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ORNL-NSIC-64

UC-80 — Reactor Technology

ABNORMAL REACTOR OPERATING EXPERIENCES 1966—1968

U. S. Atomic Energy Commission
Division of Reactor Licensing

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ABNORMAL REACTOR OPERATING EXPERIENCES 1966-1968

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
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FOREWORD

The Nuclear Safety Information Center is pleased to publish a second compilation of unusual operating experiences collected by the U. S. Atomic Energy Commission. The earlier report, ORNL-NSIC-17, entitled "Abnormal Reactor Operating Experiences" is still available for $3.00 per copy from the Clearinghouse for Federal Scientific and Technical Information (see page ii for CFSTI address).

To make this report more useful, the experiences were indexed, using keywords from the NSIC thesaurus of indexing terms. Open-literature references were added to the experiences whenever possible.

E. N. Cramer and H. B. Whetsel
Nuclear Safety Information Center

PREFACE

For a number of years the U.S. Atomic Energy Commission has published selected accounts of unusual events at reactor facilities. This has been done by the continuing series of Reactor Operating Experience (ROE) reports to individuals on a mailing list which currently numbers nearly 1400. In August 1966, the published ROE's through calendar year 1965 were consolidated in a report (ORNL-NSIC-17) prepared by the Nuclear Safety Information Center, with the thought that consolidation of such specific evaluations can lead to improvements in operational safety at reactor facilities and, on occasion, can lead to identifying and contributing to the solution of more generic reactor design problems.

In the summer of 1967, the ROE program was broadened to incorporate publication of information concerning unusual events occurring at licensed reactor facilities. Since then, the program has grown at a pace which reflects the growth in the licensed segment of the reactor industry. In early 1969, responsibility for administration of the program was transferred from the Division of Operational Safety to the Division of Reactor Licensing.

The present report serves to update ORNL-NSIC-17, and consolidates all ROE's published in the period 1966-68. The report is by no means comprehensive; far more unusual events of potential significance occurred than those included in the ROE program. Nevertheless, as representative samples, these experiences provide valuable insight into the types of operational problems which can, and in some cases, have raised safety questions of concern to both designers and operators of reactors.

We wish to acknowledge the invaluable assistance of the staffs of the AEC's field offices, contractor and licensee organizations, regional offices of the Division of Compliance, and the headquarters personnel in the Divisions of Operational Safety and Reactor Licensing, in the identification, preparation, editing and processing of these reports.

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Division of Reactor Licensing
U.S. Atomic Energy Commission

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ROE 66-1

FUEL ELEMENT RUPTURE TEST FACILITY INCIDENT AT THE PLUTONIUM RECYCLE TEST REACTOR

Summary

On September 29, 1965, the Plutonium Recycle Test Reactor (PRTR) was shut down when a fuel element failure and concomitant pressure tube failure occurred in the Fuel Element Rupture Test Facility (FERTF), a pressurized light water loop in the center of the heavy-water-moderated and cooled PRTR core. Substantial amounts of fission products were released to the test loop and the reactor core. Fission product activity spread within the containment vessel activating the containment system and requiring evacuation of the containment vessel for a period of about 12 hours. No injuries or overexposures occurred and no appreciable radioactivity was released to the environment. The heavy water was degraded.

Circumstances

The FERTF is used as a pilot irradiation facility for (1) new fuel element manufacturing processes, (2) new operating regimes, and (3) fuel defect tests, because of its separate cooling system and its capability for safe handling gross release of fission product without contaminating the primary coolant of the reactor.

At the time of the incident, a fuel element with an intentionally defected UO_2-4 wt% PuO_2 fuel rod was being irradiated in the FERTF. The facility was operating at nominal conditions of 500°F outlet temperature, 1000 psi inlet pressure, and test section power of 1765 kw. The test section was reaching equilibrium conditions following a series of temperature cycling tests which consisted of lowering the test section outlet temperature from 500°F to 280°F in nominally 50°F steps, holding at each step for temperature equilibrium. The PRTR was operating at a nominal power of 65.3 Mw with the power level being controlled automatically by varying the heavy water moderator level in the calendria.

At 5:10 a.m., September 29, 1965, the PRTR was automatically shut down due to a scram signal from the FERTF safety instrumentation. Post-incident examination showed that the pressure tube had ruptured in the vicinity of the defect in the fuel rod. The defected fuel rod had ruptured in gross manner and had lost approximately 700 grams of fuel material to the loop and to the PRTR system.

The calendria shroud tube surrounding the pressure tube was found to be intact after the incident. It is postulated that the steam-water mixture carrying fission product activity from the failed pressure tube entered the reactor dry gas system between the pressure tube and the concentric shroud tube. From here the activity spread to the containment

vessel ventilation system via seals in the dry gas system. The major inflow of H_2O from the FERTF to the calandria was through the shim rod and temperature probe seals on the calandria top tube sheet and through the interconnecting dry gas piping system.

Results

The ruptured region of the fuel tube was approximately 3-in. by 1/2-in. Considerable fuel (705 grams) was ejected from this enlarged defect. The pressure tube failed at a point adjacent to the fuel rod rupture. The shroud tube was apparently undamaged although a small amount of material was deposited on it. Considerable fission product contamination spread to the test loop and the PRTR core.

Some fission product activity spread within the containment vessel. The PRTR heavy water was degraded. No injuries or overexposures occurred and no appreciable radioactivity was released to the environment.

Conclusions

The basic mechanisms and interactions which occurred within the defected fuel rod and between the rod and the pressure tube involve a complicated and not fully understood sequence of events. Based upon the results of postmortem examinations, the following mechanisms and sequence of events are hypothesized to have occurred:

a. Preirradiated fuel rod RF-4 was defected by drilling a hole through the cladding.

b. Water entered the rod during loop activation and operation.

c. The UO_2-PuO_2 fuel oxidized during irradiation, causing the O:U ratio to increase from 2.01 to 2.1. This caused a decrease in the thermal conductivity and possibly the melting temperature of the fuel. Very recent data indicate that in the presence of water vapor, UO_{2+x} might exist at higher temperatures than previously considered possible, and that the formation can be very rapid.

d. An unexpectedly large amount of fuel melting occurred because of the reduced thermal conductivity and lower melting point of the oxidized fuel.

e. The volume expansion of the unexpectedly large amount of molten fuel could not be accommodated by the void space within the fuel rod and resulted in high internal hydrostatic pressures.

f. The high internal hydrostatic pressures and higher than normal cladding temperatures caused the cladding to swell locally and split in a ductile manner in the vicinity of the defect.

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g. High temperature fuel contacted the pressure tube in the vicinity of the ductile split. (The pressure tube did not fail at this point, probably because of coolant accessibility to the point of contact.) This caused steam formation at this point and immediately downstream for approximately 24 seconds before the pressure tube failed.

h. The localized swelling of the defected rod and steam formation caused a heat transfer disruption that permitted overheating of a larger area of downstream cladding during the 24 second period.

i. As the region of the fuel rod immediately downstream of the swelled area overheated due to steam blanketing, the adjacent Zircalloy pressure tube overheated and failed under pressure. There is no evidence of fuel contacting the pressure tube in the immediate vicinity of the rupture, and there was no evidence of fuel loss to the FERTF coolant prior to the failure of the pressure tube.

j. After the pressure tube failed, fuel (mostly molten) was forceably ejected from the rod under the pressure differential created between the inside of the rod and the decreased FERTF coolant pressure. Not all the fuel escaped through the hole in the pressure tube since an estimated 75% of the 700 g. of fuel ejected from the fuel rod remained in the rupture loop system.

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ROE 66-2

MALFUNCTION OF EQUIPMENT INVOLVING SHUTDOWN RODS IN THE PROCESS DEVELOPMENT PILE (PDP)

Summary

During a standard startup of the PDP, an equipment malfunction resulted in a single gang (5 rods) of shutdown rods being withdrawn beyond its normal limit above the top of the pile, breaking the drive cables and dropping the rods onto the tank top. Six months later during a routine shutdown of the same reactor, another equipment malfunction occurred on a different gang (4 rods) of shutdown rods, resulting in the motor continuing to drive the drums after the gang was fully inserted. This caused the cables to unwind completely and then rewind on the drum. The gang was thus again withdrawn beyond its normal limit, with the same consequences.

Circumstances

In the first instance, the operator noted that no up-limit light was obtained for the No. 2 gang during the withdrawal of the No. 1 and No. 2 gangs of shutdown rods. As soon as the nature of the problem was recognized, the moderator drain was opened, the safety rods were dropped, gangs 1, 3, and 4 of shutdown rods were driven back into the reactor, and supervision was notified.

In the second instance, the pile run was normal up to the scheduled shutdown. The shutdown was accomplished in the normal manner by turning a key in the interlock system to drive in all the control, safety, and shutdown rods and to dump the heavy water. During the post-shutdown procedures, the operator noted that the indicating lights on the shutdown rod gangs indicated that only three of the four gangs were fully inserted.

In both instances, inspection revealed that misalignment of the devices which activate the limit switches was the cause of the malfunction, although the nature of the misalighment was different in each case.

Results

The results in both instances were the same. The shutdown cable drive motor continued to run when not cut off by a limit switch, winding the cables until the rods passed their normal up-limit and until the top of the rods contacted the guides through which the cables pass, breaking the cables and dropping the rods onto the tank top. The total damage in each incident was about $30 for the replacement of the broken cables. Neither the rods nor the tank top were damaged.

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In neither case was an excursion possible. There are four independent mechanisms for reactivity change: safety, shutdown & control rods, and moderator height. Only the latter two are in use at or near critical. Thus, in the first instance, the pile was subcritical when the gang in question reached its normal upper limit. Withdrawing this gang further did not add significant reactivity. In the second instance, the malfunctioning gang was the only movable poison out of the core at the time its behavior first became abnormal.

Corrective Action

After the first instance, the limit switch actuating arm was modified to prevent recurrence of the difficulty encountered. In addition, action was initiated to design a completely separate and independent cut-off mechanism which would stop the cable drive motor in the event all other limit switches became inoperable. A prototype installation was made on the shutdown rod gang which failed. It was planned to install this system on the other three drives as soon as the new system had proved satisfactory under operating conditions. Had this installation been completed, the second incident could not have occurred even though the possibility for the second type of failure had not been recognized when the new cut-off mechanism was conceived.

After the second occurrence, the independent cut-off mechanism was installed on all four systems. In addition, special procedures were issued which require that an operator watch the movement of the shutdown rods during each startup and shutdown in order to visually detect any malfunction in their operation. Steps are being taken to further redesign the limit switch actuator mechanisms. Consideration is also being given to the feasibility and practicality of modifying the cable drums to release the cables upon full unwind as a further backup against a rewinding incident. And, the maintenance program is being reevaluated to make certain that there is no laxness in the performance of the work.

Conclusions

Although the two incidents were not precisely identical, the basic point common to both difficulties appears to be that the design of the limit switch mechanism did not adequately recognize the possibilities for mechanical interference between components. A contributing cause in either case could be personnel error in making and completing adjustments to the system.

Implementation of the previously discussed corrective action to prevent physical interference and to provide a positive backup cut-off switch should prevent any future recurrences.

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Division of Operational Safety, Malfunction of Equipment Involving Shutdown Rods in the Process Development Pile (PDP), USAEC Report TID-24698, U. S. Atomic Energy Commission, Aug. 5, 1966.

ROE 66-3

HIGH-PRESSURE RELEASE IN AN EIGHT-INCH TUBE OF THE PUERTO RICO RESEARCH REACTOR

Summary

On February 16, 1966, a high-pressure release occurred in one of the eight-inch beam tubes of the 1 Mw Puerto Rico Nuclear Center Research Reactor. The shield plug nearest the reactor core in the beam tube was found to be ruptured and the pool end of the beam tube was blown off. No damage to the reactor core or vital reactor components occurred due to this incident; no release of fission products and no injuries to personnel ensued.

Circumstances

At 12:45 p.m. the reactor scrammed, apparently from the period instrument. At the same time, a slight shock was heard in the building, but it was believed that the shock came from outside the building. A visual examination was made of the reactor components and nothing abnormal was noted. The reactor was, therefore, restarted and apparently operated normally until about 9:00 p.m. when gas bubbles were observed in the pool. The reactor was shut down and further investigation revealed that some sort of explosion had taken place in one of the eight-inch beam tubes.

The incident took place at the reactor end of the beam tube shield plug which was made of solid barytes concrete completely encased in quarter-inch-thick aluminum. The plug was located in the beam tube a few inches from the reactor core, although it was unintentional that the plug was in this location. The force of the explosion broke the circumferential weld of the end disk on the core side of the shield plug jacket and also broke out the welded disk of the beam tube itself, driving this disk up against the core fuel elements. The concrete in the shield plug was shattered, leaving a shallow conical hole.

At the time of the incident, the beam tube was not being used and only the shield plug was present in the reactor end of the tube. A plate at the outside end of the beam tube prevented water from leaking out of the pool.

The reactor was moved to the far end of the reactor pool, a gate was installed, and the experimental section of the pool was drained of water. At the same time, several of the reactor fuel elements nearest the beam tube at the time of the incident were moved to the hot cell and visually examined. There was no apparent damage to the fuel elements and no measurable release of activity. The damaged shield plug was removed from the beam tube, and studies were initiated to try to determine the cause of the high-pressure release. At the time of the removal (about 21 hours after rupture), the plug and its jacket measured 50 R/hr

at contact. A detectable but insignificant concentration of activity, presumably due to materials from the concrete, was found in the reactor pool water.

The estimated maximum dose received by the concrete in the beam tube plug was 7×10^{10} rad. This dose is based on the following assumptions:

1. The plug was adjacent to the core surface of the beam tube during the entire period of exposure to high intensity radiation.
2. The reactor had been operated at 1 Mw for about 350 hours while the plug was in this position.

Since the neutron dose was found to be negligible, the total dose received was due essentially to the gamma rays.

An identical plug had been located in a second eight-inch beam tube of the reactor in an equivalent location with respect to the core. Precautions were taken against personnel and property damage while removing this second shield plug. It was not swelled significantly, and, when a hole was drilled through the aluminum jacket of the plug, there was no evidence of internal pressure.

Causes

The cause of the pressure buildup is still not known definitely, although several possibilities have been considered. One possible cause is the formation of hydrogen and oxygen gas within the aluminum jacket from radiolysis of water, the water being either water of hydration of the concrete, or leakage water from a pin hole in the aluminum jacket. Another possibility is gas formation by corrosion at the aluminum-concrete interface.

The fact that the plug was free to move into the high flux region may have increased the chance of a high-pressure release. Experiments are planned in which a fully instrumented shield plug will be subjected to radiation under continuous observation to study the generation of gases under these conditions.

Conclusions

Although there was no fission product releases or personnel injuries as a result of this incident, it could have been potentially a very serious incident. The reactor is now back in operation at its normal position. At this time, shield plugs are located in only the normal shielding positions at the outside ends of the beam tubes. The actions of the personnel involved were commendable in keeping damage to a minimum and preventing personnel injuries.

