate, and approve safety standards associated with the operation and use of the facility. The Director (NSC) and the University Radiological Safety Officer shall be ex-officio members of the RSB. An audit group shall be composed of a minimum of one (1) member or ex-officio member of the RSB. Qualified and approved alternates may serve in the absence of regular members. The jurisdiction of the RSB shall include all nuclear operations in the facility and general safety standards.

6.2.2 RSB Charter and Rules - The operations of the RSB shall be in accordance with a written charter, including provisions for:

(a) Meeting frequency - not less than once per calendar year and as frequent as circumstances warrant consistent with effective monitoring of facility activities.

(b) Voting rules

(c) Quorums

(d) Use of subcommittees

(e) Review, approval and dissemination of minutes

6.2.3 RSB Review Function - The responsibilities of the RSB or designated subcommittee thereof include, but are not limited to, the following:

(a) Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question or a change in Technical Specifications.

(b) Review of new procedures, major revisions of procedures, and proposed changes in reactor facility equipment or systems having safety significance.

(c) Review of new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

(d) Proposed changes in technical specifications, license, or charter.

(e) Violations of technical specifications, license, or charter. Violations of internal procedures or instructions having safety significance.

(f) Operating abnormalities having safety significance.

(g) Reportable occurrences listed in 6.6.2.

(h) Audit reports.

6.2.4 RSB Audit Function - The RSB or a subcommittee thereof shall audit reactor operations at least quarterly, but at intervals not to exceed four months. Audits shall include but are not limited to the following:

(a) Facility operations for conformance to the technical specifications and applicable license conditions at least once per calendar year (interval between audits not to exceed 15 months).

(b) The retraining and requalification program for the operating staff at least once per calendar year (interval between audits not to exceed 15 months).

(c) The facility security plan and records at least once per calendar year (interval between audits not to exceed 15 months).

(d) The reactor facility emergency plan and implementing procedures at least once per calendar year (interval between audits not to exceed 15 months).

6.2.5 Audit of ALARA Program - The Director (NSC) or his designated alternate shall conduct an audit of the reactor facility ALARA program at least once per calendar year (interval between audits not to exceed 15 months). The results of the audit shall be presented to the RSB at the next scheduled meeting.

6.3 Operating Procedures - Written operating procedures shall be prepared, reviewed, and approved prior to initiating any of the activities listed in this section. The procedures shall be reviewed and approved by the Director (NSC) or his designated alternate and the Reactor Safety Board and shall be documented in a timely manner. Procedures shall be adequate to assure the safe operation of the reactor but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect for the following items.

(a) Startup, operation, and shutdown of the reactor.

(b) Fuel loading, unloading, and movement within the reactor.

(c) Control rod removal or replacement.

(d) Routine maintenance of the control rod drives and reactor safety and interlock systems or other routine maintenance that could have an effect on reactor safety.

(e) Testing and calibration of reactor instrumentation and controls, control rod drives, area radiation monitors, and facility air monitors.

(f) Administrative controls for operations, maintenance, and conduct of irradiations and experiments that could affect reactor safety or core reactivity.

(g) Implementation of required plans such as emergency or security plans.

(h) Actions to be taken to correct specific and foreseen potential malfunctions of systems, including responses to alarms and abnormal reactivity changes.

Substantive changes to the above procedures shall be made effective only after documented review and approval by the Director (NSC) and the Reactor Safety Board. Minor modifications or temporary changes to the original procedures which do not change their original intent may be made by the Director (NSC) or his designated alternate. All such temporary changes shall be documented and subsequently reviewed by the Reactor Safety Board.

6.4 Experiments Review and Approval - Approved experiments shall be carried out in accordance with established and approved procedures.

(a) All new experiments or class of experiments shall be reviewed by the RSB (Section 6.2.3) and implementation approved in writing by the Director (NSC) or his designated alternate.

(b) Substantive changes to previously approved experiments shall be made only after review by the RSB and implementation approved in writing by the Director (NSC) or his designated alternate. Minor changes that do not significantly alter the experiment may be approved by the Director (NSC) or his designated alternate.

6.5 Required Actions

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

In the event a safety limit is exceeded:

(a) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.

(b) An immediate report of the occurrence shall be made to the Chairman, Reactor Safety Board, and reports shall be made to the NRC in accordance with Section 6.6.2 of these specifications, and

(c) A report shall be prepared which shall include an analysis of the cause and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Board for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

6.5.2 Action to be Taken in the Event of a Reportable Occurrence

In the event of a reportable occurrence, the following action shall be taken:

(a) Reactor conditions shall be returned to normal or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the Director (NSC) or his designated alternate.

(b) The Director (NSC) or his designated alternate shall be notified and corrective action taken with respect to the operations involved.

(c) The Director (NSC) or his designated alternate shall notify the Chairman of the Reactor Safety Board.

(d) A report shall be made to the Reactor Safety Board which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence, and

(e) A report shall be made to the NRC in accordance with Section 6.6.2 of these specifications.

(f) Occurrence shall be reviewed by the RSB at their next scheduled meeting.

6.6 Reporting Requirements

6.6.1 Annual Report

An annual report covering the operation of the reactor facility during the previous calendar year shall be submitted to the NRC prior to March 31 of each year providing the following information:

(a) A brief narrative summary of (1) operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;

(b) Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality;

(c) The number of emergency shutdowns and inadvertent scrams, including reasons therefor;

(d) Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;

(e) A brief description, including a summary of the safety evaluations of changes in the facility or in procedures and of tests and experiments carried out pursuant to Section 50.59 of 10 CFR Part 50;

(f) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient.

1. <u>Liquid Waste</u> (summarized on a monthly basis)

    (a) Radioactivity discharged during the reporting period.

        (1) Total radioactivity released (in curies).

        (2) The MPC used and the isotopic composition if greater than 1×10^{-7} microcuries/cc for fission and activation products.

        (3) Total radioactivity (in curies), released by nuclide, during the reporting period based on representative isotopic analysis.

        (4) Average concentration at point of release (in microcuries/cc) during the reporting period.

    (b) Total volume (in gallons) of effluent water (including diluent) during periods of release.

2. <u>Gaseous Waste</u> (summarized on a monthly basis)

    (a) Radioactivity discharged during the reporting period (in curies) for:

(1) Argon-41

(2) Particulates with half-lives greater than eight days.

3. Solid Waste

(a) The total amount of solid waste transferred (in cubic feet).

(b) The total activity involved (in curies).

(c) The dates of shipment and disposition (if shipped off site).

(g) A summary of radiation exposures received by facility personnel and visitors, including dates and time where such exposures are greater than 25% of that allowed or recommended.

(h) A description and summary of any environmental surveys performed outside the facility.

6.6.2 Special Reports - In addition to the requirements of applicable regulations, and in no way substituting therefor, reports shall be made to the NRC Region IV, Office of Inspection and Enforcement as follows:

(a) There shall be a report not later than the following working day by telephone and confirmed in writing by telegraph or similar conveyance to be followed by a written report that describes the circumstances of the event within 14 days of any of the following:

(1) Violation of safety limits. (See 6.5.1).

(2) Any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;

(3) Any reportable occurrences as defined in Section 1.28 of these specifications. The written report (and, to the extent possible, the preliminary telephone or telegraph report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent reoccurrence of the event;

(b) A written report within 30 days of:

(1) Personnel changes in the facility organization involving the Director of the Texas Engineering Experiment Station (licensee) or the Director (NSC).

(2) Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

6.7 Records - Records of facility operations in the form of logs, data sheets or other suitable forms shall be retained for the period indicated as follows:

6.7.1 Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved

(a) Normal reactor facility operation.

(b) Principal maintenance operations.

(c) Reportable occurrences.

(d) Surveillance activities required by the Technical Specifications.

(e) Reactor facility radiation and contamination surveys where required by applicable regulations.

(f) Experiments performed with the reactor.

(g) Fuel inventories, receipts, and shipments.

(h) Approved changes in operating procedures.

(i) Records of meeting and audit reports of the RSB.

6.7.2 Records to be Retained for at Least One Training Cycle - Retraining and requalification of certified operations personnel: Records of the most recent complete cycle shall be maintained at all times the individual is employed.

6.7.3 Records to be Retained for the Lifetime of the Reactor Facility

(a) Gaseous and liquid radioactive effluents released to the environs.

(b) Off-site environmental-monitoring surveys required by the Technical Specifications.

(c) Radiation exposure for all personnel monitored.

(d) Drawings of the reactor facility.

GA-A16613

INTERPRETATION OF DAMAGE TO THE FLIP FUEL DURING OPERATION OF THE NUCLEAR SCIENCE CENTER REACTOR AT TEXAS A&M UNIVERSITY

by
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*Texas A&M University
**Argonne National Laboratory, Idaho Facilities
***Present Address: Battelle Memorial Institute, Columbus, Ohio

GENERAL ATOMIC PROJECT 4314
DECEMBER 1981

GENERAL ATOMIC COMPANY

CONTENTS

FIGURES

FIGURES (Continued)

TABLES

1. INTRODUCTION

In a letter report (Ref. 1) dated November 1, 1976, Professor J. D. Randall of Texas A&M University presented the history of core operation, the discovery of the damaged FLIP fuel, the description of this fuel, and the possible causes contributing to the damage of the fuel. The results of the preliminary investigations designed to interpret the damage were also presented. The following is a summary of the salient points made in that report.

1.1. HISTORY OF CORE WITH FLIP FUEL

From June 1973 to September 27, 1976, the Nuclear Science Center Reactor (NSCR) operated with a mixed TRIGA core (Core III) consisting of 35 new FLIP elements and 63 partially depleted standard TRIGA elements. The core operated 287 MW-days in the steady-state mode and underwent 725 pulses. The pulse reactivity insertion was limited to \$2.00 until July 1975, when the technical specifications were changed to allow pulse insertions up to \$2.70. All of the 30 pulses greater then \$2.00 but less than \$2.70 and 37 of the 54 total \$2.70 pulses were performed in approximately two months immediately following license approval for higher pulse insertions. The peak core temperature rise was 727°C with the \$2.00 pulse insertion and 883°C with the \$2.70 pulse insertion. These were calculated temperatures derived from observed thermocouple temperatures located not in the hottest fuel element.

1.2. OBSERVED DAMAGE IN THE FUEL ELEMENTS

During a loading operation on September 27, 1976 (about 14 months after approval of pulsing up to \$2.70*), four "lead" elements, each in a different cluster of FLIP elements, were found to be "somewhat deformed." These elements all operated in nearly the same flux. The elements in the next lower flux regions were not damaged. The four damaged elements were all positioned adjacent to the transient rod throughout their operating history. The pulsing history of Core III-A is given in Table 1. The maximum fuel deformation occurred in the west position closest to the transient rod. This element was separated from the transient rod guide tube with a water gap of approximately 0.05 in. The damage appeared to be slightly less severe in the elements positioned south and east of the transient rod (0.204-in. and 0.075-in. water gaps), and the least damage was in the fuel element in the north position with a water gap of 0.075 in. (the only element to pass the go/no-go gauge and thus not removed from service). The core configuration and details of the region under study are shown in Figs. 1 through 3. The calculated peak-to-average energy ratios in the four elements were very similar.

The visual inspection of the most damaged element revealed bulging in the cladding and a bow in the element around the fuel centerline. The data from a profilometer scan of the most damaged element indicated a displacement of approximately 0.15 in. near the midplane of the element, in the northeast direction. As shown in Fig. 4, the upper end of the element was displaced by about 0.23 in.

---

*Since there were no technical specification requirements for annual fuel inspection or for inspection after newly increased operational limits, no such inspections were performed.

TABLE 1
PULSING HISTORY OF CORE III-A[a]

Insertion ($)	Number Of Pulses	Worth
1.00	2	2.00
1.10	2	2.20
1.15	4	4.60
1.20	6	7.20
1.25	23	28.75
1.30	7	9.10
1.35	4	5.40
1.40	16	22.40
1.45	3	4.35
1.50	37	57.00
1.55	1	1.55
1.60	10	16.00
1.65	2	3.30
1.70	7	11.90
1.75	32	56.00
1.80	9	16.20
1.85	3	5.55
1.90	13	24.70
1.93	1	1.93
1.95	4	7.80
2.00	455	910.00
2.10	2	4.20
2.15	1	2.15
2.20	1	2.20
2.25	4	9.00
2.30	1	2.30
2.35	1	2.35
2.40	2	4.80
2.50	16	37.50
2.60	2	5.20
2.70	54	145.80
	Total Pulse Worth	1,413.43

[a] From Ref. 1.

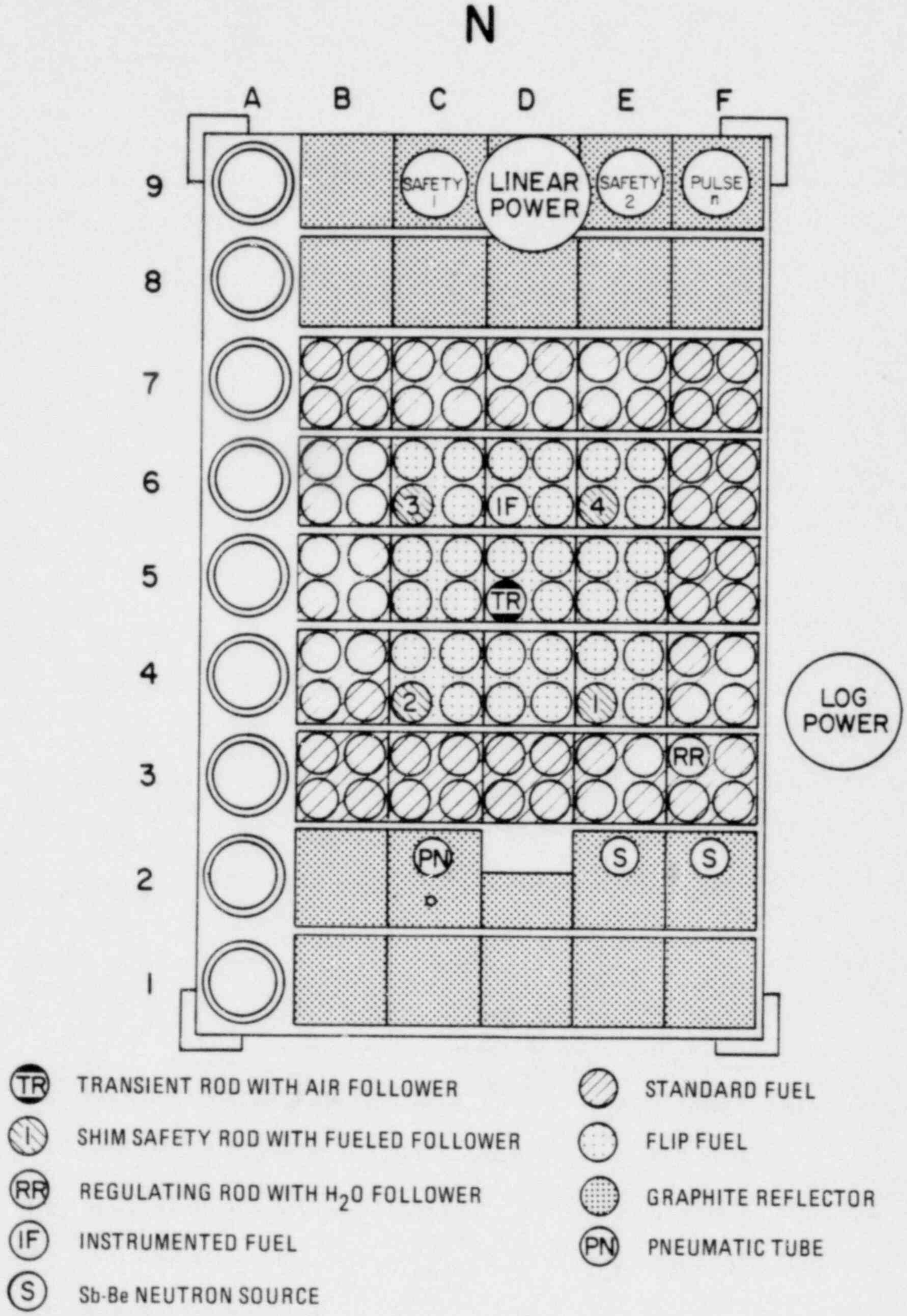


Fig. 1. Core III-A (from Ref. 1)

