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U.S. Nuclear Regulatory CommissionTraining for Research and Training Reactor Inspectors

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***Abstract*—**Currently, a large number of license activities (Early Site Permits, Combined Operating License, reactor certifications, etc.), are pending for review before the United States Nuclear Regulatory Commission (US NRC). Much of the senior staff at the NRC is now committed to these review and licensing actions. To address this additional workload, the NRC has recruited a large number of new Regulatory Staff for dealing with these and other regulatory actions such as the US Fleet of Research and Test Reactors (RTRs). These reactors pose unusual demands on Regulatory Staff since the US Fleet of RTRs, although few (32 Licensed RTRs as of 2010), they represent a broad range of reactor types, operations, and research and training aspects that nuclear reactor power plants (such as the 104 LWRs) do not pose. The NRC must inspect and regulate all these facilities. This paper addresses selected training topics and regulatory activities providedNRC Inspectors for RTRs.

***Keywords*—**Regulations, Research and Test Reactors, Training, US NRC

I. Introduction

THE safety and security of the nuclear reactor operations in any part of the world or in any application is of concern not only to the nation hosting the reactors, but also the entire world nuclear community. It has been justifiably said that a reactor accident or major incident anywhere is of concern and action everywhere. The accidents at TMI II in US (March 1979), Chernobyl (April 1985), and Fukushima (March 2011) are evidence of this concern. Safety and security, or lack of these, have technical, political, and economic consequences for all nations and ubiquitous consequences throughout the world. Principal role of the U.S. Nuclear Regulatory Commission (NRC) is securing safety of the public while licensing and regulating the design, construction and operation of nuclear reactors in the U.S.

II. Instruction Topics

A summary of the training topics presented for NRC Inspectors is provided in the following list. [1]

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1.0 Facilities & Operations at Research & Training Reactors
1.1 Administration and Staffing
1.2 Radiation Protection
- ALARA Program
- Sources of Radiation
- Personnel Radiation Monitoring
- Radiation Surveys
- Experiments
- Radioactive Waste Management

1.3 Instrumentation and Control Systems (ICS)
- Radiation Detection Systems
- Neutron Detection Systems
- Reactor Control Systems

1.4 Material Aging Management
- Aging Mechanisms in Nuclear Reactors
- In-Service Inspections (ISI) for RTRs

2.0 Reactor Physics
- Nuclear Reactions
- Neutron Balance
- Nuclear Cross Section
- Slowing Down of Neutrons

2.1 Multiplication Factor
- Reactivity
- Reactivity Temperature Coefficients

2.2 Control Rods
2.3 Fission Product Poisons
2.4 Reactor Kinetics
- Effect of Delayed Neutrons on Generation Time
- Delayed and Prompt Critical
- Prompt Jump and Prompt Drop
- Excess Reactivity and Shutdown Reactivity
- Reactivity of Experiments
- Control Rod Reactivity Worth
- Transient Rods and Inherent Feedback

2.5 Power Distribution
- Reflector Effects
- Control Rod Effects
- Void, Flux Trap, and Experiment Effects

2.6 Critical Loading and Reactor Startup
2.7 Fuel Storage
2.8 RTR Fueling Options
- Low-Enriched, Low-Concentration U
- High-Enriched, Low Concentration U
- Low-Enriched. High-Concentration U
- Physics of High-Enriched U Systems
- Physics of Mixed Cores

2.9 Physics of Reactor Pulsing - TRIGA
2.10 Physics of Heavy Water Moderated RTRs

III. US Research And Test Reactors (RTRs)

The operating organization and staffing level for Research and Training Reactors (RTRs) differ significantly from commercial nuclear power reactors (NPRs). At RTR facilities (especially at universities) regulated by the US Nuclear Regulatory Commission (NRC) there are significant staffing differences between administration, facilities, and operations depending on mission, financial support and level of utilization of the particular RTR. NPRs have a single focus, namely the safe, economical, and efficient generation of electricity for distribution and sale. The NPR site is organized and given the technical, financial, and personnel resources required to maintain high plant availability.

The NPR is staffed for continuous shift operation and expeditious on-line and corrective maintenance. All plant work is done with careful attention to regulatory requirements and oversight by the onsite NRC Inspector(s) and adequate radiation protection staff to maintain personnel radiation exposure as low as reasonably achievable (ALARA). These NPR personnel along with the support of engineering, procurement, clerical, quality assurance, and other specialist groups form an organization composed of hundreds of personnel with a mission of producing a single product, electrical power for distribution and sale.

In contrast, RTRs are usually a small component within a larger organization with a mission broader and more diverse than sustained reactor operation and production of electrical power. Thus, the mission and activities of the RTR staff differs greatly from that of the NPR. The staff size and expertise together with the mission of a small RTR are inter-related. The mission of the RTR may be limited by its small staff size and the expertise of its personnel. The same facility could have a significantly expanded mission if the staff and resources were available to manage and promote the RTRs use. In contrast to NPRs, sustained operations at RTRs is not usually essential and a forced RTR outage lasting a few days or longer is not the concern it would be for NPRs. A comparison of the major differences between power and RTRs is provided in Table 1.

