SURVEY OF NON-POWER REACTORS

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

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CONTENTS

TABLES

APPENDICES

ABBREVIATIONS

AEA	Atomic Energy Act
AEOD	Analysis and Evaluation of Operational Data, Office for (NRC)
ALARA	as low as is reasonably achievable
LOCA	loss-of-coolant accident
MTR	Materials Testing Reactor
NOTRTR	National Organization of Test, Research, and Training Reactors
NRC	U.S. Nuclear Regulatory Commission
SAR	Safety Analysis Report
SRO	senior reactor operator

EXECUTIVE SUMMARY

This study was initiated by the Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data[1] in response to occurrences at non-power reactors in 1993 involving loss of multiple scram functions and reactor protection and control problems. The purposes of the survey were to make an independent assessment of non-power reactor safety performance, and to provide appropriate recommendations or suggestions based on the study's observations and findings. The survey included review of operating events, inspection findings, National Organization of Test, Research, and Training Reactors feedback to the non-power reactor community, feedback of lessons learned, U.S. Nuclear Regulatory Commission enforcement actions, and visits to representative facilities. The survey was focused on the "higher power" (typically 2 MW or greater) Materials Testing Reactor-fueled facilities because of their relatively higher fission product inventories and TRIGA-fueled reactors because of the number of similar facilities. This particular subset of non-power reactors was chosen because it was recognized that the higher power reactors dominated the potential offsite risk associated with non-power reactors, especially in comparison to the very "low power" facilities ranging from 1W to 100 kW. One to two day visits were made to research reactors at the Armed Forces Radiobiology Research Institute, Georgia Institute of Technology Neely Nuclear Research Center, General Atomics, Massachusetts Institute of Technology Nuclear Reactor Laboratory, Oregon State University Radiation Center and TRIGA Reactor, University of Arizona, University of Illinois Nuclear Reactor Laboratory, University of Missouri Research Center (Columbia), University of Texas at Austin Nuclear Engineering Teaching Laboratory, and the test reactor at the National Institute of Standards and Technology, to understand unique reactor features, research programs, and operational performance. Individual facility design and safety analyses were reviewed as necessary to ensure understanding of the safety significance of the operating experience and the importance of root causes, corrective actions, and lessons learned. The study did not include an evaluation of the design bases of the facilities or their major design features.

During the course of performing the non-power reactor survey, many facilities promptly addressed emerging issues or lessons learned that are contained in this report. Ongoing findings of the survey were also communicated as a result of NRC presentations at several annual meetings held by the non-power reactor community. Operating experience indicates that non-power reactors have a good overall safety performance. None of the presently operating non-power reactors have had recent incidents affecting public health and safety. None of the recent operating events resulted in fuel damage or radiation releases in excess of regulatory limits. Except for one event in 1986, personnel exposures have been within 10 CFR 20 limits. No adverse effect on public health and safety was seen. Non-power reactors have been operating for many years without causing harm to the public. There is considerable inherent safety in the

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[1] Effective March 28, 1999, the Office for Analysis and Evaluation of Operational Data (AEOD) was disbanded. The work described in this report was initiated by AEOD and completed by the Regulatory Effectiveness Assessment and Human Factors Branch of the NRC's Office of Nuclear Regulatory Research.

non-power reactor community of reactors. The adequate protection of public health and safety requirement has been satisfied by all currently operating non-power reactors.

The assessments and results contained in this report support the four NRC performance goals for the nuclear reactor safety arena. These performance goals are to maintain safety, protection of the environment, and the common defense and security; increase public confidence; make NRC activities and decisions more effective, efficient, and realistic; and reduce unnecessary burden on stakeholders. Given continued appropriate regulatory attention and focusing on improved operational practices, the NRC performance goal to maintain non-power reactor safety can be achieved. The survey did not identify actions to support the performance goal to reduce unnecessary burden on stakeholders, but rather found that in accordance with the provisions of the Atomic Energy Act, the NRC has imposed for non-power reactors "... only such minimum amount of regulation of the licensee as the Commission finds will permit the Commission to fulfill its obligations under this Act to promote the common defense and security and to protect the health and safety of the public and will permit the conduct of widespread research and development." Although the survey activities did not lend themselves to any direct conclusions on the other performance goals to increase public confidence and make NRC activities and decisions more effective, efficient, and realistic; these strategic goals may be realized through sustained safety and appropriate regulation as observed by the survey.

Although large differences exist in the form and content of the Safety Analysis Reports, these reflect (1) the evolution that has occurred in licensing since 1956, (2) the large variation in non-power reactor design and function, and (3) the absence of a standard form and content for hazard analyses until relatively recently. However, the NRC issued NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," in 1996 to suggest a uniform format and to help ensure the completeness of information provided. For the non-power reactors in operation at this time, postulated reactivity insertion events were not uniformly considered, even by reactor design type. However, fundamental facility design changes or standardization are not warranted in view of the composite body of safety analyses from similar reactors, together with experimental information described in this survey.

Specific recommendations for changes to ensure or enhance continued safety of NRC licensed non-power reactors were not supported. Survey observations in Section 4 suggest some potential areas for improving operational or safety performance.

1 Introduction

1.1 Background

This study was initiated in response to occurrences at non-power reactors in 1993 involving loss of multiple scram functions and reactor protection and control problems. The purposes of the survey were to make an independent assessment of non-power reactor safety performance, and to provide appropriate recommendations or suggestions based on the study's observations and findings. The survey included review of operating events, inspection findings, National Organization of Test, Research, and Training Reactors (NOTRTR) feedback to the non-power reactor community, feedback of lessons learned, NRC enforcement actions, and visits to representative facilities. The survey was focused on the "higher power" (typically 2 MW or greater) Materials Testing Reactor (MTR)-fueled facilities because of their relatively higher fission product inventories and TRIGA-fueled reactors because of the number of similar facilities. This particular subset of non-power reactors was chosen because it was recognized that the higher power reactors dominated the potential offsite risk associated with non-power reactors, especially in comparison to the very "low power" facilities ranging from 1W to 100 kW. One to two day visits were made to research reactors at the Armed Forces Radiobiology Research Institute, Georgia Institute of Technology Neely Nuclear Research Center, General Atomics, Massachusetts Institute of Technology Nuclear Reactor Laboratory, Oregon State University Radiation Center and TRIGA Reactor, University of Arizona, University of Illinois Nuclear Reactor Laboratory, University of Missouri Research Center (Columbia), University of Texas at Austin Nuclear Engineering Teaching Laboratory, and the test reactor at the National Institute of Standards and Technology, to understand unique reactor features, research programs, and operational performance.

The main focus of the report is on the relationship between the operating events (or conditions) and the accident analysis appropriate to the reactor experiencing the event. Therefore, salient portions of the accident analyses were extracted and captured. During facility visits by AEOD, site tours were undertaken; interviews were conducted with reactor operators, reactor management, and radiation protection management; operator logs and safety evaluations were reviewed; and operating events were discussed.

A draft report written by a panel of senior NRC personnel using the non-power reactor operating experience information gathered by AEOD staff was issued for peer review in December 1996. The draft report was subsequently technically reviewed by many members of the non-power reactor community including offices within the NRC, with comments being forwarded to AEOD for resolution. During the survey, several meetings were held with the NOTRTR to discuss preliminary results, and corrective actions taken in response to the non-power reactor survey visits by AEOD. The final survey report is the product of peer review resolution and editing by Office of Nuclear Regulatory Research staff. This final report replaces the draft report.

Section 2 of this report is organized according to classes of hazards: (1) reactor protection and control related (some events involved degradation of the reactor protection system),

(2) loss-of-coolant accident (LOCA)/loss of flow event, (3) inlet flow blockage, and (4) "other." Section 2 provides a brief summary of the safety analysis practices for these categories.

Section 3 is divided into seven categories each with representative operating experience and a discussion of safety significance. Referenced operating experience summaries are included in Appendix A.

1.2 Relative Risk of Non-Power Reactors

Although power reactors and non-power reactors are not comparable, some understanding of their design, risk, and regulatory differences is necessary for some readers to understand, and place into context, the operational risk or hazard. The risk of power reactors (on the order of thousands of megawatts thermal) has been subjected to exhaustive characterization (e.g., NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants"). Non-power reactors by contrast, are much lower in power (most are less than two megawatt thermal) and generally do not run continuously for months at a time. These and other operating characteristics contribute to a much lower fission product inventory. All reactors (power and non-power) are licensed to operate as utilization facilities under Title 10 and in accordance with the Atomic Energy Act (AEA) of 1954, as amended[2]. The AEA stated that utilization facilities for research and development should be regulated to the minimum extent consistent with protecting the health and safety of the public. The NRC has promulgated these concepts in 10 CFR 50.40, 50.41, and in other parts of Title 10 that deal with non-power reactors.

The licensed thermal power levels of non-power reactors are several orders of magnitude lower than power reactors. In addition, when considering the generally intermittent operation of non-power reactors, this results in a significantly smaller inventory of fission products in the fuel. These factors have been some of the bases upon which the NRC has determined that less stringent and less prescriptive measures are required to adequately protect the safety of the public, workers, and the environment.

Other non-power reactor attributes also contribute to lower risk. For example, MTR-type fueled reactors have a fuel cladding melt temperature on the order of 649 °C (1200 °F), meaning that the less volatile elements (e.g., Lanthanum) will likely stay in the solid form during an accident. A fundamental feature of TRIGA reactors is their inherent safety that results from the large prompt negative fuel temperature coefficient of reactivity of their fuel-moderator system. The lower power reactors also have less tendency to melt following a LOCA or core uncovery. In many non-power reactors, the decay heat is insufficient to cause cladding damage under any

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[2] The AEA was written to promote the development and use of atomic energy for peaceful purposes and to control and limit its radiological hazards to the public. Congressional intentions and authority for regulation of non-power reactors are expressed in Sections 103 and 104 of the AEA.

cooling condition. Overall, the public risk associated with non-power reactors is much less than that of power reactors.

Because of their inherent lower risk, non-power reactors are required to meet different regulatory requirements than power reactors. Non-power reactors are generally not required to be housed in robust containments; have smaller exclusion radii; and, in some instances have less equipment redundancy and diversity. In addition, non-power reactors have, consistent with their reduced risk, less formal and rigorous operator training programs than do power reactors. Also, since design, function, and potential hazards vary widely among non-power reactors, the regulations have been implemented in different ways at various non-power reactors. All these requirements are consistent with the facility design and safety analyses which provide acceptable assurance that the public health and safety are protected.

The staff reviewed the ways that non-power reactors can be damaged. Based on operating experience and specific developmental testing accomplished during the design phase of early reactors, it is well-known that step or ramp injections of reactivity for certain reactor designs and power levels must be limited to prevent significant core damage. Also, the control of potential flow blockage conditions has been long recognized as an important lesson learned from several instances of partial core melting at early experimental or test reactors, usually limited to parts of a single bundle, caused by a coolant inlet flow blockage. Some brief descriptions of such events or experiments are provided in Appendix B. It is recognized that extreme events at early experimental or test reactors have only very limited implications for the normal operation of current designs used for non-power reactors since this experience has been considered in the development and approval of current designs.

2 Review of Selected Safety Analyses

The issue of what standards to use in evaluating accidents at a research reactor was discussed in an Atomic Safety and Licensing Appeal Board decision issued on May 18, 1972, for the research reactor at Columbia University in New York City. The Atomic Safety and Licensing Appeal Board stated that "as a general proposition, the Appeal Board does not consider it desirable to use the standards of 10 CFR 20 for evaluating the effects of a postulated accident in a research reactor inasmuch as they are unduly restrictive for that purpose. The Appeal Board strongly recommends that specific standards for the evaluation of an accident situation in a research reactor be formulated." The staff did not find it necessary to conform to that recommendation to develop separate criteria for the evaluation of research reactor accidents, since the majority of research reactors have been able to adopt the conservative 10 CFR 20 criteria (Ref. 1).

The staff reviewed portions of safety analysis sections for seven non-power reactors that had MTR-type fuel elements and operated at 2 MW or higher. Summary comments on these safety analyses follow.