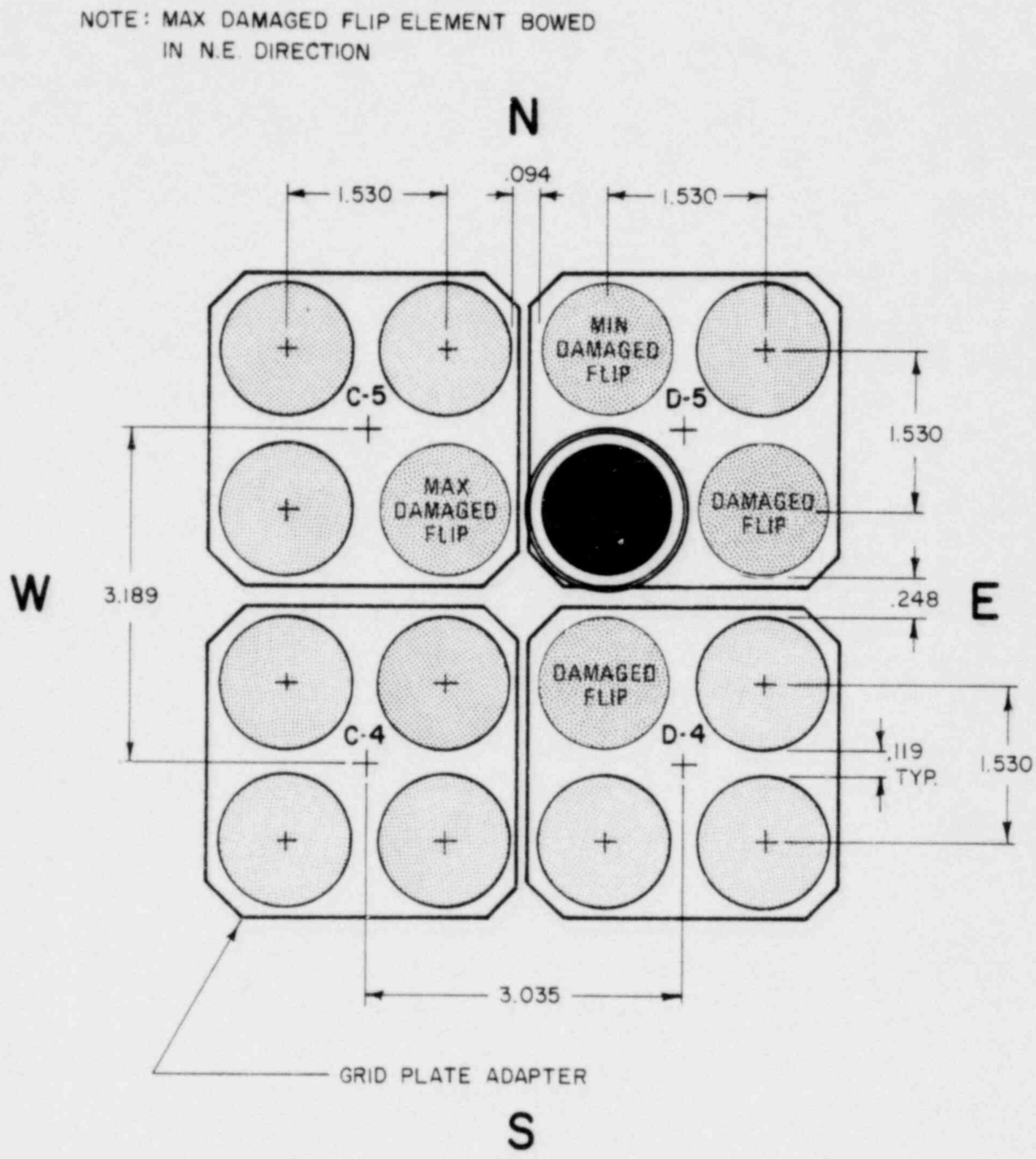
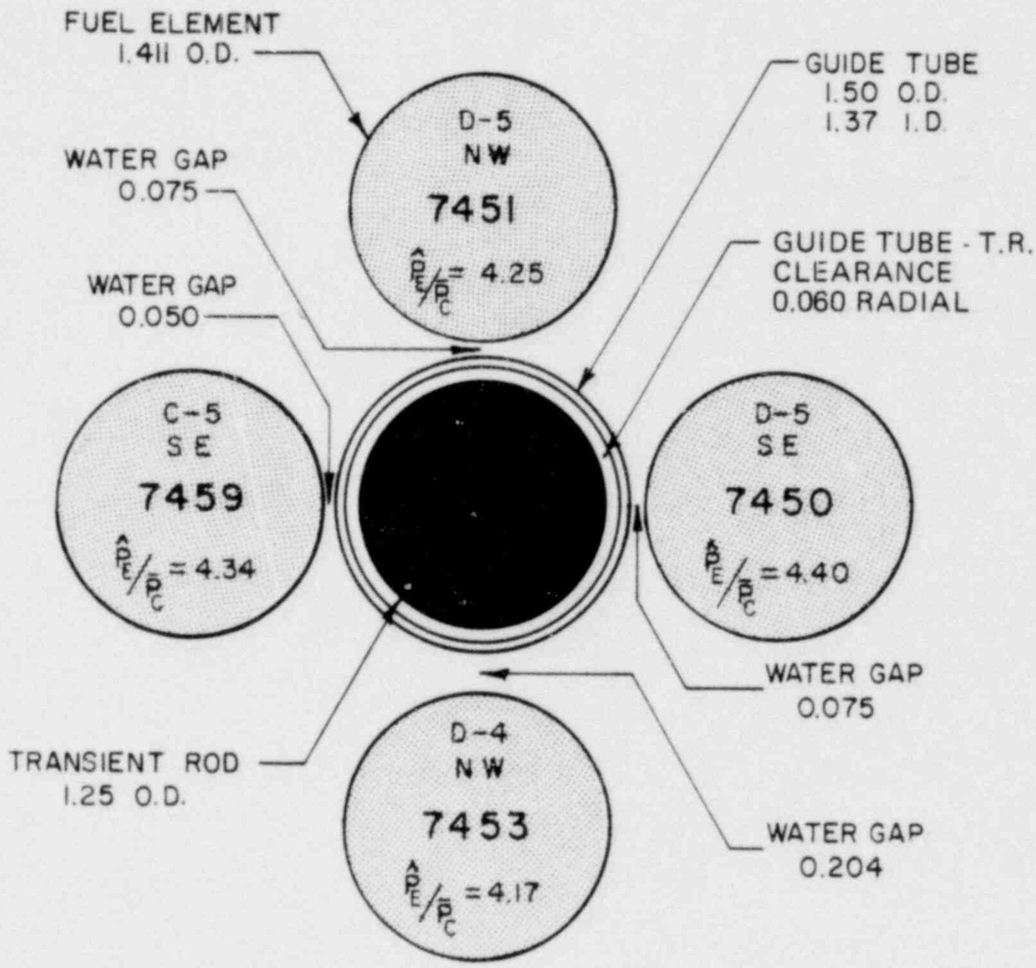


Fig. 2. FLIP fuel element spacing in the NSCR core near transient rod (from Ref. 1)



$$\hat{P}_E/\bar{P}_C = \frac{\text{PEAK ELEMENT ENERGY}}{\text{AVERAGE CORE ENERGY}}$$

Fig. 3. Water gap and fuel spacing of damaged FLIP elements adjacent to transient rod (from Ref. 1)

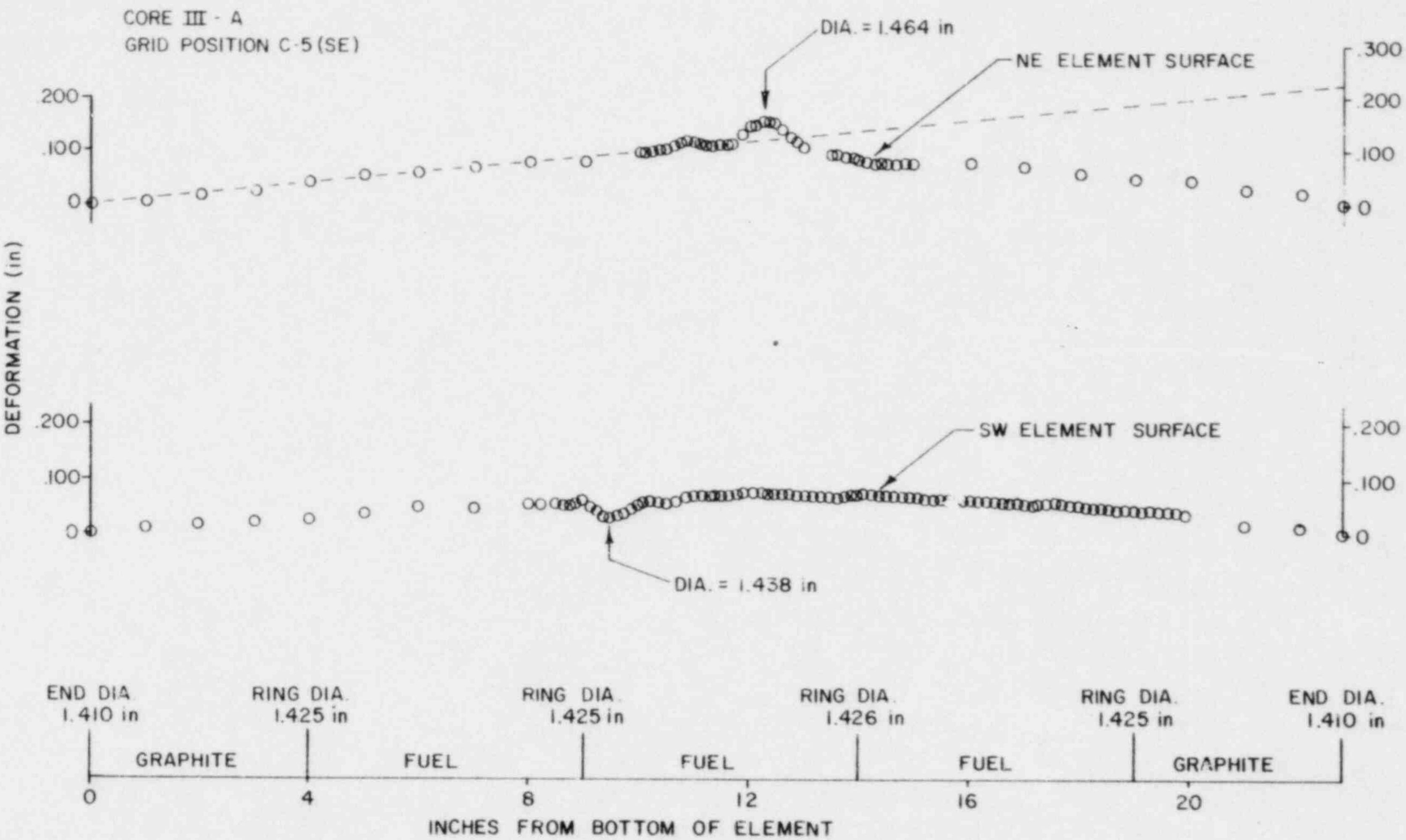


Fig. 4. Profile of maximum damaged element no. 7459 in direction of maximum bowing (from Ref. 1)

1.3. POSTULATED CAUSES OF THE DAMAGE

The following factors were considered as possible contributors to the damage of the fuel elements:

- Incorrectly calculated power distributions in the core and the elements.
- Water leakage into the air-filled follower of the transient rod.
- Improper safety limit for FLIP fuel.

It was considered statistically unlikely that four defective fuel elements were received and that these were the only ones placed in the damaging regions.

1.4. PRELIMINARY INVESTIGATIONS

The core peak temperatures were calculated from the thermocouple readings and from the ratio of the flux or power density in the peak core location to that in the thermocouple. The significant temperatures measured in Core III-A are listed in Table 2. The measurements of temperatures did not, however, include instrumented elements next to a void or a water hole. Calculated power density ratios are used to correct the measured temperature to the peak power element and also to correct for the power distribution in a specific element next to the pulse rod. The flux distribution inside a fuel element adjacent to the pulse rod (in the location of maximum damage) was measured to verify the calculation of power peaking. The calculated values of flux peaking yielded conservative results, so that the calculated flux values appear to be higher than the measured values on the fuel surface. Hence, it was concluded that the calculated adiabatic temperature distributions are also conservative values.

Visual and neutron radiographic examinations of the two followers that were used during the period in question do not support the postulate that water may have leaked into the air-filled follower of the transient rod.

TABLE 2
SIGNIFICANT MEASURED AND CALCULATED TEMPERATURES
IN CORE III-A[a],[b]

Pulsing	
\$2.00 Pulse	
Maximum observed thermocouple temperature	427°C
Maximum core temperature[c]	764°C
\$2.50 Pulse	
Maximum observed thermocouple temperature	499°C
Maximum core temperature[c]	885°C
\$2.70 Pulse	
Maximum observed thermocouple temperature	520°C
Maximum core temperature[c]	920°C
Steady State (1-MW Operation)	
Maximum observed thermocouple temperature	450°C
Maximum core temperature[c]	575°C

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[a] From Ref. 1.

[b] Using a ratio of peak core power to power at the thermocouple = 2.8.

[c] Calculated; includes ambient temperature of 37°C.

The question of whether an improper safety limit had been set for FLIP fuel is answered in the negative. The vast bulk of experience at General Atomic and elsewhere, particularly in the comparable Washington State University TRIGA, does not suggest an improper safety limit. It was tentatively concluded that the mechanism producing the damage in the fuel elements is related to some phenomena occurring during pulsing, and that the steady-state history of the fuel is not a factor.

2. EXPERIMENTAL INVESTIGATIONS

The investigation of the fuel damage described in this document has been a cooperative effort by Texas A&M, Argonne National Laboratory (ANL), and General Atomic. The examinations performed are very significant and have produced indispensable information, but the budget limitations applicable to all concerned have predictably led to the fact that not enough information is available to answer all the questions raised during the investigation. However, the necessary information for a qualitative examination and evaluation has been obtained.

The tests and measurements that have been carried out on the fuel elements at Texas A&M and at ANL-West since September 1976 have included metallography, alpha autoradiography, neutron radiography of the damaged fuel elements and the transient rod air-filled follower, diametral and bow measurements of these elements, and determinations of the axial flux distributions by means of gross gamma scans and isotopic gamma scans. The two diameter measurements were at 90° from each other and each at approximately 45° from the direction of maximum bow.

2.1. DIAMETRAL AND BOW MEASUREMENTS

The diametral variations of the most damaged fuel element are shown in Tables 3 and 4 and in Figs. 5 and 6 at 90° apart. Figure 5 exhibits a sharp increase in diameter at 9 in. from the top shoulder of the element. The diametral increase at this point is approximately 0.08 in. This peak drops rapidly to 0.02 in. at 10 in. from the top shoulder. There is then a shallower (about 0.04 in.) and wider peak in the region between 11 and 13 in. This peak diminishes sharply to a plateau of about 0.02 in. in the region between 13 and 18 in. There is then a gradual decrease in diameter up to 20 in. from the top shoulder to reach the original diameter.

TABLE 3
TRIGA FUEL DIAMETER MEASUREMENTS AT REFERENCE 0°[a]

Inches From Top Shoulder	Diameter E.F. Pin Forward
1	1.41205
1.5	1.41205
2	1.41242
2.5	1.41235
3	1.41321
3.5	1.41450
4	1.41291
4.5	1.41606
5	1.41600
5.5	1.41193
6	1.41850
6.5	1.42150
7	1.42135
7.5	1.42702
8	1.43150
8.5	1.44344
8.75	1.46272
9	1.49217
9.5	1.43195
10	1.43075
10.5	1.43358
11	1.45000
11.5	1.44712
12	1.45073
12.5	1.43834
13	1.43404
13.5	1.43225
14	1.43150
14.5	1.43149
15	1.43063
15.5	1.43005
16	1.42916
16.5	1.43000
17	1.42855
17.5	1.42872
18	1.42546
18.5	1.42283
19	1.42012
19.5	1.41708
20	1.41478
20.5	1.41480
21	1.41455
21.5	1.41323
22	1.41308
22.5	1.41312
23	1.41330

(a) From Ref. 2.

TABLE 4

TRIGA FUEL DIAMETER MEASUREMENTS AT REFERENCE 90°[a]

Inches From Top Shoulder	Diameter E.F. Pin Up
1	1.41194
1.5	1.41215
2	1.41060
2.5	1.41075
3	1.41145
3.5	1.41150
4	1.41125
4.5	1.41090
5	1.41000
5.5	1.41030
6	1.41000
6.5	1.41213
7	1.41260
7.5	1.41225
8	1.41151
8.5	1.41130
8.75	1.41081
9	1.41490
9.5	1.42668
10	1.44605
10.5	1.44433
11	1.43375
11.5	1.42926
12	1.42285
12.5	1.46073
13	1.43510
13.5	1.43070
14	1.43836
14.5	1.43152
15	1.43326
15.5	1.43865
16	1.44500
16.5	1.43649
17	1.41776
17.5	1.41628
18	1.41445
18.5	1.41290
19	1.41290
19.5	1.41158
20	1.41073
20.5	1.40850
21	1.40901
21.5	1.40951
22	1.41028
22.5	1.41070
23	1.40948

(a)

From Ref. 2.