ROE 66-3

Corrective Actions

In the future, all beam tube inserts will be secured against motion due to vibration or other uncontrolled causes. All shield plugs will have holes drilled in their aluminum jackets to prevent buildup of gases. In addition, beam tube stops will be used to prevent the shield plugs from moving too close to the reactor core.

ROE 66-4

BEARING WEAR PROBLEM ON THE HIGH FLUX ISOTOPE REACTOR CONTROL PLATES

Circumstances

The High Flux Isotope Reactor (HFIR) control plates are located in a 0.869-in.-thick annular region between the outer fuel element, 17.124 in. o.d., and the removable beryllium (Fig. 66-4-1). The control plates consist of two 1/4-in.-thick concentric cylinders which are separated from each other and from the fuel and reflector regions by three coolant channel annuli ranging in thickness from approximately 0.1 to 0.17 in. through which water flows at a velocity of about 16 fps. The inner cylinder, which is a single piece, is used as a shim and a regulating rod. The outer cylinder is divided into four quadrants, each used as a shim-safety rod and each having its own drive rod and scram mechanism. Reactivity is added by lowering the inner cylinder and/or raising the outer shim-safety plates.

The cylinder and plates are driven from beneath the reactor by drive rods which extend into the subpile room where the drive mechanisms are located. Guidance for each control plate is furnished by six Stellite ball bearings. Four bearings are attached to each outer control plate, two at each end, and are run in stationary tracks which extend above and below the core. The other two bearings are attached to the lower track assembly and bear against the control plate. The single inner control cylinder is guided by eight bearings, four of which are attached to the track assemblies at each end of the reflector and extend through slots between the four outer control plates.

During the shutdown terminating the first 100-Mw cycle, ten 3/16-in.-diameter balls were found in the primary system strainer. An inspection of the control plates in the pool revealed that the source of these balls was one of the upper bearings of the No. 4 control plate. It was also apparent that some other upper bearings had worn excessively. All four control plates and the inner control cylinder were removed from the reactor. The race on the failed bearing was retrieved as the control plate was removed. By using a periscope, a detailed inspection of all the control-plate bearings was made and indicated that considerable wear had been incurred on all the upper bearings of the control plates; and, in fact, a race and its companion balls were missing from the upper bearing of a No. 3 plate. This race and three balls were found in the pool, indicating that this bearing came apart after being removed from the reactor. There was some evidence of wear on the lower control plate bearings but to a much lesser degree than the upper bearings. The bearings attached to the track assemblies were inspected and found to be in good condition.

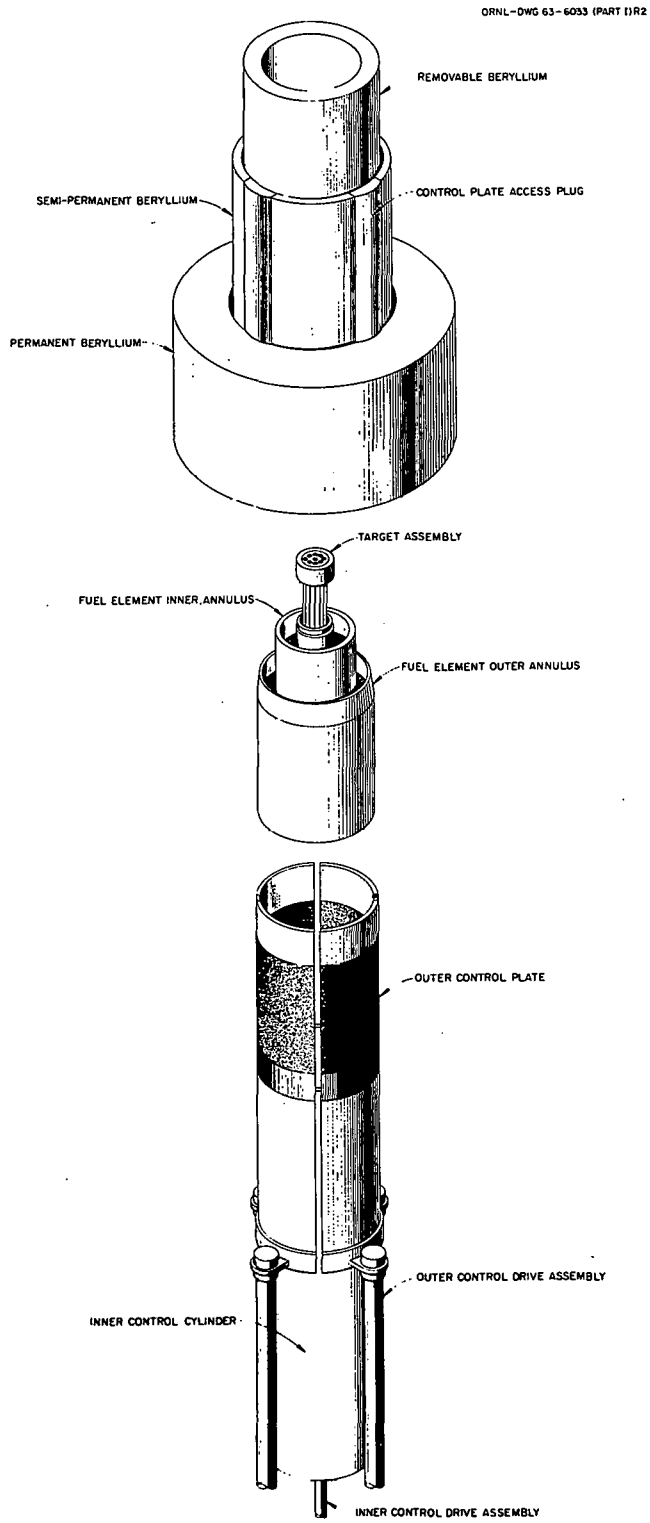

Fig. 66-4-1. HFIR Element, Exploded View.

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After the primary coolant was circulated at full flow for approximately one hour, the strainer was inspected again. Six additional balls were found in the strainer. Of the 26 balls missing from the two bearings, 19 have been accounted for--16 recovered from the strainer and 3 found in the pool. The balls unaccounted for are of no real concern, as it has been demonstrated that they are small enough to pass through the control region; and it is improbable that they can cause any trouble.

Cause

The excessive wear of the control bearings was probably caused by fretting corrosion. Plate vibration apparently results from the turbulence of the cooling water which enters the reactor vessel just above the upper end of the control plates. This vibration is probably maximum at the end of the fuel cycle when the plates are in the upper portion of their travel.

The top bearings showed much more wear than the bottom bearings. Vibration of the bottom end of the plates is reduced by spring loaded bearings which maintain contact with the lower end of the plates. Also, there is probably less turbulence in this region.

Conclusions

While the wear rate on the Stellite bearings was somewhat greater than anticipated, it should be noted that these plates, which were in use during the hydraulic and low-power testing, had been in the reactor almost twice as long as their design life. However, the accumulated power of the reactor was less than the amount which would result from operation at full power during the design life of the plates.

A routine time-of-flight check, made on all the control plates just after terminating the 100-Mw cycle, and before discovering that one of the bearings had failed, was normal, indicating that the scram response of the safety plates was not affected even though the bearings had worn excessively.

Corrective Action

To prevent a recurrence of a bearing failing while in the reactor, a new set of control plates was installed on which the bearings were modified to provide retainers for both the balls and the race. Routine tests of the time-of-flight of the new plates were normal. While this change does not alleviate the wear problem, it does prevent the bearings from coming apart. More frequent inspections of the bearings will be made to observe the rate of wear and plates will be changed before the wear becomes excessive. Also, the method of attaching the bearings to the control plates was modified to provide an easier means of replacing a bearing remotely when this becomes necessary. For the longer term, work is under way to develop better bearings and to investigate the mechanism of the wearing process.

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ROE 67-1

FAILURE OF EMERGENCY BATTERY BANK AT THE MATERIALS TESTING REACTOR

Summary

On October 4, 1966, a commercial power failure at the Materials Testing Reactor (MTR) revealed that the emergency battery bank, designed to supply uninterrupted power to vital experimental loads upon loss of commercial power, had far less than rated capacity. Routine tests and preventive maintenance checks in use prior to this occurrence included specific gravity and voltage measurements which did not detect deterioration of the batteries. In the future full discharge tests will be made periodically under rated load conditions. The loss of power was for such a short duration that no damage to equipment resulted.

Circumstances

The failure-free power system, for experiments at the MTR, supplies important experiment loads with commercial power under normal conditions. It also includes a motor-generator set which keeps a lead-acid battery bank fully charged. In the event of a commercial power disturbance, breakers isolate the system from outside power sources, the motor-generator set inverts and, driven by the bank, supplies uninterrupted power for about 30 minutes. During this time, the system can be manually returned to commercial power or to the emergency diesel system. In this particular case, the commercial outage was of a brief duration. The batteries had carried the load only a small fraction of the rated 30 minutes. The operator attempted to synchronize the motor-generator set with commercial power and return the system to normal. He was unable to synchronize properly, and a breaker in the DC portion of the system was opened, cutting off all power. The system was promptly switched to commercial power so that the loss-of-experiment power lasted approximately one minute.

Investigation showed that the battery bank voltage had dropped so low, with a resulting reduction in motor-generator speed, that synchronization was not possible.

Results

The reactor was scrammed by the commercial power failure. Battery capacity was sufficient to continue experiment coolant flow as required for several minutes. The one minute interruption of power did not result in damage to experiments. Safety of the reactor was not in danger since this emergency power system is not required for protection of the reactor.

ROE 67-1

Causes

After investigation and discussion with the battery manufacturer, it was decided that the battery plates had become coated with lead sulfate to the point that the battery capacity was seriously reduced. The battery bank was about ten years old and nearing the end of its rated life. While this condition had probably been some time in developing, it had not been detected. Routine tests had been limited to voltage and specific gravity checks which had appeared normal. The system had been required to operate occasionally on commercial power failures, but in most cases, prompt switching resulted in battery discharge times so short that no real test of battery capacity resulted.

Corrective Action

A temporary battery bank replacement was installed and procurement of a new permanent battery bank initiated. Preventive maintenance procedures for all test reactor area emergency battery banks have been modified to include a full discharge test periodically under rated load. This has been completed for all operational batteries, and they were found to be in satisfactory condition.

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ROE 67-2

DISPLACED ENGINEERING TEST REACTOR FUEL ELEMENT

Summary

On November 17, 1966, the Engineering Test Reactor (ETR) was taken critical with one fuel element inserted only nine inches into the core. If conditions had permitted the fuel element to slip into the core, it would have represented an appreciable reactivity insertion. A reactivity discrepancy was noted when the reactor was started up, and the displaced fuel element was discovered during the resulting investigation.

Circumstances

The ETR was operated for a low-power gamma heat run on November 16, 1966, following which some experiment changes were made. For this work the Standard Practices Manual required removal of four fuel elements for reactivity control. One of the four elements removed was in the K-10 core position. Later, when the four elements were reinserted, considerable difficulty was experienced in installing the element in the K-8 position. This is not unusual at ETR where slight misalignment of one of the loop in-pile tubes can often make fuel element insertion difficult. Eventually, the core assembly was completed, normal startup preparations made, and reactor startup initiated.

During the approach to critical, it appeared that there was a significant difference between the control rod positions observed during the startup and the corresponding rod positions during the preceding gamma heat run. Several checks were made; but since nothing abnormal could be discovered, the reactor was taken to N_L (1% of full power) at 0338 on November 17. At this point, the reactivity discrepancy was verified; and the reactor was held at this level while possible explanations were investigated. Finally, it was apparent that the discrepancy was unexplainable, and the reactor was scrammed at 1107. When the core was inspected, it was discovered that the K-10 fuel element was only inserted nine inches into the core, leaving the entire fueled portion above the core. Initial attempts to insert the element were unsuccessful indicating that there was little likelihood of the element falling into place accidentally. Further efforts to seat the element were abandoned when it was decided to replace the entire core for programmatic reasons.

Causes

The K-10 fuel element had been logged as properly seated by one shift. The following shift had completed a pre-startup check list which included a check of the seating of all fuel elements. Therefore, consideration was given to any possible mechanism by which the fuel element could have been lifted out of the core after the reactor top head was installed. It was concluded that no such mechanism existed, and that the

fuel element had been left elevated by personnel error. The exact circumstances have not been determined, but it is postulated that the K-10 element was lifted to permit easier insertion of the K-8 element and then was overlooked. The K-10 position is surrounded on two sides by loop in-pile tubes and on the other two sides by control rod guide tubes, making observation difficult.

Corrective Action

Pre-startup procedures have been revised to require that the position of each fuel element be determined by physical measurement under the direct supervision of the Shift Supervisor.

Conclusion

It has been concluded that the K-10 fuel element was left elevated during reactor startup because of personnel error, but that in this case there was essentially no possibility of the fuel element falling into the core, producing an excursion.

Following this occurrence, the worth of the K-10 fuel element was measured in the ETR Critical Facility as 1.5% $\Delta K/K$. A sudden reactivity insertion of this magnitude was not considered in the original ETR safety analysis, so a calculation was performed to determine what consequences would have resulted from sudden insertion of this element under full hydraulic acceleration with the reactor at N_L. If the period scrams operated properly, the excursion would have been limited to a peak power of about 4 MW and an energy release of about 0.5 MW-sec. This would not have damaged the reactor. If the period trips should not function, a level trip would be encountered at 24.5 MW. The peak power and energy release for the excursion without period scrams were calculated to be 1390 MW and 61.6 MW-sec., respectively. In this case, it is probably that some reactor damage would result.

References

E. H. Smith et al., Progress Report for Cycle No. 85, November 6, 1966—December 15, 1966 — ETR Operations Branch, USAEC Report IN-1059, Idaho Nuclear Corporation, Jan. 31, 1967.

ROE 67-3

LOSS OF REACTIVITY AT THE LOS ALAMOS WATER BOILER REACTOR (SUPO)

Summary

On August 15, 1966, a routine pre-startup reactivity measurement disclosed a reactivity loss of about 0.6% ΔK/K. This reactivity loss was large enough to prevent the reactor from being operated at full power. After checking many possible causes, it is believed that the most likely reason for the reactivity loss is the dispostion of a small amount of water in the graphite reflector.

Circumstances

On Monday, August 15, 1966, a pre-startup reactivity measurement indicated a reactivity loss of about 0.6% ΔK/K. A reactivity measurement is performed routinely the first time the reactor is brought to power during the week. The measurements have been done for many years and normally show only about ± 0.07% ΔK/K variation. The reactor had operated at full power the preceding Friday and at that time there was no evidence of a problem. During the next thirty days, daily measurements of reactivity showed a continued decrease. The lowest reactivity measured was 1.06% ΔK/K less than the values obtained prior to August 15. Also when sufficient uranium was added to compensate for the reactivity loss and the reactor was brought to full power, it was found that the actual reactor power obtained from the temperature and flow of the water coolant was considerably greater than the power indicated by the current from the control chambers.