2.1 Reactivity Related Analyses

The Georgia Institute of Technology reactor considered a sudden reactivity insertion of 1.5 percent with the reactor critical, and asserted it was a highly improbable event (Refs. 2 and 3). A postulated fuel loading event might produce as much as 2.5 percent reactivity addition, which would not produce fuel melting. An arbitrary sequence involving fuel melting and vessel rupture would yield about 135 MW-seconds of energy. The maximum internal containment pressure was estimated to be 115.9 kPa (2.11 psig) compared with a design pressure of 114 kPa (2 psig) with a safety factor of 3. This scenario, however, was considered incredible.

The University of Michigan reactor considered an instantaneous 1.6 percent reactivity addition, which would take the fuel to 482 °C (900 °F) (Ref. 4), while the melting point of the cladding is above 649 °C (1200 °F). The NRC staff was unable to identify a credible method for such a rapid reactivity insertion. The licensee had also reviewed data from the BORAX tests, which indicated that an instantaneous 1.8 percent reactivity addition would melt the meat of the hottest fuel element if the ambient coolant temperature was as low as 21.1 °C (70 °F). However, the minimum coolant temperature for the University of Michigan reactor is approximately 32.2 °C (90 °F), and, as previously noted, a credible mechanism for even an instantaneous 1.6 percent reactivity addition had not been identified.

The Massachusetts Institute of Technology reactor analyzed a spectrum of reactivity insertion accidents, including the uncontrolled withdrawal of the regulating rod and the step reactivity insertion resulting from the instantaneous failure of the highest worth experiment allowed in the reactor (Refs. 5 and 6). These events involve a maximum reactivity insertion of 1.8 percent. The analyses were based on correlation with data obtained at the SPERT facilities, and indicated the resulting fuel temperatures would be well below clad melting temperature.

At the National Institute of Standards and Technology reactor, the licensee analyzed the startup accident that might result from control rod withdrawal (Refs 7, 8, and 9). The shims were postulated to be withdrawn steadily until the reactor was scrammed by a high power trip. The initial power level was at 1E-4 MW, the reactivity insertion rate was 5E-4 Δk/second, and the high power trip was set at 30 MW (150 percent of full power). The energy of the excursion was 4.8 MJ, and the peak power was 43.3 MW. The energy required to (adiabatically) bring only the metal of the fuel element structure within the core to the melting point was calculated to be 34 MJ, a factor of seven times that calculated for the startup accident. Therefore, no core damage should result. Dropping a fuel element into a critical core was not considered credible. Broken shim arms, beam tube collapse, and the cold water accident were all considered and shown to be less severe than the startup accident.

For the National Institute of Standards and Technology reactor, the maximum ramp reactivity insertion rate is 2.6 percent Δk/second. The ramp insertion was assumed to take place while the reactor was at full power (20 MW). The reactor was tripped at 30 MW. With maximum ramp insertion, the energy excursion was 7.1 MJ, well below the 34 MJ required to melt the fuel element structure.

For the Rhode Island Nuclear Science Center Research Reactor (Refs. 10 and 11), the startup accident, as analyzed, was the most limiting credible reactivity insertion accident for the low enriched uranium core. The two most limiting accidents for this type of reactor were analyzed. The maximum reactivity insertion was assumed with the reactor in cold and clean conditions, reactor power at the source level, and the regulating rod withdrawn. The maximum reactivity insertion was the sequential withdrawal of all safety blades at the maximum rate. It was assumed that the period scram failed, and a delay of 0.5 seconds occurred before the safety blades were free to drop. The reactor would trip on the high neutron flux scram (120 percent of full power). The analysis resulted in a maximum fuel temperature of 88.1 °C (191 °F). The licensee later presented a more conservative analysis with the added assumption that the reactor did not trip on the high neutron flux scram. However, this analysis did not continue beyond the point of nucleate boiling. Although the results are correct for most operational configurations of the facility, it is not necessarily correct for the most conservative conditions.

The licensee for the University of Missouri at Columbia Research Reactor analyzed various reactivity transients to determine the maximum reactivity addition that the reactor could withstand without core damage. The analysis shows that with a reactor scram a positive step of 0.006 $\Delta k/k$ can be added to the reactor without exceeding burnout (which is predicted to occur at 25.23 MW) (Ref. 12).

For the low enriched uranium core (in operation since February 1994), the University of Virginia reactor considered a short period transient due to the addition of reactivity (Ref. 13). The analysis showed a peak power of 3.88 MW that is below the safety limit for the true value of total coolant flow of 3168 liters/minute (837 gpm). There was no credible nuclear excursion possible with the University of Virginia reactor that could exceed the safety limits for the fuel. According to the Safety Analysis Report (SAR), there was reasonable assurance that fission product activity would not be released from the fuel to the environment as a result of a reactivity insertion event.

2.2 Loss-of-Coolant Accident and Loss of Flow Analyses

The Georgia Institute of Technology considered a LOCA which, absent operation of the emergency core cooling system, would result in some fuel melting (Refs. 2 and 3). Given a containment leak rate of 0.5 percent/day, 10 CFR 100 doses would be realized near the containment. The analysis was conservative in that non-typical assumptions were made, such as assuming a prolonged operating history and the break location. For loss of flow, it was assumed that the reactor would scram on a low flow rate, and that there would be no fuel failure.

The University of Michigan considered a core uncovery following a leak (caused by the failure of a pneumatic tube) of 794.9 liters/minute (210 gpm) with no makeup (Ref. 4). The core would uncover in about 3 hours; but because decay heat would be low, core damage would not occur.

Loss of flow was not a concern at Massachusetts Institute of Technology as pool boiling was not predicted to start within one hour, which allowed time for remedial actions to be taken. Additionally, no pool drainage event which would lead to a LOCA was identified as a concern (Refs. 5 and 6) as the pool water level can only be lowered 1.22 meters (4 feet), leaving the core covered with at least 1.83 meters (6 feet) of water. Core cooling would be by natural convection.

Calculations at the University of Missouri at Columbia indicated that a 20 percent flow reduction was required to produce core boiling (Ref. 12). No releases ensued. A LOCA of the pool system would not affect the reactor because the reactor cooling system is a separate closed system independent of the pool system. The licensee analyzed a double-ended rupture of the largest core cooling pipe at the worst location between either isolation valve and the pool liner. The drop in pressure will cause the reactor to scram, the primary coolant pumps to stop, and the core isolation valves to close. The core will remain covered and decay heat will be transferred to the reactor pool without core damage.

The University of Virginia considered a core uncovery from a double-ended guillotine break in the outlet coolant pipe, which uncovered the core in about 20 minutes (Ref. 13). Two independent core spray systems were assumed not functioning. Core melt was not indicated. For a postulated LOCA in which the reactor did not scram on low pool level (equivalent to an instantaneous LOCA) and only one core spray system functioning, the core was not predicted to melt.

2.3 Inlet Flow Blockage Analyses

The Georgia Institute of Technology did not analyze inlet flow blockage; presumably, the consequences of a LOCA would bound this event (Refs. 2 and 3).

The Massachusetts Institute of Technology postulated the blockage of five channels, with four melted fuel plates (Refs. 5 and 6). Experience at the Oak Ridge Research Reactor shows that fuel melting does not propagate beyond the affected flow channels. For the flow blockage analyses, iodine would be reduced by absorption in the water (factor of 10), and about .075 percent of the iodine inventory would be available for release to the environment, at a leak rate of 1 percent per day. Thyroid and whole body doses would be well within 10 CFR 100 requirements.

The National Institute of Standards and Technology analysis considered an object blocking all flow through a single fuel element leading to the failure of a fuel element (its design basis accident) when the reactor was at 20 MW (Refs. 7, 8, and 9). The analysis had conservatisms. The internal confinement charcoal filter recirculation system was assumed to be inoperable for the accident analysis. Charcoal filters were assumed to be 95 percent efficient in the removal of iodine, but are required to be 99 percent efficient by technical specifications. The highest power element (730 kW) was assumed to melt. It was assumed that 100 percent of the blocked element's cladding would melt and release fission products that would escape from the primary water into the confinement building. It was assumed that 100 percent of noble gases and

50 percent of iodines escape into the confinement building. Calculated doses were well within regulatory limits.

2.4 Other Accident Analyses

The Massachusetts Institute of Technology analysis also considered external events, such as seismic, and decided that the facility would not be damaged for credible events (Refs. 5 and 6).

For the University of Missouri at Columbia, in July 1966, the AEC staff postulated (Ref. 14) a total core meltdown following a prolonged period of operation at 10 MW and the radiological consequences were evaluated. It was assumed that 100 percent of the nobles and 25 percent of the halogens would remain airborne and be available to leak from the containment which is initially at 115 kPa (2 psig). Under atmospheric inversion conditions the whole body dose was less than 0.2 Seivert (20 rad) and the thyroid dose was 2.8 Seivert (280 rad) at the 152.4 m (500 ft) exclusion radius for the first two hours. These accident scenarios are just within 10 CFR 100 limits. The AEC staff stated (Ref. 12) that "...the improbable nature of this type of accident and the conservative manner in which we have calculated the consequences lead us to the conclusion that in any real event, the dose would be at least one order of magnitude lower."

The University of Virginia postulated failures in fueled experiments (Ref. 14). Nominal doses resulted for both occupational and non-occupational personnel.

2.5 Summary

Although large differences exist in the form and content of SARs, these reflect (1) the evolution which has occurred in licensing since 1956, (2) the large variation in non-power reactor design and function, and (3) the absence of a standard form and content for hazard analyses until relatively recently.   Indeed, some facilities were licensed before the definitive experiments of SPERT, or the SL-1 event, or some of the noteworthy inlet flow blockage events. However, all the lessons learned from these events are currently considered in non-power reactor design and operation. Further, the NRC issued NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," in 1996 to suggest a uniform format and to help ensure the completeness of information provided. For the non-power reactors in operation at this time, the postulated reactivity insertion events were not uniformly considered, even by reactor type. However, fundamental facility design changes or standardization are not warranted in view of the composite body of safety analyses for similar reactors, together with experimental information described in Appendix B (Selected Core Damage Events at Early Experimental or Test Reactors). Further, non-power reactors have been recently relicensed, and most will relicense again before 2002.

3 Review of Selected Operating Events

Operating experience for higher powered MTR-type and TRIGA test reactors was examined, and the experience mapped into the seven areas discussed in Sections 3.1 through 3.7. Some operating events mapped into more than one area and so may be discussed under more than one area. For example, an event involving inoperable conditions of the reactor protection system might map into both the reactivity area and the attention to detail area.

Table 1 provides the number of the most significant non-power reactor events involving reactor protection and control, coolant leaks, and fuel-handling events (designated as rpc, cl, and fh respectively) at MTR-fueled reactors ≥ 2 MW. These types of events have occurred primarily at non-power reactors whose power is greater than 2 MW.

Table 1 **Reactor Protection and Control, Coolant Leaks, and Fuel-Handling Events at MTR-Fueled Reactors ≥ 2 MW**

Facility	90	91	92	93	94	95	96	Total
Georgia Institute of Technology					1rpc			1
Massachusetts Institute of Technology	1rpc			1rpc		4rpc		6
National Institute of Standards and Technology	3rpc	1fh		1rpc				5
Rhode Island						1cl		1
University of Michigan			1rpc 1fh	2rpc				4
University of Missouri at Columbia			1rpc	2rpc	1rpc	2rpc	1rpc	7
University of Virginia				2rpc	1cl	1cl	1cl	5
Total	4	1	3	8	3	8	2	29

3.1 Reactor Protection and Control Related Events

The consequences of recent operating events and occurrences involving reactor protection and control did not result in fuel damage or radiation release because of non-power reactor design characteristics, operating conditions, and licensee responses. Several reactor protection and control related events due to instrumentation failures, control rod problems, or human performance deficiencies are included below.

There have been numerous incidences affecting reactor protection and control at the larger MTR-type research reactors during the past several years. At the Massachusetts Institute of

Technology Nuclear Reactor Laboratory, the reactor was operated for 28 minutes in the power range with only one operable power safety level channel while two were required by the technical specifications. Subsequent examination of the power supplies for the period and power safety level channels found that several were improperly connected to the low voltage protection system (March 1989, event A1). Other alarmed power safety level channels were functional. The reactor was operated at low power with two inoperable flux channels (March 1995, event A2). It also was operated with a power tilt with one control rod in the core (July 1995, event A3). The core was taken critical above its estimated critical position with one control rod slipping as it was withdrawn (August 1995, event A4).

National Institute of Standards and Technology experienced shim rods not fully inserting on manual scram (three instances in 1990, events A5, 6, and 7) that appear to have been corrected by either bearing or relay replacement.