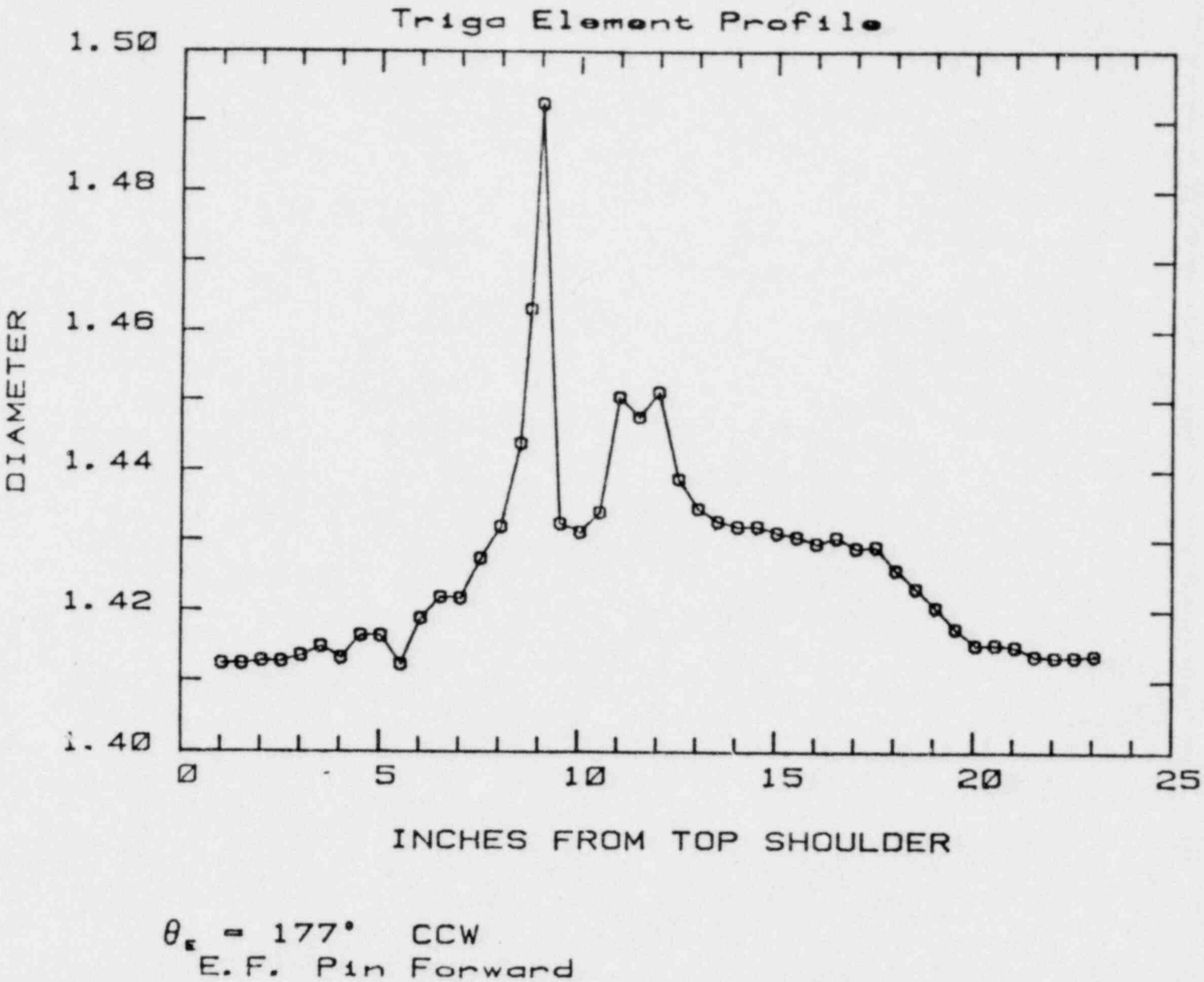


Fig. 5. Diametral variations of most damaged fuel at reference 0° (approximately 45° from direction of maximum bow) (from Ref. 2)

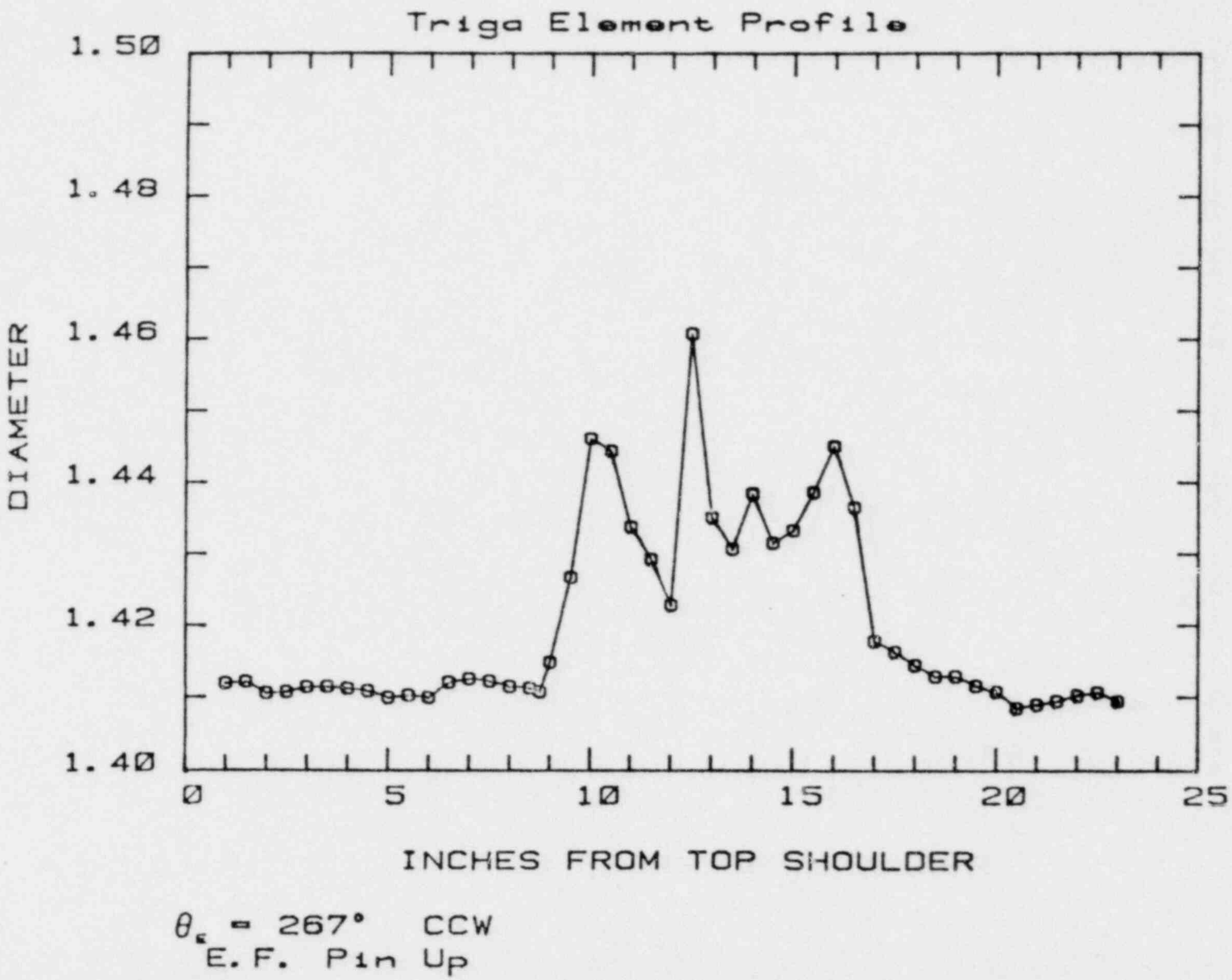


Fig. 6. Diametral variations of most damaged fuel at reference 90° (approximately 45° from direction of maximum bow) (from Ref. 2)

The diametral changes shown in Fig. 6 are at 90° to those of Fig. 5. Here the diameter increases sharply by 0.035 in. at 9 in. from the top shoulder, drops continuously by 0.025 in. to 12 in. from the top shoulder, then peaks sharply by 0.05 in. at 12.5 in., peaks modestly at 14.5 and 16.5 in. from the top shoulder, and finally drops sharply first to 16 in. and then gradually to the original diameter.

The measured bowing of the fuel element is illustrated in Fig. 7. The maximum bow is approximately 0.11 in. at a distance of 15 in. from the bottom, i.e., about halfway up the element.

2.2. NEUTRON RADIOGRAPHS

The distressed regions in the fuel appear as light spots on the neutron radiograph prints, indicating a depletion of hydrogen at these locations. Figure 8 is a print of the neutron radiograph of the most damaged element, and it indicates the locations at which samples were taken. The light areas are depleted in hydrogen, but the degree of depletion is unknown. More recent radiographs are shown in Figs. 9 and 10, depicting the third and second most damaged elements, respectively. The third most damaged element shows hydrogen depletion in the top segment of the fuel, in a fairly small region. The central axial 0.25-in.-diam region, where the zirconium rods are present, also reflects a reduced hydrogen content (difficult to see on the reproductions). Evidently, the maximum temperature was too low to drive hydrogen from the fuel to the unhydrided zirconium rods in the central axial hole. On the other hand, the neutron radiographs of the second most damaged element show significant hydrogen movement into the central zirconium rods, indicated by apparently equal hydriding of the fuel and the central zirconium axial rod (again difficult to see on the reproductions). The most damaged element exhibits large, significant depletions of hydrogen in all three segments, particularly in the central region (Fig. 8).

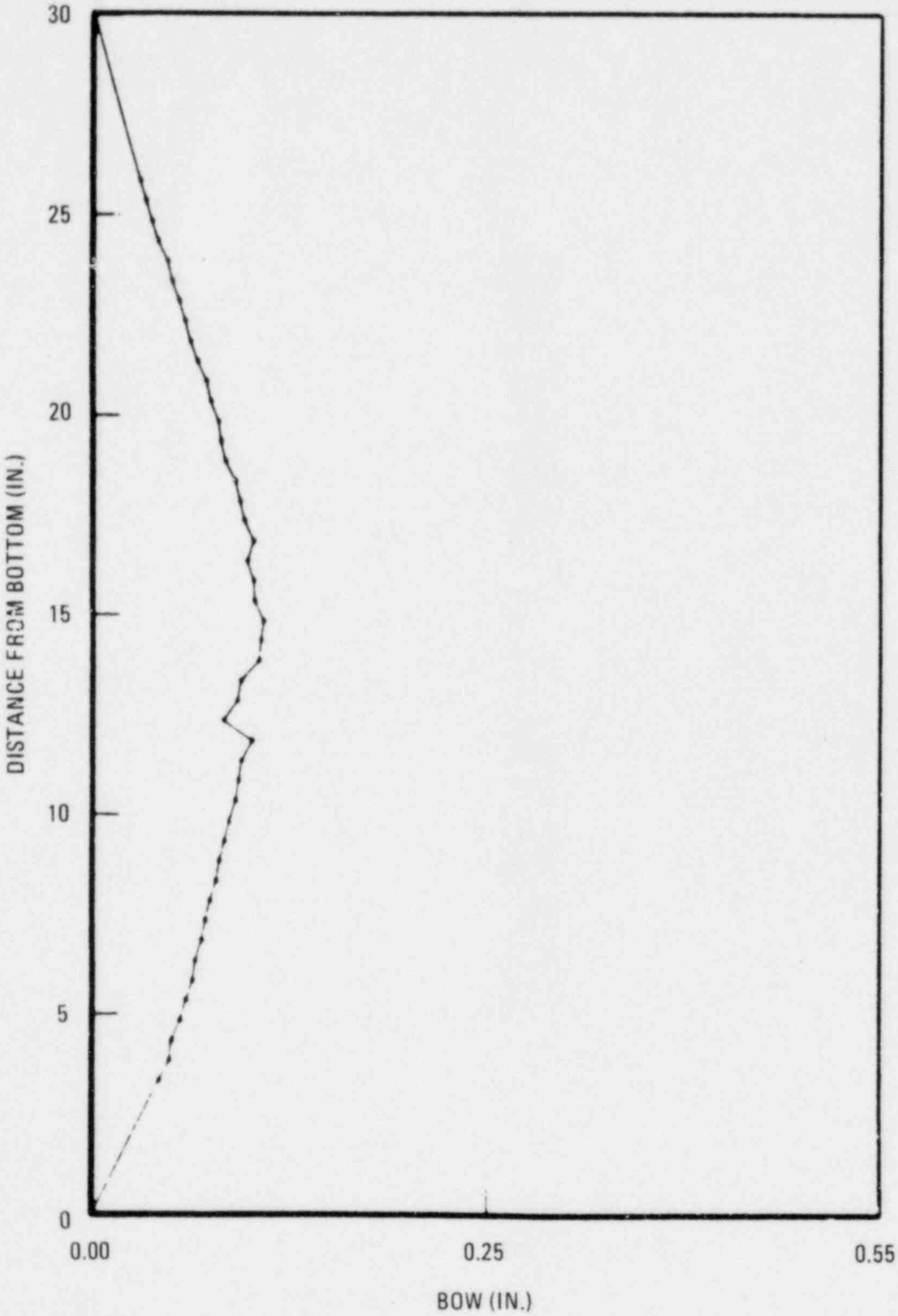


Fig. 7. Measured bowing of TRIGA element (from Ref. 2)

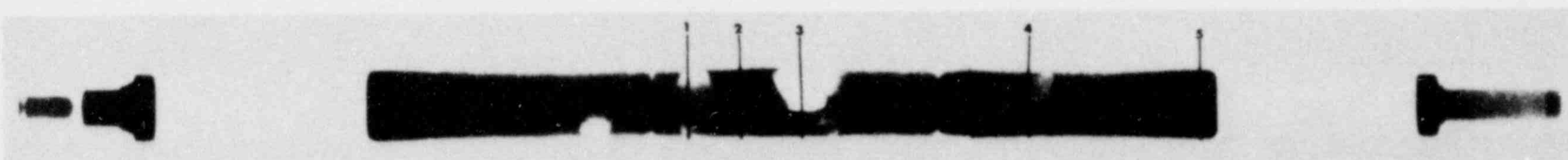


Fig. 8. Neutron radiograph of damaged TRIGA element showing locations at which samples were taken. Light areas are hydrogen depleted. (From Ref. 2.)

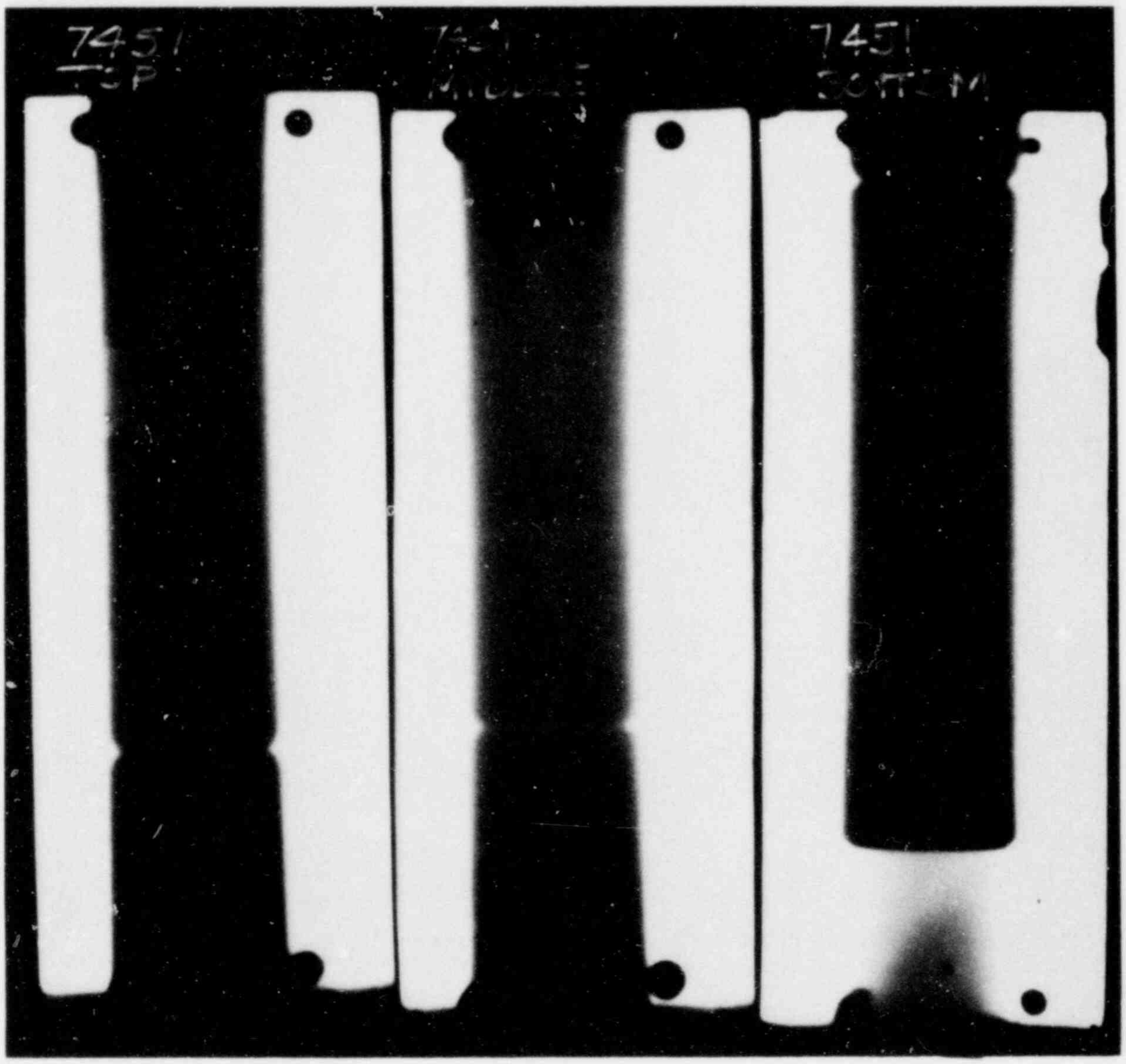


Fig. 9. Neutron radiograph of third most damaged element (neutron radiograph by Texas A&M)