Discussion

After checking many possible reasons for the reactivity loss such as uranium precipitation, solution loss, control rod malfunction, etc., it was finally concluded that the most likely reason for the loss was the deposition of water in the graphite reflector. A summary of the observations which led to this conclusion follows. When the ports were disassembled to see if there was anything near the reactor core which might account for the reactivity loss, moisture was found in some of the ports which enter the graphite reflector. The ports are inspected every three months and have always been found to be dry. At the time, it was felt that the small amount of water observed could not account for the reactivity loss. Later, after investigation of the more obvious possibilities failed to show a reason for the reactivity loss, a flux map of the core and adjacent graphite reflector was obtained. A comparison of this map with those obtained prior to August 15 showed marked flux peaking in the graphite where no such peaking had occurred before. The information obtained from the flux map furnished the incentive to make flux and reactivity calculations with and without a small amount of water added to the reflector. The results of the calculations showed that 2%

(by volume) of water added to the reflector would approximately reproduce the observed flux peaking and measured reactivity loss. The calculations also showed that the addition of water to the reflector decreased the flux at distances comparable to the position of the control chambers, thus accounting for the increase in actual reactor power relative to chamber current. To test how much water the reflector could reasonably retain, a small piece of graphite similar to the graphite used for the reflector was soaked in water for several days. The difference in weight of the piece before and after soaking corresponded to an uptake of water of about 2% by volume.

Several attempts at drying the accessible regions of the graphite reflector have recovered about half of the reactivity loss. Extensive investigation has failed to reveal the source of water. Between October 1966 and March 1, 1967, no drying efforts were made. During this period the reactivity (measured at least three times a week) was more or less constant with perhaps a slight upward trend. This prolonged test indicates either that the source of water is no longer present and the rate of natural removal of water from the graphite is quite small, or that there is a small leak in equilibrium with the natural removal rate. Further drying operations were resumed on March 2, 1967. To date, these efforts have yielded very little additional reactivity increase.

ROE 67-4

REACTIVITY SHIFT IN SPERT-III REACTOR

Summary

Routine control rod critical position checks at the SPERT III reactor indicated an erratically varying critical position. The observed variances took place over a period of several weeks and had magnitudes corresponding to reactivity changes of up to about 20¢. Disassembly and inspection confirmed the postulation that a flux suppressor in the control rod follower was loose.

Circumstances

Critical position checks before and after a test in the SPERT III reactor indicated that about 15¢ in reactivity had been lost during the test. Limited operation of the reactor was proposed in an attempt to determine the cause of this change. During these operations other changes in the control rod critical position were observed. In the meantime a study was undertaken to find a plausable reason for the anomalous behavior. As a result it was postulated that a flux suppressor in one of the control rod followers may have become displaced. This condition could cause the observed behavior since the flux suppressor would then be free to move in a vertical direction within the fuel follower section.

Critical experiments, designed to evaluate the postulated condition, were performed. These experiments indicated that one of the control rod assemblies was manifesting an anomalous reactivity worth pattern.

The fuel was removed from the reactor and the rods were radiographically inspected. This revealed that, as postulated, the lower flux suppressor in the suspect rod was free to move about twelve inches up or down in the vertical direction within the fuel follower section. Both the effects of the upward coolant flow during scram and the inertial effects of the scram could act to move the flux suppressor and therefore cause it to come to rest at a different position within the fuel follower.

Results

No damage to the reactor resulted. During the operations following the first reactivity change, the possibility of damage to the reactor from future, unexpected changes was acknowledged. Based on analyses of the reactivity change which would be required to produce damage to the reactor, the risk was judged to be small. It was decided to risk this damage in an effort to expedite clarification and correction of the problem. As an added safeguard, operations were conducted under weather conditions which would ensure negligible exposures to operating personnel, even in the event of severe reactor damage.

ROE 67-4

Causes

The flux suppressor was brazed into place in such a fashion that visual inspection was impossible. It is postulated that during the acceptance drop testing the suppressor came loose and then came to rest at its lowest possible position. This is substantiated by the unchanging cold critical rod position experienced in SPERT III during the static tests after the core was first loaded.

Corrective Action

Repair was accomplished by drilling six holes on each of the four sides of the fuel follower housing to allow welding of the flux suppressors to the housing. Drop tests were performed to ensure that the repair had been successful. As a preventative measure, the flux suppressors in each of the other seven control rods were also welded in place. In addition, an overall engineering assessment of the original control rod system was completed, and the design as modified was deemed adequate.

References

T. G. Taxelius (Ed.), Criticality Shifts in the E-Core, pp. 3-9, Quarterly Technical Report, SPERT Project, October, November, December 1966, USAEC Report IDO-17245, Phillips Petroleum Company, October 1967.

ROE 67-5

ENGINEERING TEST REACTOR
FUEL ELEMENT RUPTURE

Summary

On February 20, 1967, coolant flow through a large part of one Engineering Test Reactor (ETR) fuel element was blocked by a piece of tape. A fission break resulted which caused an automatic reactor shutdown.

Circumstances

Approximately 13 hours after the ETR had reached full power following a regular cycle shutdown, a sudden burst of activity in the primary coolant system shut the reactor down through a reverse originating in the N-16 monitoring system. The presence of fission products in the primary system was confirmed by analysis of primary system water samples and efforts to locate the source were begun. Both fuel elements and experimental capsules were checked by a series of stagnation tests, and the source of the activity was determined to be the M-12 fuel element position. When the fuel element was removed to the canal, a large piece of silver colored tape was observed blocking the inlet end of the element.

The tape was of a type which had been used extensively in the reactor during major experimental modifications made the preceding shutdown. It is used for placing radiographic film, etc. The reactor tank had been inspected four different times prior to startup, but the tape being silver colored was very difficult to see under conditions in the reactor. Even when the failed fuel element had been identified, the tape was not observed until the element was removed from the reactor.

Results

No particular external damage was visible when the fuel element was inspected in the canal. However, when the element was disassembled in the hot cell and examined, it was found that considerable melting had occurred. Some portions of nine plates showed evidence of melting, and four were essentially fused together with about 50% of the plate area melted. Detailed studies of this element are continuing.

Corrective Action

A satisfactory tape which is colored a highly visible yellow has been located and will be used exclusively in the reactor tank from now on. If tape is inadvertently left in the tank, this should make it easier to detect it during subsequent inspections.

ROE 67-5

References

E. H. Smith et al., Progress Report for Cycle No. 87, January 30-March 13, 1967 — ETR Operations Branch, USAEC Report IN-1084, Idaho Nuclear Corporation, April 21, 1967.

D. R. Deboisblanc, Test Reactor Incidents — Selected Case Histories, pp. 370-376, in Proceedings of the Incipient Failure Diagnosis for Assuring Safety and Availability of Nuclear Power Plants, Gatlinburg, Tennessee, October 30—November 1, 1967, USAEC Report CONF-671001, Feb. 20, 1967.

ROE 67-6

FAILURE OF HALVES TO SEPARATE IN A SPLIT TABLE TYPE CRITICAL ASSEMBLY

Summary

During a normal shutdown operation on a split table critical assembly, the reactor was shut down by the control-safety rods but the halves failed to separate. The overloads on the electric motor which drives the halves apart were found tripped. These were reset and a second attempt made to drive the halves apart. Again the overloads tripped. On a third attempt, the halves were successfully driven apart. Inspection did not reveal any electrical or mechanical failures except that a thrust bearing in the drive screw mechanism appeared to permit excessive end play. This bearing was replaced before operations were resumed. Lubrication of the moving half was augmented, as this may have been insufficient due to the heavy loading of the table at the time of the malfunction. More comprehensive checks of the drive system have been instituted.

Circumstances

In this critical assembly, one half is driven against the other from a 60-inch separation by a lead screw. The moving half slides along V-ways. When the two halves contact, the drive motor continues to rotate until manually stopped, but an adjustable clutch in the drive train slips. As long as no change occurs in the adjustable clutch the moving half will be brought up against the fixed half with rather closely reproducible thrust force and reproducible positioning each time. When the reactor is to be shut down, a different motor is energized through a higher speed and lower torque gear train to drive the halves apart.

There have been 16 recorded instances when the overloads controlling the out-motor have tripped so that this motor failed to drive the halves apart when called upon. Most of these occurred in the commissioning period, during the search for a satisfactory "way" lubricant. In all cases, operation of the reactor was discontinued for corrective action. In August 1958, a serious failure in the drive nut under the movable half caused the out-motor to stall and trip the overloads. In this instance complete dismantling of the machine for repair was required. In April 1961 three instances of the overloads tripping were associated with another failure of the nut. In the subsequent five years only one other instance occurred prior to the recent one in which the overloads tripped. This happened in March of 1963 when a loose set screw permitted axial movement of the clutch until it contacted the gear box housing. The pressure against the housing then tightened the clutch excessively and the torque of the out-motor was insufficient to break the halves apart until after several tries. The set screw was properly tightened and at the same time the thrust bearing (double tapered roller bearing) on the drive shaft was replaced to prevent any axial movement of the parts. The system was then operated without incident until August 11, 1966. On this date

the halves of the reactor again failed to separate when the reactor was shut down until the third attempt at starting the out-motor. Although separation of the halves is not necessary for reactor shutdown, this being effected by control and safety rods, the incident demonstrated the need for a check of the mechanical and electrical features of the carriage drive system.

Results

The results of this investigation did not reveal anything wrong with the motor or the overloads, which are properly sized for the application. Because of its previous history of failures, the nut under the carriage was a prime suspect, but inspection failed to reveal any damage. Inspection of the V-ways revealed that they were nearly dry, but a thin film of oil seemed to be present. The critical assembly under study was one of the heavier ones that has been loaded in the machine, and the weight may have been sufficient to squeeze out the lubricant.

The thrust bearing which had been installed in 1963 was observed to have .080-in. end play, which is regarded as excessive. It is possible that axial motion of the shaft to this extent could cause the clutch faces to tighten so that the halves would be driven apart harder than normally. When called upon to drive the halves apart, the torque of the out-motor might then be insufficient to start the halves apart, so that the motor overloads would be tripped.

Corrective Action

This bearing was therefore replaced with a new one having .030-.040-in. play. Lubrication to the V-ways was increased slightly.

After repeated functional checks of the carriage drive indicated that everything was working satisfactorily, normal critical operations were resumed.

Conclusions

It is concluded that certain checks of the drive and lubrication system must be incorporated into the periodic maintenance procedures. These inspections should include a check of the drive screw end play on a quarterly basis, and a weekly check of lubrication of the ways.

ROE 67-7

MECHANICAL FAILURES OF THE ANTI-CRITICAL GRID OF THE HIGH FLUX BEAM REACTOR

Summary

Two failures (breaks) in the anti-critical grid have occurred at the Brookhaven National Laboratory (BNL) High Flux Beam Reactor (HFBR). The first failure which resulted from improper welding practices was detected following initial hydraulic testing prior to reactor operation. The second failure which probably resulted from stress corrosion enhanced by improper welding practices was detected several months after the reactor was placed in operation. In both cases parts of the grid broke loose and were displaced by hydraulic forces within the reactor vessel. No serious damage resulted from either failure.

Circumstances (First Failure)

The HFBR anti-critical grid is a set of stainless steel bars mounted in the bottom of the vessel to prevent criticality of a fuel mixture resulting from a core meltdown accident. The grid is divided into four segments which are bolted to the support saddle in the base of the vessel (Fig. 67-7-1).

The first failure was detected when the primary system was opened for draining and drying after the initial system tests so that D_2O could be put into the reactor. At this time a chevron-shaped piece of steel was found lodged in the tube sheet of a primary heat exchanger. This piece of steel was identified as part of the anti-critical grid. An investigation indicated that all of the hold down lugs which attached one segment of the grid to the support spider had failed, leaving the segment free to move under hydraulic forces. This segment had subsequently repositioned itself 90° from its original location. The broken rectangular hold down lugs were found still captive under their respective bolts in the support spider. In addition another broken chevron-shaped hold down lug was found, indicating a second failed segment (Fig. 67-7-2).

Causes

The old down lugs failed because they had been poorly welded and the hydraulic forces were great enough to cause these welds to break. The weld material on the lugs was not only improperly deposited, but also lacked the required full penetration. In some cases as little as 1/6 in. of penetration was found (Fig. 67-7-3).

Conclusions

Luckily no serious damage resulted from the failures of these lugs even though there was potential danger in damaging the pumps, heat exchangers, and possibly core components. Repairs to the grid were made without concern for radiation as fuel had not yet been used.

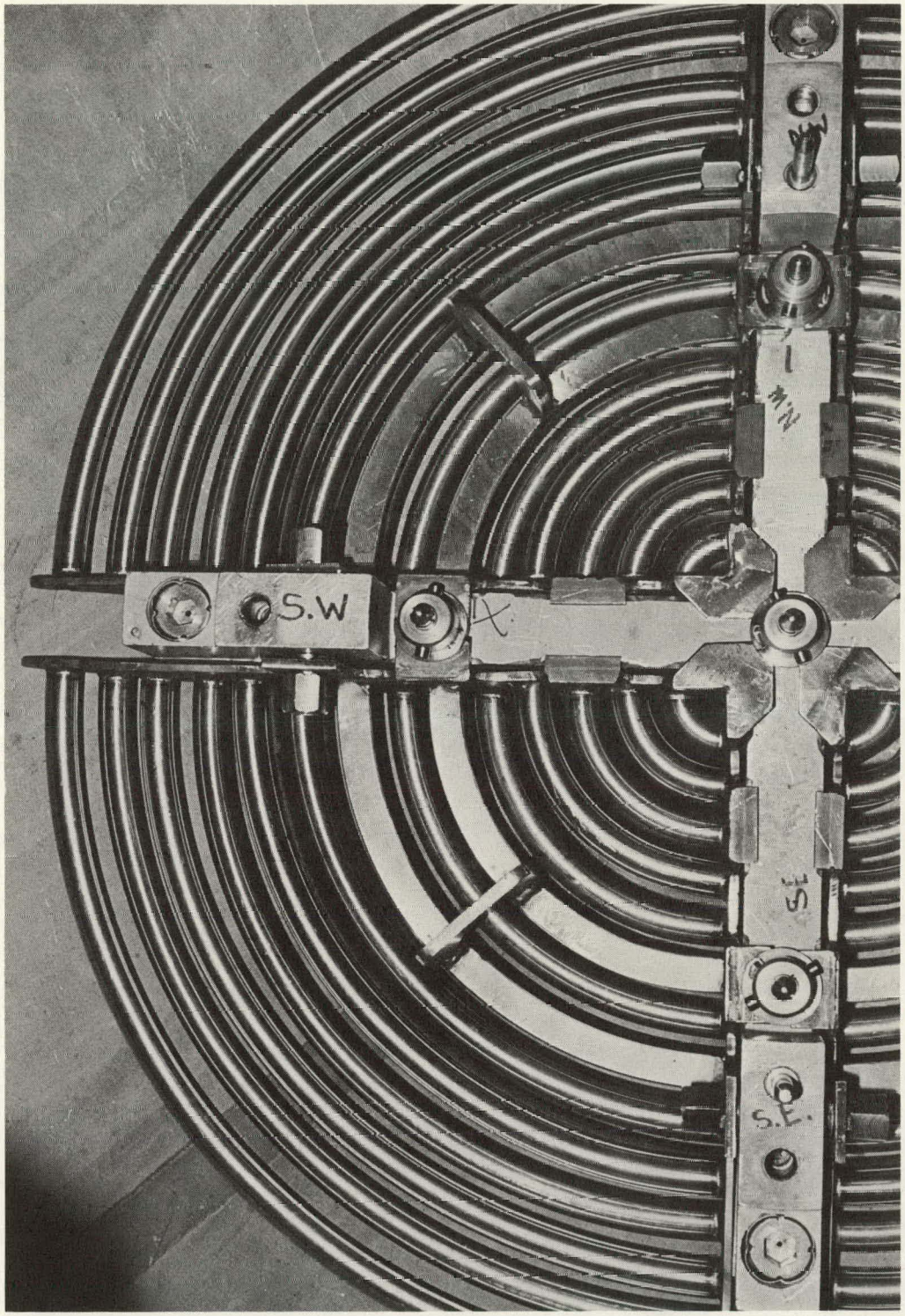

Fig. 67-7-1. Anticritical Grid.