An unlatched fuel element caused power fluctuations at the National Institute of Standards and Technology reactor (September 1993, event A8). When the power fluctuations occurred, power was reduced to 1 MW, but the fluctuations continued, leading to a manual reactor shutdown.

The University of Missouri at Columbia had seven failures of the regulating blade between 1988 and 1996 (events A9, 10, 11, 12, 13, 14, and 15). The primary root cause was component failure for five of the events, an inadequate procedure for one event, and inadequate corrective action for one event. The regulating blade was used to automatically control reactor power, but was not part of the reactor safety system defined in technical specifications. The 0.0017 Δk worth of this blade was not considered in any safety analysis as contributing to the reactor shutdown margin of 0.02 Δk with any one shim blade fully withdrawn. Thus, this blade provides an additional safety factor beyond what is required for reactor shutdown. When a reactor scram or rod run-in occurred, the regulating blade automatically shifted to manual control to prevent it from trying to maintain power by shimming to ensure termination of a transient. An inoperable regulating blade mechanism also made the technical specification rod run-in associated with the regulating blade inoperable. While not affecting the ability to automatically shut down the reactor, the facility was manually shutdown or power was reduced to investigate each event. The nonsafety-related automatic control system functions were lost or degraded.

At the University of Michigan, operations staff removed a fuel element from the core while the core was at low power (8 kW) (June 1992, event A16). The NRC dispatched an Augmented Inspection Team to investigate the event. Two violations were issued (collectively characterized as a Severity Level III problem); one for violating the technical specifications which required having the reactor subcritical during fuel movements and another for violating the procedure to have all rods fully inserted during fuel movements which resulted in a civil penalty. A Severity Level IV violation was also issued for failure to report the event to the NRC within 24 hours.

A minor event at the University of Michigan involving a failed shim-safety rod control switch required the reactor to be shut down by using the manual scram (February 1993, event A17). A more significant event (March 1993, event A18) occurred one month later when the reactor exceeded its licensed power by 15 percent, operating at 2.3 MW, with a high power scram set above its technical specifications limit for 11 minutes. An NRC Special Safety Inspection found that the root causes of this second event were a lack of management oversight of operations and poor communications between the shift supervisor and the console operator, deficient operator knowledge regarding the power level instrumentation, operator failure to review or use the startup procedure, an unclear procedural step, and ineffectiveness of previous corrective actions.

The University of Virginia operated for 5.5 hours with both power-level scrams, the intermediate-range scram, the low primary coolant flow scram, the loss of power to pump scram, the range switch scram, and the key switch scram (all technical specification-required protective scrams) inoperable because of errors during maintenance (April 1993, event A19). The manual scram remained available.

An examination of these events indicates no inadequacies in the basic designs of the systems. Until the Oregon State University event (February 1998, event A20), there have been no known failures to scram on demand. This event was outside the survey's time period. In the Oregon State event, the reactor operator quickly reacted to the situation illustrating the importance that licensed operators can play in minimizing the consequences of events. Some of these events show the benefits of redundancy and diversity in the design of safety systems. However, the number of events could have been reduced through more extensive maintenance, greater procedural compliance, enhanced corrective action implementation, and more rigorous post-maintenance or post-modification testing. These refinements would help to ensure that equipment is operable and reduce the occurrence of future equipment failures.

3.2 Loss of Coolant and Loss of Flow Events

Several primary system leaks have occurred over the past several years, none of which was beyond the capability of the makeup system. These had little radiological consequences because of the usual low concentration of radionuclides in the water.

Relatively minor pool leakages have occurred at the Rhode Island Atomic Energy Commission Nuclear Science Center (July 1995, event A21) and at the University of Virginia (January 1994, event A22, and November 1995, event A23). The minor pool leakage at the University of Virginia continued in 1996 and 1997. The leak was occurring around the gate that divides the pool into two parts, and was approximately 151.4 liters/day (40 gallons/day). There have also been small primary to secondary leaks in the heat exchanger at the University of Virginia (August through October 1995, event A24). At any point in time, none of these leaks exceeded approximately 454.2 liters/day (120 gallons/day). Two safety concerns were considered. First, for research reactors at or above 2 MW in power, if the loss of coolant led to core uncovery, fuel damage could result. Second, the water draw-down will also lead to increased radiation levels above the core as the depth of the water shield is reduced. However, these pools contain tens of thousands of gallons of water. Draining at the rates experienced (or even many times those rates) should be easily detectable and correctable.

The more likely way in which fuel may be damaged in reactors of this type appears to be from loss of core cooling due to inlet flow blockage or from loss of forced flow when operating at high power. The event at the University of Virginia (March 1996, event A25) in which a paper towel on top of a fuel element led to 200 kW power fluctuations (i.e., 10 percent of rated power) is illustrative of the potential effect of minor blockage. Core inlet blockages from things as simple as paper towels or gaskets can lead to damaged fuel. The area around the reactor pool was cleared of any materials that could inadvertently get into the pool, and a checklist was developed to identify anything that could get into the pool. The significance of inlet flow blockage is reflected in several of the accident analyses submitted by these licensees.

During an AEOD site visit to the University of Missouri at Columbia facility in 1994, members of the refueling crew were observed to have materials in their shirt pockets while moving fuel, creating the potential to drop foreign material on top of the core. The licensee took immediate corrective actions to address this observation.

The loss of coolant events reported by licensees were the result of aging of facilities as opposed to licensee error. Many licensees have established water inventory programs that allow for the quick identification of pool or heat exchanger leakage along with response procedures. Also, licensees now analyze and the NRC staff reviews these types of events during the license renewal process so that if a licensee has a pool or heat exchanger leak, they are analyzed events. Although flow blockage events have been rare, continued vigilance by licensees to prevent the introduction of material into the reactor pool that could block coolant flow is warranted.

3.3 Fuel Handling Events

Very few events have occurred involving fuel handling.

At the University of Michigan, operations staff removed a fuel element from the core while the core was at low power (8 kW) (June 1992, event A16). The NRC dispatched an Augmented Inspection Team to investigate this event. The Augmented Inspection Team concluded that this event had no safety consequences, however, two violations were issued (collectively characterized as a Severity Level III problem); one for violating technical specifications for the reactor to be subcritical during fuel movements and another for violating the procedure to have all rods fully inserted during fuel movement. A Severity Level IV violation was issued for failure to report the event to the NRC within 24 hours. The NRC reported this event to Congress as an Abnormal Occurrence.

At the National Institute of Standards and Technology (June 1991, event A26), a new fuel element dropped off the handling tool and landed on the top grid of the core.

Because of the potential to quickly add a large amount of reactivity to the reactor core, the NRC considers fuel handling equipment and the potential fuel handling problems as a high priority issue. Fuel handling issues are considered in the SERs for all licensed non-power reactors. Also, selected procedures are reviewed and activities are observed for potential fuel handling problems by NRC inspectors.

3.4 Radiation Protection Events

The observations indicated that sample handling and fuel movement were the most probable sources of personnel exposure and contamination. Other than one event at the University of Missouri at Columbia (June 1986, event A27), radiation protection practices have been sufficient to prevent exceeding regulatory dose limits. Several past notices of violation (Table 2) were considered to determine the significance of radiation protection incidents. Sample handling and shipping practices were also reviewed. The examples of weak sample handling and shipping practices primarily involved the failure to identify and list the amounts of isotopes contained in shipping packages and to adhere to or have appropriate procedures which could prevent such occurrences. In addition to the review of past violations, the results of site visit survey observations and experiences were reviewed.

As low as is reasonably achievable (ALARA) practices were very good at most facilities. However, there have been some instances of inattention to detail towards radiation safety. During an AEOD site visit in 1994, it was discovered that the University of Missouri at Columbia facility allowed the use of normal street clothes during refueling despite several personnel contamination events during the prior year.

At Ohio State University, one researcher received an 0.089 Seivert (8.9 rem) dose to his fingers (March 1996, event A28). In another event, two operators at the Massachusetts Institute of Technology Nuclear Reactor Laboratory received 0.092 Seivert and 0.115 Seivert (9.2 and 11.5 rem) to their hands (June 1994, event A29) in an experiment, and left the area without surveying themselves thereby subsequently contaminating the control room. During an AEOD site visit in 1994 to the University of Missouri at Columbia, it was learned that radioactive particles of gold were found on the pants of two operators during the previous year.

3.5 Design Basis Control

Design basis control and continuing safety analyses of events and modifications are part of the process that ensures that appropriate attention is given to equipment and processes so that safety margins used in the facility’s SAR are not degraded. The low safety significance of the events within this study does not support the need to revise the regulations concerning the maintenance of a current SAR. The relative simplicities of non-power reactor SARs and the few number of significant changes that are made to non-power facilities obviate the need for a prescribed periodic update of SARs. SARs for non-power reactors are not required to be updated on a periodic basis as is the case with power reactors (10 CFR 50.71[e]). Instead, the requirement for non-power reactors is to submit an SAR as part of the initial license application and to submit revisions to the SAR in conjunction with any license renewal.

Table 2 Selected Health Physics Violations

Facility	Description
University of Missouri at Columbia	Several violations were collectively categorized as a Severity Level II problem with a $4000 civil penalty (EA86-191, 1987). The violations (occurring in 1985 and 1986) resulted from the licensee failing to adequately assess the hazards of radiation exposure associated with the handling of thulium-170 pellets. The oversight led to an unplanned extremity overexposure of 1.15 Seivert (115 rem) to an individual's hands.
Georgia Institute of Technology	Several violations were collectively categorized as a Severity Level III problem with a $5000 civil penalty (EA88-32, 1988). The violations (occurring in 1987) were issued for failure to follow or have approved procedures during topaz irradiation resulting in a contamination event.
University of Virginia	Several violations were collectively categorized as a Severity Level III problem with a $1250 civil penalty (EA87-155, 1987). The violations (occurring in 1987) were issued for failure to perform surveys necessary to identify a high radiation area and to take appropriate action to provide written procedures for the installation, operation, modification, and surveillance of experimental facilities, resulting in an individual's exposure of up to 2.7 mSeivert (270 mrem), which had the potential for more significant exposure.
Texas A & M	Several violations were collectively categorized as a Severity Level III with a $5000 civil penalty (EA88-92, 1988). The violations (occurring in 1987) were issued for failure to provide dosimetry to personnel and to establish proper controls and failure to use and wear personnel monitoring equipment in high radiation areas. ALARA principles were not employed when working in high radiation areas.
University of Missouri at Rolla	A Severity Level IV violation (occurring in 1987) for bypassing a frisker station when leaving the reactor bay area. Several individuals occasionally ate, drank, or smoked in the bay area.
University of Texas at Austin	A Severity Level V violation (1990 inspection) for failure to have available records to document the results of radiation surveys performed to determine dose rates .
Massachusetts Institute of Technology	A Severity Level IV violation (for activities occurring between 1988 and 1991) for (1) failure to conduct quarterly inspections of radiation safety activities from 1988 to 1991, (2) not conducting or recording refresher training for health physics technicians, and (3) failing to record radionuclides on a shipment burial manifest for a December 1990 waste shipment.
University of Maryland	A Severity Level IV and one Severity Level V violations (1992 inspection) for failure to perform adequate radiation surveys necessary to determine that individuals were not exposed to airborne concentrations in excess of 10 CFR 20.102. In addition, the licensee was cited for failure to maintain radiation survey records used to determine postings for neutron beam port experiments.
University of Michigan	A Severity Level IV violation (1993 inspection) for failure to follow health physics procedures in accordance with their technical specifications to use calibration source standards traceable to National Institute of Standards and Technology in their pool water analysis since 1982.
Georgia Institute of Technology	A Severity level IV violation (1994 inspection) was issued for failure to make a proper evaluation of the extent of neutron radiation present following a survey underestimating the dose rate by a factor of 100.

The requirement that licensees submit an annual report to the agency summarizing all changes made to the facility under the allowances of 10 CFR 50.59 assures that the information needed by the NRC to fulfill its statutory obligations will be available. However, licensees are required to maintain the information necessary to properly and safely operate the facility, such as data, procedures, and reference documents needed to conduct safety evaluations, submit licensing requests, and provide proper and effective training, and are encouraged to maintain current SARs on file with the NRC. Non-power reactor licensees are required to review all facility changes and modifications for their potential to result in unreviewed safety questions. Changes which do not result in such conditions can be implemented by the licensee under the allowances of 10 CFR 50.59. A report which summarizes the changes made under the allowances of 10 CFR 50.59 is required to be submitted annually (Ref. 15).