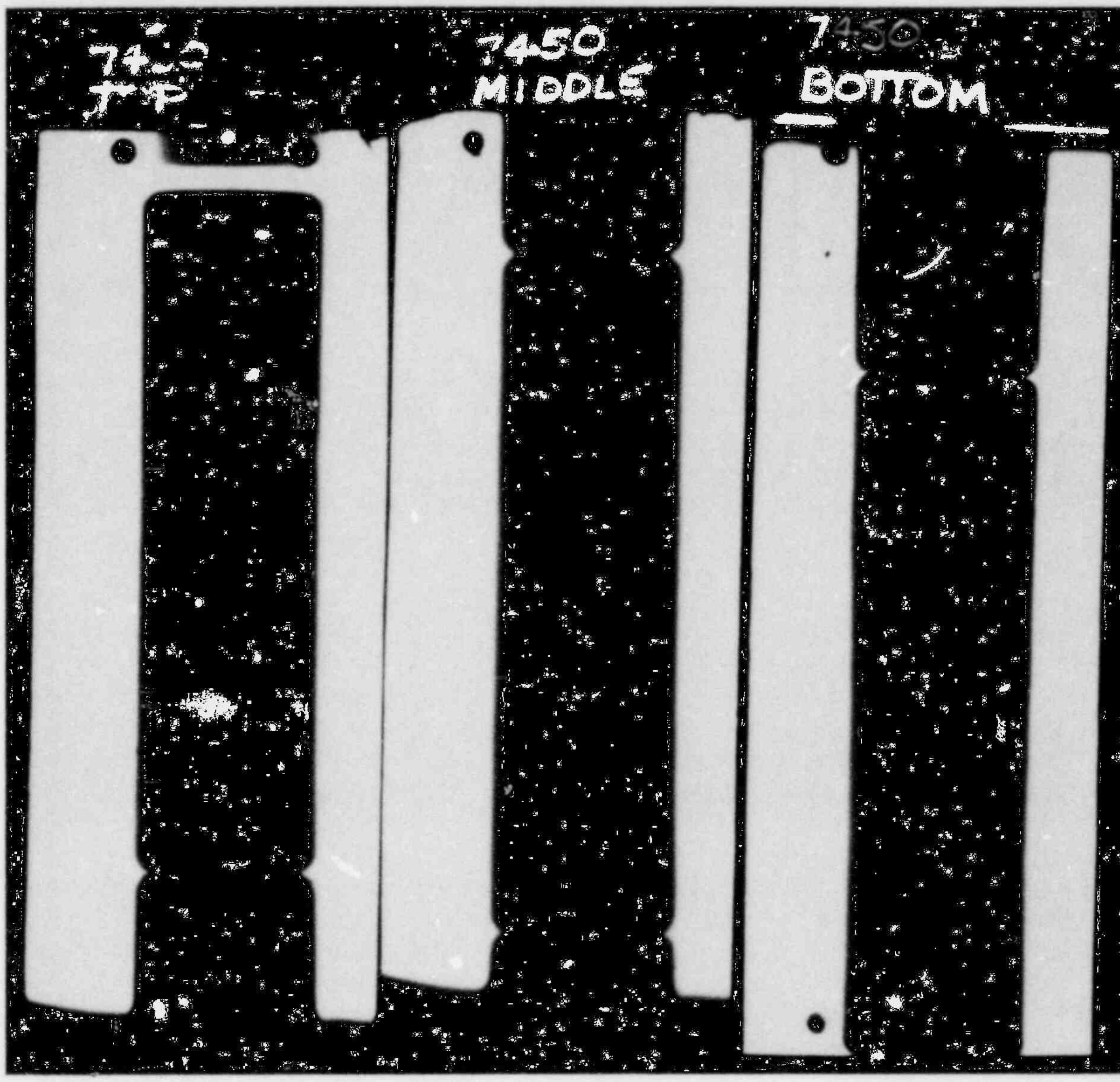


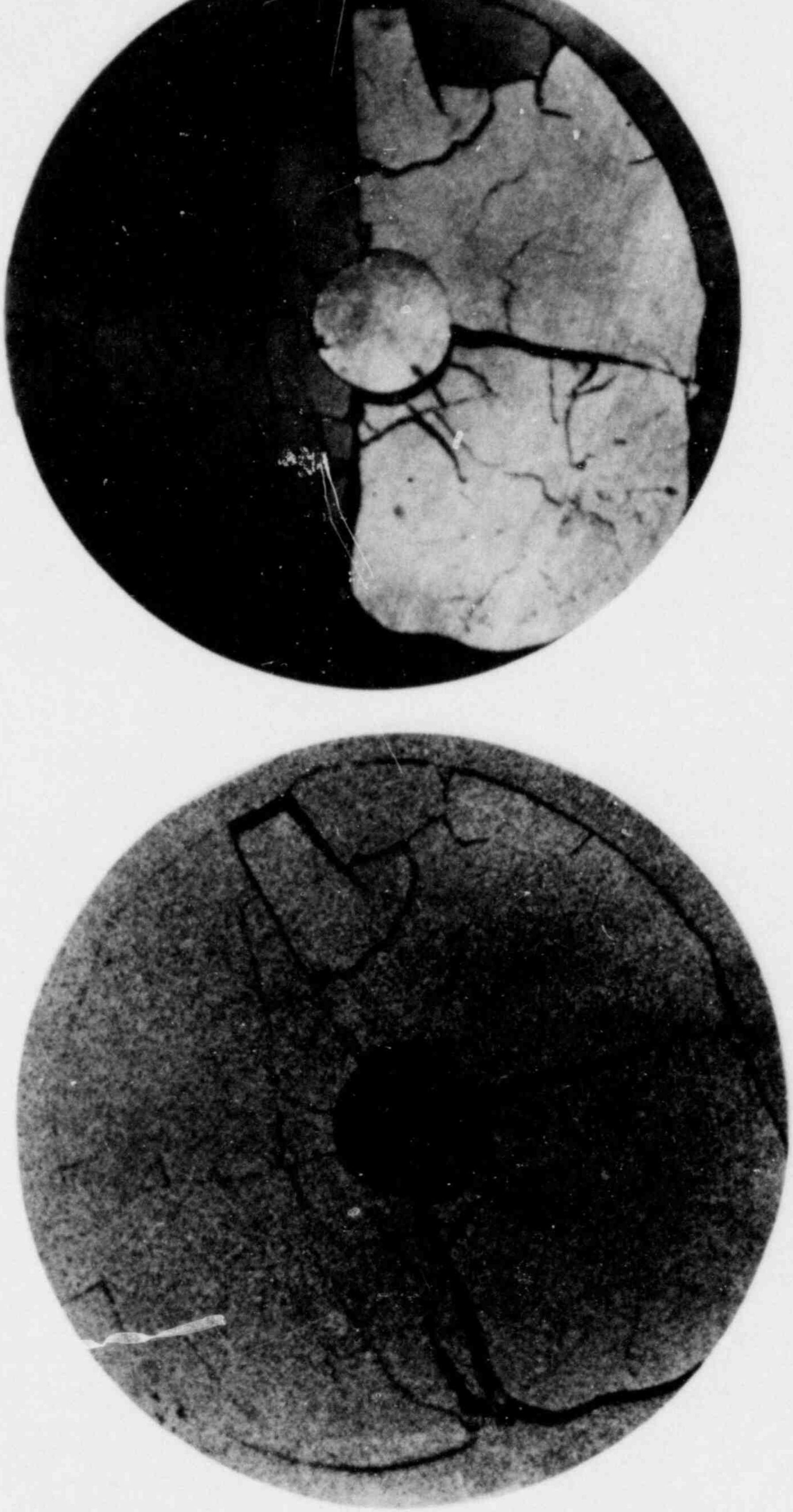
Fig. 10. Neutron radiograph of second most damaged element (neutron radiograph by Texas A&M)

2.3. METALLOGRAPHY AND ALPHA AUTORADIOGRAPHY

The metallography and alpha autoradiographics of the most severely damaged element are shown in Figs. 11 through 29 (from Ref. 2), which also show metallography of the cladding. The areas examined in the fuel include four from distressed areas and the fifth from an essentially undamaged area at the bottom of the fuel column. The undamaged area was used as a control sample for comparative purposes. The distressed regions were selected from the areas that appear as light spots on the neutron radiograph prints, indicating depletion of hydrogen at those locations. The greatest concentration of hydrogen-depleted material was found in the area surrounding the cladding bulge. Alpha autoradiographs were made of the five metallographic samples to determine the uranium distribution in the fuel. No significant changes in uranium concentration were observed.

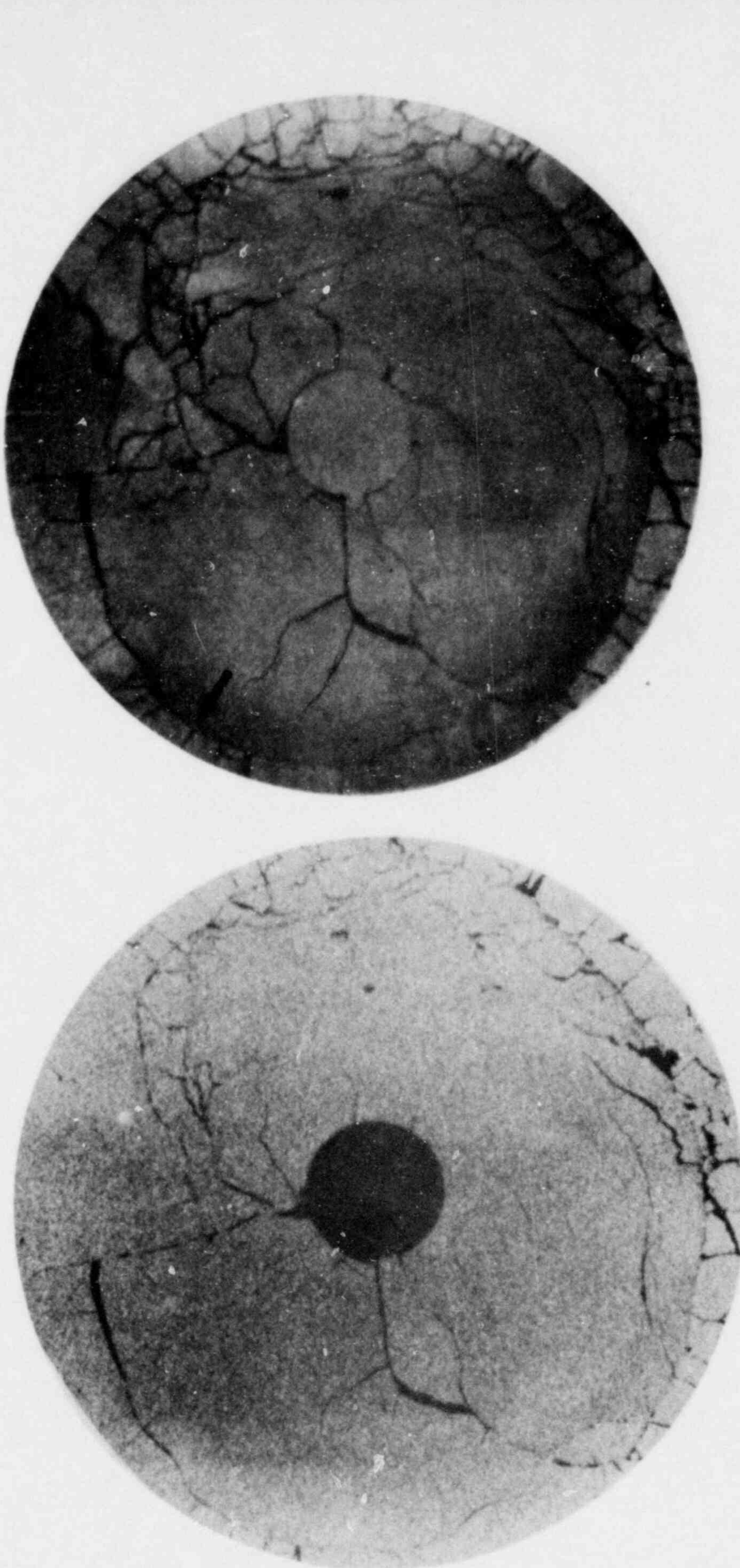
The metallographic examination revealed circumferential cracks near the clad on all fuel samples, as well as extensive radial and random cracking from distressed locations. The unaffected (control) region from the bottom of the fuel column had only two radial cracks and a partial circumferential crack. The condition of the zirconium rod which runs axially through the center of the fuel column is shown in several of the figures. This rod appears to be undamaged in the control sample. In the other samples the central rod is either deformed or partly bonded to the fuel, or cracked. It is most damaged (cracked) in Sample 4, where it appears to be straining against the fuel in the least damaged of those samples that show hydrogen depletion. Swelling of the central rod by hydrogen absorption increased its diameter by 10 to 12% over that of the control sample. The microstructure of samples of the central zirconium rod shown in Figs. 20 and 25 present evidence of transformation to the epsilon hydride phase, indicating a $ZrH_{1.7}$ material.

The distressed fuel areas exhibited both a low hydrogen alpha phase mixed with delta, and a normal delta phase ($ZrH_{1.6}$). The region shown in Fig. 22 (Sample 2) has a highly porous area near the cladding, separated



3.5X

Fig. 11. Periscope macro of Sample 1 with alpha autoradiograph below. It is likely that the light-reflective material contains significant quantities of alpha zirconium. Zirconium rod was deformed by adjacent fuel. Location of clad weld seam is at top on all five macros. (From Ref. 2.)



3.4X

Fig. 12. Macro and alpha radiograph of Sample 2. Fuel is extensively broken up. Material on the clad side of the circumferential crack shows different appearance. Porous area shown in Fig. 19 does not show, but location is indicated by arrow. (From Ref. 2.)

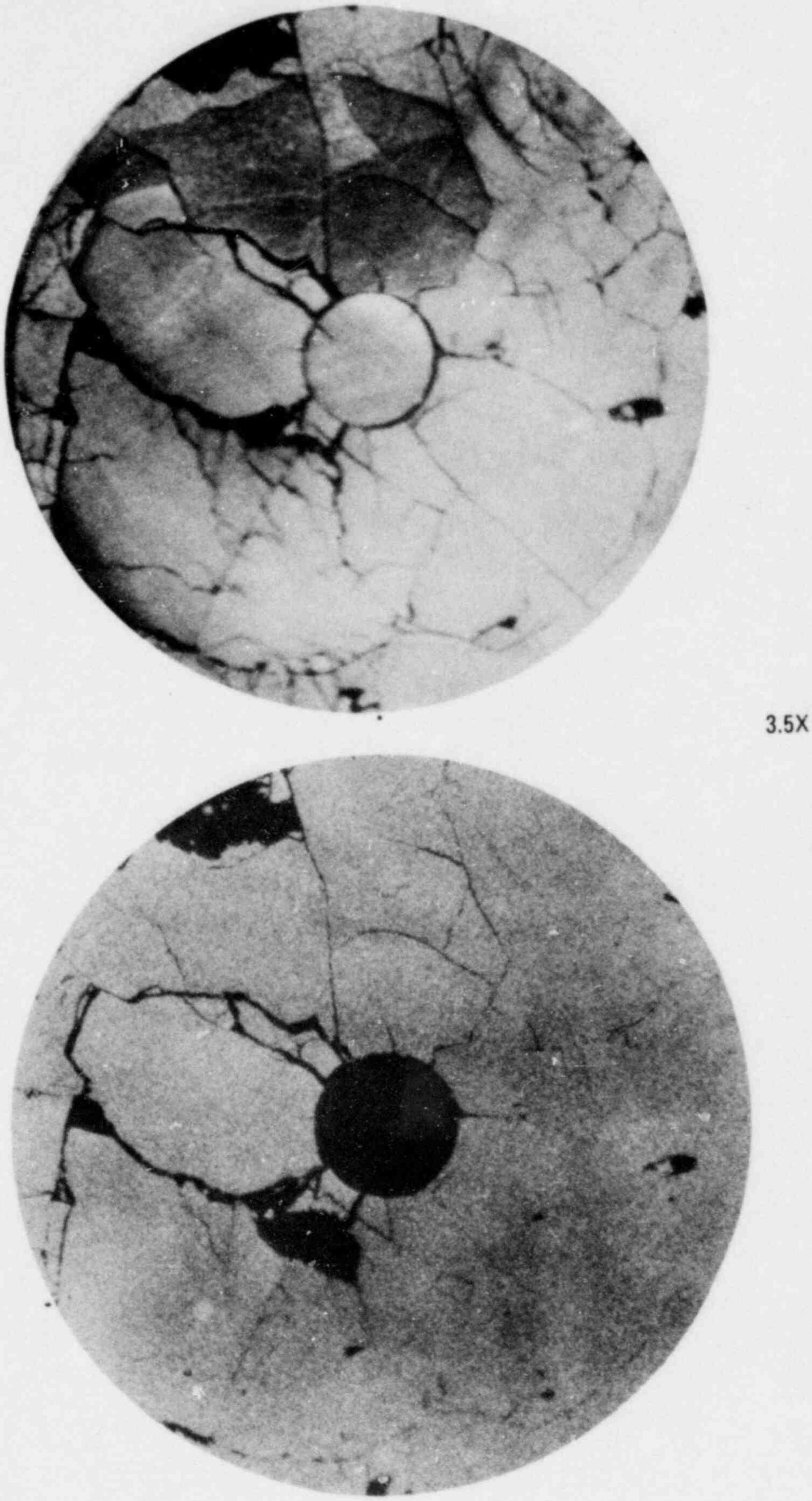


Fig. 13. Sample 3 macro shows fragmentation typical of distressed areas. (From Ref. 2.)

