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August 6, 2009

U.S. Nuclear Regulatory Commission
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Mail Station P1-37
Washington, DC 20555-0001

Reference: Docket 50-186
University of Missouri – Columbia Research Reactor
Amended Facility License R-103

Enclosed is a request to amend the Technical Specifications appended to Facility License R-103 pursuant to 10 CFR 50.59(c) and 10 CFR 50.90.

If you have any questions, please contact Leslie P. Foyto, the facility Reactor Manager, at (573) 882-5276 or foytol@missouri.edu.

Sincerely,

Ralph A. Butler, P.E.
Director

RAB/djr

Enclosures

NOTARY PUBLIC
NOTARY SEAL
STATE OF MISSOURI

MARGEE P. STOUT
My Commission Expires
March 24, 2012
Montgomery County
Commission #08511436

Margee P. Stout

A020
NRR

August 6, 2009

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REFERENCE: Docket 50-186
University of Missouri – Columbia Research Reactor
Amended Facility License R-103

SUBJECT: Written communication as specified by 10 CFR 50.4(b)(1) requesting U.S. Nuclear Regulatory Commission approval to amend the Technical Specifications appended to Facility License R-103 pursuant to 10 CFR 50.59(c) and 10 CFR 50.90

Introduction

The University of Missouri Research Reactor (MURR) is requesting a change to the facility Technical Specifications (TSs) that would allow the implementation of an engineered safety device that would prevent operation of the reactor unless the Center Test Hole Canister is inserted and latched onto the inner pressure vessel. TS 3.3, "Reactor Safety System," lists the safety system instrument channels that are required to be operable during the different modes of reactor operation. This modification would add two new instrument channels to the reactor safety system and allow a change in the methodology that is currently used to calculate the reactivity contribution of samples, or experiments, loaded in the center test hole. By implementing this change, the engineered safety device would account for the reactivity contribution of the Center Test Hole Canister in the flux trap reactivity balance calculation because the canister would now be considered a part of the reactor, but its reactivity worth would not be included within the experiment reactivity limit of TS 3.1.h. This would allow greater flexibility and capacity in the center test hole for the irradiation of high specific activity radioisotopes that are used for radiopharmaceutical research and cancer treatments.

Background

The *center test hole* is defined by TS 1.3 as "*The volume in the flux trap occupied by the removable experiment test tubes.*" Additionally, the *flux trap* is defined by TS 1.6 as being "*That portion of the reactor through the center of the core bounded by the 4.5 inch inside diameter tube and 15 inches above and below the core vertical center line.*" The "*removable experiment test tubes*" is also referred to as the Center Test Hole Canister in the Hazards Summary Report and as the Flux Trap Holder among the operating staff. These terms are well understood and are used interchangeably at MURR.

The number and the volume of samples which can be irradiated in the center test hole are limited mechanically by the design of the Center Test Hole Canister. Three Center Test Hole Canisters are designed and approved for use at MURR, those being a six-tube, a three-tube, and a single-tube. Additionally, if the reactor is to be operated without a Center Test Hole Canister, then the center test hole Strainer is installed to preclude the possibility of any foreign objects from entering the flux trap region.

The three-tube canister consists of three aluminum tubes, each being 10 feet 2 inches (3.1 m) long with an internal diameter (I.D.) of 1.334 inches (3.4 cm). The tubes are arranged in a clover leaf pattern and spot welded together to form a single assembly. The tubes are clearly identified as A, B, and C, both physically on the Center Test Hole Canister and in all sample loading documentation. Stainless steel bands wrapped around the aluminum tubes provide redundancy for the spot welds. A support rod and base piece are attached to the bottom of the tube assembly. The overall length of the canister is 14 feet 2½ inches (4.3 m).

The six-tube canister is similar in design to the three-tube with a few exceptions. There are an added four vertical inches of irradiation capacity in the three 1.334-inch (3.4 cm) I.D. tubes and the addition of three smaller diameter irradiation tubes [0.68 inch (1.7 cm) I.D.]. The small diameter tubes are designed to allow movable and unsecured experiments to be irradiated in the flux trap (Approved by License Amendment No. 31). The small diameter tubes are clearly identified as 1, 2, and 3, both physically on the canister and in all sample loading documentation.