Currently there are 32 NRC-licensed RTRs that fall into one of three categories:

- 3 RTRs in Private industry: Dow Chemical Company, Aerotest Operations, Inc., and General Electric Company Nuclear Test Reactor
- 3 RTRs at US Federal facilities: US National Institute of Standards & Technology, Armed Forces Radiobiology Research Institute, US Geological Survey, US Department of Interior
- Academia. (26 RTRs)

The Armed Forces Radiobiology Research Institute (AFRRI) studies the biological effects of radiation. The National Institute of Science and Technology (NIST) RTR provides specialized neutron beams for its world class Center for Neutron Research where basic science research and materials studies are performed. The mission of the US Geological Survey (USGS) RTR is primarily to characterize geological specimens to fulfill the USGS overall mission.

Table I
Major Differences between NPRs and RTRs

Nuclear Power Plants (NPRs)	Research and Test Reactors (RTRs)
Operated as heat source for power production	Operated as intense neutron source
Operated continually at full power during entire fuel cycle	Small RTRs: Frequent power cycling. Large RTRs: Continuous Operation
Slow, methodical, infrequent power changes	Frequent start-ups and rapid power changes
Core power shaped for maximum fuel cycle burnup	Power within limits for experimental needs.
Access restricted to small operating staff	Intended to support many students and researchers
Productivity measured by electrical energy generated at high capacity factor	Productivity measured by number of students trained and research supported
Secondary plant facilities affect reactor safety	Reactor operator aware of activities by users
Fission product inventory decay heat may damage fuel	Only large RTRs exhibit decay heat limits
Fuel enriched to 3 to 10%	LEU fuel (3% to 20%). HEU fuel (>20%) replaced by LEU
Shielding integral to design to avoid neutron escape	Neutron beams extracted for irradiations and experiments
Standardization of NPRs to produce electrical power	RTR range in core size, design, experimental facilities
NPR power range from ~1500 to 4000 MW(th)- 2.67range	RTR power range from 5 W to 20 MW- range of 4 million

The 26 US University RTRs have operating organizations ranging from a one or two member operating staff to several dozen full and part time staff. The usual reason for a small staff at an RTR, insufficient to provide significant research services, is that the RTR is also a significant pedagogical tool requiring minimal investment and operating expense.

At the higher end of the academic spectrum of utilization are the university RTRs engaged in basic nuclear research that provide academic and outside users with neutron activation and irradiation analysis, neutron radiography, radionuclide supplies, neutron irradiation services, medical applications, material studies, and basic science research supported by nuclear reactor facilities.

The current NRC Licensed RTRs as of 2010 are shown in Table II. The table provides the Docket and License Number, Reactor Type, Facility, and Licensed power level in kW. The 26 US University RTRs have operating organizations ranging are the university RTRs engaged in basic nuclear research that provide academic and outside users with neutron activation analysis, neutron radiography, radionuclide supplies, neutron irradiation services, medical applications, material studies, and basic science research supported by nuclear reactor facilities.

TABLE II
NRC LICENSED RTRS AT US UNIVERSITIES

#	NRC Docket	NRC License	Reactor Type	Facility U (University)	Power Level (kW)
1	50-184	TR-5	Tank/Plate	National Institute of Standards & Technology	20,000
2	50-186	R-103	Tank	U of Missouri–Columbia	10,000
3	50-20	R-37	Tank	MIT U	5,000
4	50-607	R-130	TRIGA	UC, Davis McClellan Nuclear Research Center	2,300
5	50-193	R-95	Pool/Plate	Rhode Islanded	2,000
6	50-602	R-129	TRIGA	U of TX	1,100
7	50-5	R-2	TRIGA	Pennsylvania State U	1,100
8	50-243	R-160	TRIGA	Oregon State U	1,100
9	50-170	R-84	TRIGA	Armed Forces Radiobiology Research Institute	1,100
10	50-274	R-113	TRIGA	US Geological Survey (USGS)	1,000
11	50-27	R-76	TRIGA	Washington State U	1,000
12	50-156	R-74	TRIGA	U of WN	1,000
13	50-128	R-83	TRIGA	Texas A&M U	1,000
14	50-297	R-120	PULSTAR	NC State U	1,000
15	50-223	R-125	Pool/Plate	U of Massachusetts Lowell	1,000
16	50-150	R-75	Pool/Plate (MTR)	Ohio State U	500
17	50-264	R-108	TRIGA	Dow Chemical	300
18	50-326	R-116	TRIGA	U of California-Irvine	250
19	50-288	R-112	TRIGA	Reed College	250
20	50-228	R-98	TRIGA	Aerotest Operations, Inc.	250
21	50-188	R-88	TRIGA	Kansas State U	250
22	50-166	R-70	TRIGA	U of MD	250
23	50-123	R-79	POOL	U of Missouri - Rolla	200
24	50-113	R-52	TRIGA	U of AZ	110
25	50-407	R-126	TRIGA	U of UT	100
26	50-73	R-33	NTR	GE Nuclear Test Reactor	100
27	50-83	R-56	Argonaut	U of FL	100
28	50-182	R-87	Pool/Plate (MTR)	Purdue U	1
29	50-225	CX-22	Critical Assembly	Rensselaer Polytechnic Institute U	0.1
30	50-59	R-23	AGN-201	Texas A&M U	0.005
31	50-284	R-110	AGN-201	Idaho State U	0.005
32	50-252	R-102	AGN-201	U of NM	0.005