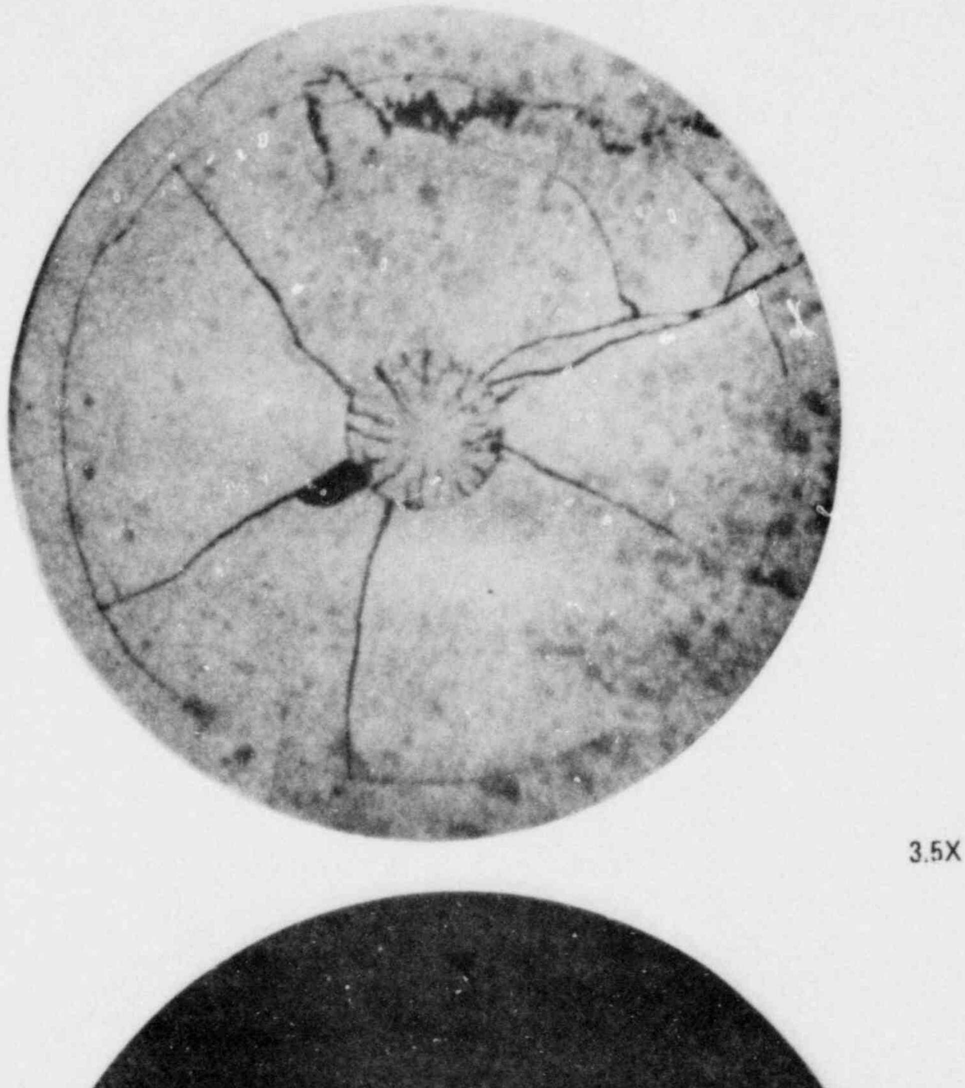
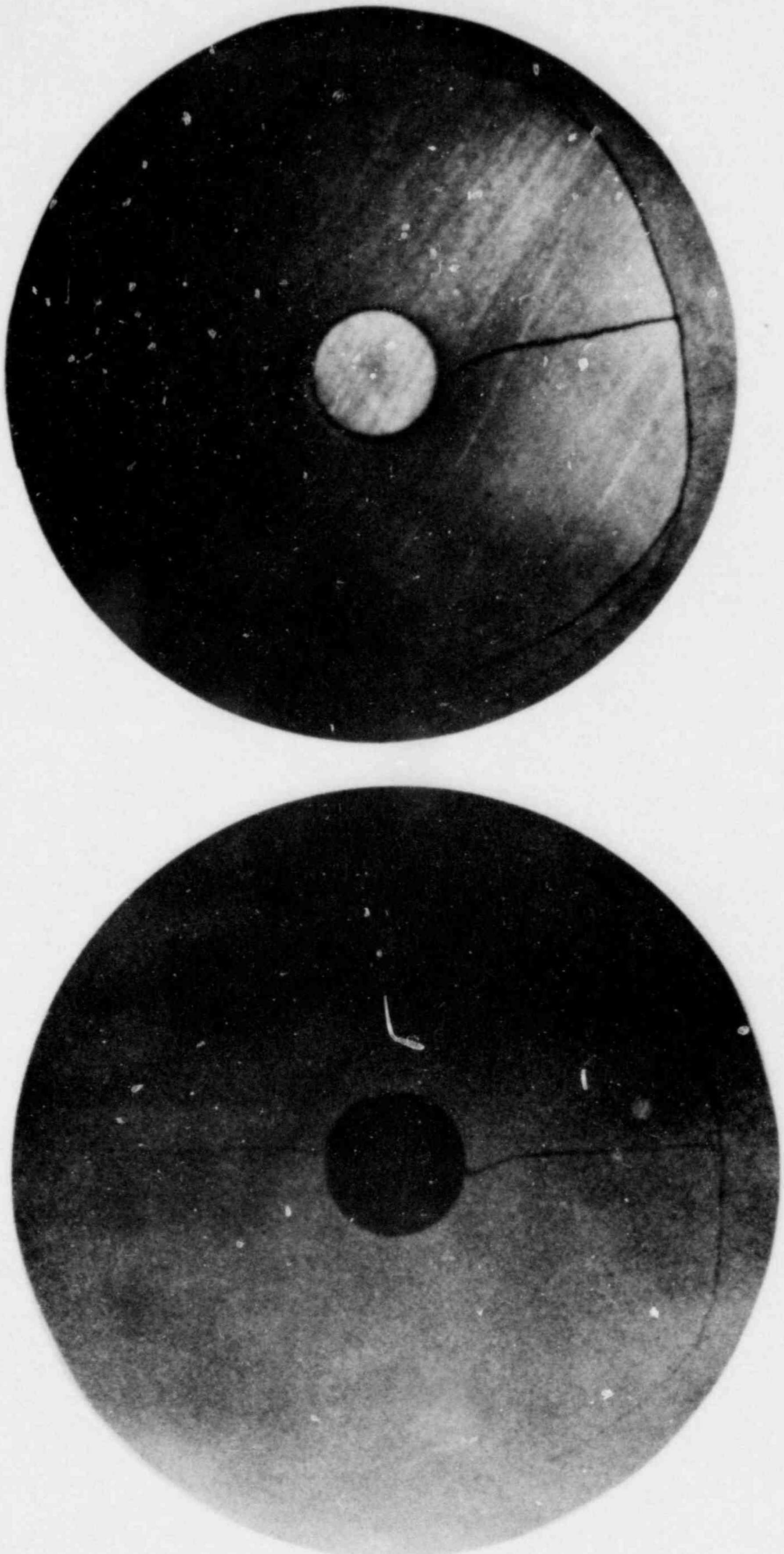
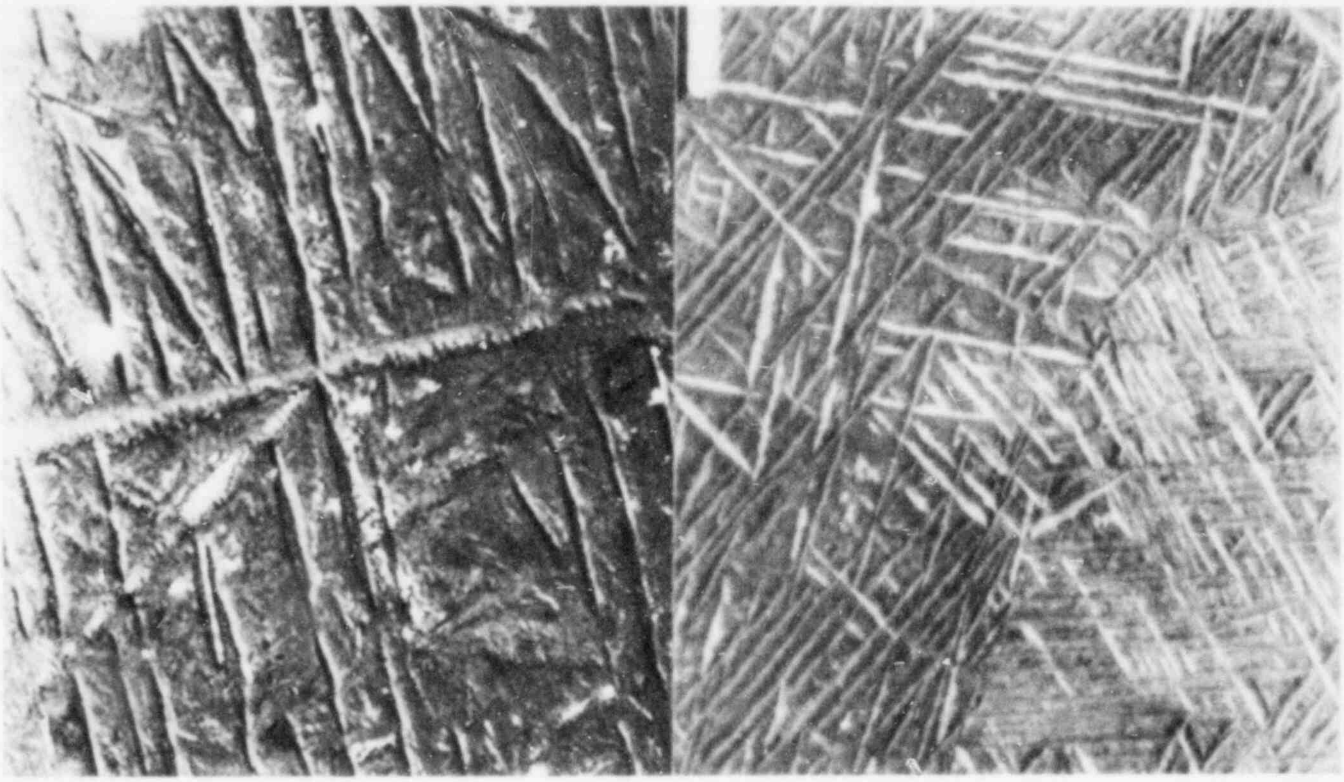


Fig. 14. Sample 4 appears the least damaged of those samples that show hydrogen depletion. Central zirconium rod is cracked more than in any of the other sections and appears to be straining against fuel. (From Ref. 2.)



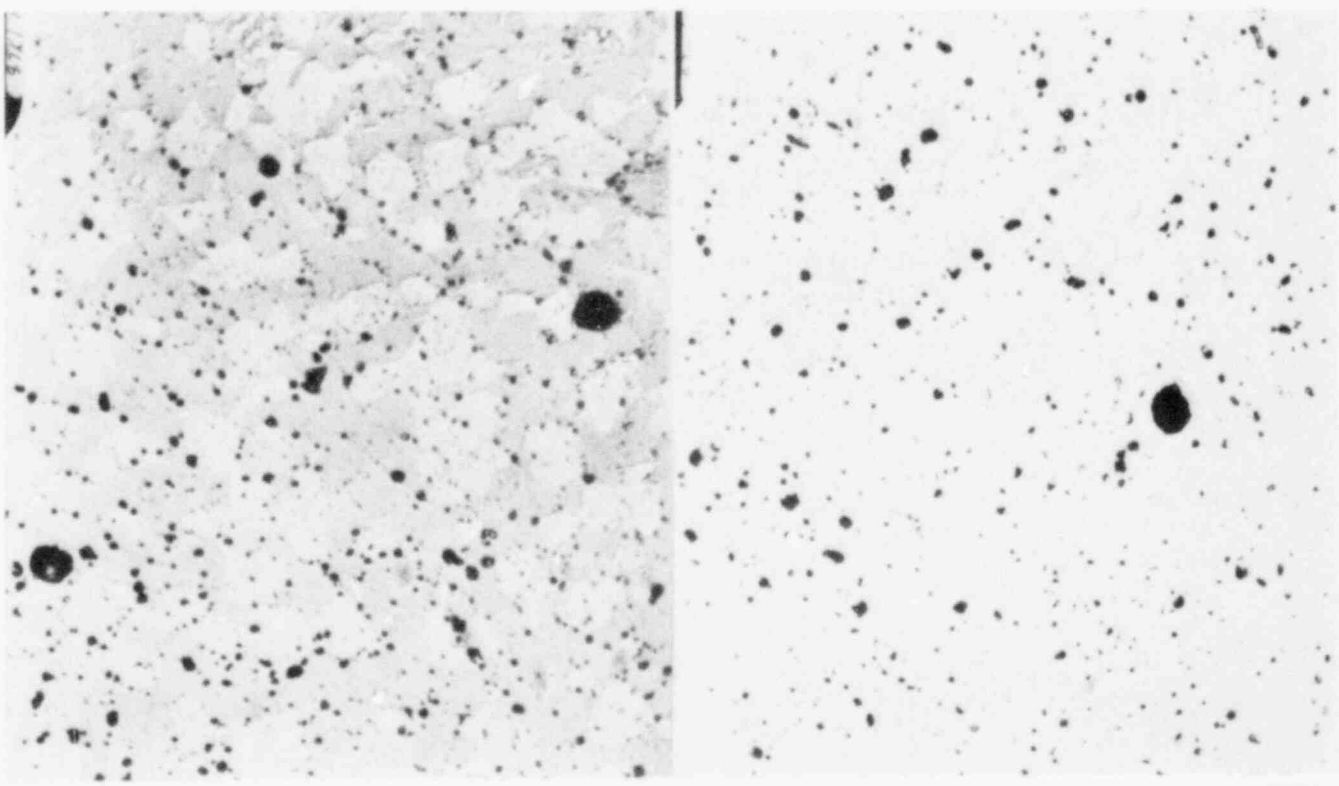
3.5X

Fig. 15. Control Sample (5) is not badly cracked and central rod appears undamaged in this macrograph. (From Ref. 2.)



Pd LIGHT 800X 500X

Fig. 16. Microstructure of central zirconium rod from Sample 1. (From Ref. 2.)



Pd 800X B.F. 800X

Fig. 17. Fuel from the hydrogen-depleted area of Sample 1 is shown at left. Structure is believed to contain significant quantities of low hydrogen alpha phase. Fuel at right is delta phase $ZrH_{1.6}$. (From Ref. 2.)

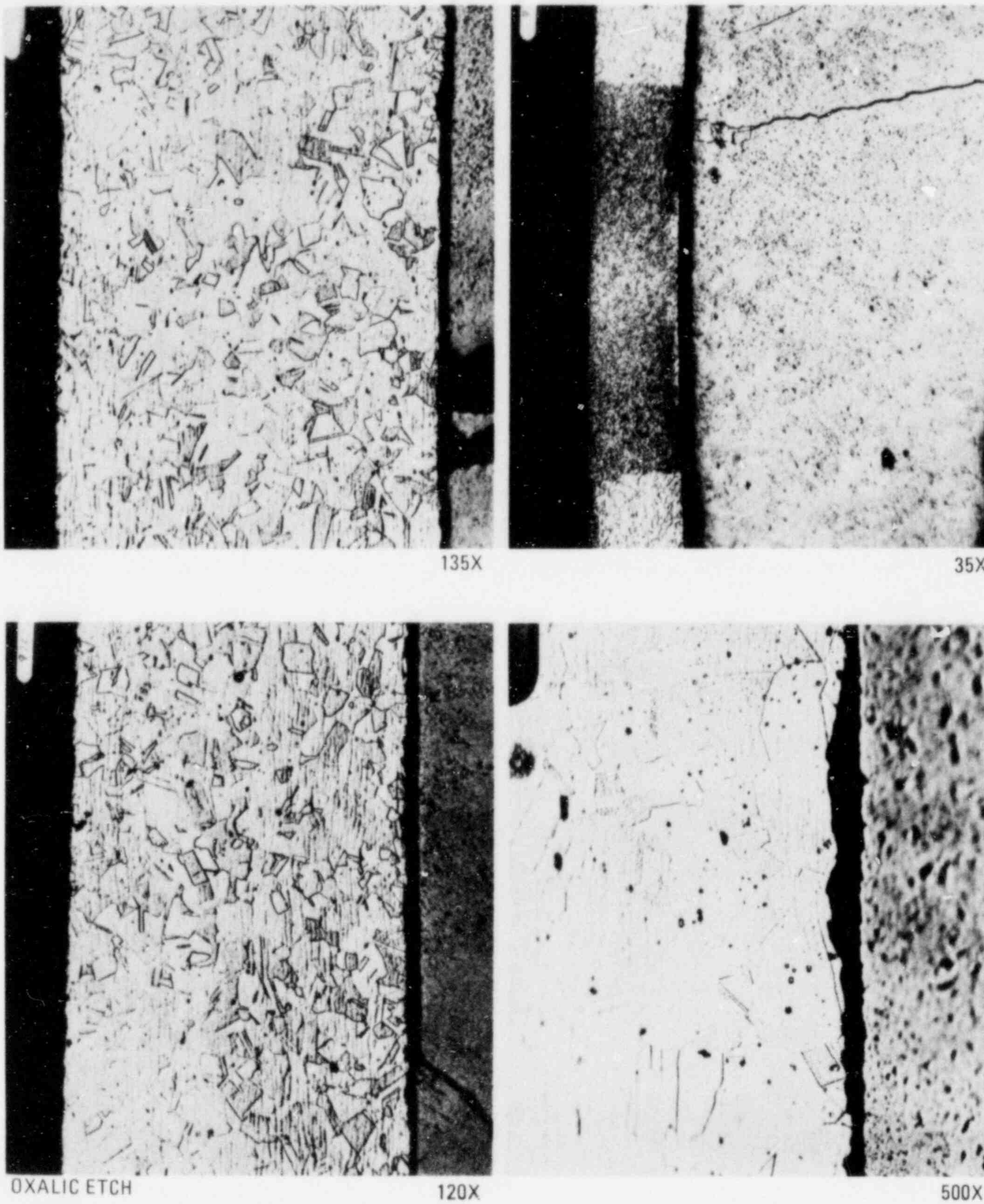
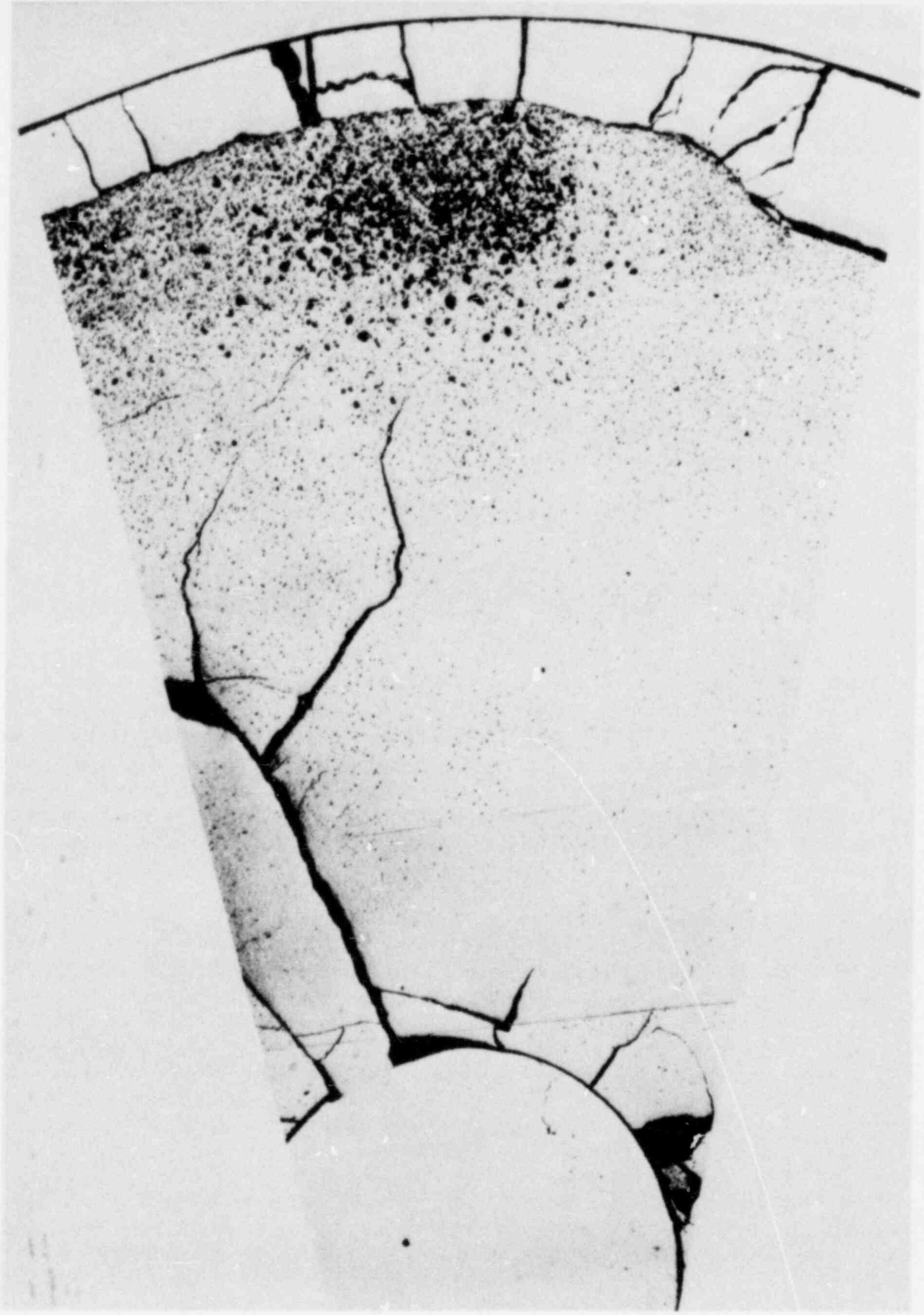