In some instances, a lack of design control contributed to the occurrence and consequences of events. Selected event summary information is provided below:

At the University of Michigan during a routine maintenance period (November 1992, event A30), the shim range-control rod interlock system was removed from the reactor control system for a modification that had been reviewed and approved by the facility Safety Review Committee. The senior reactor operator misinterpreted system response during the subsequent post modification testing. This error resulted in a failure to identify that the power level deviation interlock was inoperable due to a wiring error.

The University of Missouri at Columbia had a series of seven operational failures from 1988 to 1996 (events A9, 10, 11, 12, 13, 14, and 15) of the regulating blade that required the reactor to be shutdown or reduced in power. The primary root cause was component failure for five of the events, an inadequate procedure for one event, and inadequate corrective action for one event. The functioning of this nonsafety system component could have been improved at least in the instances of inadequate procedure and inadequate corrective action.

During maintenance activities at the University of Virginia (April 1993, event A19), two mixer-driver modules were changed in the scram logic drawer. Although the two new mixer-driver modules appeared to be identical, they had been altered internally by facility staff, and were not identical. This resulted in the inoperability of both power-level scrams, the intermediate-range scram, the low primary coolant flow scram, the loss of power to the primary pump scrams, the range switch scram and the key switch scram for a period of 5.5 hours. All of these scram functions were required by the technical specifications. The component changes were made during troubleshooting activities that were being conducted to determine the cause of a spurious reactor shutdown earlier in the day. A system verification was not performed after the maintenance due to an inappropriate decision by the shift supervisor. The loss of scram function(s) was not self-revealing, and would have been discovered only by a proper post-modification/post maintenance test such as a scram system check. A Severity Level II violation and a $2000 civil penalty were assessed for this event by the NRC staff.

A shim rod failed to fully insert into the TRIGA reactor at the United States Geological Survey (January 1996, event A31) during a normal shutdown. Shim rod 2 had been replaced during a routine control rod inspection conducted in December 1995. Prior to the installation of the new shim rod, the fuel-follower control rod had been measured and found to be about 2 inches shorter than the rod it was replacing. Calculations at the time of installation indicated that the new rod was of sufficient length to remain below the top of the bottom grid plate when retracted to its uppermost configuration. In fact, the rod did not remain below the top of the grid plate, but instead, "caught" on the plate's edge, thereby hanging up several inches above the fully inserted position. The licensee did not perform a 10 CFR 50.59 evaluation of this modification.

Non-power reactor licensees in general have performed acceptable evaluations of facility changes. However, the events at the University of Virginia, (April 1993, event A19) and the United States Geological Survey, (January 1996, event A31) indicate that non-power reactor licensee evaluations are not always thorough or documented, though these instances have been infrequent. Non-power reactor licensees are not relieved from the requirements of 10 CFR 50.59. However, these two events do not, in and of themselves, support an overhaul of the 10 CFR 50.59 process. They do suggest that the NRC may need to continue to emphasize the importance of effective and comprehensive licensee evaluations.

3.6 Operating Experience Feedback

The "NOTRTR Newsletter" highlights NRC inspection emphasis and problems identified, and calls attention to some important events so that others can learn more by contacting the facility directly. Many informal communications occurred between licensees. In addition, industry conferences, generally held on an annual basis, addressed selected operating experience. These processes disseminated information on what was considered the most important events. While this survey was underway, a NOTRTR computerized bulletin board server was established which provides a more rapid dissemination of information on some events and conditions. Other forums for operating experience reviews included peer reviews by NOTRTR, audits of facilities by nearby licensees, and reviews of NRC generic communications.

Despite the significant design differences between non-power reactors, facilities can benefit from each other’s experiences, whether they are positive or negative. Consequently, a greater event distribution and discussion of event root causes and resultant corrective actions could be of use. At least one of the events discussed in this report which had which had reactive occurrences could have been precluded by a more detailed and insightful root cause analysis of the initial (or subsequent) event. This survey found differences of opinion in the non-power reactor community toward systematically sharing operating experience. While some facility personnel expressed a desire to increase the sharing of operating experience, a few appeared reluctant to share their operating experience with others. The vehicles are in place to effectively learn from operating experience; however, they could be used more. A more public, expanded exchange of operating experience has the potential to further improve operational or safety performance.

It should be recognized that events which do not violate the technical specifications are not reportable, and given that non-power reactor technical specifications were not standardized, similar events occurring at different facilities may or may not be required to be reported. In keeping with the Atomic Energy Act requirement for minimal regulation of non-power reactors, the purpose of reporting events in non-power reactors is to inform the NRC of safety significant events quickly. Because of this, the focus of reporting is limited to those events that violate or have the potential to violate the technical specifications or regulations, or those events that can cause an unsafe condition regardless of the technical specifications or regulatory requirements. Thus it is possible that some events may not be reportable events, such as small reactor leaks without radiological consequence, operation with scrams inoperable that are not required by the NRC, an unexpected control rod withdrawal, high core excess reactivity, or power fluctuations caused by debris on the core without safety consequence that do not violate any technical specification or procedural requirement. Licensees often give a "courtesy call" or otherwise voluntarily report important events or conditions that fall below the legal threshold of the reporting requirements in their technical specifications. In a few instances, further review of the event by the NRC resulted in the voluntary event being considered reportable. Events that are not reportable may be reviewed during the inspection process regarding reporting requirements and event resolution.

3.7 Attention to Detail

Inattention to detail was a common thread in some of the events discussed in previous sections of this report, and contributed to the occurrence or consequences of several events. Examples (not inclusive) are:

At the University of Michigan (November 1992, event A30) during a routine maintenance period, the shim range-control rod interlock system was removed from the reactor control system for a modification that had been reviewed and approved by the facility Safety Review Committee. However, subsequent post-modification testing was inadequate to identify that the power level deviation interlock was inoperable due to a wiring error. Additionally, the post-modification check list was done by a trainee and monitored by an senior reactor operator (SRO). An electrical engineer was present during the initial performance of the Control System Startup Checklist, but neither he nor any other member of the quality assurance team was observing the step-by-step performance of the checklist. The startup checklist, which had previously worked satisfactorily, was recently revised to reduce the possibility of misinterpreting system response. However, during performance of the startup checklist, the SRO misinterpreted a portion of the system response.

At the University of Missouri at Columbia, a series of seven operational failures (1988 through 1996, events A9, 10, 11, 12, 13, 14, and 15) of the regulating blade necessitated manual reactor shutdowns. In at least one case where the failures were related, adequate corrective actions had been identified, but not effectively implemented.

During maintenance activities at the University of Virginia (April 1993, event A19), two mixer-driver modules were changed in the scram logic drawer. The component changes were made during troubleshooting activities that were being conducted to determine the cause of a spurious reactor shutdown earlier in the day. Although the two new mixer-driver modules appeared to be identical, they had been altered internally by facility staff, and were not identical. This resulted in the inoperability of both power-level scrams, the intermediate-range scram, the low primary coolant flow scram, the loss of power to the primary pump scrams, the range switch scram and the key switch scram for a period of 5.5 hours. All of these scram functions were required by the technical specifications. A system verification was not performed after the maintenance due to an inappropriate decision by the shift supervisor. The loss of scram function(s) was not self-revealing, and would have been discovered only by a proper post-modification/post maintenance test such as a scram system check.

A shim rod failed to fully insert into the United States Geological Survey TRIGA reactor (January 1996, event A31) during a normal shutdown. Shim rod 2 had been replaced during a routine control rod inspection conducted in December 1995. Prior to the installation of the new shim rod, the fuel-follower control rod had been measured and found to be about 2 inches shorter than the rod it was replacing. Calculations at the time of installation indicated that the new rod was of sufficient length to remain below the top of the bottom grid plate when retracted to its uppermost configuration. In fact, the rod did not remain below the top of the grid plate, but instead, "caught" on the plate's edge, thereby hanging up several inches above the fully inserted position. The licensee failed to perform a 10 CFR 50.59 safety evaluation of this shim rod modification indicating a less than rigorous safety attitude.

The events at the Massachusetts Institute of Technology Test Reactor (March 1990, event A32, December 1993, event A33, and January 1995, event A34) show weaknesses in the control of reactivity. In the 1990 event (A32) of an improperly calculated estimated critical position, the console operator and reactor supervisor did not check a trainee's estimated critical position calculation, placed excessive reliance on the estimated critical position to identify criticality, and did not closely monitor indications as the reactor was approaching criticality. In the 1993 event (A33), the reactor automatically scrammed while a technician was investigating why one of two low-range neutron flux power level safety channels had failed. The reactor had been started with one of two low-range neutron flux power scrams inoperable. Preoperational testing failed to identify this equipment inoperability caused by a nonself-revealing relay failure. In the 1995 event (A34), the reactor was operated for a short period of time with one of the three normally in service core low flow scrams inoperable (two are required to be operable). An operator had addressed a core temperature pin oscillation during a shutdown by inadvertently turning off both the core flow and temperature recorder instead of just the temperature pin. Although the restart procedure contained a step to restart any instrument which had been secured, the operators did not have a list itemizing which instruments had been secured.

While the above events illustrate individual examples of lack of attention to detail, the survey found that generally non-power reactor staff members were knowledgeable and attentive to their responsibilities. Therefore, although continued focus by non-power reactor management and NRC staff is appropriate to reduce the potential for events such as those described above, no specific changes were identified to ensure the continued safety at non-power reactors licensed by the NRC.

4 Observations

Specific recommendations were not supported for changes to ensure or enhance continued reactor safety of NRC licensed non-power reactors. Survey observations suggest some potential areas for improved safety or operational performance. Overall observations regarding safety performance, SARs, procedure compliance and testing, and use of operating experience are provided below.

4.1 Operating Experience Indicates Good Overall Safety Performance

None of the presently operating non-power reactors have had recent incidents affecting public health and safety. None of the recent operating events resulted in fuel damage or radiation releases. Except for one event in 1986, personnel exposures have been within 10 CFR 20 limits. No adverse effect on public health and safety was seen. Non-power reactors have been operating for many years without causing harm to the public. There is considerable inherent safety in the non-power reactor community of reactors. The adequate protection of public health and safety requirements has been satisfied by all currently operating non-power reactors. Given continued appropriate regulatory attention and focusing on improved operational practices, adequate safety can be sustained.

4.2 Large Variations in Safety Analysis Reports

Although large differences exist in the form and content of SARs, these reflect (1) the evolution which has occurred in licensing since 1956, (2) the large variation in non-power reactor design and function, and (3) the absence of a standard form and content for hazard analyses until relatively recently. As a result, large variations exist in the SARs. However, the NRC issued NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" in 1996 to suggest a uniform format and to help ensure the completeness of information provided. For the non-power reactors in operation at this time, postulated reactivity insertion events were not uniformly considered, even by reactor design type. However, fundamental facility design changes or standardization are not warranted in view of the composite body of safety analyses for similar reactors, together with experimental information described in Appendix B. Further, non-power reactors have been recently relicensed, and most will relicense again before 2002.

4.3 Procedure Compliance and Testing

The majority of events discussed were reactor protection and control related caused by either operator actions or by component failures. Inattention to detail was apparent in some events. Operating experience indicates an opportunity for improvement in procedural compliance (particularly in reactor operation) and in the verification of operability of important instrumentation. Post-maintenance or modification testing sometimes failed to ensure that equipment was operable, resulting in additional equipment failures.

4.4 Greater Dissemination and Analysis of Operational Experience

An expanded event distribution and discussion of event root causes, and resultant corrective actions could be of use. In addition, at least one of the events discussed in this report which had repetitive occurrences could have been precluded by a more detailed and insightful root cause analysis of the initial (or subsequent) event. Lessons learned from non-power reactor operating experience can be used to enhance operational or safety performance and to reduce the likelihood of more significant events in the future.

5 References

1. U.S. Nuclear Regulatory Commission, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," NUREG-1537, Part 2, February 1996, p. xiv.

2. Georgia Institute of Technology application to the NRC, December 1994.

3. U.S. Regulatory Commission Order Modifying Licensee Georgia Institute of Technology, June 1995.

4. Safety Evaluation Report related to the renewal of the operating license for the training and research reactor at the University of Michigan; NUREG-1138, Docket No. 50-2, July 1985.