A single-tube canister is also designed for use in the center test hole. It was used during initial operation of the MURR and replaced by the three-tube assembly when an increase in material irradiation capacity was required. It is no longer used.

When installed, the Center Test Hole Canister position is positively determined by a latching mechanism located at the top of the assembly. Two Inconel metal latching fingers secure the canister to the upper portion of the inner reactor pressure vessel. To provide additional vertical alignment and support, the base piece engages into a test hole slot which is welded to the base flange of the reflector tank.

The experiment volume of the larger diameter tubes of both the three- and six-tube canisters is filled at all times with either aluminum or titanium spacers, water cans or experiment samples. The spacers, water cans and samples are maintained in position by a hold down rod assembly which is secured at the top of the canister by a head pin and hair pin keeper.

Three 1/16-inch (1.59 mm) thick standoffs attached to the internal wall of the sample tubes, and spaced 120 degrees apart, ensures that a minimum cooling channel exists on all sides of the samples. The small diameter tubes of the six-tube canister may or may not contain experiment samples during operation.

TS 3.1.h states that "*The absolute value of the reactivity worth of all experiments in the center test hole shall not exceed 0.006 ΔK.*" The current method of calculating the reactivity worth of all experiments in the center test hole during reactor operation includes not only the samples but the reactivity contribution of the Center Test Hole Canister itself. This is a conservative interpretation of TS definition 1.5, which states that an *Experiment* is:

"*a. Any device which is exposed to significant radiation from the reactor and is not a normal part of the reactor.*
b. Any operation designed to measure or monitor reactor characteristics or parameters."

This conservative approach of including the reactivity contribution of the Center Test Hole Canister, although it could be argued that when it is installed and latched it is a "*normal part of the reactor*," severely limits the total available reactivity worth of samples loaded into the flux trap region. Since the flux trap region has a positive void coefficient of reactivity (approximately $+0.865 \times 10^{-5}$ ΔK/K/cc void), the empty aluminum canister appears as a void when it is inserted into the flux trap region, thus adding a significant amount of positive reactivity to the calculation methodology. The reactivity contribution of the canisters has been well defined over the years through numerous measurements at low power. The reactivity worth of the three-tube and six-tube canisters is 0.0036 ΔK and 0.0050 ΔK, respectively. The MURR would like to implement an alternative methodology in which the reactivity contribution of the Center Test Hole Canister is not included within the limit of TS 3.1.h and only the reactivity contribution of the samples, or experiments, is. In order to conservatively implement this methodology, instrumentation [named the Flux-trap Irradiations Reactivity Safety Trip (FIRST)] will be installed such that insertion or removal of the Center Test Hole Canister or Strainer cannot physically be performed during reactor operation. This instrumentation would provide an input signal to the reactor safety system such that the reactor could not operate unless the Center Test Hole Canister, or Strainer, is properly installed – inserted and latched. Therefore, the Center Test Hole Canister would clearly be interpreted as "*a normal part of the reactor.*"