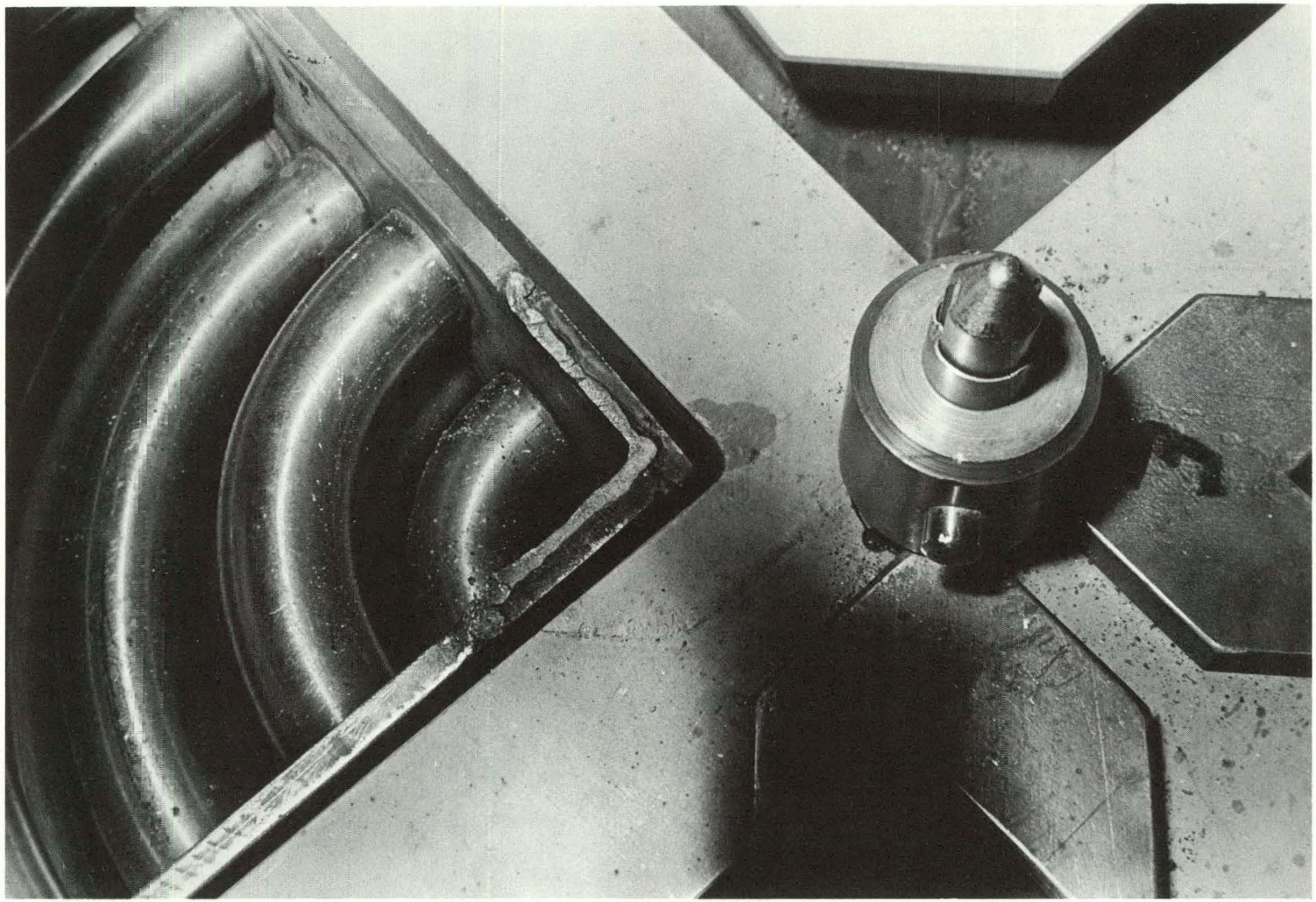

Fig. 67-7-2. Failed Segment of Anticritical Grid.

Fig. 67-7-3. Broken Hold-down Lug on Anticritical Grid.

ROE 67-7

Corrective Action

All broken or poorly welded lugs were replaced with first-class full penetration welds from both sides and all accessible welds on the grid were rewelded. An additional vertical support lug was added. However, studies of the various hydraulic forces on the grid incidated that the original hold down design was adequate.

Circumstances (Second Failure)

Several months after the initial failure was repaired, an unusual rattling noise developed within the vessel. Because the source of the noise could not be identified the reactor was operated at reduced power and flow to eliminate most of the noise. This type of operation was continued for several months until a major plant modification afforded time to carry out a detailed inspection of the vessel and vessel components. It was then found that one of the outer bars of a segment of the anti-critical grid had broken completely free of the grid and one end had moved up the side of the vessel. A close examination of the vessel revealed visible marks attributed to the motion of the bar under hydraulic forces. The bar was removed to a hot cell where it was thoroughly examined. Removal of the bar eliminated the noise.

Causes

The failure of the bar was probably caused by stress corrosion helped along by improper welding. One end of the bar broke loose by stress corrosion initially, then the other end broke by fatigue failure brought about by hydraulic forces.

Conclusions

No serious damage resulted from the broken bar. Other bars, although they appear sound under test, may be undergoing this same type of stress corrosion and they may ultimately fail.

Of major concern is the potential for beam tube puncture which in turn would cause significant coolant loss. Tests were performed by dropping a bar of the same length and weight as the biggest anti-critical bar from various heights and angles (corresponding to credible hydraulic forces) on a mock-up of a beam tube. These tests clearly showed that the beam tubes would not be broken.

Corrective Action

The reactor is again running at full power and flow. The vessel is being continuously monitored for noise so that any abnormal noise level can be automatically detected and corrective action can be taken to insure safety of the reactor. A committee has been established to continue investigations into the various aspects of the need, strength, and

performance of the anti-critical grid. This committee performed an initial analysis which showed that the whole outer ring of bars of the grid could be lost without having significant effect on the computed reactivity in the event of a core meltdown. Another grid is being designed in the event additional failure occurs and the grid must be replaced.

ROE 67-8

HEAT EXCHANGER PROBLEMS AT THE HIGH FLUX BEAM REACTOR

Summary

Both primary heat exchangers for the High Flux Beam Reactor (HFBR) were constructed without adequate internal supports for the tube bundles. As a result, it was necessary to operate the reactor at reduced power and flow (i.e., about 75% design conditions) to prevent tube rattling and possible tube failure which could lead to the loss of a million dollar D_2O inventory.

The heat exchangers were modified to eliminate the vibration problem, and the reactor was returned to full design operating conditions.

Circumstances

The HFBR has two primary heat exchangers of the U-tube, counterflow, heavy-water to light-water type, mounted horizontally.

A rattling sound was detected in both primary exchangers at flow rates in excess of 2,800 gpm per exchanger on the secondary side. (The rated flows per exchanger are 4,000 gpm on the secondary [shell] side and 9,000 on the primary [tube] side.) Consequently, the flow was restricted to a subrattling level (less than 2,500 gpm), and the power level was held at 30 Mw instead of the design power of 40 Mw. Transducers installed on the exchangers were used to detect rattling. Inspection ports with lucite cover plates were installed in order that the tube bundle could be observed under flow conditions. Accelerometer tests were also made in various tubes to determine the point of peak vibration. Calculations were made of the natural frequencies of the various internal components and frequency analyses were performed to compare the actual with the calculated frequencies.

Causes

The tube rattling resulted from the U-bend ends of the tubes striking together under full-flow conditions. This condition could have led to tube wall rupture. The excessive vibration arose because of the large unsupported length of the tubes between baffles and the excessively long unsupported length of the U-bend end of the tubes. This deficiency was similar to that found in the heat exchangers of the Advanced Test Reactor (ATR); the exchangers were built by the same firm. The Tubular Exchanger Manufacturers Association (TEMA) has established standards, to which these exchangers were supposedly made, which limit the unsupported tube length (length between baffles) to 60 inches. The HFBR unsupported length is 134 inches when the entire length of the U-bend is considered.

ROE 67-8

Conclusions

Although the reactor could be operated at subrattling flow rates and reduced power level (i.e., no tubes had failed), it was imperative that the exchangers be repaired. The tube vibration seemed to get worse with time (i.e., rattling was occurring at lower and lower flow rates) and a million dollar D_2O inventory could be lost in the event of tube failures. Also research was being hampered by the reduced power level.

Corrective Action

The exchangers were converted from a counter-current 2-pass flow to a straight single pass flow on the shell side. This was accomplished by adding two secondary outlet connections at the U-bend end of the shell. The original inlet and the original outlet at the tube sheet end both became inlet connections. The outlets were located over a baffle support which not only gave the tubes more stability, but also left the U-bend end of the tubes in relatively motionless water. Flows were thus reduced to 2,000 gpm on each exchanger half, a flow rate at which no vibrations occurred. The exchangers were conservatively rated for heat transfer and thus they were able to handle 40 Mw at a total shell side flow of 4,000 gpm with temperatures remaining within the operating limits. This modification was made one exchanger at a time while the reactor was operated on the other exchanger. The entire conversion cost under $50,000.

The reactor was returned to full design power and flow on the converted exchangers, which are being monitored continuously for any excessive or unusual noises. There appear to be none.

References

G. C. Kinne and P. R. Tichler, Heat Exchanger Tube Vibration and Repair in the High Flux Beam Reactor, USAEC Report BNL-11585 (CONF-670713), Brookhaven National Laboratory, July 19, 1967.

C. E. Dickerman, Highlights of the AIX Conference on Fast Reactor Safety, Reactor and Fuel-Processing Technology, 11(1): 1-3 (Winter 1967-1968).

ROE 67-9

FUEL HANDLING HOIST RUNAWAY

Summary

At a nuclear power station, a fuel handling grapple hoist ran away while rechanneling new fuel elements. The hoist control circuit experienced two electrical grounds, shorting out the "raise" control contact and causing an uncontrolled continuous upward drive. The hoist grapple held no load and no personnel radiation exposures resulted; however, this hoist is also used to lift irradiated fuel.

Circumstances

The hoist is used to install irradiated channels on new fuel elements and to remove channels from irradiated fuel elements since the channels are reusable in the reactor. Prior to hoist use, the prescribed prestartup checkout had been successfully performed to confirm the operability of all control and safety functions. When the runaway occurred, rechanneling of new fuel was in process.

The grapple hoist control switch can cause the hoist grapple to move up or down at variable speeds and has a spring return to the "off" position. The control circuit also contains a limit switch whose function is to prevent the grapple hook from raising above the 10 ft water depth. The runaway occurred while the operator was performing other related work. Normally, release of the handle removes power from the hoist drive motor, but the two grounds in the circuit bypassed both control and safety functions and caused the grapple hoist to operate in its upward movement at maximum speed. An assisting operator opened a nearby main power disconnect. However, by then, the empty hoist had reached its maximum up-travel, stalling the hoist motor.

Results

No injuries or damage resulted from this event. Serious personnel radiation exposure could have resulted if irradiated fuel had been attached to the hoist grapple.

Cause

Electrical grounds in the grapple control circuit caused the malfunction.

Corrective Action

To prevent recurrence of a similar event:

Mechanical stops were installed in both grapples to prevent lifting an element above the five-foot water depth.

ROE 67-9

An additional electrical stop to prevent lifting an element above the six-foot water depth was installed to prevent contacting the mechanical stop during normal operation.

References

Puerto Rico Water Resources Authority, Selected Bonus Operating Experiences, Bonus Nuclear Power Plant DRL Monthly Report, No. 9, Docket No. 115-4, September 1967, available at USAEC Public Document Room.

Commonwealth Edison Company, Dresden 1 Fuel Cleaning and Piping Inspection, Dresden 1 Annual Report of Station Operation for 1967, Docket 50-10, Jan. 23, 1968, available at USAEC Public Document Room.

ROE 67-10

IMPROPER UTILIZATION OF A BF_3 SENSOR FOR LINEAR POWER MEASUREMENT

Summary

Recent startup operations involving a critical assembly disclosed that BF_3 proportional counters had inadvertently been installed as detectors for the linear power level channels. This situation could possibly have prevented the power level channels from performing their safety function. The anomalous behavior between log channels and the linear channels was noted at a relatively low power level, and led to shutdown and checkout without any adverse effects. No personnel exposures or injuries resulted.

Circumstances

A reactor startup was made in preparation for flux mapping at a power level of 200 watts. During the scheduled rise to power, it was observed that the log channels indicated approximately 100 watts while the four linear power level channels were indicating approximately 50 watts. The reactor was shut down to evaluate the problem. Operation under restricted conditions was resumed to obtain additional information, and the current to two linear power level channels was plotted against power level up to and including 15 watts (based on log channels). Results indicated that nonlinear signals were being received from the four linear power level detectors.

Upon examination of the detectors, it was determined that instead of ionization chambers, each of the power level channels was connected to BF_3 counters. The BF_3 counters were removed and replaced with ioninization chambers.

The counter assemblies had been obtained from another critical facility with assurance that they were ionization chambers. The lending facility was not aware of the discrepancy since its operations were at milliwatt levels for which the detector response was apparently linear.

Results

No injuries or damage resulted from this event. However, inadvertent use of the BF_3 counters without adequate initial startup procedures and checks could have compromised the designed safety function of the linear power level channels.

Cause

A lack of a requirement concerning checkout of delivered material prior to use in a reactor safety circuit was the cause of this unusual experience.

ROE 67-10

Corrective Action

The BF_3 counters were replaced with ionization chambers.

To prevent recurrence, more detailed evaluation of key instrumentation components will be required prior to installation.

ROE 67-11

POLYURETHANE THERMAL INSULATION FIRE

Summary

Polyurethane thermal insulation covering piping and heat exchangers at a nuclear power station ignited and burned for a period between 1-3/4 and 4 hours. The fire is believed to have been started by spontaneous combustion. It was extinguished 1-3/4 hours after detection. The power station's scheduled power resumption was delayed 11 days. Equipment damage was slight; soot and fire derbis were spread in the basement of the containment building.

Circumstances

Charcoal beds operating at low temperatures are used to delay and hold up fission products purged from the reactor fuel elements and a side stream from the reactor main helium coolant. Polyurethane thermal insulation was sprayed on the pipes, delay bed containers, and associated equipment to improve fission product retention. No difficulty was encountered with this insulation.