IV. RADIATION PROTECTION

Operations at RTRs inherently carry the potential for personnel exposures to elevated levels of ionizing radiation in excess of national background levels. The US National Council on Radiation Protection and Measurements (NCRP) in Report 160 reported in 2006 that the average radiation exposure of US residents is now 6.3 mSv. Significantly, 48% of this US average exposure is now associated with medical applications of radiation and less that 0.1% is associated with all industrial uses including all nuclear reactors.

ALARA, which is a result of this conservative regulatory philosophy, is defined as the policy of making reasonable efforts to maintain exposures to ionizing radiation as far below regulatory dose limits as is practical. The standard operation of ionizing radiation facilities and activities at near regulatory limits is an unacceptable practice. ALARA is thus a philosophy of practice associated with applications associated with ionizing radiation. It is not a numerically defined regulatory limit nor is it a mandated series of procedures that must be implemented without exception. ALARA is also a practical policy in that it recognizes that radiation and radioactive materials provide beneficial products and services for humans including medicine, science, energy, industry and other benefits. The basis of ALARA also assumes that any level of exposure to radiation must be assumed to have some associated level of risk to human health. This implies that for any activity resulting in human exposures to radiation, it is necessary to ensure that:

- Benefit of activity greater than potential harm from radiation
- Risk level from occupational exposures doesn't exceed risks acceptable in other occupations with high safety standards
- Public risk less than or equal to other normal risks accepted by society.

A. ALARA Applied to RTRs

ALARA programs at RTRs are established and documented in accordance with Title 10 of the US Code of Federal Regulations, Part 20.1101 (10CFR20.1101) [2]. The typical goal for most RTR facilities is to limit radiation levels in unrestricted areas to about 10% of 10CFR20.1301(a)(1)[3] using training, shielding and operational procedures with periodic review of these activities. Records documenting ALARA activities, personnel exposures, and reviews are standard items for NRC inspections.

Of course the original source of most of the radiation sources found at a RTR are those associated with the nuclear fission process and the succeeding radiation and radioactive materials produced as a result of the fission process in the RTR core. This cascading process of nuclear and radioactive reactions results in a complex chain of reactions generating a broad spectrum of electromagnetic radiation (EM) spanning the energy spectrum from visible (Cherenkov) UV to MeV gamma rays

Besides the radiation emanating from the RTR during operation and even during shutdown, there are many other potential sources of radiation at RTRs such as fixed point sources, airborne, liquid, and solid sources. In general, radiation sources found at typical research and test reactors can be classified into the following general classes:

- Calibration and check sources
- Startup, and other sources used for instrumentation and nuclear support functions

- Airborne, liquid, and solid radiation sources from operations
- Radiation sources produced in experimental facilities
- Fission products as applicable

Liquid radioactive material is not routinely produced or used in normal operations of RTRs, with the exception of neutron activation of impurities in the primary coolant. A filter(s) and demineralizer resins remove the majority of these impurities. Other radioactive waste can be generated from decontamination, maintenance, or laboratory activities. Radionuclides and their concentrations in the environment of the RTR depend on reactor power, reactor operating time and time since reactor shutdown. Typical sources of these radiation sources are shown in Table III.

TABLE III
TYPICAL RTR SOURCES OF RADIOACTIVITY AT RTRS

Major Radiation Sources Associated with RTR Operations		
Airborne	**Liquid**	**Solid**
Ar^{41}, N^{16}, H^3, C^{14}	H^3, Ag^{110m}, Cu^{64}, Cu^{66} (soluble irradiated nuclides)	Co^{60}, Fe^{55}, Fe^{59}, Zn^{65}, Na^{24} (irradiation assemblies)

B. Reactor Fuel

Unirradiated fuel at RTRs poses very low external radiation exposure to workers. The principal issues regarding new fuel are criticality issues and safe and secure storage. The fuel classified as Special Nuclear Material (SNM) is controlled under the operating license of the RTR and must be handled, stored, and properly documented to satisfy license and NRC requirements. Documented inventories of all SNM are required on a prescribed basis.