5. Massachusetts Institute of Technology Safety Analysis Report (MITR-II); October 22, 1970.

6. U.S. Nuclear Regulatory Commission Docket No. 50-20 Massachusetts Institute of Technology, Notice of Issuance of Amended Facility License, July 1975.

7. Final Safety Analysis Report on the National Bureau of Standards Reactor, Addendum 1, November 1980, U.S. Department of Commerce, National Bureau of Standards.

8. Final Safety Analysis Report on the National Bureau of Standards Reactor — NBSR9, July 1996.

9. Safety Evaluation Report Related to the License Renewal and Power Increase for National Bureau of Standards Reactor (NUREG-1007, September 1983, U.S. Nuclear Regulatory Commission.

10. Issuance of Order Modifying License No. R-95 to Convert from High to Low-Enriched Uranium (Amendment 0.17), Rhode Island Atomic Energy Commission, March 17, 1993.

11. Safety Analysis Report for the Low Enriched Fuel Conversion of Rhode Island Nuclear Science Center Research Reactor, November 1991.

12. University of Missouri Research Reactor Facility (Columbia), Addendum 3, August 1972.

13. Safety Analysis Report for Low-Enriched Uranium Fueled University of Virginia Reactor, January 1991.

14. AEC staff Safety Evaluation, the University of Missouri Research Reactor Facility (Columbia), July 1966.

15. U.S. Nuclear Regulatory Commission, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," NUREG-1537, Part 2, February 1996, p. xix.

APPENDIX A

SELECTED NON-POWER REACTOR EVENTS

Event A1

Facility: Massachusetts Institute of Technology (5 MW)

Event Date: March 8, 1989

Event Description:

The reactor was operated with two of three power level safety channels inoperable in violation of technical specifications and four of six period and power level safety channels improperly connected to the low voltage protection circuit.

The technical specifications require that at least two nuclear safety (power level) channels be operable prior to the reactor being brought critical. Nuclear safety channel No. 4, one of three channels providing an automatic shutdown signal on high reactor power, had been serviced during the maintenance period preceding the reactor startup and was considered out-of-service pending observation of its performance during startup and power operation. The reactor was started up with stepwise increases in power to monitor the readings on the safety channels. While doing this, it was noted that safety channel No. 5 was not responding properly and the reactor was immediately shut down from 4 MW. The reactor had operated above 1 MW for 28 minutes.

One power level channel was operable and three redundant scrams on the core outlet temperature were operable.

Cause of Event:

It was found that the channel No. 5 high voltage power supply had been switched off unknowingly, even though the voltage had been checked and recorded as part of the instrumentation checklist.

Subsequent analysis of the low voltage protection circuit that causes an automatic shutdown in the event of a loss of chamber high voltage or lack of continuity on the chamber signal cables in any of the nuclear safety channels found that it was operational. However, it was found that the power supply for channel No. 5 had not been connected to this circuit and had been interchanged with another level channel.

Licensee Corrective Actions:

Safety channel No. 4 was calibrated and returned to service. Safety channel No. 5 was reconfigured so that it properly interfaced with the low voltage protection channel. The power supplies for both of the other level channels and the three period safety channels were inspected. Four of the six period and power level safety channels were found to be improperly connected to the low voltage protection circuit. The "low voltage protection" circuit was not a technical specification requirement. The licensee instituted a requirement to document the proper interfacing of safety channel power supplies to the low voltage protection circuit whenever a detector or associated cabling was serviced.

Event A2

Facility: Massachusetts Institute of Technology (5 MW)

Event Date: March 20, 1995

Event Description:

The reactor was operated at up to 70 kW for flux measurement experiments with two inoperable low-range neutron flux channels that were required by Technical Specifications for natural circulation operation below 100 kW (also see December 7, 1993 event). Three period channels, a 5.5 MW high-range channel, and two primary coolant outlet temperature scrams (that could have scrammed the reactor had reactor power approached 100 kW) were operable at the time.

Cause of Event:

Although the low-range amplifiers were set in accordance with facility procedures, the amplification factor was found to be in error resulting in actual scram setpoints of 420 kW and 230 kW, instead of being correctly set at or below 100 kW.

Licensee Corrective Actions:

As a corrective action to address the root cause, the amplification factors of the low-range amplifiers were to be verified on an annual basis.

Event A3

Facility: Massachusetts Institute of Technology (5 MW)

Event Date: July 19, 1995

Event Description:

The reactor was started up and operated with one control blade fully inserted in the core. This condition was not discovered until after approximately seven hours of operation.

At 1017 hours, the reactor was started up by an operator-in-training, under the supervision of two licensed SROs, with two shim blade "blade in" indications (Nos. 3 and 4) known to be out of commission because of malfunctioning proximity switches. The startup was halted by procedure after the control blade positions exceeded the estimated critical position by 1.27 cm (0.5 inches) and the reactor was subcritical. Rather than following the procedure, which required the operators to confirm that the control blades were coupled to their magnet drives, the two licensed SROs decided that the reason for the reactivity difference was that the reactivity contribution of a newly installed Boiling Coolant Chemistry Loop experiment had not been properly estimated. The reactor was taken critical at 4.32 cm (1.70 inches) above the estimated critical position and run at low power for an hour.

After power was increased to 4.8 MW, the supervisor noticed that the ΔT across the core was higher than would be expected for that power level. A heat balance calculation was performed, which found that once equilibrium conditions were attained, the thermal power would be 5.04 MW, slightly above licensed power level, and reactor power was immediately lowered to 4.5 MW. Operation continued at 4.56 MW until 1720 hours, when an operator performed a reshim, noticed that movement of control blade No. 4 caused no reactivity effect, and manually scrammed the reactor.

The NRC was notified of the technical specification violation of operation without all blades within 5 cm (2 inches) of the operating position.

Cause of Event:

The subsequent event review found the main cause was the failure of licensed operators to follow written procedures for investigating a mismatch between calculated and observed estimated critical positions. Control blade magnetic currents were routinely lowered to about 60 milliampere compared with the normal setpoint of 80 milliampere, to avoid electrical interaction between the magnet current circuitry and one of the period channels. It was determined that the minimum current necessary to reliably pick up blade No. 4 was 64 milliampere. Because the "blade in" indication was out of commission, there was no immediate indication that blade No. 4 was not attached to its magnet. It was not unusual for these proximity switches to fail due to the hostile environment. Blade No. 4 had not been picked up during a startup on July 11, 1995, but this event had been properly diagnosed and corrected before the reactor was brought critical. However, this condition had not been communicated to all of the reactor operators, including those on duty during this event.

Licensee Corrective Actions:

No evidence of fuel damage or mechanical binding of blade No. 4 was found.

Power density and thermal hydraulic calculations determined that none of the core power distribution limits had been violated during operation with blade No. 4 full in.

Although the two failed proximity switches were replaced, another switch subsequently failed.

The reactor safety committee recommended development of a method for better communication among all operators, management action to ensure all personnel follow facility procedures, determination of the minimum current required to pick up each blade, development of a checklist to guide operators when the reactor is not critical within 1.27 cm (0.5 inches) of the estimated critical position, development of a special procedure for verification that a blade was connected to its magnet when the "blade in" indication was out of commission, and management action to resolve electronic equipment problems.

Event A4

Facility: Massachusetts Institute of Technology (5 MW)

Event Date: August 9, 1995

Event Description:

The reactor was taken critical above its Estimated Critical Position and operated, when a shim blade position indication was incorrect after the blade mechanism slipped during withdrawal.

On August 1, 1995, a rebuilt shim blade mechanism, magnet and shim blade were installed in shim blade position #4, tested satisfactorily and the reactor operated normally for 5 days. On August 9, 1995, the reactor was started up after installation of the Boiling Coolant Chemistry Loop experiment. The estimated critical position of 22.98 cm (9.05 inches) was reached, but the reactor was still subcritical. Each blade was verified to be coupled to its magnet drive per procedure. The Director of Reactor Operations attributed the reactivity difference to the Boiling Coolant Chemistry Loop experiment and startup was continued, with critical blade position being reached at 24.7 cm (9.73 inches). Operation was continued until the reactor was shutdown on the morning of August 10, 1995. At that time blade #4 indication was 10.8 cm (4.25 inches) with the blade fully inserted.

Cause of Event:

Examination found the blade mechanism slipped intermittently when the blade was withdrawn, but no slippage occurred during insertion. Since the drive position indication was coupled with the drive mechanism, the position indication in the control room showed a withdrawal even if no actual blade withdrawal occurred. Ex-core tests showed consistent slippages of 10 to 15 cm (4 to 6 inches) during withdrawals. The drive mechanism had a vespel nut rotated on the lead screw, driving the blade in or out. Upon disassembly of the drive mechanism, the vespel nut pin was found to be worn and scoring marks were found on the inside of the vertical shaft. The licensee speculated that the pin was improperly positioned outside its slot and that the friction between the pin and the vertical shaft held the vespel nut in place until the pin wore and slipped. Although slippage could occur in either direction, the licensee noted that because of the shape of the vespel nut, a downward movement of the vertical control blade shaft would create a greater friction force on the vespel nut and pin, reducing the likelihood of slippage while driving the blade in.

Licensee Corrective Actions:

The blade drive mechanism assembly procedure was modified to include a step for two person verification to ensure the vespel nut and pin were properly rotated, seated, and locked at maximum depth.

Event A5

Facility: National Institute of Standards and Technology (20 MW)

Event Date: April 17, 1990

Event Description:

Shim arms did not fully insert on a manual scram because of bearing problems. During a routine shutdown, Shim No. 1 failed to drop from 12° upon a manual scram. The scram was reset and the shim was driven to the lower limit. Surveillance tests had been normal two weeks before and the shim had scrammed normally from 12° five times previously. Background information: During routine shutdowns, shim arms are driven to 12°, at the shock absorber edge, and then manually scrammed, cutting off current to the clutch, allowing it to disengage, and the shim to fall. If the clutch is not released, a rundown signal associated with the scram drives the shims in. The manual scram circuit is independent of the nuclear logic scram.

Cause of Event:

The Shim No. 1 clutch plate appeared difficult to turn by hand and the mechanism did not seem to turn freely during bench inspection. Upon disassembly, the ball nut assembly had some specks of "dirt" but no sign of excessive wear. The lower taper bearing seemed a little rough.

Licensee Corrective Actions:

The two bearings were replaced and the mechanism was cleaned and tested several times. Five degree rod drop tests were performed satisfactorily. The reactor was restarted within a few hours with all indications normal.

Event A6

Facility: National Institute of Standards and Technology (20 MW)

Event Date: September 18, 1990

Event Description:

After a routine shutdown, Shim No. 4 returned to about 1.9°, above its normal position of < 1°. The lower limit switch setting was at 2.5°. Otherwise, the shim operated normally and met all release and drop requirements. During 1989, the same phenomenon had been observed on Shim No. 2.

Cause of Event:

After investigation and testing, the Shim No. 2 problem was traced to the inner bearing of the shaft, which was replaced. The licensee intended to replace the bearings on the remaining three shims at the next shim arm replacement, and stainless steel bearings were ordered to replace the carbon steel bearings. Shim No. 4 symptoms were similar to Shim No. 2.

Licensee Corrective Actions:

The licensee replaced the bearing-seal assemblies on Shim Nos. 1, 3, and 4 with stainless steel bearings.

Event A7

Facility: National Institute of Standards and Technology (20 MW)

Event Date: September 18, 1990

Event Description:

On the same day that Shim No. 4 remained above its normal position after a scram because of a bearing problem, Shim No. 1 failed to release completely upon manual scram because of a relay failure.

Shim No. 1 failed to release completely upon manual scram from 12°, falling to about 9°, after a routine shutdown for maintenance and an attempted restart that could not override xenon. From there it drove in to about 6° at which time the manual scram button was depressed for a second time and held, resulting in the shim dropping to the bottom. During this period of time, it was noticed that the current to the clutch of Shim No. 1 did not cut off even after the reactor keys were removed. The manual scram button was depressed for a third time, and the current did cut off.

Cause of Event:

The problem was traced to a mercury-wetted-contact relay in the manual scram circuit of Shim No. 1, the contacts of which failed to break completely.

Licensee Corrective Actions:

The relay was replaced and the shim operated normally. All other similar relays in the system were checked and found to operate normally.

Event A8

Facility: National Institute of Standards and Technology (20 MW)

Event Date: September 10, 1993

Event Description:

The reactor was manually scrammed because of power fluctuations caused by an unlatched fuel element.