Some of the insulation was later removed to permit a modification to the delay bed system. Following the equipment modification, polyurethane insulation was again sprayed on the equipment. Workmen finished applying the insulation at approximately 9:00 p.m. the evening before the fire. At 10:00 p.m. a routine inspection by a plant operator revealed nothing abnormal. The routine inspection at 12:45 a.m., some three hours later, revealed black smoke coming from the low temperature delay bed room and the neighboring cell which contains interconnecting piping from this delay bed system.

Proper authorities were notified and fire fighting procedures were initiated. The local fire department arrived at 1:00 a.m. Two plant operators equipped with a hose from a dry powder extinguisher cart and wearing breathing apparatus attempted to enter the room, but were stopped by dense smoke. Potassium bicarbonate from a large portable extinguisher was then emptied into the cell from above; the fire continued.

CO_2 from the fire protection system was discharged into the room containing the delay bed piping, but the smoke did not clear. This system was actuated a second time, and a high pressure water spray was directed into the low-temperature delay bed area. Two men subsequently entered the area at 2:00 a.m. with breathing apparatus and determined that the fire was out. The source of the fire was found to be newly applied polyurethane insulation.

ROE 67-11

Results

Fire damage was limited to the insulation and a few low-temperature piping lines. Smoke and soot accumulation was limited to the containment basement areas and the ventilation filter system (the absolute filters were plugged by the dense smoke). Water damaged the exposed insulation and the tin vapor barrier of the low-temperature delay bed regenerative heat exchangers. The reactor operating schedule was delayed 11 days.

Conclusions

The fire is believed to have started from heat released by exothermic curing of the insulation. The insulation, Iso-foam B2444, is rated as fire resistant and in the non-burning class. It is made up of two components and applied with a spray gun.

The insulation is usually applied in layers with sufficient time between applications to permit the chemicals to cure. This application was made without curing time between layers. Apparently the heat trapped in the insulation led to self ignition.

Corrective Action

This insulation was removed from all equipment, and will not be reapplied.

Recommendations

1. Care should be taken when specifying thermal insulation to assess the combustible properties of the insulation to be used.

2. When applying materials with unfamiliar characteristics, manufacturer's precautions and directions should be followed very carefully.

References

Additional information pertaining to sprayed-on polyurethane insulation fires can be found in AEC's Health and Safety Information Issue No. 254, July 14, 1967, "Recent Foamed Polyurethane Fires on Non-AEC Facilities".

Philadelphia Electric Company, Polyurethane Insulation Fire Due to Improper Installation, Peach Bottom Atomic Power Station, Monthly Operations Report No. 14, Docket No. 5-1717, April 1967, available at USAEC Public Document Room.

ROE 67-12

HIGH FLUX ISOTOPE REACTOR TARGET FAILURE

Summary

The aluminum cladding of some of the ^{242}Pu targets being irradiated at the High Flux Isotope Reactor (HFIR) to produce transplutonium elements has cracked, apparently due to fission-gas-induced expansion of the fuel pellets and embrittlement of the aluminum jackets. The mechanism of the aluminum embrittlement is being investigated and the target fuel pellets are being redesigned to reduce their expansion.

Circumstances

The HFIR was constructed primarily to produce isotopes of the higher elements by irradiating plutonium in the target or trap region. This is a cylindrical hole in the center of the fuel where the peak thermal flux is 3×10^{15} n/cm^2/sec. Plutonium oxide is mixed with aluminum powder and the mixture is compressed inside aluminum capsules to form aluminum-clad pellets with 20-mil-thick walls and ends. The pellets are assembled inside aluminum tubes 3/8 of an inch in diameter and with a wall thickness of 60 mils. Space was left in the top and bottom of the tubes for accumulating fission gas. The tubes, extruded with six fins, have all of the fins removed except three short sections which were left to center a water-flow tube approximately 3/4 of an inch in diameter.

Seventeen of the target rods were irradiated for about a year, prior to startup of the HFIR, at a flux of about 1×10^{15} n/cm^2/sec; and these, together with 14 new targets were inserted in HFIR and irradiated approximately five months.

During routine checks in February 1967, small amounts of alpha activity were observed in the HFIR circulating primary water. No appreciable change was noted in the fission gas or other fission-product activity in the water. The buildup of alpha activity was followed until it was clear that the activity was increasing steadily. The reactor was shut down and the targets were removed.

A procedure was developed for circulating water over each individual target and through a filter and ion-exchange column, while keeping the target in the canal. This procedure proved effective in locating five targets which had failed, and these were sent to hot cells for inspection. One target contained a gross failure approximately 2 inches long and 1/16-inch wide. This tube was cut and sections were inspected in a metallographic hot cell. It was determined that brittle failure had occurred.

The reactor was started up with those targets which appeared to be intact since it was desired to obtain as much production as possible before processing the targets. The demineralizers were not effective in removing the activity from the primary coolant, practically all of which

was ^{244}Cm. It appeared that the curium was in some unionized form which was not affected by the demineralizers, and attempts to filter it were unsuccessful. Testing with the very finest filters gave very poor removal. While the alpha contamination in the water could be lowered by several days' operation of the cleanup system, it appeared that the removal was more likely due to plating out on the system surfaces rather than the result of filtration or ion exchange.

Operation of the HFIR continued with several subsequent ruptures of targets until July 8, 1967, when the reactor was shut down for the annual inspection. At that time, all of the 12 remaining targets from the original 17 which had been preirradiated were examined and all but one of the 12 showed visible cracks. This one target was somewhat scarred during removal of its water jacket so that it was difficult to tell whether cracks existed or not.

It is estimated that the failed elements had received an integrated fast flux (>0.8 MEV) of approximately 1×10^{22} nvt. The integrated thermal flux is estimated at 6.3×10^{22} nvt.

Cause

Inspection of the first failed target tube indicated that the failure probably had two causes. First, the fission gas did not escape from the plutonium oxide-bearing pellets and induced expansion of the pellets. Second, the aluminum jacket became brittle and could not expand the small amount which the pellets required (about 1 percent). The brittle fracture of the aluminum occurred with no detectable plastic deformation.

Conclusion

The unusual embrittlement of the aluminum coupled with the expansion of the plutonium-bearing pellets caused cracking of the target tubes.

Corrective Action

Investigation of this radiation embrittlement of aluminum is underway, including a program to evaluate different types of aluminum. At present, insufficient evidence is available to determine the embrittlement mechanism.

For the immediate future, targets are being fabricated with a lower density than the original ones--75-80 percent versus 90 percent. This should give the plutonium oxide more room to expand and allow a longer irradiation before failure occurs.

References

R. V. McCord and B. L. Corbett, Target Rod Clad Rupture Detection, pp. 20-26, High Flux Isotope Reactor Quarterly Report, January, February, March of 1967, USAEC Report ORNL-TM-1895, Oak Ridge National Laboratory, July 6, 1967.

ROE 67-13

ACCIDENTAL INJECTION OF SUPPLEMENTARY SAFETY SYSTEM POISON INTO A PRODUCTION REACTOR AT SAVANNAH RIVER PLANT

Summary

The supplementary safety system of a production reactor at the Savannah River Plant was accidentally activated, injecting a neutron poison into the reactor. At the time of the incident the reactor was operating at full power. The reactor was shut down by the poison. Moderator cleanup was successful and subsequent reactor operation was normal. The downtime for the reactor resulting from this incident was 22 hours. All operator and system actions subsequent to the accidental activation were correct. There was no adverse safety implication in the event.

Circumstances

Each production reactor at the Savannah River Plant is equipped with an auxiliary shutdown device called the supplementary safety system (SSS). This system, which functions by injecting gadolinium nitrate solution, $Gd(NO_3) \cdot D_2O$, directly into the D_2O moderator, provides a means for shutting down the reactor and keeping it subcritical in the unlikely event of an incident in which the reactor control and safety rods become ineffective. It has never been necessary to utilize this system.

The SSS for each reactor consists of two identical subsystems. Each subsystem injects the poison into the reactor at three points, and consists of a poison tank pressurized with nitrogen, two parallel valves, and piping leading to the three injection points in the reactor. One valve on each subsystem is an explosive type that can be fired by either of two switches in the control room, and the other is a pneumatic valve which can be operated from the control room by a pull ring. The electrical system for the explosive valves is battery operated (24v) and independent of all other electrical systems.

Just prior to the accidental injection, a maintenance crew was performing a periodic check of the battery monitoring voltmeters on one subsystem of the SSS. In connecting a calibrated check voltmeter to the monitoring meter terminals, using alligator clips, one of the clips slipped off and momentarily shorted an unrelated 110-volt terminal to the meter terminal. This applied 110 volts AC to the detonator of the explosive valve, resulting in the release of poison solution into the reactor. This is the only accidental release ever experienced on a Savannah River Production Reactor.

Results

When the explosive valve opened, the reactor was at full power. The readings on the four high level flux monitors immediately began to de-

crease rapidly. The reactor console operator promptly initiated a manual scram, as required by operative procedures.

It was noted that the SSS nitrogen injection pressure gage read zero. As soon as the situation was fully appraised, maximum flow through the deionizer beds was established to remove the poison from the moderator. Reactor startup and continued operation was initiated when sufficient reactivity was available and the moderator chemistry was within established limits. This required 22 hours. No operating difficulties were encountered.

Conclusions

This incident was primarily the result of a circuit layer deficiency. A source of 110 volts AC was in close proximity (1/4 inch) to the monitoring meter terminal.

The results and consequences of this unplanned operation of the SSS completely supported the results of a controlled test of this system in 1964. The shutdown capability of this system is adequate. A complete cleanup of the primary system is not only possible, but can be accomplished in a short period of time, by circulation of the moderator through the normal deionizer beds.

ROE 67-14

TURBINE CONTROL SYSTEM FAILURE

Summary

The turbine admission valve at a boiling water reactor failed closed, producing a neutron flux excursion. Rapid closure of the valve caused reactor pressure to rise, which in turn collapsed voids that added reactivity and led to the flux excursion. It terminated in a high flux scram.

Circumstances

Rapid turbine admission valve closure was caused by a malfunction of the governor control system. The governor control hydraulic system, as well as turbine bearing lubrication and turbine hyrogen seal systems, are supplied by a pump driven from the turbine shaft. Poor electrical grounding of the turbine shaft forced the pump bearings to become a part of the discharge path for the static charges which accumulate on the shaft. This action resulted in failure of the oil pump bearing.

Bearing material in the oil system lodged in the primary pilot valve in the governor control system. This valve controls the position of the turbine admission valve, and when the material blocked pilot valve oil flow, the admission valve slammed shut. The turbine bypass valve which automatically bypasses steam to the condenser for reactor pressure control did not respond rapidly enough to prevent the resulting pressure and flux transients. The bypass valve is not designed for complete load rejection except when the rejection signal is generated by a loss of electrical line monitoring relay. Therefore, closure of the turbine admission valve resulted in a pressure and flux transient which terminated in a high flux scram. The pressure rise was not sufficient to lift safety valves.

Results

Although unrelated, the reactor could not be restarted subsequent to the scram due to a faulty control rod. During the ensuing shutdown the initial pressure regulator was recalibrated since this was thought to be the cause of rapid valve closure. (The initial pressure regulator normally causes the valve to close down on reduced pressure in an attempt to regulate steam pressure.) After control rod repair, the reactor was restarted and the generator was synchronized to the line. Again erratic turbine control system behavior required a shutdown (controlled) after about an hour of electrical generation. The control system was thoroughly investigated, the underlying cause was identified, and the system was repaired. Several days of power generation were lost because of the control system failure. No other equipment damage was apparent as a result of the turbine control system malfunction.

ROE 67-14

Cause

Poor electrical grounding of the turbine shaft caused failure of the turbine control system as described in "Circumstances".

Corrective Action

Existing grounding brushes were replaced and additional brushes were added to improve turbine shaft grounding.

ROE 67-15

EMERGENCY GENERATOR FAILURE AT A RESEARCH REACTOR

Summary

The emergency generator at a research reactor failed to assume load following an electrical power failure during a lightning storm. The power outage lasted 1.67 hours; the generator was returned to service 9 minutes after normal power was restored. Failure to assume load is attributed to excessive loads connected to the emergency bus. No core damage or equipment damage resulted from the generator failure.

Circumstances

When the maintenance engineer arrived after the power failure, he found the emergency generator overload breaker open indicating an overload on at least one phase of the three phase system. The engine was not running and was presumed to have stalled or shut down because the temperature of the generator stator was excessive. (A thermal overload switch is actuated by high temperature in the generator stator.)

Two nonessential air compressors were disconnected from the emergency bus and the unit was subsequently started, load tested, and ready for service a few minutes after normal power was restored. The unit was then successfully started and loaded several times.

Previously, the emergency power unit, a natural gas 50 KW system, had proven reliable and relatively trouble free. It had been load tested under simulated or actual power failure conditions several times each year. (Six successful load tests were experienced during the year prior to this failure.) Further, it had been functionally tested without load weekly. The day following the occurrence, the power failure was duplicated during a test by connecting the two air compressors with the other equipment on the bus. When the two compressors were disconnected, the generator unit started and accepted the load satisfactorily.

In retrospect it became apparent that gradual addition of equipment to the emergency bus had exceeded the capacity of the system. Past tests were successful because power demands by the equipment were not simultaneous. As assessment of the power demand indicated loads of 67 KW; the unit is rated at 50 KW.

Results

Emergency generator operation is not required to protect the reactor core. However, power was temporarily lost to the neutron monitors, radiation monitors and the emergency air exhaust system.

ROE 67-15

Cause

Excessive power demand by the loads connected to the emergency bus overloaded the generator and prevented load assumption.

Conclusions

Care should be taken each time equipment is added to the emergency bus to:

1. Determine whether the load is essential.

2. Determine that the newly connected load in concert with existing loads does not exceed the power generating capacity of the power source. Starting current of equipment should be included in this assessment.

3. Determine that the newly connected load does not degrade the power source by substantially increasing the probability of bus failure.

References

Letter, Industrial Reactor Laboratories, Inc., to USAEC, Emergency Generator Failure at IRL, Docket 50-17, July 14, 1967, available at USAEC Public Document Room.

ROE 68-1

LOSS OF POOL AND CANAL WATER

Summary

The shielding water covering spent fuel, capsules and other radioactive material at a test reactor was lowered about six feet when a temporary water seal, installed in the pool pump suction line to permit pump relocation, was expelled from the line. The radiation dose rate at the top of the water increased by a factor of about 130 to 2 Rem/hr. Fourteen thousand gallons of water spilled onto the basement floor and flooded the floor to a maximum depth of about 8 inches. No personnel overexposures occurred and property damage was slight.

Circumstances

The reactor core is contained in a pressure vessel which, in turn, is located within a pool. The reactor vessel and its surrounding pool have separate and independent cooling systems. The reactor pool joins the fuel storage canal, but can be isolated from the storage canal by means of a 3-section vertical gate with a full-length inflatable seal. Each section is about 3-ft wide x 6-in. thick and 6-ft, 6-in. long (Fig. 68-1-1).