A typical irradiated TRIGA fuel element in a 1 MW TRIGA Reactor has a radiation field greater than 1Gy/hr in air at 1 meter when the element is removed from the reactor tank. Radiation dose rates from handling these elements are the primary concern for radiation protection of personnel. Thus, spent fuel element transfers involving irradiated fuel are performed by reviewed and written procedures and with adequate shielding to meet appropriate radiation limits. Typically, such fuel management operations produce a High Radiation Area and a fuel element transfer cask is employed to reduce personnel exposures.

An important, potential source of radiation and radioactivity in sources at RTRs, other than irradiated fuel, is the release of fission products from the irradiated reactor fuel. Those detected radionuclides through chemical analyses and detectors used in helium sweep systems of the primary coolant system include radioactive gases of xenon, krypton, and Cs-138 (a daughter product of Xe-138. Radioactive isotopes of iodine (e.g., I-131) are of particular concern. However, using the typical makeup rate for the helium system at NIST, for example, less than 3.7 GBq of these radionuclides are released annually. These release concentrations are low (less than 0.01Bq/liter) and thus represent a negligible contribution to the total gaseous emissions.

V. EXPERIMENTAL RADIATION SOURCES

A. Neutron Beams

The majority of research at a research and test reactor use neutrons of various energies to study material constituents, processes, and structures. These neutrons are often extracted from the reactor core via neutron beam tubes (i.e., channels along which neutrons can travel to the experiment). Neutron beams at an RTR typically range from a few square mm to 200 square cm. Beams with an in-beam dose rate in excess of 1 mSv/hr and are accessible (have an open path in excess of 30 cm) are designated as High Radiation Areas. A characteristic of well-designed neutron beam tubes is that the radiation field outside of the beam is usually low, less than 0.05 mSv/hr. However, experimental samples and equipment at the beam stop can result in Radiation Area or even High Radiation Area conditions. These areas must be controlled as required by 10CFR20 Sections1601 and 1902 [4]. Non-beam related and short-term experiments are also shielded and controlled to keep personnel exposures ALARA.

B. Pneumatic Systems and In-Core Exposure Facilities

Experiments and their configurations utilizing pneumatic and in-core facilities are highly variable, frequently producing multi-curie activity sources. All elements of the activity, facility usage, experiment management, disposal, and potential personnel exposures are addressed by technical review and administrative authorization processes. Typically, holding the irradiated experiments in a shielded configuration to allow sufficient decay prior to direct manipulation, processing, or analysis is a primary ALARA policy used in these situations.

Both Ar-41 and N-16 are produced in the section of the pneumatic transfer system that is located in the reactor core. During operation of the pneumatic transfer system, air containing very small amounts of these two radioisotopes is exhausted from the system through a HEPA filter to the facility stack. Experience at RTRs has shown that even after repetitive operations of this system, there have been no detectable increases in the release of these two radioisotopes. Therefore, the Ar-41 and N-16 from the pneumatic transfer system is not generally considered to be a measurable contributor to the radioisotopes released or exposure rates associated with reactor operations. However, operating records must demonstrate that this is true.

C. Radiation Surveys

The main purpose of the radiation survey program is to assure radiological surveillance over selected reactor facility work areas in order to provide information and trending characteristics and assessment of the existing ALARA program. Data of this type is used to confirm that safe radiation working conditions exist within the various operational areas under surveillance and to reduce personnel exposures where possible.

The first objective of the radiation survey program is to assure that the monitoring program is organized such that routine radiation level and contamination level surveys of specific designated areas and activities within the facility are performed. Also special radiation surveys are performed as necessary to support non- routine facility operations.

A second objective of the program is to make frequent on-the-spot personal observations (including recorded data) of radiation work areas. These observations may provide advance warning of needed corrections in order to ensure safe use and handling of radiation sources and materials.

A third objective is to use the information that has been gathered through completion of the first two objectives in order to ensure (and document) that all phases of the operational and radiation protection programs are consistent with the goal of keeping radiation doses to personnel and releases of radioactivity to the environment ALARA.

D. Operations Radiation Levels

Depending on the class/research workload, a typical RTR operates for only one shift per day or less (40 hours per week). Facilities such as the University of Missouri RTR and NIST will typically operate a 24-hour shift schedule. An occupationally exposed individual only spends a fraction of the time in areas where there is a potential for measurable radiation levels. Radiation surveys of a reactor facility are usually performed within the restricted area during full-power operations to ascertain an exposure rates for personnel working in the vicinity. Typical values of radiation levels at various locations at a typical 1 MW TRIGA are provided in Table 4. Taking into consideration the limited occupancy times, the relatively low dose rates observed, and typical personnel doses received by the reactor staff, it is common that occupational doses can be maintained below the regulatory limits given in 10 CFR 20.