Following refueling the day before, a routine reactor startup was made with required steps to 0.1 MW, 1 MW, and 10 MW. When the operator noticed power fluctuations on all three power channels, he immediately reduced power to 1 MW and when the fluctuations continued, he shut down the reactor.

Cause of Event:

A check of the entire core revealed that a fuel element was unlatched.

Licensee Corrective Actions:

The fuel element was removed from the core and inspected; the element was intact but appeared more discolored than normal. All other fuel elements were double checked and a replacement fuel element inserted. Primary flow was continued for 2 hours, stopped for 30 minutes, then restarted again while nuclear instrumentation was monitored, and the fuel element was rechecked again before restart. The reactor restart had an additional stop at 5 MW and 2 days at 10 MW before increasing power.

Special instructions were issued to operators on fuel handling, monitoring, and testing and response to instrument fluctuations. The detent in the grid under the unlatched position was examined with a boroscope and found to be acceptable. The orientation of the latching bar on all elements was later reverified and found not to be fully rotated in three instances. As a result, a special tool was developed that could only be rotated in the locking direction and used to confirm that the latching bar was in the proper orientation in following refuelings.

Event A9

Facility: University of Missouri at Columbia (10 MW)

Event Date: September 21, 1988

Event Description:

With the reactor operating in the automatic mode, a high power rod run-in occurred after reactor power reached approximately 10.9 MW. The operator returned to the licensed power level of 10 MW, found that the regulating blade failed to operate in either the automatic or manual mode, and shutdown the reactor.

Cause of Event:

The unexpected increase in reactor power was caused by positive reactivity from slowly decreasing primary temperature after the cooling tower fans had been shifted to fast speed, while the regulating blade did not respond automatically to the power level increase until the trip set point for the rod run-in was reached. The reactor was operated for approximately 15 minutes with the regulating blade inoperable because of a loose set screw in the gearbox.

Licensee Corrective Actions:

A spare gearbox was installed and tested. The licensee also added a semiannual preventative maintenance requirement for the regulating blade drive mechanism that included disassembly and visual inspection of the gears and gear set screws.

Event A10

Facility: University of Missouri at Columbia (10 MW)

Event Date: November 28, 1988

Event Description:

The reactor was manually scrammed from full power in the automatic mode, after an alarm annunciated after reactor power decreased to 95 percent. The reactor had operated for less than three minutes with the regulating blade inoperable.

Cause of Event:

The regulating blade gearbox output shaft was found to have sheared because of a misalignment between the gearbox and the coupling.

Licensee Corrective Actions:

The licensee considered changing the technical specifications so that a failure of the regulating blade was not automatically a deviation from the limiting condition for operation and would allow for a timely reactor shutdown as an action statement, to alleviate the generation of a licensee event report. Also, in a February 22, 1996, report, and a January 26, 1996, licensee event report concerning the regulating blade being inoperable, the licensee stated that they are developing a safety analysis for such a technical specification change.

Event A11

Facility: University of Missouri at Columbia (10 MW)

Event Date: June 3, 1989

Event Description:

The reactor was shutdown from full power in the automatic mode, after an operator found that the drive chain had fallen off the drive gear for the regulating blade rotary limit switch assembly, which provided position alarms as well as the rod run-in function.

Cause of Event:

The drive chain had fallen off the drive gear for the regulating blade rotary limit switch assembly.

Licensee Corrective Actions:

The chain was reinstalled and the rod run-in functions were tested. The licensee added a quarterly visual inspection of the drive chains for the position indication transmitter and rotary limit switch assembly on the regulating blade drive mechanism.

Event A12

Facility: University of Missouri at Columbia (10 MW)

Event Date: November 4, 1992

Event Description:

The reactor was manually scrammed from full power in the automatic mode, after a downscale alarm annunciated when reactor power reached 95 percent.

Cause of Event:

The set screw engaging the regulating blade motor shaft to the gearbox had come loose, making the regulating blade inoperable for seven to eight minutes.

Licensee Corrective Actions:

The licensee filed a flat on the motor shaft and used Loctite on the screws as a corrective action.

Event A13

Facility: University of Missouri at Columbia (10 MW)

Event Date: April 26, 1994

Event Description:

With the reactor operating at full power in the automatic mode, the shift supervisor noted that the wide range chart was showing a downward trend. Manual operation of the regulating blade switch revealed that the regulating blade mechanism motor was responding, but the gearbox shaft was not, as reactor power steadily decreased due to xenon buildup. The regulating blade had been inoperable for a total of six to seven minutes. Tightening the loose set screw returned the regulating blade to operability. The missing set screw was replaced. Full reactor power was recovered approximately ten minutes later.

Cause of Event:

The problem was identified as one loose and one missing set screw in the motor to gearbox coupling. The semiannual regulating blade preventative maintenance procedure that had been performed twice since the November 24, 1992 failure, had not been changed to include the use of Loctite on the set screws.

Licensee Corrective Actions:

The licensee drilled the motor shaft to accept a pin to provide a more positive mounting of the coupling to the shaft.

Event A14

Facility: University of Missouri at Columbia (10 MW)

Event Date: December 27, 1995

Event Description:

The reactor was manually shutdown after an operator heard a difference in the sound of the regulating blade operation when he was on the reactor bridge, even though a wide range monitor indicated that the regulating blade had been maintaining reactor power within its normal range. The regulating blade was found to drive in normally, but was slower than normal when driving out.

Cause of Event:

A technician found that the dowel pin had failed in the gearbox coupling to the drive motor resulting in a friction fit between the gearbox input shaft and the gearbox coupling that allowed the shaft to slip in the coupling when the regulating blade was driven out. Subsequent shop investigation indicated that the dowel pin was missing and had presumably broken shortly before the operator noticed the difference in its sound of operation.

Licensee Corrective Actions:

The dowel pin and input shaft bearing were replaced, the mechanism tested, and regulating blade rod run-ins were checked. Since the technical specification limiting condition for operation considered the facility in non-compliance, regardless of prompt operator action to shutdown the reactor whenever the regulating blade was found to be in a degraded condition, the licensee decided to develop a safety analysis to support a request for a technical specification change that would allow a timely reactor shutdown as an action statement for the failure of the regulating blade, consistent with ANS-15.1 and ANS-15.18, where special reports would not be required when a research reactor momentarily operates outside its limiting condition for operations if prompt remedial action is taken.

Event A15

Facility: University of Missouri at Columbia (10 MW)

Event Date: January 23, 1996

Event Description:

The reactor was manually scrammed after the regulating blade had been inoperable for approximately 5 minutes as a result of a seized bearing on the gearbox input shaft.

Cause of Event:

This bearing had been a replacement as a result of the December 27, 1995 event (A14), but may have lost its original lubricant during storage, judging from its unusual failure mechanism. The licensee identified the maintenance procedure as deficient as it did not provide directions for lubrication or replacement of bearings.

Licensee Corrective Actions:

The licensee rebuilt the spare gearbox with new sealed (self-lubricating) bearings, placed it in service, and changed the regulating blade drive preventative maintenance procedure to specify that the gearbox be replaced with a rebuilt gearbox every two years. The licensee continued preparing their technical specifications revision to eliminate the requirement to issue a licensee event report for conditions which they considered did not pose a safety concern for the reactor or the public.

Event A16

Facility: University of Michigan (2 MW)

Event Date: June 8, 1992

Event Description:

The assistant manager (an SRO) sent two other operators to move fuel, while he performed the operational checks of control room instrumentation done every 2 hours. While the reactor was still critical at 8 kW in automatic control rod control, an operator yelled "coming out" and pulled the next designated fuel element out of the core. This was contrary to fuel handling procedures and technical specification requirements that the reactor be subcritical by at least 0.025 $\Delta k/k$ during fuel loading changes. The reactor became subcritical when the fuel element was withdrawn and the control rod that had been in automatic control audibly switched to manual control. Another operator yelled "stop" and went into the control room and began driving the rods into the core. The facility's radiation monitoring equipment indicated that there had been no release of radioactivity as a result of this event. The operators determined the event was not reportable to the NRC. While the assistant manager searched for the project director, who was unavailable, the rest of the operators continued with the final fuel movement without considering positive reactivity insertions as possible consequences or developing any corrective actions based on the implications of the event. The next day, the assistant manager and the project director again concluded the event was not reportable to the NRC. On June 16, 1992, the Safety Review Committee recommended that the NRC be notified of the event, and the licensee notified the NRC on the following day.

Cause of Event:

Subsequent analyses determined that replacing the same fuel element into the same position could have added 0.0054 $\Delta k/k$ to the core, while a fresh undepleted fuel element could have added 0.01 $\Delta k/k$. Both possibilities were within the SAR's analyzed reactivity addition of 0.016 $\Delta k/k$ that could occur without fuel damage. The event had no immediate effect on the health and safety of the public. However, the event was the result of personnel error involving two apparent violations of the facility's technical specifications: (1) safety margin for fuel movements, and (2) a failure to report the event within the time frame defined by the technical specifications. Weaknesses identified included poor communications during the event, operators not reviewing or using procedures, a lack of a thorough review and full analysis of the event prior to further fuel moves, and the determination of reportability of the event.

Licensee Corrective Actions:

Licensee corrective actions were required prior to restart of operations. These corrective actions included modification of procedures and installation of light indicators on the rod drive housing located on the bridge (the indicators are lit only when the rods are fully inserted). Training and these procedural and equipment changes provided improved communication, control of fuel changes and visual indication on the status of the control rods. Additionally, the licensee requested and received an NOTRTR review (Review 92-02, Docket 50-2) to prevent recurrence and to strengthen the overall operation of the facility.

NRC Actions:

The NRC dispatched an Augmented Inspection Team to investigate this event. Two violations were issued; one for violating the technical specifications which required having the reactor subcritical during fuel movements and another for violating the procedure to have all rods fully inserted during fuel movements. Those violations were characterized as a Severity Level III problem and assessed a $1250.00 civil penalty. A Severity Level IV violation was also issued for failure to report the event to the NRC within 24 hours. The NRC reported this as an Abnormal Occurrence to Congress.

Event A17

Facility: University of Michigan (2 MW)

Event Date: February 24, 1993

Event Description:

The reactor was shutdown by inserting the control rod and depressing the manual scram button to insert the shim-safety rods, after the shim-safety rod control switch failed during a routine shutdown margin and excess reactivity check following shim-safety and control rod calibrations.

Cause of Event:

The switch failed and could not be placed in the insert position.

Licensee Corrective Actions:

The shim-safety rod control switch was removed, repaired, and tested satisfactorily. Following a pre-startup checkout, the reactor was restarted normally.

Event A18

Facility: University of Michigan (2 MW)

Event Date: March 24, 1993

Event Description:

The MTR-fueled 2 MW reactor exceeded its licensed power by 15 percent, operating at 2.3 MW, with a high power scram set above its TS limit for 11 minutes. The shift crew (two senior reactor operators) conducted a routine startup. At an indicated 1 MW power level (50 percent of full power), the crew conducted a calorimetric power determination which found that the reactor was actually at 1.156 MW. At that point the shift crew should have adjusted the automatic rod control mode used for automatic power control to 86 percent which, while in the automatic control mode, would have reduced actual power to 1 MW. However, the crew continued to raise power to 2 MW (100 percent indicated power). That resulted in the actual power being 2.3 MW, 15 percent above the licensed limit. Approximately 10 minutes later, the Assistant Reactor Manager for Operations arrived in the control room, reviewed the calorimetric data, and immediately ordered the crew to return the reactor to 1 MW indicated power. The NRC Preliminary Notification noted that the crew had also failed to adjust the overpower reactor scram setpoint, a high power level scram would not have occurred at the TS limit of 2.4 MW, but at about 2.6 MW.

Cause of Event:

An NRC Special Safety Inspection found that the root causes of the event were a lack of management oversight of operations and poor communications between the shift supervisor and console operator (including reluctance by some operators to ask questions when uncertain, based on experience of being chastised by shift supervisors when questioning their actions); deficient operator knowledge regarding the power level instrumentation; operator failure to review or use the startup procedure; an unclear procedural step covering the evolution (had it been used); and ineffectiveness of previous corrective actions from a July 22, 1992 event.