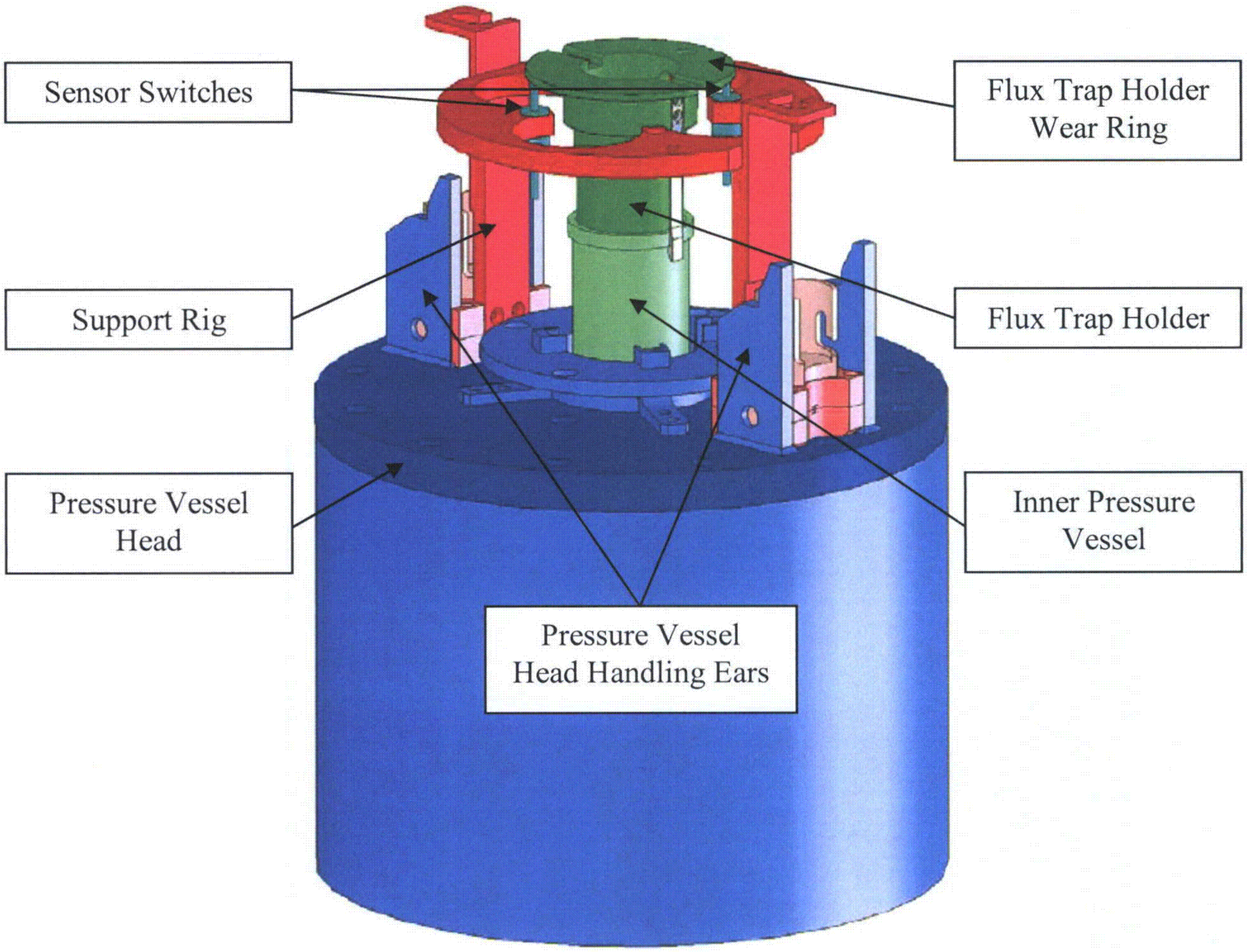
Theory of Operation

Modification Record 08-5, "Install Temporary Prototype of the Flux-trap Irradiations Reactivity Safety Trip (FIRST) Device," documented the installation of a testing prototype that was not connected to the reactor safety system. The prototype was installed in December 2008, and tested almost continuously since that time (see "Test" Section).

The FIRST instrument channeles will use two independent sensors to detect the position of the Center Test Hole Canister. When a sensor is 'made-up,' a relay coil is energized, maintaining a normally open contact in the closed position. This contact is one of several in a series that make up one leg of the 24 VDC input to the reactor safety system. This configuration is duplicated for the other sensor. When a sensor is no longer 'made-up,' the relay coil de-energizes, opening a contact, thus interrupting the 24 VDC input signal to the safety system. The resulting trip of the Non-Coincidence Logic Unit (NCLU) will de-energize the Trip Actuator Amplifiers (TAAs) and scram the reactor. This action ensures that if the position of the Center Test Hole Canister or Strainer changes from its secured and latched position, reactor operation will cease. Conversely, if the sensors are not 'made-up,' the TAAs cannot be reset. This ensures that the Center Test Hole Canister must be installed in order to start up and operate the reactor.

Description of the Reactor Safety System Center Test Hole Instrument Channels

The FIRST is comprised of the following major components: Flux Trap Holder Wear Ring, Sensor Switches, Support Rig, Relays and Power Supply, Safety System Interface, and FIRST Bypass Switch.



The wear ring is an existing component of the Center Test Hole Canister, or Flux Trap Holder. The wear ring is the top-most component of the Flux Trap Holder. Its position is fixed with respect to both the holder and the inner reactor pressure vessel when the holder is installed. The wear ring will be modified to provide a flanged radial extension at its top. This extension will be the reference point for contact with the sensor switches.