E. Authorization and Conditions for Experiments

Administrative requirements exist at RTRs to assure that all experiments are performed in a manner that will ensure the protection of the public. Experiment review meets the requirements of Regulatory Guide 2.2 [5}, and Standard ANSI N401-1974 (ANS-15.6) [6]as modified by Regulatory Guide 2.4 [7]. The two Regulatory Guides identifies the considerations that should be addressed to define limits and other requirements are included in the technical specifications for the RTR.

TABLE IV
TYPICAL RTR RADIATION LEVELS RTRS AT 1 MW

Facility Location	Typical Dose Rate Equivalent on Contact (mSv/hr)	Typical Dose Rate Equivalent @ 30 cm (mSv/hr)
Reactor Pool Surface	1 (typically N-16)	0.65 (typically N-16)
ReactorBay Floor	1	1
Demineralizer Tank	25	1
Primary Water Pipes	10	2
Primary Water Filter	3	<1

Considerations and safety analyses for experiments should address:

(1) Interaction of an experiment with the reactor system that has the potential for breaching any primary barrier for fission product release from fuel, interaction of an experiment with the reactor system that has the potential for breaching any primary barrier for fission product release from fuel
(2) Interaction of an experiment with the reactor system that has the potential for breaching any primary barrier for fission product release from fuel, interaction of an experiment with the reactor system that has the potential for breaching any primary barrier for fission product release from fuel
(3) Any interaction of experiment with reactor system that could adversely affect engineered safety features or control system features designed to protect public from fission product release
(4) Any inherent feature of an experiment that could create beams, radiation fields, or unconfined radioactive materials
(5) Potentially adverse interaction with concurrent experimental and operational activities.
(6) Reactor Control Circuits

F.RTR Control Rods

Control rods are used to adjust flux within the core. Often, it is desirable to adjust the height of the individual control rods. Most RTRs contain have a rod position indication device connected (most often chain driven) to the rod drive motor. The rod position indication device generates a signal proportional to the distance traveled by the control rod.

Most RTRs have two classes of reactor control rods (i.e., thermal neutron absorption rods), safety rods and regulating rods. Safety rods provide safe shutdown capability for the reactor and generally have a large negative reactivity capacity (several $). Safety rods are connected to the rod positioning equipment via an electromagnet. Upon receipt of a scram signal, a current amplifier supplying current to the electromagnet, de-energizes (i.e., fails safe), causing the safety rod(s) to drop into the core and shutting down the reactor.

Regulating rods (usually with significantly lower reactivity worth) are used to accurately adjust and control reactor power. A third class of control rods (referred to as a transient rod) is associated with those TRIGA reactors licensed for pulsing operations. The transient rod is designed to move very rapidly (“fires”), exiting the core and producing a large positive reactivity increase within the reactor core (reactivity > 1$).For pulsing operations, often the regulating rod is directly connected to the positioning motor to prevent the regulating rod from scramming. Safety rods on the other hand are required to scram if an unsafe condition develops during the pulsing operation. The transient rods also retain their scram capability. In the late 1980s instrumentation designers introduced reactor control systems that convert the analog information from radiation detectors into digital signals that can be readily processed by digital controllers and computers. Many RTRs have installed such digital based systems to replace aging analog systems. In general, digital based systems are more accurate, reliable and exhibit improved output for operational control of the reactor.The important

physical and nuclear properties and signals to be measured and controlled in a RTR include the following as shown in Table 4

TABLE V
MEASUREMENT PARAMETERS IN RTRS

Neutrons (e.g., counts and counts/s)
Neutron flux (e.g., nts/cm^2-s)
Gammas (e.g., counts/s, mSv/hr, Gy)
Charged particles (alphas, betas, fission fragments in counts/s)
Fission events (e.g., kW(thermal)/cc)
Temperature (e.g., ^oC, ^oF)
Pressures (e.g., psi, kPa, atmospheres)
Fluid flow of air and water (e.g., cfm, gpm)
Component movement and position (e.g., control rods, etc.)

Reactor startup and operating ranges for the power levels for RTRs cover a range from about 0.1 mW to 100 MW.A fission chamber is used for reactor startup when the neutron signal is very low and the large amplification produced in the fission chamber.

VI. RTR FUELING OPTIONS

In contrast with Light Water Reactors (LWR), there is wide variation in the physics and engineering principles underlying the designs and operating characteristics of research reactors. The reason is that different RTR designs and uses emphasize different reactor operational objectives. Among the objectives, some RTRs maximize in-core thermal neutron fluxes and external neutron beam physics, while others are concerned with building a fool-proof safe training facility. In the latter category, an organization concerned with building one unit for its own use probably emphasizes high-neutron economy and versatility. A group primarily concerned with becoming a training reactor supplier and vendor would emphasize safety features, competitive costs, standardization, and unique patentable design features.