Licensee Corrective Actions

An NRC Special Safety Inspection noted that the licensee permanently removed the shift supervisor from licensed activities; required that neutron channel adjustments necessary to match thermal power be made at 1 MW prior to proceeding to 2 MW: revised the "Reactor startup" operating procedure to provide a smooth transition to the "Power Level Determination" operating procedure and back; conducted an unannounced oral and written examination of the reactor operations staff; and issued a memo emphasizing adequate review of procedures prior to their use.

NRC Actions:

The NRC performed a reactive team inspection and issued a Severity Level III violation and a $3,750 civil penalty.

Event A19

Facility: University of Virginia (2 MW)

Event Date: April 28, 1993

Event Description:

During maintenance activities, two mixer-driver modules were changed in the scram logic drawer of this MTR-fueled 2 MW pool reactor, which resulted in the inoperability of both power-level scrams, the intermediate-range scram, the low primary coolant flow scram, the loss of power to the primary pump scram, the range switch scram and the key switch scram (all technical specification-required protective scrams) during 5.5 hours of reactor operation.

A test of the reactor trip system had been successfully performed earlier that morning, as required by procedure. The exchange was made during troubleshooting to determine the cause of a spurious reactor shutdown earlier in the day. Although the two new mixer-driver modules appeared to be identical, they had been altered internally by the facility staff. A system verification was not performed after the exchange because the operator and his supervisor had decided that the exchange of mixer-driver modules as well as the earlier temporary switching of solid state relays did not require any checks. The console operator had all the normal alarms, instrumentation, and manual shutdown capability available to him. The loss of the scram function was not self-revealing, so that only a check of the scram system with the reactor shutdown would have found these inoperable trips. During the period the reactor was operated with some scrams inoperable, no operational parameters were exceeded, no safety limits were violated, and no damage was caused to the reactor or other electronic components in the console.

Cause of Event:

A peer review of the event on behalf of the NOTRTR found the root and immediate causes to have been "the lack of a focal point for controlling operations at the facility compounded by the failure to recognize that a modification had been made with no subsequent checks or tests to verify operability of the safety system functions." The peer review found the licensee had neither a requirement to make a determination that a change did not involve an unreviewed safety question nor an adequate restart checklist.

NRC Actions:

This event resulted in a reactive inspection from NRC Region II, a Severity Level II violation for operating without five safety system channels required by technical specifications and failing to verify that the safety system channels were operable following maintenance as required by the technical specifications, and a $2,000.00 civil penalty. The NRC reported this as an Abnormal Occurrence to Congress.

Event A20

Facility: Oregon State University (1MW)

Event Date: February 17, 1998

Event Description:

The TRIGA had completed a routine 14 minute run at 15 watts power to perform core excess reactivity measurements. An attempt was made to manually scram the reactor using the scram button at the end of the run. When the manual scram button did not work, the operator's next step was to turn off power to the scram circuit using the reactor three-position key switch. As the operator touched the switch, the switch moved from a position between OPERATE and RESET to the OPERATE position. The operator then tried the manual scram button and the reactor scrammed.

Cause of Event:

The licensee determined that a buildup of dirt prevented the three position switch from returning to the OPERATE position. When the switch is in the RESET position, the scram bus is disabled. However, the console is designed so that magnet power is cut off when the switch is in the

RESET position. This switch dates to original console installation in 1967. The switch operated properly during preoperational testing before start-up.

Upon further investigation, the licensee discovered that the wiring of the scram circuit was different from that shown in the Instrument Maintenance Manual provided by the reactor vendor. The licensee's investigation led to a conclusion that the location of a jumper was probably modified during initial installation of the reactor console in 1967. This modification bypassed the design feature that cut off magnet power when the three-position switch is in the RESET position.

Licensee Corrective Actions:

The licensee took a number of corrective actions. The three position switch was removed, cleaned, and relubricated, which restored proper operation, and the switch was reinstalled in the console. The reactor console wiring was restored to its as-designed condition. A physical, electronic check of the wiring in the scram circuitry and other non-scram related circuits was performed to demonstrate that the wiring in the console is as designed. The reactor startup procedure was rewritten to test that the magnet power is cut off when the three position switch is placed in the RESET position. The reactor console was subject to routine startup checks and the semiannual console check procedure. The reactor vendor was contacted to obtain check out procedures to confirm that all suggested surveillances are done before reactor operation.

NRC Actions:

The NRC performed a reactive inspection and issued two violations characterized as a Severity Level III problem. A civil penalty was considered but not imposed based on prior enforcement history and credit for licensee corrective actions. NRC issued Information Notice 98-14.

Event A21

Facility: Rhode Island Atomic Energy Commission Nuclear Science Center (2 MW)

Event Date: July 1995

Event Description:

The reactor pool leak increased from 1993 to 1995 from 1-2 liters per day to 18.9 liters (5 gallons) per day to 94.6 liters (25 gallons) per day. The leaking water was collected in drain lines and pumped to a holding tank. After sampling, the water was discharged. Most of the leakage came from the area below the thermal column door on the main reactor floor level. This leakage had not been present for a few years but reappeared.

Cause of Event:

The most probable source of the leakage was from cooling tubes that were welded to the pool liner that made a u-shape into the concrete.

Licensee Corrective Actions:

The leak is the top priority for funding under the Department of Energy Grant Program. The licensee developed a program to determine the source of the pool water leakage. Lessons learned at other facilities served as a basis for a plan that included cutting grooves in the floor to collect leakage and using the sodium-24 produced during routine operations as a decay-corrected tracer. These actions were documented in a 1996 NRC inspection report.

Event A22

Facility: University of Virginia (2 MW)

Event Date: Condition existed for several months until January 1994

Event Description:

The reactor pool had a variable leak rate of up to 379 liters (100 gallons) per day into a retention pond outside the building. The leak was within the makeup capability. Because there was no leaking fuel, the pool water was within 10 CFR 20 limits for release to unrestricted areas.

Cause of Event:

Pool leak

Licensee Corrective Actions:

The licensee shut down the reactor and analyzed water samples in the pool and pond weekly for about 4 months while the leak was repaired by pumping liquid grout into several areas of the pool wall where moisture was observed and painting the top 1 m (3 feet) of the pool wall.

Event A23

Facility: University of Virginia (2 MW)

Event Date: November 1995

Event Description:

An NRC morning report from Region II noted that the licensee detected leakage from the reactor pool at a rate of 379 to 454 liters (100 to 120 gallons) per day, as determined by pool level decrease. Over a period of months, this decreased to about 151 to 189 liters (40 to 50 gallons) per day. Measured radioactivity levels in water samples from the pool were generally below 10 CFR 20 limits for release to unrestricted areas except for sodium-24.

Cause of Event:

Suspected pool leak around the center buttress.

Licensee Corrective Actions:

The licensee monitored the pool levels daily and analyzed pool and pond samples on a weekly basis. The leak appeared to be self-sealing, with makeup levels (including pool evaporation) dropping to 265 to 302.8 liters (70 to 80 gallons) per day during the first two months of 1996.

Event A24

Facility: University of Virginia (2 MW)

Event Date: August – October, 1995

Event Description:

During the week of August 7, 1995, the secondary side of the heat exchanger was cleaned. Some of the tubes had scaling and pitting, although there was no evidence of a leak. The tubes were cleaned and the heat exchanger reassembled and the reactor was operated at 2 MW on August 17, 18, and 22. On August 22, low levels of sodium-24 were detected in the secondary water, indicating a possible primary to secondary leak.

Cause of Event:

Leaking heat exchanger tubes

Licensee Corrective Actions:

The licensee closed the isolation valves between the primary and secondary systems and restricted reactor operation to less than 200 kW on natural convection until the problem was evaluated and corrected. Eight tubes were plugged on October 25, 1995, and a static pressure test revealed no other leaks. A new surveillance requirement for weekly monitoring of the secondary water for radioisotopes was added to the technical specifications.

Event A25

Facility: University of Virginia (2 MW)

Event Date: March 13, 1996

Event Description:

An operator manually shut down the reactor during a startup in response to 0.2 MW power fluctuations on both linear and intermediate power channels. Power level had not exceeded the authorized maximum power level of 2 MW.

Cause of Event:

The licensee subsequently found a paper towel lying on top of the core. Visual inspection of the core and water sampling confirmed that no fuel damage had occurred. The licensee theorized the paper towel had permitted water to pass through it with some increased resistance and attributed the power fluctuations to reactivity changes resulting from water temperature changes by the affected fuel element.

Licensee Corrective Actions:

The licensee instructed the reactor staff to keep foreign materials away from the pool border, searched the pool for other debris (and found none), examined the areas around the pool and removed materials that could have been inadvertently dropped into the pool, and developed a checklist to identify anything that could get into the pool and remove the item as necessary.

Event A26

Facility: National Institute of Standards and Technology (20 MW)

Event Date: June 25, 1991

Event Description:

While remotely inserting a new fuel element into position in the core, the element fell off the tool and landed on the top grid about a foot below.

Cause of Event:

Not given in the brief information letter to the NRC.

Licensee Corrective Actions:

A special tool was made and the element was retrieved. A thorough inspection showed the element was undamaged. The element was cleaned, minor external scratches polished, and reinserted into position.

Event A27

Facility: University of Missouri at Columbia (10 MW)

Event Date: June 1986

Event Description:

An individual who handled radioactive Thulium-170 (Tm-170) pellets in a restricted area received a dose of approximately 1.15 Seivert (115 rem) to the hands while transferring the pellets into a container for shipment.

Cause of Event:

The licensee failed to adequately evaluate the potential consequences of handling curie quantities of Thulium pellets. The licensee did not perform exposure rate calculations and pre-operational dry runs.

Licensee Corrective Actions:

The licensee completed a study to determine the specific survey meters best suited for monitoring Tm-170 operations. The licensee planned to use thulium wafer holders specifically designed to prevent significant beta fields in any direction except through a tool opening. The licensee adopted a procedure for thulium wafer transfer which provided greater provisions for radiological controls.

NRC Actions:

NRC conducted a reactive inspection in response to this event. NRC assessed a civil penalty of $4000 for two violations which were together characterized as a Severity Level II problem. The NRC also issued three Severity Level IV violations.

Event A28

Facility: Ohio State University (500 kW)

Event Date: March 6, 1996

Event Description:

A researcher received an extremity dose of 0.089 Seivert (8.9 rem) while performing a tissue bath experiment.

Cause of Event:

The researcher noted that one of the vial caps had popped off. The researcher, without gloves, replaced the cap on the vial and became contaminated.

Licensee Corrective Actions:

The researcher was instructed to wear protective wear at all times when dealing with radioactive materials. The importance of wearing appropriate protective wear and other radiological control practices was communicated to all users of radioactive materials via several means.

Event A29

Facility: Massachusetts Institute of Technology (5 MW)

Event Date: June 29, 1994

Event Description:

Two reactor operators received extremity doses of 0.092 Seivert and 0.115 Seivert (9.2 and 11.5 rem), respectively. A vial containing Dysprosium-165 (Dy-165) ruptured during handling and contaminated the reactor operators.

Cause of Event:

Attempts to remove the vials resulted in contamination spread. One of the reactor operators incorrectly removed anti-contamination gloves. The reactor operators failed to survey themselves after handling an irradiated sample.

Licensee Corrective Actions:

The licensee purchased longer tongs for Dy-165 handling.

NRC Actions:

None Identified

Event A30

Facility: University of Michigan (2 MW)

Event Date: November 24, 1992

Event Description:

The reactor was operated without the power level deviation interlock operable. This could have allowed continuous withdrawal of a control rod.

On November 24, 1992, during a routine maintenance period, the shim range, control rod interlock system was removed from the reactor control system for a modification that had been reviewed and approved by the facility Safety Review Committee on October 6, 1992. Following the wiring modification, a reactor pre-start checklist was conducted to test the modification. A wiring error made during the modification inadvertently disabled the interlock that drops the reactor out of automatic control if the linear level neutron detection system indicated power 5 percent below the automatic control setpoint. The fact that this interlock was disabled was not discovered during the performance of the post modification test. An SRO misinterpreted a rundown as the expected drop-out-of-auto when a trainee apparently increased the setpoint too slowly during the checklist. Neither the electrical engineer nor any other member of the quality assurance team observed the performance of the checklist step-by-step. Following "successful" completion of the checklist, the reactor was started up to approximately 5 kW to perform a shutdown margin and excess reactivity check. The control rod was withdrawn to 61 cm (24 inches) and the shim rods to criticality, but the reactor was never placed in automatic control. Following the reactivity measurement, the reactor was shutdown.