During a scheduled shutdown, modifications were to be made to the reactor pool cooling system which required cutting the suction line and relocating a pump. The reactor was shut down, and all fuel except one control rod fuel section was removed from the core and placed in the fuel storage racks in the canal. In preparation for pump relocation, the reactor pool water level was lowered to the pool cooling suction line level. A regulation basketball, wrapped in rubber tape to increase the diameter by 2 inches, was inserted into the intake of the suction line and inflated. The basketball had been fitted with a wire harness and the harness was fastened to a nearby pipe in the pool. The reactor pool water level was then raised five feet, to the level normally maintained during shutdown, to permit capsule and irradiated component handling in the reactor tank. The ten-inch suction line was parted and work on pump relocation began. Only the basketball plug prevented leakage out of the open suction line (Fig. 68-1).

Unanticipated maintenance problems developed during pump relocation which extended the job. Capsule transfer from the reactor pool into the canal had been scheduled, and the shift supervisor decided to proceed with the transfer while the work on the cooling system piping was in progress. The reactor pool water was raised to the normal level, approximately 16 feet above the level of the basketball, the gate seal was deflated, and one section of the vertical gate between the reactor pool and fuel storage canal was removed. A second section, immediately below the top section, was to be removed to facilitate the capsule transfer, when the basketball plug was forced through the pipe and out the open

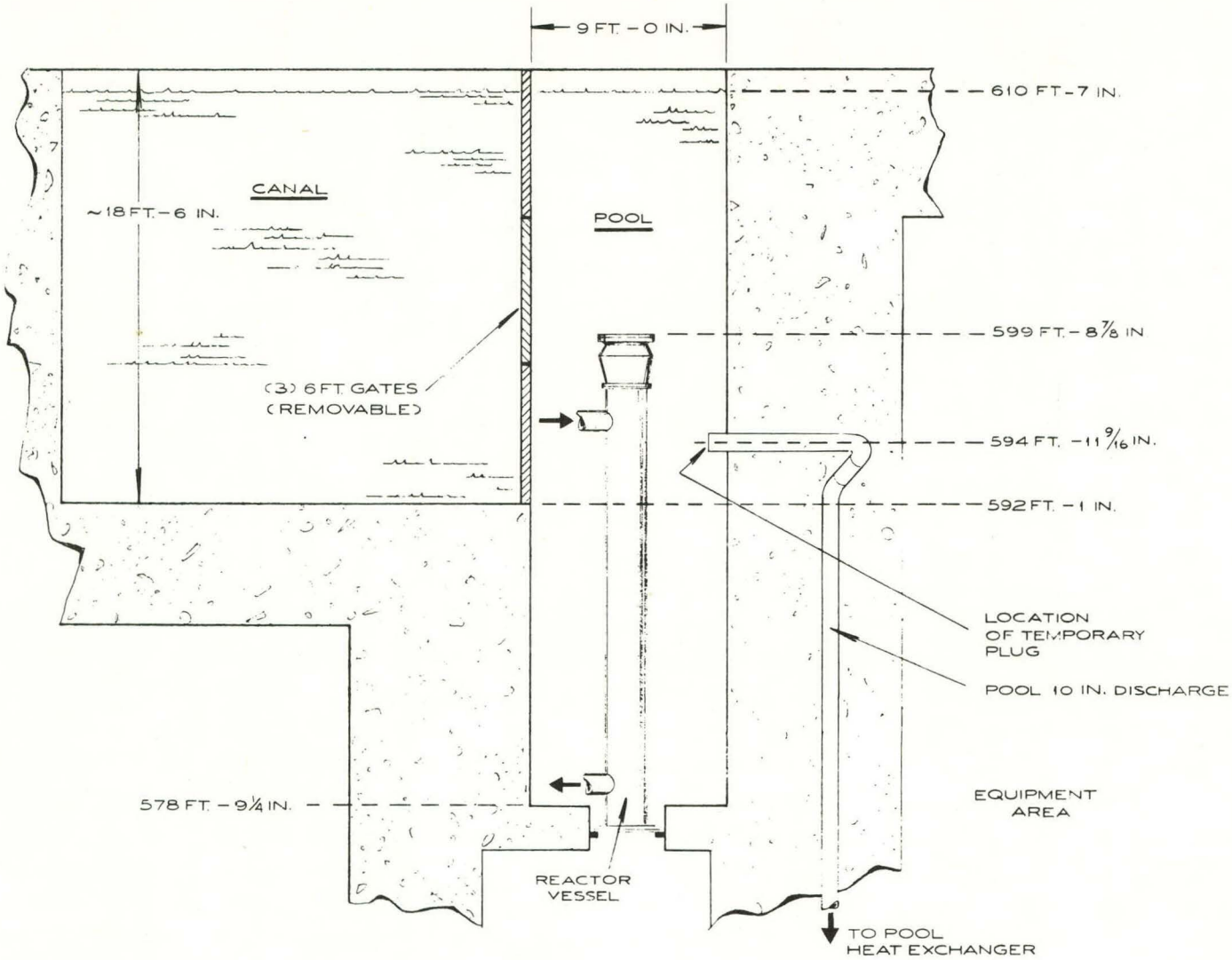

Fig. 68-1-1. Canal and Pool.

end by the water static head pressure (a force of about 500 pounds). Fourteen thousand gallons of water spilled into the basement in approximately five minutes. The reactor pool water level dropped to the level of the suction line, and the canal level dropped in excess of six feet to the top of the second gate section. The first gate section was replaced and the seals inflated to prevent further loss of water from the canal. The basement flooded to a maximum depth of about 8 inches.

The canal water depth was 12 feet, and a radiation monitor at the edge of the canal indicated a dose rate of approximately 2.0 R/hr. If the second gate had been removed as planned, the water would have dropped to a level just above the spent fuel stored in the canal. Since the pneumatic seals were deflated, the water level could have dropped further to a level leaving only the bottom one-third of the fuel covered.

Results

The maximum indicated personnel exposure was 0.040 rem. Splashing by the spilled water temporarily shorted out a lighting transformer and the sump pumps. Sump pumps were able to be started about 15 minutes after the flooding; they removed the water. An additional 1-1/2 inches of water would have caused short circuits in the power supply to the two pool water emergency recirculation pumps.

Corrective Action

A more conventional seal utilizing a metal pedestal with a rubber-faced flanged top, which forms a better seal with increased pressure, was substituted for the basketball.

Furthermore, the facility plans to make modifications to prevent any possibility of fuel melting in the event of a complete loss of water from the canal.

Conclusions

Procedures written for the performance of work such as described in this experience should include positive measures to assure recognition of the hazards involved in possible draining of fuel storage facilities; e.g., locking and tagging the canal gates.

The shutdown schedule should be reviewed and modified as necessary whenever scheduled maintenance is not completed as originally planned.

A plug working against hydrostatic pressure should be blocked from being forced into or out of the pipe which it plugs.

Where risks of fuel melting and personnel safety are involved, consultation with knowledgeable people should be made prior to questionable operation.

ROE 68-1

References

Letter, General Electric Company, to USAEC, GETR Loss of Pool Water During Maintenance, Docket 50-70, Sept. 19, 1967, available at USAEC Public Document Room.

Letter, U. S. Atomic Energy Commission, to L. S. Moody, GE-Vallecitos, Noncompliance Citation for GETR Pool Water Loss, Docket 50-70, Oct. 13, 1967, available at USAEC Public Document Room.

ROE 68-2

RAPID TURBINE CONTROL VALVE CYCLING AT A PWR

Summary

The four turbine control valves at a PWR simultaneously cycled from the fully open position to the fully closed position and then returned to the fully open position, all within about 12 seconds. The closure induced a severe load transient and caused other plant parameters to experience transients of a lesser magnitude. The neutron flux varied only 10% and remained well below the 103% alarm condition. No equipment damage resulted from the transients.

Circumstances

While the reactor plant was operating normally at an electrical load of 86% of rated power, a scheduled load increase to 97% of rated power was inititated by control rod withdrawal. Reactor power was increased, followed by a large increase in turbine noise and vibration which was most severe at the high pressure turbine near the turbine control valves. When load and average coolant temperatures were reduced, the noise and vibration subsided. Other attempts were made to increase load with similar results.

The turbine manufacturer was contacted, and a service representative came to the plant the following day. A program was developed to increase load. The average temperature of the reactor coolant was reduced to permit the turbine control valves to open fully at less than rated power. Load was increased to 89% of rated (at the reduced temperature) by opening the control valves fully. Reactor average temperature and reactor power were then slowly increased by control rod withdrawal to match the load until rated power was achieved.* Slight turbine noise appeared intermittently. After operation at full load for about 50 minutes, the noise level began increasing. Ten minutes later, all four control valves closed simultaneously and then reopened. The transients that resulted are shown in Fig. 68-2-1.

When the valves closed the average coolant temperature began to rise because the heat sink was lost; the operator reacted rapidly to insert a

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*When generator load changes, reactor power changes in the direction to match load demand because of the negative reactivity temperature coefficient. Changes in load cause the average coolant temperature to increase or decrease to the point at which the temperature coefficient overcomes the power difference if no steps are taken to adjust reactor power manually. The reactor control operator normally adjusts control rods to match power to load and maintain a constant average reactor coolant temperature.

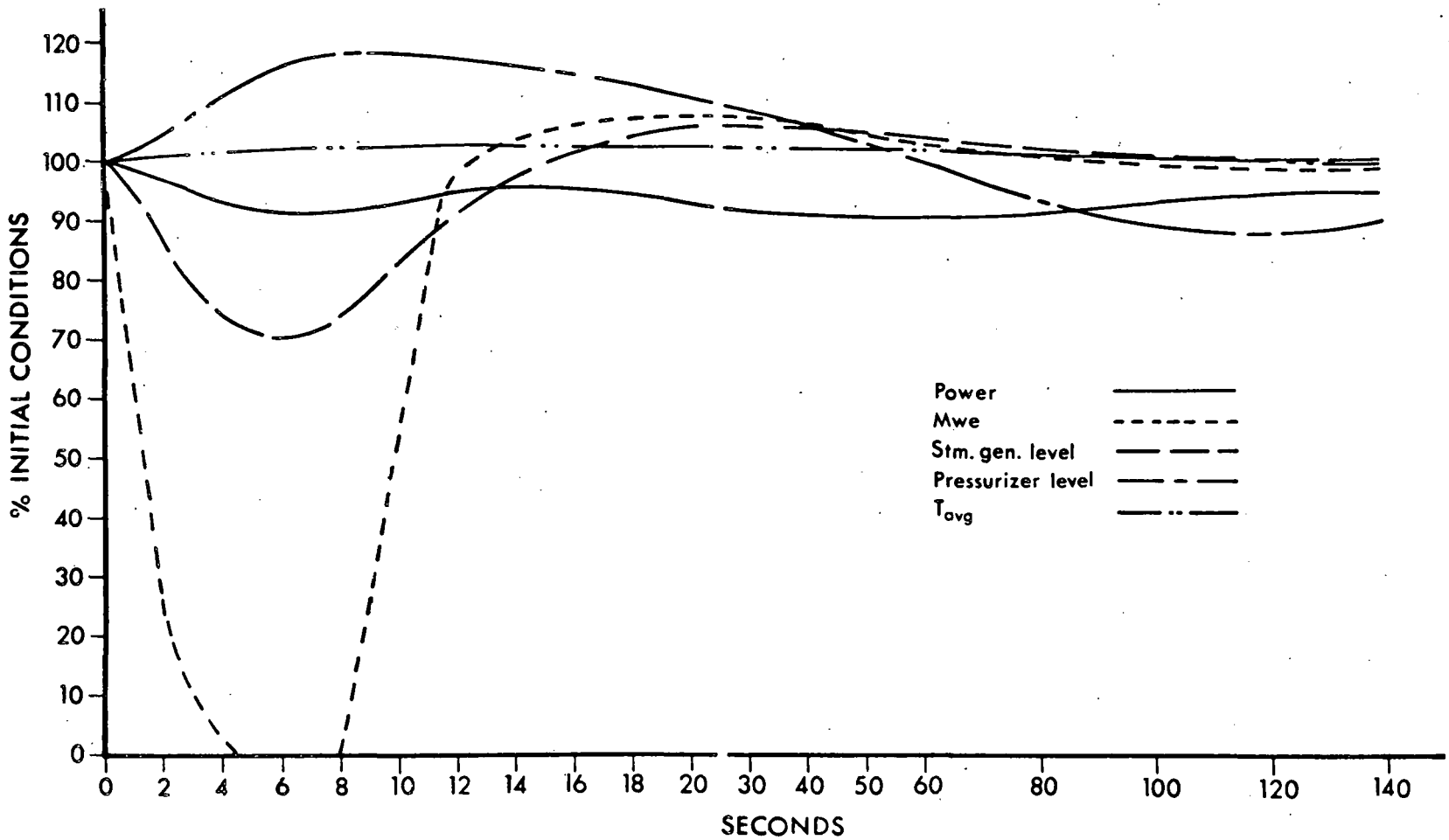

Fig. 68-2-1. PWR System Response to Turbine Control Valve Cycling.

group of control rods.* As pressurizer pressure and level surged upward because of the temperature increase, the pressurizer heaters were shut off (heaters normally cycle on and off to maintain constant primary system pressure). The operator attempted manually to initiate sprays in the pressurizer (a technique used to lower primary system pressure). The spray valves did not open immediately; the operator selected "All Rods In" and partially inserted all control rods. The close-open valve cycle took place within about 12 seconds; the operator actions described above began at valve closure and were completed within about 1 minute. His actions minimized the transients and aided in preventing a larger neutron flux swing.

Steam generator water level in each generator dropped about 15-in. because of the sudden loss of steam flow. The levels operate in a 4 out of 4 coincidence to trip the turbine. The turbine was not tripped because the coil of the stop valve trip solenoid was open. In a later test of the generator water level sensors all responded properly and tripped at the proper water level.

Results

No apparent equipment damage resulted from the transients. When the load passed through zero, the generator operated as a motor for a few seconds without adverse affects on the system. Extended operation of the generator as a motor can cause damage to the turbine.

Conclusions

Had the valves remained closed for a longer period, the flux transient would have been much more severe. An overpower flux scram probably would have occurred when the valves opened.

Simultaneous low level in the steam generators will trip the turbine stop valve producing a scram, but the solenoid trip coil was defective. Had the stop valve tripped, steam flow to the turbine would not have resumed when the control valves reopened, and train on the turbine going from zero to full load in about 2 seconds would have been prevented. Also the reactor would have shut down, thereby removing the heat source.

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*The operator was aware of the possibilities of uncontrolled load changes and remained alert. Normally, the reactor operator would scram the reactor on a complete load rejection, but in this case he reacted to his first impression that this was simply a partial load rejection such as he had experienced before. Partial load loss had produced similar instrument response.

ROE 68-2

Apparent Causes

As in most cases of complex experiences of this type, several events combined to produce an unexpected result. Had any of these causes been eliminated, the course of the experience would have been altered significantly.

The initial closing of the turbine control valves probably occurred because of a release of hydraulic pressure which holds the valves open. Subsequent inspection revealed that hydraulic pressure (normally maintained at 45 psi by the turbine auxiliary governor) was improperly adjusted to only 22 psi. The governor is set to release pressure to the turbine control valves at 20 psi.