The sensors chosen for this service are waterproofed plunger-type switches with molded cables of sufficient length to ensure all connections are made above the pool water line. These switches have been tested over several months and numerous cycles under their intended service conditions. Further testing description for these switches is provided later in the Section entitled "Testing." The material used in the fabrication of the instrument cable jacket is a form of neoprene called "Carolprene." As stated in the report <u>Radiation Resistance of Elastomers</u>, neoprene has good radiation resistant properties up to a fluence of 1 x 10^8 rads and does not display signs of brittleness until approximately 3 x 10^8 rads. As measured directly at 10-MWs, the radiation field on the reactor pressure vessel head is approximately 1,300 rads/hr gamma and essentially no neutron. Assuming that the reactor operates on average 150 hours per week, 52 weeks/year at 10-MWs, it would take approximately 10 years to reach a fluence of 1 x 10^8 rads.

The support rig will attach to the reactor pressure vessel head via the existing vessel head handling ears, and will support and vertically align the two independent switches that close when the Center Test Hole Canister is in its secured and latched position. The support rig will be fabricated from 6061-T6 Aluminum and 304 Stainless Steel and designed using rack and pinion engagement pins to engage the existing handling ears of the pressure vessel head, provide a means to be installed, removed and stored, and allow physical access to all bolts of the pressure vessel head should the device prove unable to be disengaged and removed underwater. The rack and pinion engagement pins will be operated using a 2-inch rotating rod (a common handling tool in the pool), and will have opposed direction of rotation to counter any induced moment.

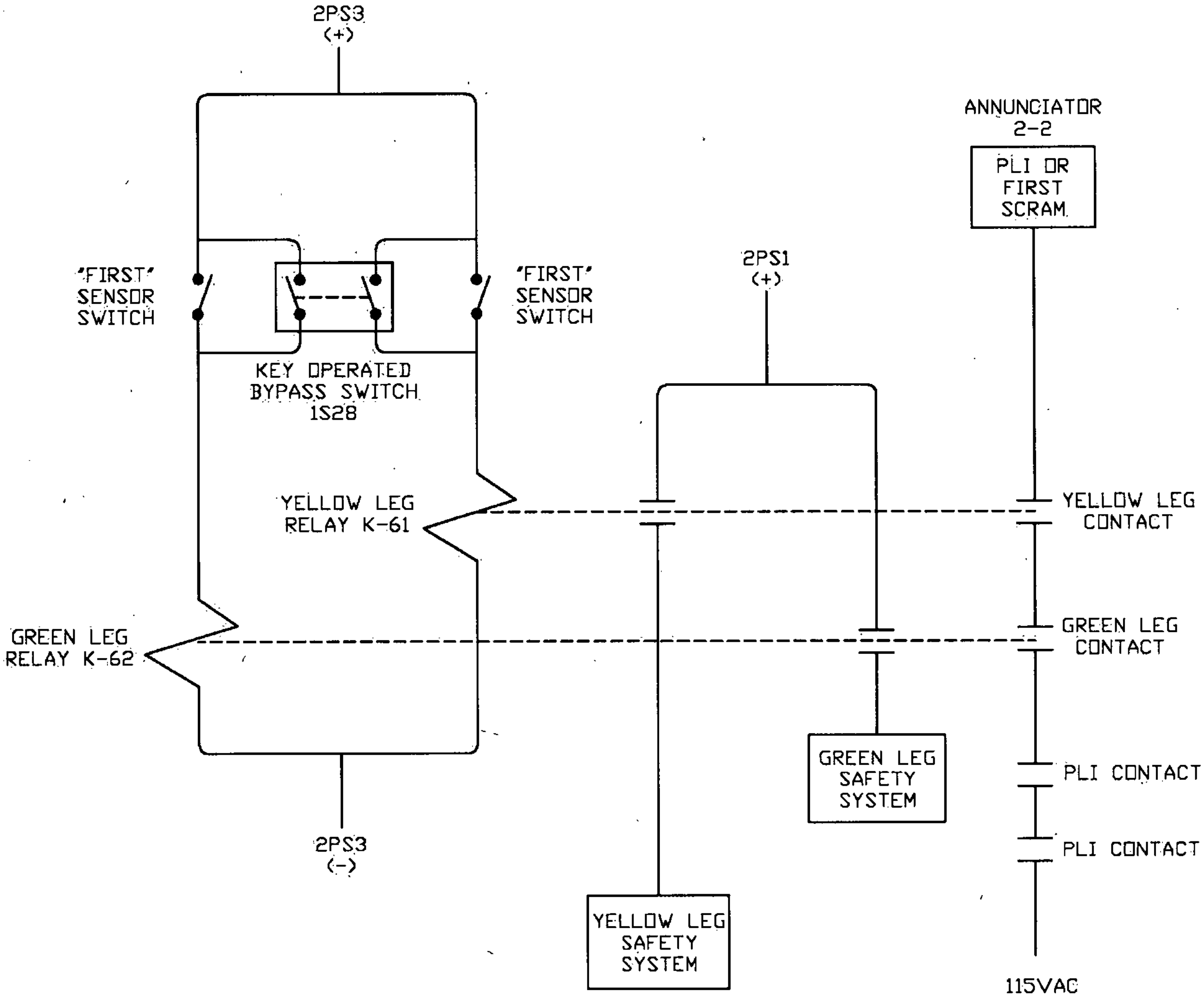
The switches will be adjustable in height, held in place using jam nuts, and be provided protective collars to prevent over travel of the switches. This condition would only occur in the event of improper handling or improper installation of the support rig.