A. Low-Enriched, Low-Concentration Uranium

Most university research and training reactors (RTRs) can be classified as low enriched (less than 20% U-235) and low concentration uranium The majority of these RTRs are hydrogen moderated, usually water, however the AGN reactors are moderated by polystyrene, CH_2 (or H_2C), very similar to H_2O in hydrogen density. These reactors are essentially "undermoderated" so that reduction in moderator from temperature increase or boiling will provide a negative temperature and void coefficient.

A prime example of a RTR supplier is the TRIGA Reactor line designed and marketed worldwide by Gulf General Atomic (GA) in San Diego, CA. The most notable and unique feature of this reactor is the special reactor fuel used in TRIGAs. In 1956, a team of distinguished physicists at General Atomics, including Frederic de Hoffmann, Freeman Dyson, and Ted Taylor, developed a novel reactor fuel that exhibited such a large, prompt negative reactivity temperature coefficient, that the TRIGA fuel could experience a prompt critical condition without fuel failure. The following description will not trace the long historical basis for developing and testing this "new fuel" (see GA-471, Technical Foundations of TRIGA, August 1958), but describes briefly the nucleonics and thermal-hydraulics of the U-ZrHx-fueled reactors. As shown below, TRIGA fuel rods incorporate a high concentration of chemically combined hydrogen and zirconium, so these rods are actually fuel-moderator rods.

The traditional standard TRIGA fuel element contains uranium enriched to 20% U-235 with about 8.5% weight as uranium uniformly mixed in the zirconium hydride matrix. There is nearly the same density of hydrogen in the ZrH1.6 as in the H_2O, so for most of the neutron slowing down energies the fission neutrons are moderated essentially as if the moderator were pure water (the order of 70% to 80% volume fraction. However, as the neutron energies approach the binding energy of the hydrogen in the zirconium lattice (about 1 eV), the apparent hydrogen mass is similar to that of zirconium, so any further thermalization must occur primarily in the water surrounding the TRIGA fuel element. The bound hydrogen atoms in the ZrH vibrate in place and can even transfer some kinetic energy to a colliding neutron. When this occurs, the average neutron energy in the fuel rod is increased above the average neutron energy in the surrounding water, and these faster neutrons have a greater chance of leaking out of the fuel rod without causing fission. When the total fission rate increases and raises the fuel temperature, there is increased vibration of the bound hydrogen and a larger fraction of the thermalized neutrons will gain energy through collisions. The energy of the average neutron in the fuel rod will thus be higher, with a higher probability of not causing fission (because the fission cross section is inversely proportional to velocity) and thus having higher probability of leaking into the surrounding water. The net effect of this rather complex process is a decrease in the thermal utilization f as the temperature rises in the uranium-zirconium hydride. Because the uranium is intimately mixed with the ZrH, the heat transfer from the uranium and temperature rise of the ZrH is instantaneous, so the result is a highly useful, prompt negative temperature coefficient of reactivity that dominates the other temperature dependent reactivity terms, as discussed below.

Because there is a high ratio of U-238 to U-235 in the standard TRIGA fuel (about 4 U-238 atoms per U-235 atom), the Doppler broadening of U-238 resonance absorption (that directly affects the resonance escape probability p) also produces a prompt negative temperature coefficient of reactivity that acts for epithermal neutrons independently of the thermal utilization effect described previously. These two negative reactivity mechanisms (f and p) are additive and together contribute about 80% of the total negative temperature coefficient of low-enriched TRIGA fuels. The remainder of the reactivity temperature coefficient is due to increased neutron leakage from the core at higher moderator temperatures.

This prompt temperature coefficient is inherent in the TRIGA fuel and highly stabilizes reactor operation around the steady-state mode of operation. This coefficient has such a large negative reactivity effect upon some RTRs that it is the basis for allowing TRIGA reactors to safely and routinely operate in the pulse mode routinely with reactivity insertions greater than one dollar. Of course power reactors (e.g., LWRs) do not have such TRIGA fuel properties, viz., the resultant

large, prompt, negative temperature coefficient, and should never operate in a pulsed mode.

Finally, in summary, there are three components of the prompt negative temperature coefficient for TRIGA fuel that all lead to reduction in the effective multiplication factor in the reactor core.

- As the temperature of Zr-H in the fuel increases (this change is immediate with the fuel temperature) neutrons in the fuel leave the fuel and enter the surrounding water with higher energy.
- As the fuel temperature increases, absorption resonances (principally in U-238) Doppler broaden and the resonance escape probability decreases, decreasing the effective multiplication factor.
- The increase in the coolant moderator (water) increases the mean free path for neutrons increasing core leakage.