Fig. 18. Stainless steel clad from Sample 1. Microstructure of welded clad has not been sensitized and appears to be undamaged. (From Ref. 2.)



~10X

Fig. 19. Segment of Sample 2 showing porous region. Fuel between porous area and clad (clad not shown) is a uniform delta phase material; that of porous region has a two-phase structure. (From Ref. 2.)

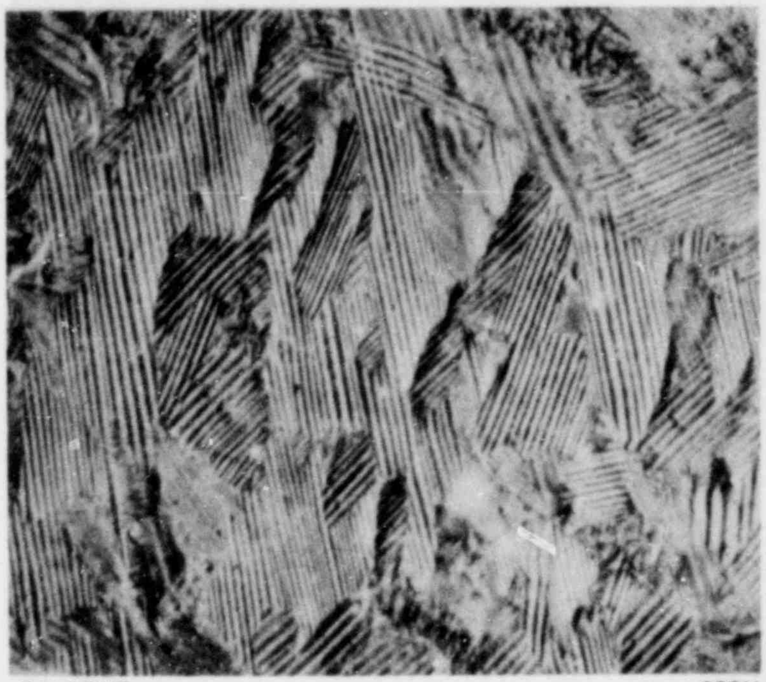


Fig. 20. Microstructure of zirconium rod in Sample 2. Chevron structure is that of epsilon phase, indicating a hydrogen content of $ZrH_{1.7}$+. (From Ref. 2.)

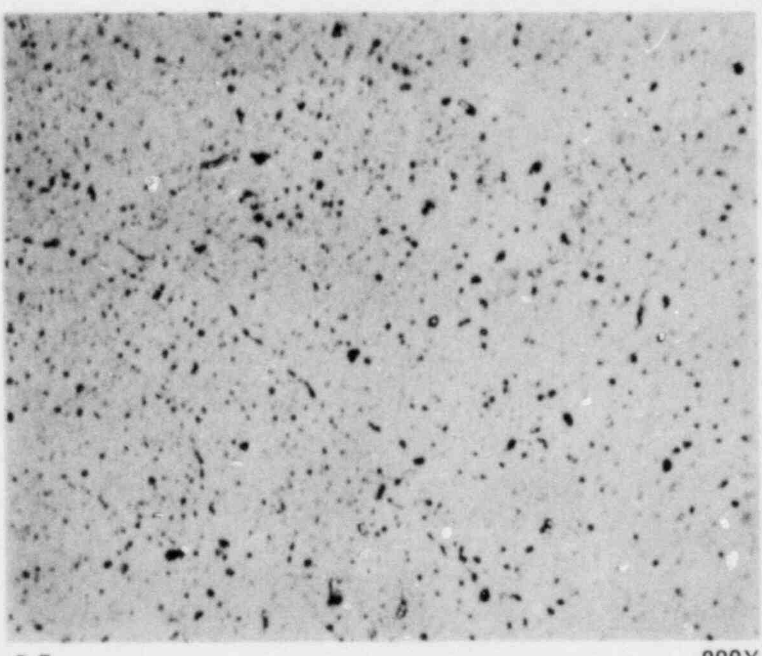


Fig. 21. Typical fuel structure in areas away from porous region of Sample 2 (from Ref. 2)

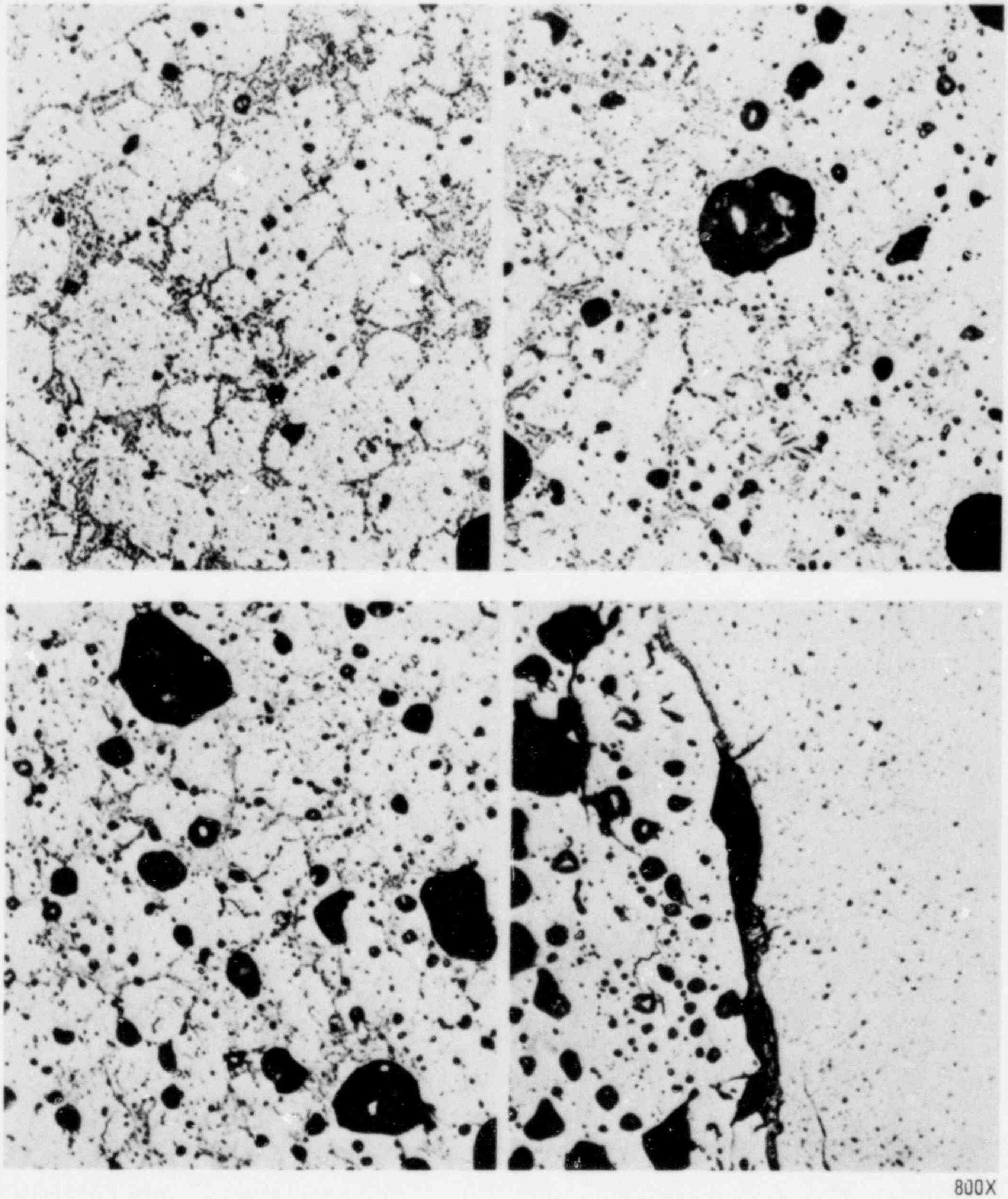


Fig. 22. Sample 2 fuel from porous area. Two-phase material is probably delta with alpha-delta eutectic on grain boundaries. Bottom right micrograph is of porous to sound material interface. (From Ref. 2.)

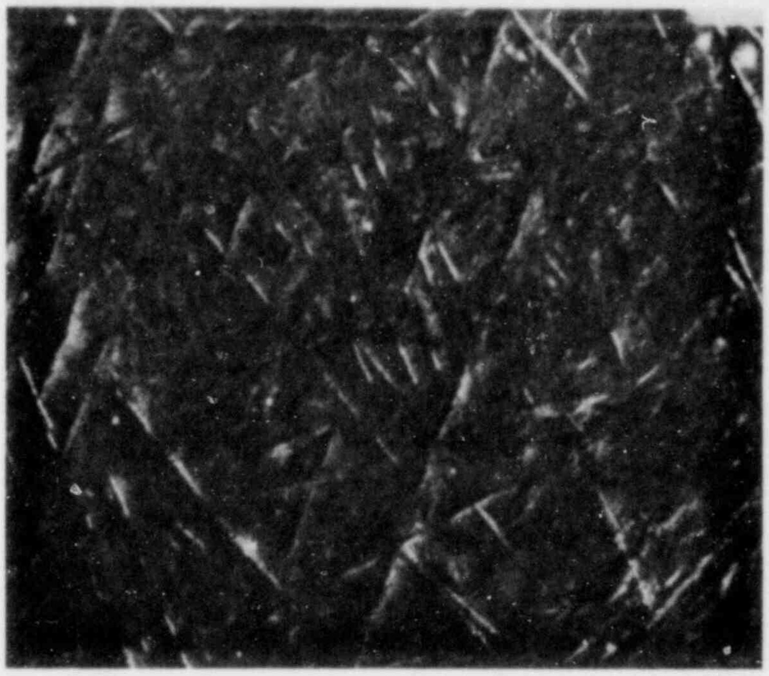


Fig. 23. Microstructure of zirconium rod from Sample 3 (from Ref. 2)

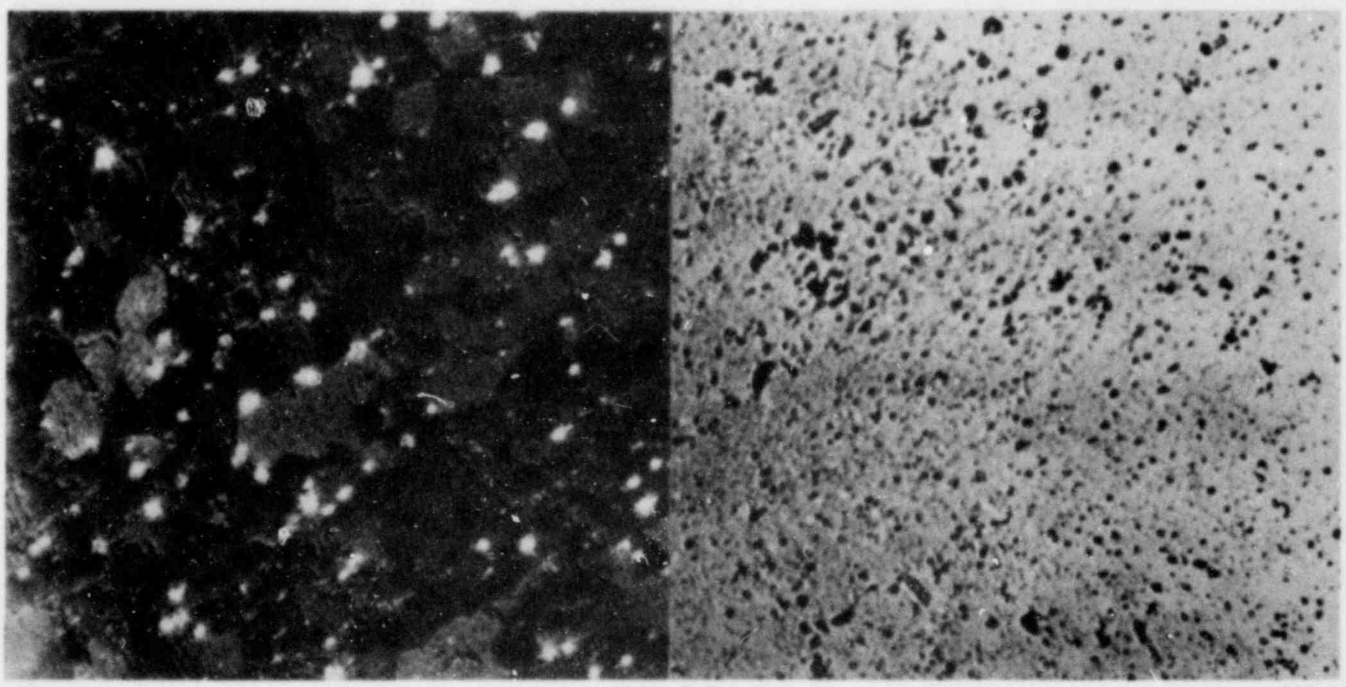


Fig. 24. Two fuel structures were found in Sample 3 similar to Sample 1. The alpha (left) comprises about 20% of the area and delta the balance. (From Ref. 2.)

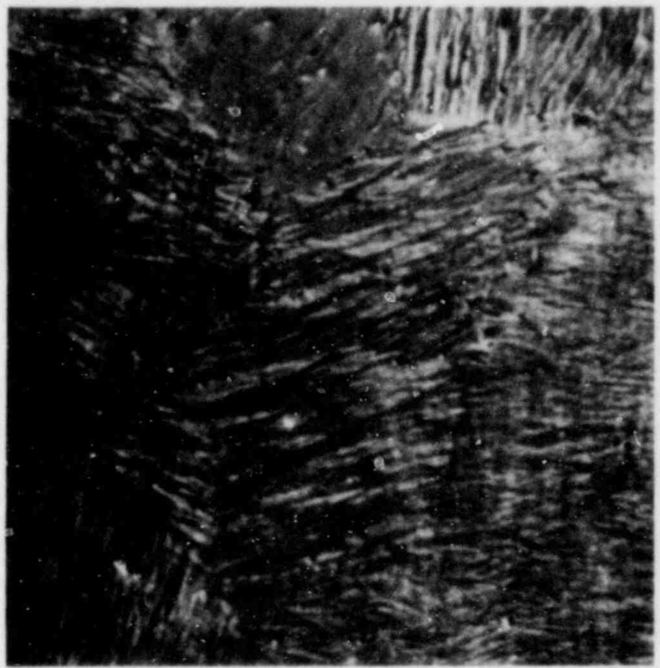


Fig. 25. High hydrogen microstructure is present in the central zirconium rod of Sample 4. (From Ref. 2.)