The isolation relays will be of the standardized K-relay type, commonly used in the reactor safety system, which have an excellent service history. The relays, K-61 in the Yellow Leg and K-62 in the Green Leg, will be located in the existing K-relay drawer. This enables them to be accessible, use a standardized shorted relay if needed for testing and troubleshooting, and to provide a configuration consistent with existing reactor safety system components and operation.

The isolation relay coils will be powered via 24V Power Supply 2PS3. This power supply has ample capacity, is already present in the K-relay drawer, and currently powers all existing K-relays in the K-relay drawer. This approach will greatly simplify the installation of the new relays and again provide a configuration consistent with existing safety system components and operation.

The normally open Contact 1 from each relay will be connected into the existing Annunciator logic for the Power Level Interlock Scram. The contacts will be connected in series with the existing annunciator function to provide a scram annunciation in the event that any of the four contacts should open. The existing annunciator window "POWER LEVEL INTERLOCK SCRAM" will be replaced with a new annunciator window "PLI OR FIRST SCRAM." The normally open Contact 2 from each relay will be connected into the existing Yellow and Green Legs as described below.

Two contacts will be added to the existing Yellow Leg and Green Leg series inputs to the reactor safety system. These normally open (meaning contacts will open when the coil is de-energized) contacts (K-61, Contact 2, in the Yellow Leg, and K-62, Contact 2, in the Green Leg) will provide the ability to interrupt the 24 VDC input signal to the reactor safety system. The resulting trip of the respective NCLU will de-energize the TAAs and scram the reactor.



Simplified Diagram of FIRST Wiring Scheme

The FIRST Bypass Switch (1S28) will provide the following two functions: (1) allow testing and troubleshooting capability, and (2) provide the use of both the proposed methodology and the current methodology for calculating the reactivity contribution of samples and the Center Test Hole Canister in the flux trap region.

When it is desired to use the current methodology of including the reactivity contribution of the Center Test Hole Canister, the FIRST Bypass key will be inserted and the switch placed in the "BYPASS" position.

The Bypass Switch will be of the standardized keyed-type, currently used in the reactor safety system with an excellent service history, and labeled 1S28. It will be located on the Reactor Console adjacent to the existing safety system bypass switches. The switch directly jumpers both FIRST sensor switches to energize relay coils on K-61 and K-62. This configuration provides the Reactor Operator a direct and unmistakable indication at the Reactor Console that this reactor safety system function is bypassed.