B. High-Enriched, Low Concentration Uranium

Gulf General Atomic (GA) has also developed a TRIGA fuel using high-enriched uranium (HEU) (70% U-235) but still containing about 8.5 wt% total uranium in the zirconium hydride. This was designated as Fuel Lifetime Improvement Program (FLIP) fuel. The objective was to increase the U-235 loading, thereby increasing the fuel lifetime and neutron flux intensity and reducing core size. To maintain an approximately constant reactivity over the extended life, a burnable poison, erbium, was added to the fuel. Erbium was chosen because of its strong absorption resonance for neutrons of about 0.5 eV to maintain a large prompt negative temperature coefficient of reactivity. With a resonance this close to thermal neutron energies, some of the neutrons that have been slowed down in the water to below the resonance energy are scattered back up at high fuel temperature. This results in another chance for neutron absorption by erbium.

The net effect is that the thermal utilization of slow neutrons decreases as the fuel temperature increases. Because these neutron processes all occur within the fuel rods, there is again a large prompt negative temperature coefficient of reactivity. As before, the higher the fuel temperature, the larger the fraction of neutrons scattered back up to higher energy within the fuel rod. As the erbium is depleted however, the magnitude of this temperature coefficient decreases. In the FLIP fuel, there is much less U-238 to contribute to the Doppler-effect component of the temperature coefficient because of the high-uranium enrichment. Careful design has produced a FLIP uranium-zirconium hydride fuel with a prompt negative temperature coefficient whose magnitude is near that for the standard TRIGA fuel but is based partly on a different mechanism that changes slowly with fuel burn up.

C. Low-Enriched, High-Concentration Uranium

Because of "Reactor Safeguard concerns" over the use of highly enriched HEU fuel in research reactors, GA has recently developed an additional modification of the FLIP concept. This design returns to the 20% U-235 enrichment. But to obtain the desired lifetime longer than the original TRIGA fuel, the loading of uranium is increased from 8.5% to 20 or 30 wt% uranium. As with the FLIP fuel, the reactivity is initially suppressed and maintained approximately constant during fuel use by incorporating the same burnable poison, erbium. The loading of uranium and erbium are chosen so that the temperature coefficient of reactivity is still large, negative, and prompt, with about 20% to 25% of the reactivity effect due to the Doppler absorption broadening in the U-238 resonances. This newer fuel may become a TRIGA fuel standard in the future and has essentially the same inherent safety features as the original TRIGA fuel, but now satisfies Reactor Safeguard considerations.For both the high-enriched uranium (HEU) FLIP fuel and the high-loaded low-enriched uranium (LEU) fuel, as the burnable poison depletes and the magnitude of the temperature coefficient decreases, the magnitude of reactivity change that can be safely compensated tends to decrease. This is a slow process, but the licensee must be aware of this change and adjust operating procedures accordingly. One licensing consequence is that Technical Specifications that authorize pulsing operation in a TRIGA reactor must limit fuel temperature or energy excursion per pulse, rather than excess reactivity for the new LEU fuel.

C. Physics of High-Enriched Uranium Systems

A principal objective of research reactors is to obtain high neutron fluxes with the least impact from dissipating the high heat generated. It is apparent that the lower the concentration of U-235, the higher the neutron flux must be for the same power density. But a critical mass of U-235 must be maintained to make the reactor operable. It can be shown that the minimum critical mass is obtained when the thermal utilization is at a maximum for the system and occurs for minimum non fission absorptions of thermal neutrons. This implies minimizing the nonproductive neutron absorbers that inherently accompany the U-235. Because the U-238 is a major neutron absorber, this further implies using the highest enrichment of U-235 possible. When research reactors were first designed in the 1940s and early 1950s, designers favored the use of HEU. Safeguard concerns were not as evident then as they are today. HEU was generally available from the former Atomic Energy Commission (AEC) and was preferred by most designers of research reactors.

The development of HEU reactor cores follow closely with the general concepts of criticality considered earlier. However, because a reduced fraction of uranium fuel is U-238 in HEU, there is reduced absorption of neutrons in resonances and the negative reactivity (Doppler absorption) effect on heating of the fuel is reduced. The first research reactor designed to use enriched uranium was a liquid homogeneous reactor designed and operated at Los Alamos. The first licensed nongovernment-owned research reactor in the US was designed and built at North-Carolina State University based on the design of this Los Alamos reactor.

Another major design criterion was to build research reactors to operate at the highest power densities feasible. This led to the thin-plate design of HEU metal dispersed in an aluminum complex. This type of fuel was developed for the Materials Test Reactor (MTR) reactor in Idaho and later adapted for submarine nuclear power reactors for the US Navy. After the MTR had operated successfully for about a year, adoption of this fuel for general purpose research reactors also became the choice for future RTRs. However, the thin-plate HEU fuel had no significant prompt negative

temperature coefficient of reactivity to counteract rapid increases in reactivity, so an experimental program was initiated at Idaho to investigate excursion-limiting and possible shutdown mechanisms. This program, first called Borax and then SPERT, both located at Idaho, confirmed the absence of any significant prompt inherent mechanism for HEU plate fuel and showed that expansion and expulsion of coolant/moderator that was relatively slow was the principal reactor shutdown mechanism. These experiments became the bases of safety considerations for HEU plate-type research reactors.