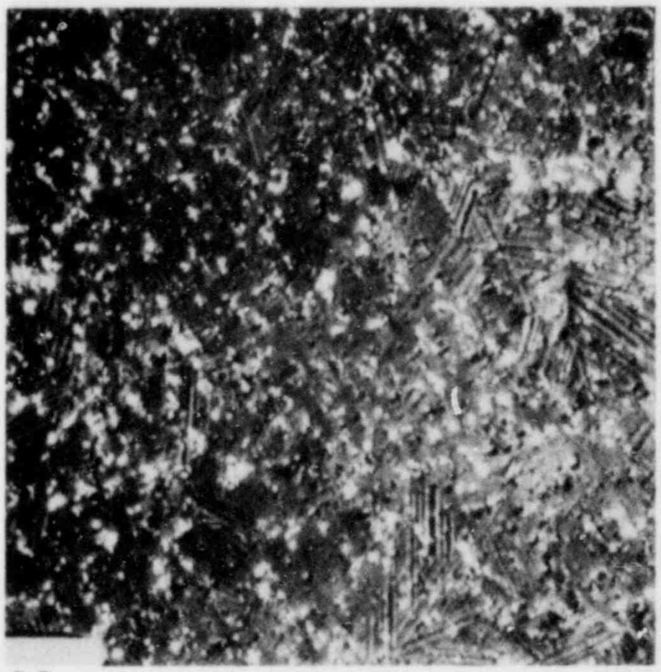


Fig. 26. Sample 4 fuel is shown in this micrograph. Structure is primarily delta with some epsilon also present. Hydrogen ratio is probably close to as-manufactured specifications. (From Ref. 2.)

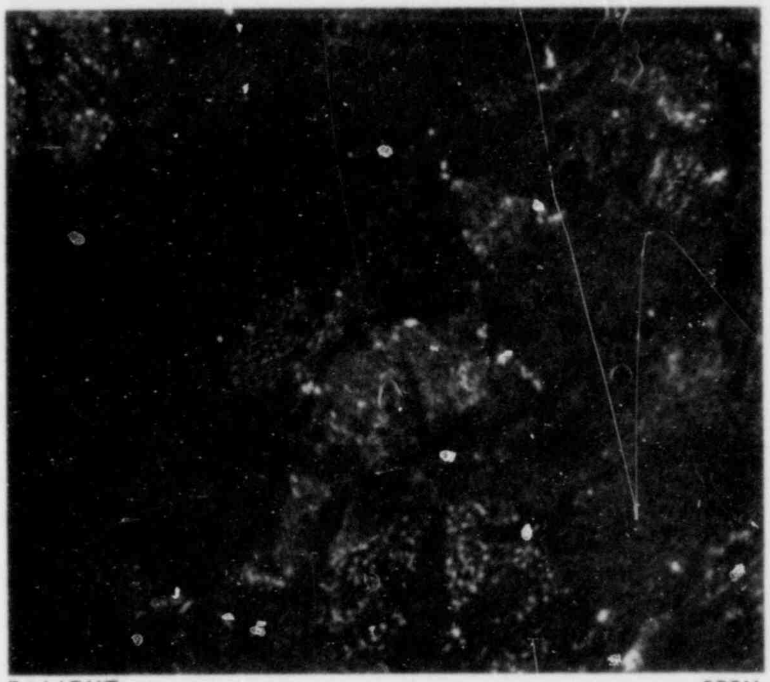


Fig. 27. Sample 5, central zirconium rod (from Ref. 2)

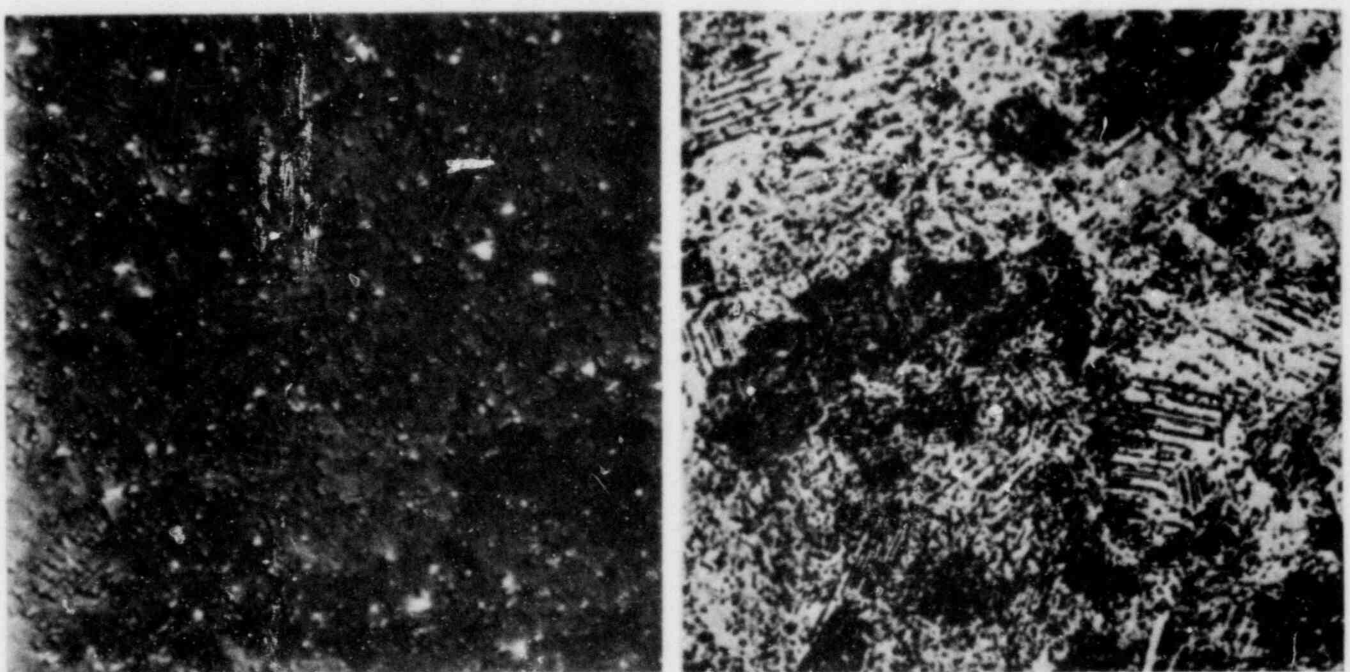
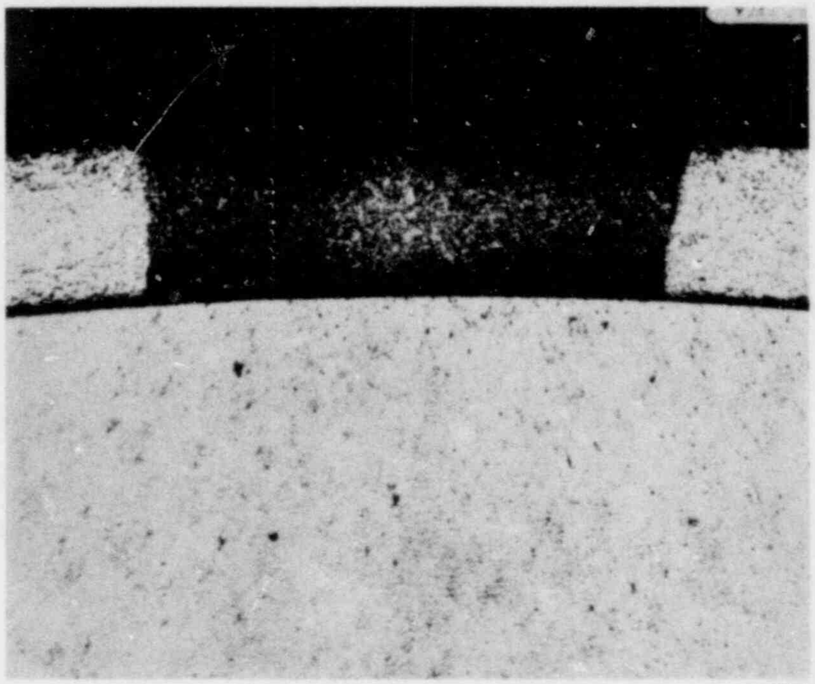
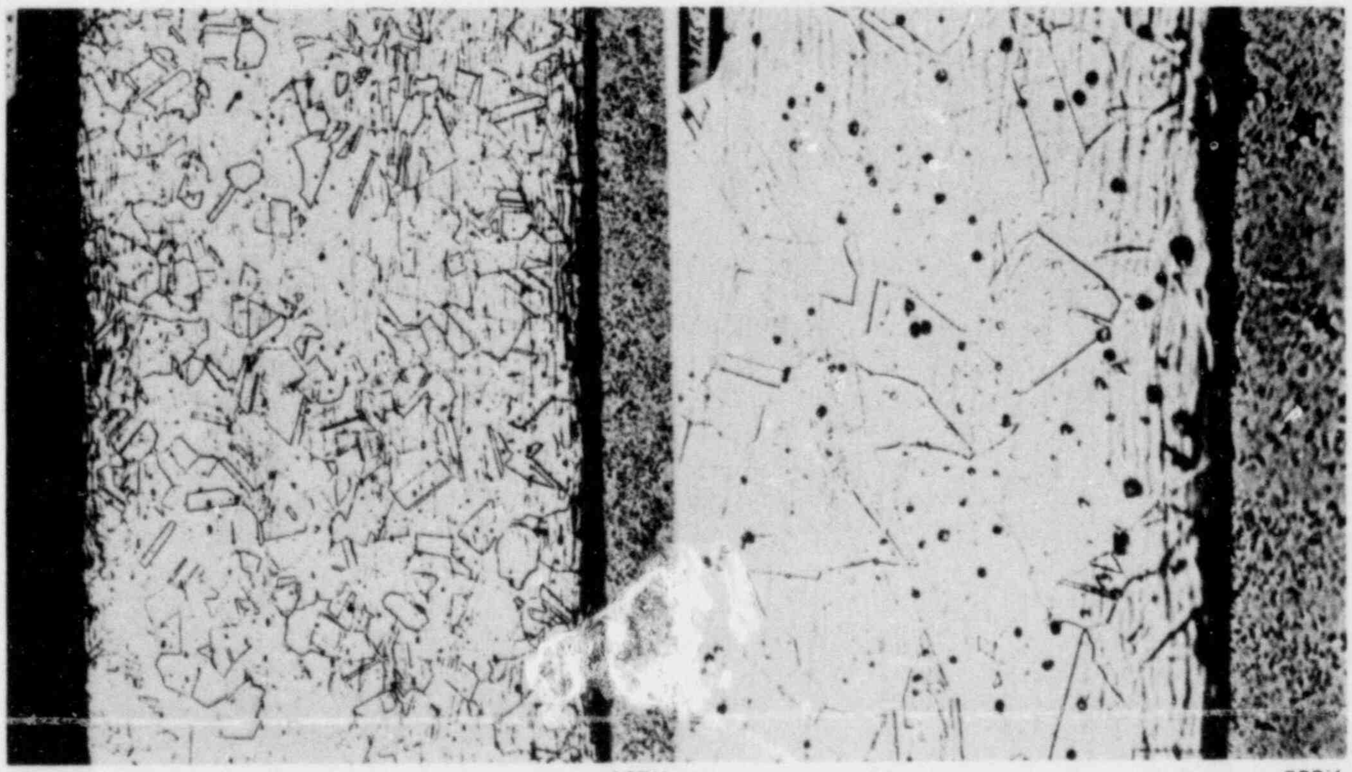


Fig. 28. Typical fuel from Control Sample (5). Microstructure is delta with some epsilon. (From Ref. 2.)



35X



120X 500X

Fig. 29. Stainless clad from Control Sample (5) appears undamaged. (From Ref. 2.)

from the cladding by a cracked nonporous delta phase layer which extends circumferentially but nonuniformly. The two-phase porous area is delta phase with alpha-delta eutectic on the grain boundaries. Approximately 20% of the fuel in Sample 3 is primarily alpha phase, the remainder being delta. The Control Sample (5) from the bottom of the fuel consists of a delta matrix with epsilon-phase inclusions.

The stainless steel cladding is structurally sound and does not appear to have reacted chemically with the fuel rods. A small gap is present between the fuel and the cladding.

2.4. GROSS AND ISOTOPIC GAMMA SCANS

The gross gamma scan of the most damaged fuel element is shown in Fig. 30. The shape of this curve reflects the burnup of the fuel as a function of position, using a slit width of 49.9 mils. The burnup appears as a broad distribution peaking in the center and decreasing toward the fuel ends. Small peaks occur at the ends of the fuel element; two small reductions in the gamma intensity occur at positions of fuel rod interfaces (three fuel sections, each 5 in. long, contained within the element clad).

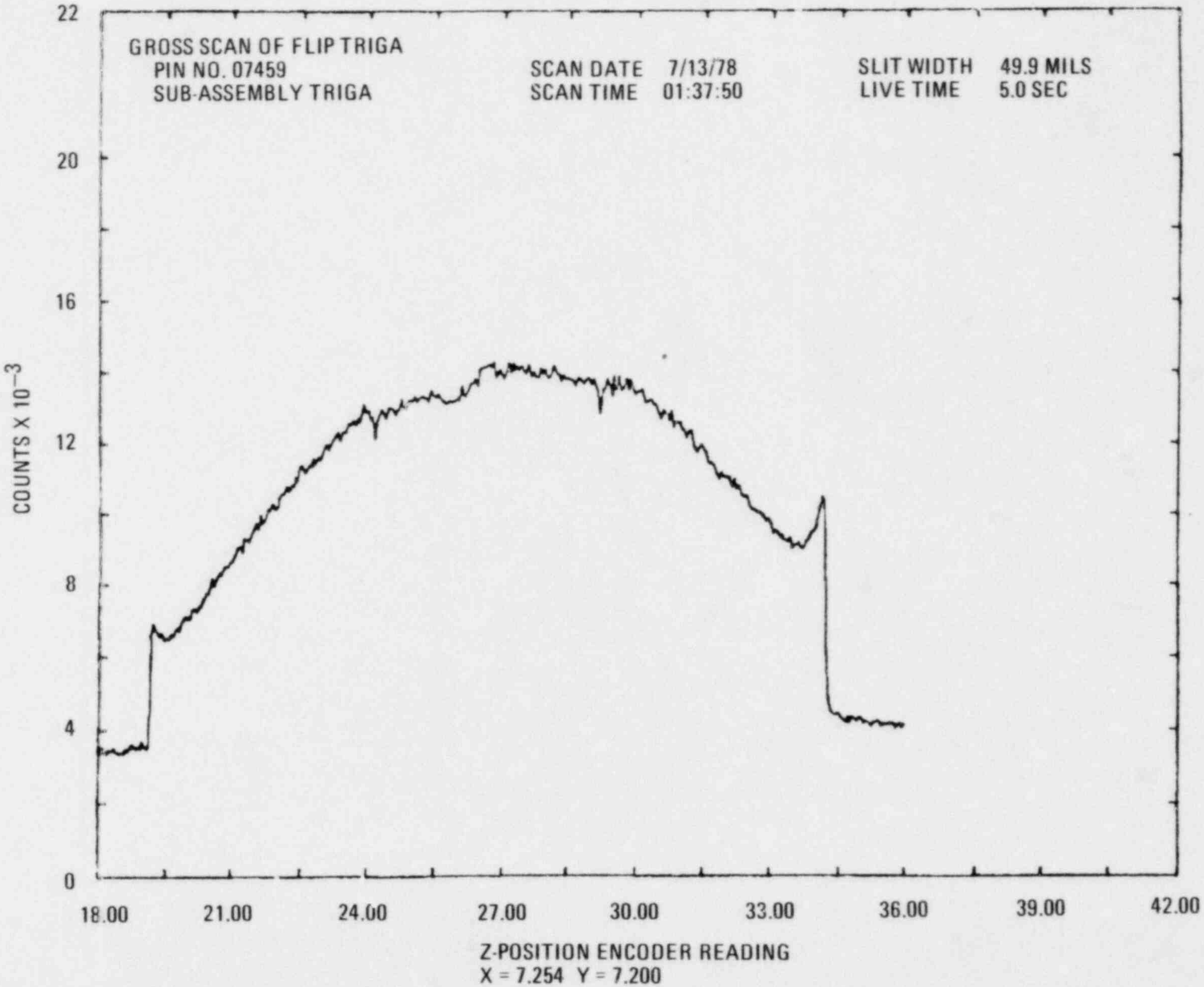


Fig. 30. Gross gamma scan of most damaged fuel element (from Ref. 2)

3. COMPARATIVE OPERATIONS AND RESULTS AT OTHER FACILITIES

Of the several reactor facilities using TRIGA-FLIP fuel, only three have the similarities of core configuration and/or pulse size that place them in a category comparable with the Texas A&M facility. These are the reactor facilities of the University of Wisconsin, Washington State University, and General Atomic at San Diego.

3.1. UNIVERSITY OF WISCONSIN

This facility has been operated with a core configuration very similar to that at Texas A&M in which the fuel damage occurred. While the Texas A&M facility has control rods and a central pulse rod, Wisconsin's has control blades and a pulse rod. The Texas and Wisconsin facilities both operated with mixed cores containing 35 centrally located FLIP elements. Wisconsin's generated about 2 MW-days of energy and pulsed 283 pulses, with its mixed core containing 35 FLIP elements. The major differences in operation between the Wisconsin core and the Texas A&M core are the relatively small energy generation at Wisconsin and, most important, the fact that the pulse size was never over \$2.00. No problems have been experienced with the fuel, and the reactor is operating with a full core of FLIP elements (since June 1979). The core has operated ~24 MW-days and pulsed ~74 times.

3.2. WASHINGTON STATE UNIVERSITY

The Washington State University reactor core is of particular interest because of its operation with a mixed core configuration similar to that at Texas A&M and with a pulse size up to \$2.50 compared with the Texas A&M \$2.70 maximum pulse. The mixed core with 35 centrally located FLIP elements was used with control blades and a pulse rod. An important difference is that Washington State's pulse rod was water followed, causing larger power

peaking factors than the air-filled aluminum follower rod used with the Texas A&M pulse rod. To date, Washington State has pulsed over 55 times with insertions of \$2.50. This is about the same number of pulses Texas A&M performed with insertions of \$2.70 (these pulses being included in the approximately 80 pulses greater than \$2.00 at Texas A&M). Temperatures measured in the instrumented element at Washington State, which is in a location similar to the instrumented element at Texas A&M, compare favorably with the measured temperatures (~500°C) for a \$2.50 pulse in the Texas A&M core. Calculations performed at Texas A&M indicate that a \$2.50 pulse with a water-followed pulse rod would produce higher peak fuel temperatures in an adjacent FLIP element than a \$2.70 pulse with an air-followed rod.