System Layout and Routing:

a. The Flux Trap Holder wear ring engages the sensor switches.
b. The switches are supported and aligned by the support rig.
c. The support rig is held in place by the lifting ears of the pressure vessel head.
d. Switch leads are routed through the pool, through an access on the south side of the pool bridge, and down to a junction box on the Biological Shield Mezzanine Level.
e. Switch leads will continue on through existing chases to the keyed bypass switch, 1S28, on the Reactor Console.
f. Switch leads will continue on through existing chases to the Instrument Panel area.
g. The switch contacts control the coil of relays K-61 and K-62 in the K-relay drawer.
h. The normally open contacts of these K-relays provide an additional series contact in each of the Yellow and Green Legs of the reactor safety system.
i. An additional normally open contact in the isolation relay provides input to the "PLI OR FIRST" Scram Annunciation.

As addressed previously in Appendix A of Addendum 4 to the Hazards Summary Report, the design of the FIRST device meets all of the applicable criteria of the Institute of Electrical and Electronics Engineers (IEEE) Standard IEEE-279, such as single failure criterion, channel independence, derivation of system inputs, etc. in regard to reactor safety system instrument channel design. The proposed instrument channels are most similar to existing instrument channels that provide a reactor scram in the event that any one of the three primary or pool coolant system isolation valves V507A, V507B, or V509 come off their open seat. Only failure modes that are operationally limiting, and not safety-related, can occur due to the implementation of these instrument channels. Additionally, American National Standard ANSI/ANS-15.15, "Criteria for the Reactor Safety Systems of Research Reactors," was used to verify the appropriate design requirements of the new instrument channels.

Testing

A prototype FIRST device was installed in December 2008. This device was instrumented with the same sensor switches proposed here, but was connected to a digital chart recorder to monitor the operation and reliability of the switches and their support rig. The rig was installed and removed weekly, just as it would need to be done during routine operation. Additionally, these switches operate in a radiation field that is very comparable to the radiation field in the mechanical equipment room, an area that houses a significant number of reactor safety system sensors, such as limit switches, flow transmitters, resistance temperature detectors, etc. Detailed assessments of the switches and its support rig have been performed on several occasions since initial installation. To date, no failure or degradation of switch function or support rig integrity has occurred.

Choosing a suitable and reliable sensor switch was a major focus area for implementing these instrument channels. Extensive bench and in-situ testing was performed to ensure the chosen switch could perform well. Initial testing of the switch was done in a deionized (DI) water bath, where the switch was mechanically cycled numerous times to verify water-tight integrity. Next, repeatability testing was done to ensure the switch travel and contact break point could be consistently achieved. Excellent precision was obtained in this testing. As stated above, full travel of the switch is 3/8-inch (9.53 mm), is limited by the protective collars to be roughly 1/4-inch (6.35 mm), and is expected to be compressed under service conditions to roughly 3/16-inch (4.76 mm). Testing revealed that the contact closed at roughly 1/8-inch (3.18 mm) of compression, and opened at roughly 5/64-inch (1.98 mm) of compression. Testing was extremely consistent for any one switch, and varied between switches by as little as 1/32-inch (0.79 mm). These bench tests plus the in-pool support rig testing support the choice of these switches for their intended service.

The intended precedent for establishing switch height in the rig is to allow 3/16-inch (4.76 mm) travel between fully compressed at the protective collar and 'contact-open.' This will allow roughly 1/8-inch (3.18 mm) of vertical travel prior to initiating a FIRST scram if the Center Test Hole Canister is inadvertently removed. The reactivity affect of this amount of removal prior to initiation of the scram is negligible.

Safety Analysis

The transient analysis for 10-MW operation to determine the maximum safe step reactivity insertion is described in Section 3.5 of Addendum 3 to the Hazards Summary Report. Implementation of the FIRST device does not alter the assumptions or results of this accident analysis but provides a basis for not including the reactivity contribution of the Center Test Hole Canister as part of the limit of TS 3.1.h when the FIRST device is not bypassed. With the FIRST device not bypassed, the position of the Center Test Hole Canister or Strainer is sensed by redundant sensor switches which are required by the reactor safety system to be operational. This is similar to the redundant safety system pressure sensors that sense an intact reactor pressure boundary at operation. The FIRST device eliminates the reactivity administrative controls on the Center Test Hole Canister so that it does not need to be included as a potential reactivity insertion mechanism in any accident scenario associated with the center test hole.