The small margin over the setpoint was probably insufficient to hold the valves open during the vibration from the turbine.

The reopening of the control valves would have been prevented by a trip of the turbine stop valve, except for a failed trip solenoid.

Corrective Action

An auxiliary governor oil pressure gauge was installed.

Turbine auxiliary governor oil pressure was increased to the recommended value of 45 psi.

Procedures were revised to require routine checks of the governor oil pressure.

The turbine auxiliary governor was inspected.

The turbine stop valve trip solenoid was replaced.

A scram circuit using switches mounted on the valves as scram actuators, was installed to produce a scram when any combination of turbine stop and control valve closure shut off steam to the turbine.

Discussion

This experience demonstrates that quite severe load transients can be tolerated under certain circumstances, provided an alert control operator takes prompt and proper corrective action. It further demonstrates a chain of events which resulted from the interaction of several apparently unrelated misadjustments and failures. Finally, this experience demonstrates again the sensitivity of the reactor system to malfunctions of the turbine control system.

ROE 68-3

SMALL DIAMETER PRIMARY PIPING WELD HEAT AFFECTED ZONE CRACKS FOUND IN A BWR PLANT

Summary

Late in 1965, during an outage for repair of a pump flange leak at a Boiling Water Reactor Facility, a 6-in. schedule 80, type 304 stainless steel seamless pipe bypass line was found to be leaking from a crack through the pipe wall in the heat affected zone near a weld. The plant had been in operation approximately 5 years. The affected section of the line was removed for examination and it was replaced with type 304L seamless stainless steel pipe.

Further inspection of other primary system small diameter pipe by ultrasonic testing showed six other crack indications. All cracks found were in the weld heat affected zone and with one exception were circumferentially oriented. No defects were found in the large primary piping ultrasonically inspected to date (approximately 25% of the system).

Circumstances

Following the satisfactory completion of hydrostatic testing of one of the secondary steam generator loops after pump repairs, a leak was found in a 6-in. bypass line around a motorized discharge valve. The leak was in a crack adjacent to a weld. The affected loop was isolated and the reactor continued operating at approximately 190 Mwe on the three remaining loops.

The defect was dye checked and appeared as a crack approximately 1/16-in. long. The piping section was removed for metallurgical examination and new schedule 80, seamless type 304L stainless steel pipe was installed using a welding procedure which limits interpass temperature to 300°F maximum.

A second loop was removed from service and its 6-in. diameter bypass piping was ultrasonically tested. Two flaw indications, each less than one inch in length and 10% of the pipe wall thickness, were found in heat affected zones. These flaws were left in place to monitor their growth or change of dimensions.

Leaks in a 4-in. decontamination stub pipe welded to the main loop piping were found by visual inspection. These cracks were about 1/2-in. long on the external surface and 3/4-in. long on the inner surface. The pipe section was removed and also sent for metallurgical examination.

At a later date, a third loop was isolated for examination of its small diameter piping. Ultrasonic testing of this 6-in. bypass line indicated a crack in the heat affected zone which did not penetrate the

external surface but which extended to a depth of approximately 50% of the wall thickness. Alternate grinding and dye penetrant checking confirmed its existence, as indicated by the ultrasonic tests. The defective area was removed to within 1/16-in. of the inner surface of the pipe and was weld repaired.

Subsequent isolation and ultrasonic examination of the fourth loop also indicated a flaw in a weld heat affected zone. A circumferential crack about 1-in. long and extending about 30% of the pipe wall thickness from the inner surface was traced and removed by grinding. An axial crack starting at the circumferential crack and extending out into the pipe wall was found while the circumferential crack was being ground out. This axial crack was about 3/4-in. long and did not surface on the inner or outer walls of the pipe. It was also ground out.

A later reexamination of the first loop revealed a 3-in. long crack in the weld heat affected zone of one of the 6-in. diameter welds. This pipe section was removed for metallurgical examination.

Reexamination of the second loop showed that the two cracks left in situ earlier had not changed dimensions. However, the pipe containing the cracks was ultimately replaced with new schedule 80, seamless, type 304L stainless steel pipe.

During a hydrostatic test of the primary system early in 1967, a leak was discovered in the lower header tie line which also serves the reactor cleanup demineralizer system. This 6-in. line contains 18 welds and all were inspected. Seven defect indications were found by ultrasonic tests. All defects were circumferential and in the pipe weld heat affected zone. The entire pipe section was replaced with new 6-in. schedule 80, seamless, type 304L stainless steel pipe.

Conclusions

Metallurgical examination showed that the first crack had a small amount of transgranular failure but that the general failure pattern was intergranular. The failure pattern of the second crack appeared to be entirely intergranular. Further examination of other affected areas is in progress.

The existence of flaws in the 6-in. and smaller seamless piping adjacent to circumferential and axial welds is being examined in depth. The absence of flaws in forged ells and fittings made from seamless pipe similar to that containing flaws indicates differences in composition or manufacturing methods peculiar to the faulty seamless pipe. The leaking cracks, cracks found ultrasonically, and ultrasonic flaw indications have all occurred in seamless type 304 pipe made to ASTM specification A376-56T fabricated offsite by one supplier in 1958 or fabricated onsite during 1958 and 1959. All cracks started on the inner pipe wall in the area machined to match adjoining internal diameters.

ROE 68-3

Cause

Completion of the first phase study showed that the heat affected zone weld areas of the small diameter piping were sensitized during the welding process. Stress corrosion attack then occurred in the grain boundaries of the sensitized regions. It should be noted that the welding process used in fabrication of the large diameter primary piping should not result in similar sensitization of the heat affected zones. The large diameter piping (12-in. to 22-in. in diameter) was manufactured from rolled and seam welded plate and the temperatures in the heat affected zones were kept well below the 800°F interpass temperature specified. The welding of small diameter seamless tube piping was performed at temperatures high enough to precipitate chromium carbide at the grain boundaries in the heat affected zones during the long cooling periods experienced.

The flaws in the lower header tie line have been also attributed to poor process control during manufacture.

Corrective Action

Greater emphasis has been placed on welding procedures to reduce temperatures in the weld heat affected zone for small diameter pipe and minimize weld zone sensitization. Leak detection procedures have been evaluated and found to be reliable and adequate.

References

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Letter, P. A. Morris, AEC, to Commonwealth Edison Company, Dresden 1 Shutdown to Investigate Flaws in Primary Piping, Docket 50-10, Oct. 25, 1967, available at USAEC Public Document Room.

Letter, Division of Reactor Licensing, AEC, to Commonwealth Edison Company, DRL Concludes Dresden 1 may be operated Following Primary System Tests, Docket 50-10, Nov. 3, 1967, available at USAEC Public Document Room.

Commonwealth Edison Company, Dresden 1 Fuel Cleaning and Piping Inspection, pp. 7, 28, 35, and 43 of Dresden 1 Annual Report of Station Operation for 1967, Docket 50-10, Jan. 23, 1968, available at USAEC Public Document Room.

R. H. Holyoak, In-Service Inspection of Dresden-1, Nucl. News, 11(11): 32-43 (November 1968).

ROE 68-4

RELIANCE ON A SINGLE INFORMATION SOURCE LEADS TO OPERATING ERROR

Summary

An erroneous signal being supplied to the thermal power calculator of an oxide fueled pressure tube reactor yielded a low indicated power level. As this instrument was being relied upon exclusively for determining reactor power level, the planned actual tube power level was exceeded. The abnormality was detected before established safety limits were reached. Power level during reactor startups is now being double-checked by comparing neutron flux level indications with the thermal power calculator reading.

Background

The basic heat generation limit of a UO_2-PuO_2 fuel element in this reactor is a peak fuel centerline temperature of 2790°C. (This temperature has been found to be reached at a maximum specific rod power of slightly more than 20 kw/ft under intended operating conditions.) Verification of the specific rod power corresponding to this centerline temperature was being made by a series of irradiations at predetermined tube powers corresponding to various specific rod powers, followed by destructive examinations of fuel rods. By the use of flux peaking factors, this specific rod power is converted to an allowable tube power.

The steps at 16.8 and 17.7 kw/ft had been completed and the reactor was next to be operated at 18.6 kw/ft. The desired tube power for this step had been established as an average of 1,127 kw for the six pressure tubes which surround the central tube of the reactor (referred to as the Ring One Average Tube Power).

The reactor's pressure tubes have inlet flow monitors connected to the safety circuit, and outlet resistance temperature detectors (RTD's) which provide high temperature alarms. The temperature and flow measured by this instrumentation, in conjunction with the bulk coolant inlet temperature as measured by the inlet RTD's, are used to calculate manually the tube power in each pressure tube.

The reactor's thermal power calculator uses as inputs the primary coolant bulk flow, obtained from a venturi near the reactor inlet, and the bulk coolant inlet and outlet temperatures, as measured by RTD's.

Events

In preparation for the 18.6 kw/ft operation, the primary system was brought to normal operating temperatures at a power level of 8 Mw. The power level was then increased stepwise to an indicated 45 Mw on the thermal power calculator. This value was about 80 percent of the power

level at which the desired tube power levels were expected to be reached.

A check of the linear flux monitors indicated a higher than expected flux level. The Ring One Average Tube Power was calculated as a check against the high flux level. The average was found to be 1,138 kw, which was about 1% above the ultimate level desired in the test being carried out (1,127 kw).

The reactor power level was reduced, and further checks made which revealed that the indicated primary coolant bulk flow was gradually decreasing and had reached a value of 8,200 gpm from the expected 10,600 gpm. The sum of the individual tube flows showed a normal primary coolant flow of 10,600 gpm. The reactor was then shut down to determine the cause of the low indicated bulk flow.

Causes

The low primary system bulk flow indication was traced to a partially open equalizing valve in the bulk flow measuring system. This valve is used in testing and calibrating the instrument and to establish a flow through the sensing line to sweep out any air bubbles. The partially open valve allowed a flow through the sensing line, and permitted high temperature water to enter. Both of these effects (flow and temperature) reduced the venturi's differential pressure as seen by the high range primary flow transmitter. The resulting low input signal to the power calculator caused it to indicate a lower-than-actual power level. Since the calculator was being relied upon exclusively to adjust the reactor's power level, a higher power level than expected was achieved because of this incorrect input signal.

Corrective Action

Reactor rise to power following heatup is now being guided by neutron flux instrumentation as well as the thermal power calculator. In addition, at power levels of 50, 75, and 90 percent of the planned power cross checks are made of the Ring One Average Tube Power and the power calculator output. In this manner, independent observations are used to limit reactor power until manual calculation of tube powers gives assurance of the reliability of the power calculator.

In addition, the importance of correct valving on all reactor systems has been reemphasized to all operating and maintenance personnel.

ROE 68-5

RUPTURE DISC FAILURES AT A TEST REACTOR

Summary

Failures of a rupture disc in the primary coolant system of a test reactor occurred twice within less than two months. Improper installation was the fundamental cause for both failures. The rupture disc design has been modified and operating procedures contributory to the failure have been revised to minimize the possibility of future failures.

Background

A three-inch rupture disc is located in a primary system valve gate to prevent a complete blockage of flow in the event of an accidental closure of the valve. The disc used for this application is made of two-mil thick stainless steel. A separate stainless steel vacuum backup plate is used to prevent the disc from collapsing if a back pressure develops while the valve is closed. Both pieces are maintained in the spare parts inventory; however, the disc and the vacuum support plate are cataloged and maintained as separate items.

During a routine integrity test after more than four years of satisfactory service, the disc was found to have failed. No investigation to determine the specific failure mechanism was attempted because of the nature of the service period for that disc. After replacement of the disc, the reactor was discharged and the primary system flushed at higher than normal flow through temporary screens in an attempt to recover the missing pieces of the broken disc. The recovery operation was terminated after a total of 150 hours of flushing when it was evident that only 72% of the missing disc material was recoverable. The potential problem of operating the reactor with disc material in the primary system were analyzed. It was concluded that the risks were of low probability and low consequence, and therefore acceptable for reactor operation.

The second disc failure was detected less than two months later. A similar recovery process, with 566 hours of flushing resulted in recovery of 64% of the missing material. Reevaluation of the risk analyses again indicated acceptable operating conditions. An investigation into the cause of the second failure was made.

Causes

Both failed discs had been installed without the required vacuum support plate. Bench tests showed that a disc of the type used would collapse and wrinkle at one-half psi backpressure in the absence of the vacuum support plate. Primary coolant system tests performed to simulate the conditions of unusual system operation that existed during the short life of the second disc showed that the disc had been subjected to backpressures of one-half to one psi on several occasions. It was

therefore concluded that the rupture disc was seriously weakened by cyclic collapse and attendant wrinkling. Subsequent flow pulsations and eddies at high flow probably resulted in further weakening of the fattened and wrinkled disc through crack propagation. This caused it to rupture at the relatively low test pressure during a later routine disc integrity test. Additional primary system tests were performed to simulate other possible failure mechanisms. None could be identified as a probable cause.

Corrective Action

The rupture disc was changed to a type which contains an integral vacuum support plate, thereby eliminating the needless separation of the required parts in inventory. Operating Procedures were revised to avoid environmental conditions which might accidentally threaten disc integrity. In addition, the disc is now scheduled for replacement on an annual basis.

ROE 68-6

INADVERTENT DISARMING OF SAFETY CIRCUIT INSTRUMENTATION

Summary

During the startup testing phase at a pressurized water nuclear power station, an instrumentation test signal used for instrumentation calibration was not removed at the completion of the test. This condition later caused a reactor trip. Although the reactor was not operating at an appreciable power level at the time, and safe operation of the plant was not jeopardized, safety circuitry did not function entirely as designed.

Circumstances

A reactor trip was experienced while establishing a vacuum in the main condenser. The trip immediately followed the opening of all steam dump valves in the line from the main steam line to the main condenser. These valves opened when the condenser low (less than normal) vacuum block was cleared. At this time, the reactor coolant system was near operating temperature, about 525°F. When the valves opened, steam was dumped to the main condenser and the primary water temperature decreased about 90°F in five minutes because of the rapid heat removal by the secondary system. The transient was terminated by closing non-return valves in the main steam lines which had been opened slightly for secondary plant startup.

Investigation of the steam dump valve control circuit revealed the presence of a locked-in test signal of 570°F to the average primary coolant temperature averaging unit. A temperature of 570°F (much above normal) coupled with normal condenser vacuum are circumstances that appear to the steam dump valve control circuit as if the turbine had experienced a load rejection and therefore, the valves properly opened in anticipation of a pressure rise in the main steam system. The sudden increase in steam flow, coupled with low feedwater flow, properly produced a reactor trip (a situation which occurs on a large steam line break and for which the Steam Flow - Feedwater Mismatch Circuit is designed to protect the reactor).

Results

No injuries or damage resulted from this event. Safe operation of the plant was not compromised although had the situation not been brought under prompt control, serious plant damage and possible injury may have resulted. Since the primary system cooldown occurred early in core life when the moderator temperature coefficient of reactivity is very small, a severe reactivity transient was not experienced. However, had this cooldown occurred at the end of core life when the coefficient is strong-

ly negative, the reactivity insertion could have amounted to about 1% delta k/k in five minutes.