During the midnight shift on November 25, 1992, the crew performed the prestart checklist, but could not successfully complete the drop-out-of-auto and rod insertion verification. The Assistant Manager and the electrical engineer were notified, the procedure retried, the wiring mistake was discovered and corrected, and the checklist satisfactorily completed. The personnel who had conducted the original checklist were interviewed and a series of errors and misinterpretations resolved. At 9:00 a.m. November 25, 1992, the Reactor Manager gave permission to startup the reactor for power operation.

The licensee noted that the worst possible consequence of operating with the power level deviation interlock out of service would be failure of the automatic control system followed by continuous withdrawal of the control rod. If the control rod was withdrawn from 0.0 to 61 cm (0.0 to 24 inches), 0.00475 $\Delta k/k$ reactivity would be inserted, with a resultant period of 6 seconds. At a 10 second period, an automatic rundown of the shim-safety rods would occur. The reactor period scram setpoint is 5 seconds.

Cause of Event:

A wiring error made during the modification process. A weak reactor startup checklist used as the post-modification test led the operator to misinterpret the results of that test.

Licensee Corrective Actions:

The licensee changed the prestart checklist to ensure proper verification of the drop-out-of-auto and rod insertion. When a modification is made, the functional changes related to the modification will be tested and verified by the quality assurance team members directly responsible for the modification.

Event A31

Facility: U. S. Geological Survey (1 MW)

Event Date: January 4, 1996

Event Description:

Shim rod 2 failed to insert during a reactor shutdown.

Cause of Event:

Shim rod 2 did not remain below the top of the bottom grid plate, but instead caught on the plate's edge, thereby hanging up several inches above the fully inserted position. Calculations at the time of installation had indicated that the new rod was of sufficient length to remain below the top of the bottom grid plate. The licensee failed to perform and document a safety evaluation of the new, shorter replacement control rod.

Licensee Corrective Actions:

The licensee replaced shim rod 2 with the original control rod, and satisfactorily completed all control rod tests.

NRC Actions:

NRC issued a Severity Level IV violation for failure to perform and maintain records of a safety evaluation for the replacement of a control rod.

Event A32

Facility: Massachusetts Institute of Technology (5 MW)

Event Date: March 12, 1990

Event Description:

The reactor was automatically scrammed on a short period because of an error in calculating the Estimated Critical Position.

An estimated critical position was calculated independently by a trainee, the console operator, and the reactor supervisor. However, the operator and the reactor supervisor relied on the trainee's use of critical data from a xenon-equilibrium condition, instead of the xenon-free condition that should have been used. The result was an estimated critical position that was 28 cm (11.04 inches) vs. the correct value of 21.8 cm (8.60 inches). As the startup progressed, the operator observed an increasingly short period, but attributed it to noise on the instrument channels. The operator and supervisor also failed to recognize the significance of several other indicators approaching criticality including the need to upscale both the startup channels and an audible count rate meter. The trainee had operated the reactor on the unusually short reactor period of 10.4 seconds, contrary to operating procedures that allowed a minimum 30.0 second period, and an automatic safety system shutdown occurred at 340 watts. The technical specification limiting safety system setting was a 3 second period.

The reactor supervisor reported the cause of the shutdown as instrument noise and obtained approval for a restart from the Reactor Superintendent. During the restart, the operator and supervisor again observed an increasingly short period. This time they recognized it as a true signal and immediately made the reactor subcritical in accordance with existing written procedures covering an error in the estimated critical position calculation. The Reactor Superintendent was again notified, the reactor was shut down, and the Director of Reactor Operations was notified.

Cause of Event:

Human error was the cause of the event. The console operator and reactor supervisor had not checked the second portion of the trainee's estimated critical position calculation, placed excessive reliance on the estimated critical position as the means of identifying criticality, did not notice the indications that the reactor was approaching criticality, and attributed the short period indication to instrument noise.

Licensee Corrective Actions:

Licensee management stressed the need for attention to details for safe operation, the importance of not relying on trainees, and the need to investigate all abnormal instrument readings to all operators. The reactor startup procedure was modified to require two independent calculations of the estimated critical position by licensed operators. Electronics personnel were directed to investigate noise effects on the reactor startup instrument.

Event A33

Facility: Massachusetts Institute of Technology (5 MW)

Event Date: December 7, 1993

Event Description:

The reactor automatically scrammed while a technician was investigating why one of two low-range neutron flux power level safety channels required by technical specification for natural circulation operation below 100 kW had failed to 0 during operation at 50 kW.

Cause of Event:

The low-range amplifier circuit was found to have failed.

Licensee Corrective Actions:

The low-range amplifier circuit was replaced. Corrective actions also included modifying the existing low-range amplifiers so they were no longer dual-range to allow all of the circuitry associated with each amplifier to be tested without removal of the input signal cable at the rear of each amplifier.

Event A34

Facility: Massachusetts Institute of Technology (5 MW)

Event Date: January 25, 1995

Event Description:

During a survey visit, it was learned that the reactor was operated with one of three normally in service core low flow scrams inoperable (two are required). An operator addressed a core temperature pin oscillation on a recorder during a shutdown by inadvertently turning off both the core flow and temperature recorder instead of just the temperature pin. After starting up, the reactor operator turned the recorder back on and the low flow signal scrammed the reactor. The operators noted the occurrence in the operating log book.

Cause of Event:

Malfunctioning temperature recorder and operator error. Although the restart procedure contained a step to restart any instrument that had been secured, the operators did not have a list itemizing which instruments were secured.

Licensee Corrective Actions:

This was not a reportable event. Corrective actions taken by the licensee were not reported to the NRC.

APPENDIX B

SELECTED CORE DAMAGE EVENTS AT EARLY EXPERIMENTAL OR TEST REACTORS

Note: References to other documents in this appendix appear at the end of this appendix.

1 Early Fuel Damage Incidents to Test Reactors*

Several test reactors have experienced fuel damage due to inlet flow blockages, dating back to the mid-1950s.

Materials Testing Reactor

The Materials Testing Reactor (40 MW) actually had two such events. In 1954, some fuel buckled and there was some local melting. Then, in 1962, one plate partially melted due to some rubber seal material becoming lodged in a coolant channel.

Engineering Test Reactor

According to Thompson (Ref. 1), the Engineering Test Reactor (ETR) also had an incident of fuel melt as a result of flow-restriction in 1961. A plastic sight-glass had been left in the reactor tank. Eighteen plates, in six elements, melted.

Oak Ridge Research Reactor

There was an incident at the Oak Ridge Research Reactor (ORR) (30 MW) (Ref. 2) on July 1, 1963. The reactor was at 24 MW at the beginning of cycle startup. A large neoprene gasket had become lodged in the upper end box of a fuel element. The procedure which was followed called for an inspection of the core for foreign objects as part of the startup sequence. However, the affected bundle could not be seen easily. During the power ascent past 9 and 12 MW, there was some slight increase in noise on the nuclear channels. In retrospect, it was concluded that boiling was occurring at this juncture. At 24 MW the neutron noise increased considerably; so did the radiation monitors. Power was reduced to 12 MW. Radiation levels further increased, and the reactor was scrammed. Water radioactivity had increased by factors of about 200. Subsequent examination showed one plate in one element had some melting, corresponding to about 30 to 50 percent of the fueled area, or about 3 to 5 grams of fuel. About 150-200 mCi of iodine was released at the stack. Several direct causes were noted: the gasket lodged in the system; failure to detect this during visual examination; and, failure to recognize boiling. The partial fuel melt resulted in a minor release to the site, but no personnel over exposures.

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* It is recognized that extreme events at early experimental or test reactors have only very limited implications for the normal operation of current designs used for non-power reactors since this experience has been considered in the development and approval of current designs.

Westinghouse Testing Reactor

On April 3, 1960, there was a fuel element failure at the Westinghouse Testing Reactor (60 MW) at Waltz Mills, Pa. (Ref. 3). This was a modified MTR-type fuel system. There was a release of fission products to the primary coolant system, and some gaseous fission products released to the atmosphere. The reactor was in the process of being uprated from 20 MW to 60 MW. Some low flow boiling tests had been performed to explore the limits of operation. While the reactor was at 30 MW, the primary flow rate was deliberately reduced, but no boiling was observed. The power was then raised to 34 MW and leveled off. However, the power level started to drop; the shift supervisor ordered rods to be withdrawn to raise power to 40 MW. While proceeding to 40 MW, there were radiation alarms, and soon thereafter the reactor was scrammed.

Apparently, at one location, there was a poor metallurgical bond between the fuel matrix and cladding such that most of the heat generated had to be conducted through only one side of the fuel, instead of both. As a result, there was localized melting. The molten material, in turn, blocked a flow channel. Upon subsequent review of this event, it was noted that the operations staff should not have continued to increase power following the initial downturn in power that was experienced. Rather, a questioning attitude might have prevented the fuel failure altogether.

2 Reactivity Experiments Using Test Reactors

BORAX-1

A series of experiments was performed on the Borax-1 reactor (no thermal rating) in 1954 to investigate subcooled reactivity transients (Ref. 4). The final runaway experiment was a rod ejection in the amount of 4 percent reactivity. The resulting reactor period was 2.6 milliseconds, and about 135 MW-seconds of energy was released. Most of the fuel plates melted, and the reactor tank failed. It is possible that some of the damage resulted from a subsequent steam explosion (with the molten aluminum).

Self-limiting Power Excursion Tests

The Self-limiting Power Excursion Test (SPERT) program consisted of several facilities designed to explore reactivity transients. Silver (Ref. 5) provides a status report. In 1962, the first destructive test on SPERT 1 (no thermal rating) was done. A 3.5$ reactivity insertion was accomplished, with a resultant reactor period of 3.2 milliseconds. After the first power peak decayed, apparently there was an unexpected secondary reaction of water and molten aluminum (i.e., steam explosion) which produced a large pressure pulse. The core was destroyed.

SPERT-IV (1 MW) was a large pool facility investigating instability from reactivity transients for pool-type reactors. Tests were done with forced flow as well as stagnant conditions. Further information was provided on SPERT-IV by Silver in 1965 (Ref. 6). In SPERT-IV, the effects of various water heads and forced flow were investigated in the context of reactivity transients. The most significant result was the observation of oscillatory reactor power behavior following an initial burst. With increasing flow rate, both the frequency and amplitude of the oscillations increased. It appears that under some conditions a 4$ reactivity insertion (above critical) can produce oscillations at 2 Hz, with divergent amplitudes.

3 Reactivity Excursion Accident at the Stationary Low Power Plant No. 1

An unplanned reactivity event occurred at the Stationary Low Power Plant No. 1 (SL-1) (3 MW) reactor in January 1961 (Ref. 7). It was estimated that 2.4 percent excess reactivity was inserted, producing a 4 millisecond period. Fuel was probably vaporized in the center of the fuel plate. The total energy generated was on the order of 130 MW-seconds. The steam that was produced resulted in the acceleration of water from the core and deceleration at the lid of the pressure vessel, which in turn lifted the vessel nine feet. Piping was sheared along with the lid shielding. About 20 percent of the core was destroyed. About five percent of the gross fission products were ejected from the vessel. Apparently the steam pressure was about 3401 kPa (500 psi). Three operators on top of the tank at the time were killed.

Within the first hour of the event, the dose rate just outside the building was 2.5 mSeivert (250 mrem) per hour, and was more than 5.0 Seivert (500 rem) per hour near the reactor floor (Ref. 8). According to Thompson (Ref. 2) 14 people received occupational doses in excess of 50 mSeivert (5 rem). Four days after the event, the dose at the guard house (about 61 meters [200 feet] from the reactor building) was about 0.25 mSeivert (25 mrem) per hour. At 609.6 meters (2000 feet) in all directions the dose rate was less than 0.02 mSeivert (2 mrem) per hour. In all, about 100 curies of I-131 was released to the environment over a several day period.

Appendix B References

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4. Dietrich, J. R., Experimental Investigation of the Self-limitation of Power During Reactivity Transients in a Subcooled, Water-moderated Reactor, AECD-3668.

5. Silver, E. G., SPERT Program Status Report, Nuclear Safety, Vol. 5, No. 1 (Fall 1963).

6. Silver, E. G., Status Report on SPERT Program, Including Power Burst Facility, Nuclear Safety, Vol. 6, No. 3 (1965).

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