While the probability is highly unlikely, the only scenario that can be constructed in which all of the experiments in the center test hole are either rapidly inserted or extracted is the inadvertent insertion or removal of the Center Test Hole Canister by an Operator while the reactor is in operation. Implementation of the proposed change accomplishes two functions: (1) the reactor can not be started up without the Center Test Hole Canister or Strainer inserted and latched, and (2) removal of the Center Test Hole Canister while at power will cause a reactor scram.

The most likely accident is the failure of a single experiment in the center test hole. The worst case scenario is the sudden bursting of a sample can and the resulting discharge of its contents, with the possible damage to an adjacent sample can. Experiments in the center test hole will continue to be limited by TS 3.1.h such that the failure of any single experiment cannot introduce a reactivity change of greater than 0.006 ΔK.

Attached are the TS pages that will implement the requested change. If there are questions regarding this request, please contact me at (573) 882-5276. I declare under penalty of perjury that the foregoing is true and correct.

	ENDORSEMENT:
Sincerely,	Reviewed and Approved,
Leslie P. Foyto	Ralph A. Butler, P.E.
Reactor Manager	Director

Attachment: Revised Technical Specification 3.3, Pages 3 of 5 and 5 of 5

xc: Reactor Advisory Committee
Reactor Safety Subcommittee
Dr. Robert Duncan, Vice Chancellor for Research
Mr. Craig H. Bassett, U.S. NRC
Mr. Alexander Adams, Jr., U.S. NRC

NOTARY PUBLIC NOTARY SEAL STATE OF MISSOURI

MARGEE P. STOUT
My Commission Expires
March 24, 2012
Montgomery County
Commission #08511436

Margee P Stout

SUBJECT: Reactor Safety System (continued)

Manual Scram	1	1	1	Push Button at Control Console
Center Test Hole	$2^{(6)}$	$2^{(6)}$	$2^{(6)}$	Scram as a result of removing the center test hole removable experiment test tubes

(1) Flow orifice or heat exchanger ΔP (psi) in each operating heat exchanger leg corresponding to the flow value in the table.

(2) Not required below 50 KW operation if natural convection flange and pressure vessel cover are removed or in operation with the reactor subcritical by a margin of at least 0.015 ΔK.

(3) Trip pressure is that which corresponds to the pressurizer pressure indicated in the table with normal primary coolant flow.

(4) Flow orifice ΔP (psi) corresponding to the flow value in the table.

(5) Core ΔP (psi) corresponding to the core flow value in the table.

(6) Not required if reactivity worth of the center test hole removable experiment test tubes and its contents is less than the reactivity limit of specification 3.6.h.

Bases

a. The specifications on high power, primary coolant flow, primary coolant pressure, and reactor inlet temperature provide for the safety system settings outlined in specifications 2.2.a, 2.2.b, and 2.2.c. In Mode I and II operation the core differential temperature is approximately 17°F.

TECHNICAL SPECIFICATION

UNIVERSITY OF MISSOURI
RESEARCH REACTOR FACILITY

SUBJECT: Reactor Safety System (continued)

The scrams from the primary and pool coolant isolation valves (507A/B and 509) leaving their full open position provide a first line of protection for a loss of flow accident in that system initiated by an inadvertent closure of the isolation valve/s.

The power level interlock (PLI) scram provides assurance that the reactor cannot be operated with a power level greater than that authorized for the mode of operation selected on the Mode Selector Switch. The PLI scram also provides the interlocks to assure that the reactor cannot be operated in Mode I with a pool or primary coolant flow scram by-passed.

The facility evacuation and reactor isolation scrams provide assurance that the reactor is shutdown for any condition which initiates or leads to the initiation of an evacuation or isolation.

The manual scram provides assurance that the reactor can be shutdown by the operator if an automatic function fails to initiate a scram or if the operator detects an impending unsafe condition prior to the automatic scram initiation.

The center test hole scram provides assurance that the reactor can not be operated unless the removable experiment test tubes or strainer is inserted and latched in the center test hole. This is required any time the reactivity worth of the center test hole removable experiment test tubes and the contained experiments exceeds the limit of specification 3.1.h. (Ref. Section 3.5 of Add. 3 to HSR). The center test hole scram may be by-passed if the total reactivity worth of the removable experiment test tubes and the contained experiments does not exceed the limit of specification 3.1.h.