No problems have been experienced with the fuel at Washington State. It is significant to point out, however, that the steady-state usage of the Texas core is very much greater than that of the Washington State core. Prior to the high power pulses at Texas A&M, the core had about 170 MW-days of operation, while the Washington State reactor core had approximately 8 MW-days of operation. Washington State still operates with a mixed core, but the number of FLIP elements has been increased to 47. Typical operation is ~2 MW-days/month and only 5 to 10 routine (\$2.00) pulses have been performed since the pulsing program in 1976.

3.3. GENERAL ATOMIC COMPANY

TRIGA-FLIP fuel has been operated extensively at General Atomic. Only the initial tests of FLIP fuel were done in a mixed core, which contained 18 centrally located FLIP elements surrounded by standard fuel. All other operations have been performed with a full core of FLIP elements. This core was initially operated in the Mark III reactor for long-term steady-state operation at up to 2 MW. It was then transferred to the Mark F reactor for general-purpose (steady-state and pulsing) operations. The core had about 750 MW-days of burnup prior to any pulsing operations. Pulse insertions as high as \$3.20 have produced measured peak temperatures up to 550°C. A

summary comparison of the operational histories of the Texas A&M core and the General Atomic core is given in Table 5.

The differences in core characteristics, such as peaking factors and temperature coefficient caused by the mixed core versus full core, are responsible for the fact that different pulse sizes and steady-state powers produce very similar operating temperatures. The primary differences noted in Table 5 are that the General Atomic core had ~5 times more steady-state operation than the Texas A&M core prior to high power pulsing, whereas the Texas A&M core had 3.5 times as many high power pulses and the Texas A&M calculated peak pulse temperature is slightly higher.

3.4. ANNULAR CORE PULSE REACTOR FUEL TEST PROGRAM

The only previously observed fuel damage similar to that seen for the Texas A&M fuel occurred during a high power pulse testing program at General Atomic where five or six centrally located special elements containing 8.5 wt %, 93% enriched uranium were pulsed 426 times to peak (calculated) temperatures of ~1150°C. This was a program for the design of the optimized pulsing fuel for the annular core pulse reactor. No significant amount of steady-state operation was conducted prior to the pulse tests. Fuel swelling occurred to the extent that the 15-mil radial gap between the fuel and the clad was filled and the clad diameter was increased about 60 to 65 mils. The swelling occurred in the region of the localized hot spot area of the fuel. Extensive cracking of the fuel was observed as well as a porous region very similar to that in the Texas A&M fuel. The porous area extended to the outer surface of the fuel, however, not terminating prior to an essentially undamaged, thin surface area as is the case in the Texas A&M fuel. No neutron radiography nor any other examinations were made of the fuel to determine loss of hydrogen from the porous region. It was concluded that the hydrogen pressure (~90 atmospheres equilibrium hydrogen pressure for a temperature of 1150°C) generated in the fuel matrix at high pulse

TABLE 5
SUMMARY COMPARISON OF OPERATIONAL HISTORY

	Texas A&M	General Atomic
Core	Mixed 35 FLIP 63 Std. TRIGA	Full core FLIP 63-100 elements
Maximum steady-state power	1 MW	2 MW (most operation at 1.5 MW)
Steady-state burnup prior to high power pulsing	~170 MWD[a]	~750 MWD
Total number of pulses	725	158
Maximum pulse size	\$2.70	\$3.20
Number maximum pulses	54	15
Other pulses	~80 >\$2.00	~25 ≥\$3.00
Pulse $\hat{T}_{measured}$	520°C	550°C
Pulse $\hat{T}_{calculated}$[b]	~870°C[c]	~850°C
Steady-state $\hat{T}_{measured}$	450°C[d]	550°C
Steady-state $\hat{T}_{calculated}$	~575°C[e]	~550°C

(a) Additional ~100 MWD before discovery of damaged fuel.
(b) Adiabatic at fuel OD.
(c) Not same element that contains thermocouples.
(d) ~400°C prior to \$2.70 pulses.
(e) Higher-power location compared with location of temperature measurement.

temperature caused the porous fuel structure and resultant swelling. Pressures measured within the clad but outside the fuel matrix never exceed 40 psi.

4. DISCUSSION OF POSSIBLE MECHANISM

The information presented in Section 3 shows that while other TRIGA reactor facilities have had operations very similar to those at Texas A&M, each facility in total has a unique combination of steady-state and pulsing experience for its core. Texas A&M shows a long steady-state history coupled with more extensive high power pulsing than any other facility. While the operating history at General Atomic has been very similar to, and in some cases more extensive than, that at Texas A&M, there are differences in configuration, fuel temperature, operational procedures, and the number of pulses that may be significant to fuel damage. A possible mechanism for fuel damage (other than direct attainment of temperatures of ~1150°C) that fits the general characteristics observed in the damaged fuel is described in the remainder of this section.

The postulated mechanism of swelling and bowing in the damaged fuel elements may be discussed as a number of interrelated phenomena:

1. Hydrogen migration under thermal gradients from regions of higher temperature to regions of lower temperature in the fuel is a governing factor in causing swelling of the fuel.

2. Generation of high local gas pressures in the fuel matrix during very high power pulsing results in swelling and increased pore size. The gas pressure is produced by hydrogen evolution in the hot spots near the surface of the fuel rod where increased hydrogen concentrations exist. These increased hydrogen concentrations result from relatively long-term steady-state operation, during which the hydrogen tends to redistribute radially and axially by migrating to the cooler surface regions. However, there is a threshold temperature just above the temperatures in the thin

region immediately adjacent to the surface of the fuel that would limit the hydrogen migration to some distance below the surface. The higher concentration of hydrogen in the subsurface region would lead to much higher internal gas pressures at the hot spot during pulse operation than would occur with the nominal hydride composition ($ZrH_{1.6}$). To produce pressures equivalent to those in $ZrH_{1.6}$ at 1150°C, the H/Zr ratio would have to increase to about 1.85 and be subjected to a temperature of about 880°C. If the H/Zr ratio were 1.75 the required temperature would be about 1010°C.

3. The "hydrogen-depleted" regions shown in the neutron radiographs result from the loss of hydrogen to the cooler parts of the fuel. This hydrogen evolved from the hot spots during high power pulsing and appears to have been absorbed by the cooler regions, especially the central zirconium rod, thereby depleting the hot spots of hydrogen. Under high-power steady-state operation, the hydrogen in the depleted hot spots would be replenished (but at a much slower rate) by diffusion in the solid state and by migration in the gas phase from neighboring regions.

4. The central axial zirconium rod in the center of the fuel element appears to be a source of stress generation under the conditions encountered in the hottest parts of the fuel. These rods can swell up to ~15% in volume upon absorption of hydrogen to give an H/Zr ratio $\geq$1.7. Under extreme swelling conditions, the initial clearance between the zirconium rod and the fuel appears to be too small and the zirconium rod will swell and press against the fuel. It appears as if stresses were generated by the swelling zirconium rod which were large enough to crack the fuel. The neutron radiographs and metallography of the most damaged highest-temperature portions of the fuel element indicate complete hydriding of the central zirconium rods, whereas the low-temperature, undamaged fuel shows little hydriding in the zirconium rod. In some cases

the expanded zirconium rod actually bonds to the fuel at points of contact.

5. The structure of the hydrogen-depleted region in the distressed fuel is shown in Fig. 17. This region apparently contains significant quantities of alpha-phase material formed by loss of hydrogen from the original delta phase. It is of interest to note that the fine pores in this region are largely in the form of a maze of pores at the grain boundaries. As the original delta phase transforms to the alpha phase, the pores that are present in the alloy are swept to the newly formed grain boundaries of the alpha phase. Also, the change in density upon transformation from delta to the denser alpha phase will favor the formation of voids which will be trapped at the new grain boundaries.

From the foregoing discussion, it is possible to ascribe the bowing and swelling phenomena observed in the damaged fuel elements to the following series of events (also see Fig. 31 for additional descriptive information):

1. Hydrogen migrates toward the cooler surface regions of the fuel in an unsymmetrical configuration governed by thermal gradients and temperatures. The migration occurs over long periods of steady-state operation. See Fig. 31(a).

2. During steady-state operation, essentially no hydrogen migrates to the immediate surface region of the fuel or to the central zirconium rod (which is in the hottest part of the element) because of the temperature gradient. The high central temperature forces hydrogen away from the center, and very low migration rates at the immediate fuel surface region temperature prevent hydrogen build-up in this zone. See Fig. 31(a). (There is possibly some very small degree of hydriding of the central zirconium rod during high-power steady-state operation by reaction with hydrogen in the gas phase in the surrounding spaces.)

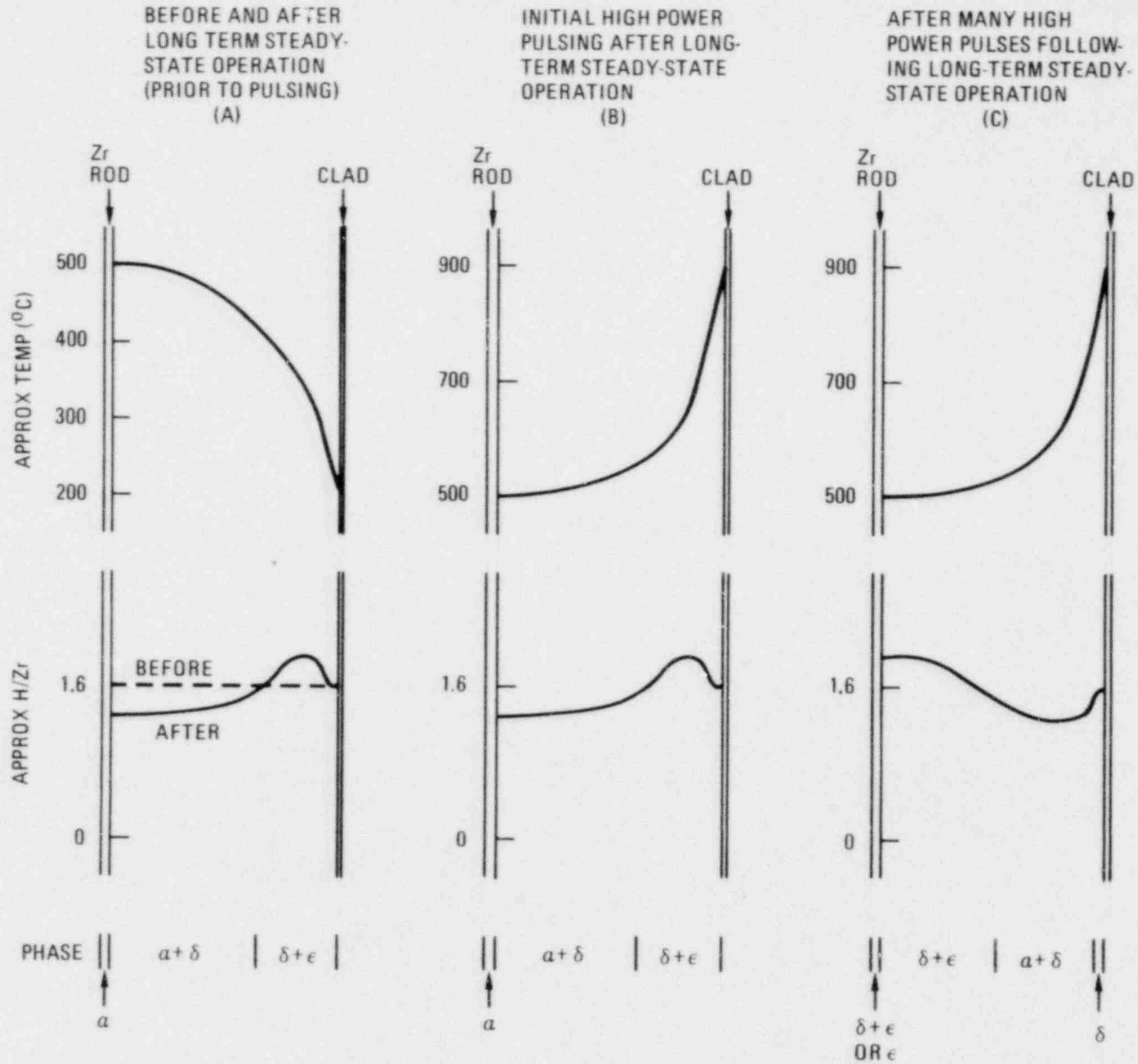


Fig. 31. Qualitative graphic description of possible changes in hydrogen distribution leading to fuel damage

3. Concentration of hydrogen takes place in the fuel internal to the cool surface area. This may set up the conditions for circumferential cracks near the surface, possibly during later high power pulse operation. The circumferential cracks can be postulated to occur either before or after pulsing. See Fig. 31(a) and/or (b).

4. The concomitant increase in hydrogen pressure in the hot spots during high power pulsing results in swelling and an increase in pore size. See Fig. 31(b).

5. Progressive loss of hydrogen takes place in the hot spots and is absorbed in the cooling regions of the fuel during and following the pulsing. See Fig. 31(c).

6. The unhydrided axial zirconium rod in the center of the fuel element can act as a crack initiator and stress-raiser by swelling as it hydrides in situ during very high power pulses and presses against the surrounding fuel element. See Fig. 31(c).

5. RECOMMENDED OPERATIONAL PROCEDURES

To help identify potential fuel damage problems prior to the time that the extent of damage demands removal of the fuel element from the core, the following operational procedures are recommended.

1. Physically inspect and measure fuel carefully after all pulsing (or steady-state) operation in new domains for a given reactor facility to insure continued successful operation. Routine fuel inspection should be on an annual basis. Inspections resulting from operations in new domains could even be limited to the immediate core regions most affected by the new operations, such as hot spot regions. For cores that require for fuel inspection the disassembly of fuel clusters involving possible damage to fittings and threads from excessive handling, the annual inspection could include only the central, high power region of the core, elements adjacent to water holes, and a selective sampling of lower power elements.

2. If not precluded by experimental or structural conditions, place the thermocouple-instrumented element in the hottest position to minimize as much as possible the uncertainty in the maximum temperature.

3. When pulsing performance is increased to new levels, correlate new and old performance as much as possible through period, energy, power, and temperature measurements to relate to Δk. Relative energy measurements can also be made with gold foils if successive conventional detector values do not behave as expected, or as an additional correlation.

It is useful to note that there are numerous examples of minor bending being detected in fuel elements during an inspection but disappearing at the next inspection. This is probably due to reinserting the fuel element in the core in a different angular orientation after the fuel inspection. The removal and reinsertion of an element is highly likely to result in a new rotational orientation. This change in orientation would likely be all that is necessary to reverse any possible bending caused by long-term steady-state hydrogen migration (or other possible thermal gradient stress).

However, progressive damage of the kind experienced at Texas A&M, and manifested by the local porosity and swelling of the fuel, is terminated only by reducing the high power pulsing temperature to which the fuel is subjected. Since the hydrogen pressure increases nearly exponentially with fuel temperature, small temperature changes can make a very significant difference in fuel damage. In fact, this relationship makes the fuel damage mechanism act, for practical purposes, as if there were a threshold temperature for damage. If damage begins to appear, a reduction in measured temperature as small as 20°C (~40°C in peak pulsing temperature) will reduce hydrogen pressures by about 33% and is likely to stop any progressive damage.

It also seems evident that the damage threshold was just slightly exceeded in the Texas A&M core. The calculated spread in the power generation among the four damaged fuel elements was about 5% and the damage ranged from very slight to significant. Also, the maximum temperature in the Texas A&M core was calculated to be only slightly greater (~20°C) than that in the FLIP core at General Atomic, where no fuel damage has occurred.

6. FUEL TEMPERATURE SAFETY LIMIT

The fuel temperature safety limit is unaffected by the events and processes discussed in the foregoing sections because it is set by the average H/Zr ratio in the element. The safety limit is based on the resulting hydrogen pressure exerted on the fuel element clad, tending to rupture it. While redistribution of hydrogen within the fuel material can result in damaging pressures localized within the fuel matrix, the pressure exerted on the clad is determined by the overpressure exterior to the fuel matrix and is set by the average H/Zr ratio in the fuel element. When the maximum pulse size is increased by relatively small increments, large numbers of pulses would be required for damage to become evident.

It is also pointed out that, in the very few cases of fuel damage of the type discussed in this report, there was never any danger of the clad rupture that would be necessary for fission product release.

7. OPERATIONS AT TEXAS A&M SINCE SEPTEMBER 1976

Since the discovery of the fuel damage at Texas A&M in September 1976, pulsing operations have been suspended but steady-state operations continued without change. Since November 1979 the core has consisted entirely of FLIP elements. Routine operations produce ~100 MW-days of burnup per year. No additional fuel damage has been observed and, in fact, the least damaged of the four bowed fuel elements has continued operation and no longer exhibits any sign of damage.

8. REFERENCES

1. "Status Report on Damage to FLIP Fuel During Operation of the NSCR at Texas A&M University," a letter report to the Director, Division of Reactor Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545, from John D. Randall, Director, Texas A&M University Nuclear Science Center, November 1, 1976.

2. Carlson, R. G., "TRIGA Element Metallography," memorandum to R. S. Wisner, Argonne National Laboratory-West, August 13, 1980.

GENERAL ATOMIC COMPANY
P. O. BOX 81608
SAN DIEGO, CALIFORNIA 92138