D. Physics of Mixed Cores

The fuel and poison loadings of TRIGA fuel rods have been chosen so that they all have about the same net effect on reactivity. This means that the new highly loaded 20% enriched uranium rods can replace either 20% enriched lightly loaded or the 70% enriched lightly loaded rods without a significant change in reactivity conditions.

However, since the power density in a fuel rod is related to the U-235 concentration, a single high-fueled rod located within an array of low-fueled rods could cause higher power densities and much higher temperatures in that rod than in its low-fueled neighbors. Because of this concern, careful analyses must precede and accompany any plan to make rod-by-rod changes from one fixed uranium loading to a different one. If the entire core is changed from one uranium enriched loading to a significantly different one, the problem does not arise because the power density distribution will now be approximately uniform over the core. Significantly, from a safety standpoint, this problem is not unique to TRIGA fuel but could occur for any fuel type for which a wide range of uranium enrichment loadings is available for core insertion.

VII. RTR PULSING OPERATIONS

In addition to TRIGA uranium-zirconium hydride fuel, another reactor fuel has been developed for research reactors that also have a large prompt, negative temperature coefficient. This reactor fuel, very similar to light-water power reactor fuel, is composed of sintered uranium oxide pellets enclosed in a sealed cladding tube. This fuel has been named PULSTAR because its intended use was in a dual-purpose steady-state and pulsing research reactor. Like the TRIGA fuel, the PULSTAR fuel has a large negative temperature coefficient that is contained within the fuel itself and so acts promptly and independently of the moderator. Such a mechanism acts to decrease reactivity if the fuel temperature rises and reduces the rate of fission activity. If the magnitude of the coefficient is large enough, the reactivity effect can compensate for an excess reactivity that is larger than one dollar, so the reactor might be operated in the pulsing mode without fuel damage.

For the PULSTAR reactor, the only significant temperature-dependent neutron absorption mechanism is Doppler broadening of the U-238 resonances. To attain a sufficiently large negative temperature coefficient, it is necessary to increase the ratio of U-238 to U-235 by a factor of about three or four above that for the 20% enriched TRIGA fuel. Thus, PULSTAR fuels that are currently in use consist of U-235 enrichments of only 4% and 6%, respectively.

These fuels provide a prompt negative temperature coefficient of reactivity comparable to TRIGA fuels. Not only are the PULSTAR reactors very stable in steady-state operation, but they also are designed for safe pulsing. Both of the NRC Licensed PULSTAR reactors have operated safely and successfully in the pulsed mode, but it has not been found to be an important feature to retain, so this provision has been deleted from PULSTAR reactor licenses. The process for pulsing a reactor requires a neutron absorbing control rod that can be ejected rapidly, inserting a step increase of reactivity in the reactor greater than one dollar ($\rho\$ > \$1$). This puts the reactor on a prompt period (i.e., operating on prompt neutrons alone), and the power level increases exponentially with time with a reactor period associated with prompt neutrons (i.e., several milliseconds) only. As the power level rapidly increases, the fuel temperature also increases until at some elevated fuel temperature the prompt negative temperature coefficient of reactivity reverses the excess reactivity and terminates the reactor pulse. This inherent event returns the reactor back to critical state, but at a high-power level. The continuing generation of power raises the temperature of the fuel higher and the reactor becomes subcritical and power level decreases to final equilibrium level. If the rate of heat transfer from the fuel is adequate and fuel does not fail, the power pulse produces approximately the same temperature change in the fuel during the power decrease as during the prompt power increase. Both the TRIGA fuels and the PULSTAR fuels are designed with these characteristics. These fuels can be safely pulsed and are essentially invulnerable to damage from inadvertent or accidental insertions of reasonable values of excess reactivity. GA has demonstrated non damaging pulses in TRIGA fuel with reactivity insertions up to \$5.0, resulting in power peaks up to 7140 MW, integrated energy production of 46 MW-sec (MJ) per pulse, and peak temperatures up to about 1150 ^{o}C.

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REFERENCES

[1] O. Research and Test Reactor Technology Course, HRDT Course R-106B, U.S. Nuclear Regulatory Commission, dated 2010, Washington D.C.
[2] Title 10 of the US Code of Federal Regulations, Part 20.1101 (10CFR20.1101).
[3] Title 10 of the US Code of Federal Regulations, Part 20.1301(a)(1) (10CFR20.1301(a)(1).
[4] Title 10 of the US Code of Federal Regulations, Part 20.1601 and 1902 (10CFR20 Sections 1601 and 1902).
[5] US Nuclear Regulatory Guide 2.2. (See reference 7)
[6] Standard ANSI N401-1974 (ANS-15.6) (See reference 7).
[7] US Nuclear Regulatory Guide 2.4