Cause

Human error and inadequate administrative control resulted in this accident. This experience highlights the ever present problem facing the reactor operator - maintaining positive control in the use of test signals during calibration of safety circuit instrumentation.

Corrective Action

To prevent recurrence of a similar event:

a. A system of administrative control has been instituted requiring instrumentation personnel performing the work (e.g., calibration) to discuss details of the work with the instrument supervisor to assure control of test signals used before returning the instrument system to service.

b. Restrictions or limitations of the safety circuit or system's operability are entered in the station night order book.

Recommendations

a. Reactor protection circuitry should be designed whenever possible so that switches or other test devices are readily visible to the reactor operator at all times.

b. Calibration procedures should require a cognizant sign off when completed. Specifically an action should be included in the procedure which requires a check of all valve positions and all bypass mechanism before returning the system to normal operating status and before sign off that the calibration is complete.

References

Bruce A. Irving, Operation Report No. 67-9 for the Month of September 1967, USAEC Report NYO-3250-14, Connecticut Yankee Atomic Power Company, Oct. 25, 1967.

ROE 68-7

FUEL CLADDING FAILURE

Summary

The aluminum cladding of several spent fuel elements awaiting shipment to a reprocessing plant was found to be "blistered" and, in at least one case, fuel material exposed. This failure was the result of chemical attack of the aluminum cladding by untreated storage pool water.

Circumstances

Fuel elements of the ETR type, containing 168 grams of 93% U^{235} alloyed with aluminum and clad with 0.015 inches of aluminum in 19 flat plates, are normally exposed in the reactor to levels of about 200,000 MWD/T before being retired. Spent fuel removed from the reactor core was normally stored in cadmium-lined holsters within the reactor pressure vessel to allow decay time to elapse prior to return for reprocessing. Due to a change in the fuel handling procedures, it became necessary to utilize all of the available holsters for the refueling operation, thus requiring 11 spent fuel elements be moved from these holsters into an adjacent storage pool. In the storage pool, the elements were placed into similar cadmium-lined holsters.

This storage pool is filled with water taken from the reactor primary system. The pH in the storage pool is maintained at approximately 6, and the water is circulated through a filter but no other water treatment is performed. After several months in the storage pool (longest time of the 11 elements was approximately 13 months), a fuel element was removed from the storage pool to be replaced in the reactor core as part of a special experiment. Inasmuch as this element had a very low burnup and, as a result, was reading about 5R/hr at contact, it was brought up for a visual inspection. Since the visual inspection revealed evidence of corrosion of the aluminum, a cleaning was undertaken prior to introducing the element into the reactor primary water. During this cleaning, it was noted that one outside fuel plate had what appeared to be a blister that had been broken, revealing fuel material. Swipes taken of this area substantiated the fact that fuel was indeed exposed. Other elements in the storage pool were subsequently visually inspected without finding any other defects although corrosion was noticed on all elements.

Results

All of the fuel elements contained in the storage pool were prepared for shipment to the reprocessing plant by shearing off the end boxes. Elements were then individually tested by placing them in a shielded cask through which water could be circulated. Water samples taken at intervals during the circulation procedure were sampled for fission product activity, and the maximum leak rate found to be one millicurie per month.

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Under these circumstances, no problems were encountered in shipping these elements to the reprocessing plant.

Conclusion

In spite of efforts to filter this water and control its pH, corrosion proceeded at a significant rate. The corrosion observed in this situation is believed to have been caused by prolonged exposure to water with a high conductivity. The discovery of this defect further substantiated the need for inspection of fuel elements whenever their disposition is changed.

Corrective Action

Additional fuel storage holsters have been fabricated to provide sufficient storage capacity within the primary coolant system for both expended elements and refueling requirements. In this way, elements will not be exposed to water with uncontrolled conductivity conditions.

ROE 68-8

CONTROL ROD MALFUNCTIONS

Summary

During the calendar year 1967 there were numerous reported experiences of control rod malfunctions at licensed reactor facilities. None of these rod problems represented any threat to the health and safety of the public, nor in any case was there a question regarding the ability to shut the reactor down safely.

A review of these experiences shows that a substantial number of them had a similar cause: foreign material in the system. The other malfunctions were caused by various, apparently unrelated conditions.

Circumstances

A brief description of representative rod malfunctions caused by foreign material in the systems follows.

1. At a BWR, a stuck rod problem was encountered during a prestartup testing period. Investigation revealed that a one-half inch nut had lodged in the control rod's drive mechanism. This nut was identified as a missing item from a replacement program for beam clamp bolts conducted two months previously.

2. At another BWR, six control rod malfunctions were caused by foreign material interferring with the function of the collet assembly of the control rod latching mechanism.

3. A control rod stuck at the fully inserted position at a PWR while the plant was being checked out in its shutdown condition. The cause of the stuck rod was foreign material in contact with a control rod guide block on the upper grid plate.

4. During test operation at another PWR, a control rod stuck 16 inches from the fully inserted position because of a metal object in the control rod snubber. This metal object appeared to have come from the lower grid plate and was probably cut from the plate during construction.

5. At a Triga research reactor one of the transient rods stuck in the fully withdrawn position following initiation of a scram during pulse operation. The cause was dirt which entered the pneumatic cylinder and caused excessive friction.

Conclusion

The recurrence of malfunctions of control rods due to foreign material in the system continues to emphasize the importance of system clean-

liness and accountability of parts. The reactors at which these experiences occurred have all operated for two years or more. This emphasis, while obviously important during construction stages, must be continued throughout the plant life during normal operation and maintenance periods.

ROE 68-9

MOVEMENT OF AN INSTRUMENTED FUEL ROD AT A RESEARCH REACTOR

Summary

Movement of an instrumented fuel rod by a control blade magnet occurred in a converted TRIGA core of a pool-type research reactor. During installation of an experiment, the upper end of the instrumented rod was inadvertently moved over a control blade magnet; subsequent movement of the magnet lifted the instrumented rod 8 to 12 inches. No core damage or personnel injury resulted from the experience.

Circumstances

In this TRIGA core, an instrumented fuel rod is attached to a stainless steel conduit through which the thermocouple leads are brought above the surface of the pool. The instrumented fuel rod is not clamped into its fuel cluster, as are the other three rods in that cluster, so that it can be inserted or removed without moving the entire cluster. The top of the conduit makes a right angle bend above the pool surface in the vicinity of the three motor-driven magnets which hold the control blades for the reactor. The conduit was tied to the transient rod guide tube with a short length of polyethylene rope.

During the installation of an experiment, the right angle bend at the top of the conduit on the instrumented fuel rod was apparently pushed over one of the magnets, but was not noticed. The rope was not tight enough to prevent the twisting of the conduit.

The reactor was brought to criticality at a low power for a reactivity measurement. The instrumented rod was apparently lifted 8.2 inches at this time and again the rope was not tight enough to prevent this fuel rod from being raised.

The blades were run in while completion of the experimental setup was made. It is not known whether the instrumented rod remained in the up position at this time.

The reactor was brought critical and pulsed for the experiment. One of the magnets had apparently lifted the instrumented rod about 12 inches at this time. An anomalously low fuel temperature was observed during the pulse. An attempt was made to repeat the critical rod positions used in the pulse, but the new settings were different.

Blades were run up and down individually in an attempt to detect any malfunction of drive units or movement of objects in the core. The instrumented rod was observed to be down, in its normal position, with the right angle bend close to one of the magnets, but not engaging it at this time. It was postulated that the instrumented rod was raised during the

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reactivity measurements and was up during the pulse. The reactivity measurements were repeated, being certain that the rod was down. The results indicate the lifting of the instrumented rod by about 8.2 inches had decreased the core reactivity by $0.95. Analysis of criticality data indicates the instrumented rod worth would have decreased $1.70 to $1.90 when lifted 12 inches. The conduit is now clamped rigidly in place with a metal clamp.

Results

A fuel rod was unknowingly raised during operation of the reactor. The possibility existed that an unplanned and uncontrolled insertion of reactivity could have taken place had the fuel rod dropped back into the core. Although the TRIGA core is designed so that the temperature rise of the fuel for this insertion would not cause any damage, other types of reactors may not be able to withstand such an increase in temperature. Mechanical damage due to dropping of the fuel rod could also occur.

Causes

There are four conditions which led to the occurrence of this event:

1. Improper design of the upper section of the conduit.
2. Improper fastening of the conduit.
3. Insufficient care in placing the experiment in the core.
4. Inattention to the significance of a substantial variation in the observed reactivity measurements.

References

Letter, H. W. Dodgen, Washington State University, to Division of Reactor Licensing, AEC, Movement of Instrumented Fuel Element by a Control Rod Magnet, Docket 50-27, Feb. 7, 1968, available at USAEC Public Document Room.

ROE 68-10

CORROSIVE FAILURE OF UNDERGROUND ALUMINUM PIPE

Summary

At a pool-type research reactor, corrosive failure of aluminum reactor primary coolant piping in its underground installation outside the reactor building was discovered during a routine scheduled pressure test of the primary system holdup tank. Radioactivity in the leakage water was below MPC for discharges to unrestricted areas as listed in 10 CFR 20, Appendix B. Except for the affected piping, which was replaced with stainless steel pipe, no equipment damage was involved.

Circumstances

Six days after the reactor was shut down for a scheduled maintenance period, a routine pressure test of the primary system holdup tank was conducted. The holdup tank was pressurized for 17 hours by the static head of pool water to determine the gaseous leak rate from the upper portion of the holdup tank. Leaking water forced its way underground along the core exit line and into a valve pit where it was discovered and determined through analysis to be primary system water.

A check on the water balance in the primary system indicated that essentially all of the leakage - amounting to several hundred gallons - was accounted for in the valve pit. The holdup tank leak rate test was terminated, and a section of the core exit line was isolated and subjected to a hydrostatic pressure test using raw water to confirm the existence of leakage and to try to establish its location.

Subsequent excavation and inspection revealed corrosion of the 6061-T6, schedule 40, extruded aluminum pipe. The extent of damage to the pipe was beyond reasonable repair and a decision to replace the aluminum pipe with stainless steel pipe soon followed.

Close inspection revealed that the corrosion originated from the outside surface following deterioration of the single layer of glass-reinforced protective craft paper which had been applied over two coats of bituminous enamel at the time of the original installation approximately nine years earlier. Although the core exit line had a radiation exposure history of about 10^6 rads, radiation damage did not seem to be a contributing factor since corrosive pitting was present to an equal extent on the pool return line and demineralizer lines, which by comparison had experienced negligible radiation exposure.

Soil conditions around the piping during service were sandy, generally wet because of the high water table, and acidic (pH 5 to 5.5). Average temperature of the primary water under flow conditions was approximately 90-95°F.

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The leak, resulting from a single corrosion pit which penetrated the 0.364-in. pipe wall, was discovered in the 10-inch core exit line approximately ten feet from the reactor building, just downstream of a Tee-fitting joining the two core exit lines with a single line leading to the holdup tank. Pitting in several other areas had progressed to depths of approximately 0.25-in., and the smaller demineralizer system lines, although corroded to a lesser degree, were nevertheless in a sufficiently deteriorated condition to warrant replacement.

Results

Portions of aluminum pipe requiring replacement were all outside the reactor building. These included:

1. 160 feet of 10-inch core exit and sump return lines.
2. Seventy feet of 4-inch suction line for the scavenger demineralizer pump between the pool and the pump room.
3. Seventy feet of 3-inch discharge line between the scavenger demineralizer pump and the pool high gutter.

Replacement of faulty pipe and testing to verify the integrity of the newly installed stainless steel pipe required five calender weeks, four of which were unscheduled down time.

Though some contamination of the soil in the vicinity of the pipe occurred, the levels in the leakage water were below the MPC for discharges to unrestricted areas specified in Appendix B of 10 CFR 20. The highest level of soil contamination detected was 7.5 picocuries per gram in soil samples from one foot below grade near the leaking pipe.

Corrective Action

These lines were replaced with schedule 10, type 304 stainless steel welded pipe because of its availability and improved corrosion resistance characteristics. For protection against electrolytic action, the flanged connections between dissimilar metals, none of which occur underground, incorporate dielectric inserts in the bolt holes which, along with neoprene gaskets, should minimize corrosion from this source. The new lines were coated with a vinyl-based primer and wrapped with a double layer of vinyl tape.

Discussion

The conditions leading to this failure were recognized by facility management to be similar to those experienced at ORNL, where aluminum piping has failed under several different conditions. These include conditions where the pipe was buried and where it was encased in concrete with the annular space between the pipe and the concrete filled with glass wool. In every case the corrosion occurred from the outside of the pipe and not from the inside, which was in contact with high

quality reactor primary coolant water. The failure always occurred from pitting-type corrosion where localized corrosion occurred in deep pits which eventually penetrated the pipe wall. Also, in every case the pipe was wet; and, in practically all cases at ORNL, the pipe was bare, i.e., it was not tarred and coated.

The Oak Ridge experience has led to the conclusion that until the mechanism of and methods of preventing aluminum corrosion are better understood, it is not worthwhile to take the chance of piping failure in systems as important as reactor primary coolant systems. The cost of repairing pipe failures coupled with the cost of reactor downtime outweighs the added cost of stainless steel piping. Aluminum should be used only where it is accessible for inspection or where it can be easily replaced.

Operators of facilities at which aluminum pipe is used underground should consider instituting a surveillance program for such piping.

References

Letter, Industrial Reactor Laboratories, Inc., to USAEC, IRL Reports Corrosive Failure of Primary Piping, Docket 50-17, Jan. 11, 1968, available at USAEC Public Document Room.

ROE 68-11

ACCIDENTAL CRITICALITY EXCURSION

Summary

On May 18, 1967, an accidental excursion of 4×10^{16} fissions took place in the critical mockup of a high power density reactor. There was neither damage to the equipment nor significant exposure of persons; nevertheless, the incident indicated poor practice and an undesirable interpretation of operating procedures which has been corrected.

Circumstances

The reactor mockup is fueled with elements composed of fully enriched uranium in a graphite matrix, and a smaller number of graphite moderating elements. This permits a relatively small core volume (250 liters). The core, housed in a graphite cylinder, drops out of its Be reflector for loading. Control and safety drums are within the annular reflector.

Before the incident, fuel along the core axis was replaced by additional moderating elements to investigate flux-trap effects. Instead of the usual step-wise interchange of elements, the entire moderating island was installed. Then, instead of step-wise multiplication measurements while inserting the core into reflector, which is proper for initial approaches to criticality, there were no measurements during interrupted insertion. It had been inferred from the behavior of different moderating elements in an earlier mockup that the overall reactivity change would be minor. This was a serious mistake, for the actual change proved to be about $10. Before complete closure was achieved, a very short period and scram (dropping the core and actuating the safety drums) occurred.

Results

Exposures as a direct result of the excursion are estimated to be about 5 mrem. Several personnel who investigated the assembly afterward received nearly 200 mrem. Except for the induced radioactivity, the mockup was unaffected by the incident. Based on radiochemical analysis of fuel samples, the excursion yield was established at 4.1×10^{16} fissions.

Corrective Action

Although the excursion was well within the protective capability of the facility, the facility operating procedures were intended to deter practices such as those involved. Technically, the procedures were deficient in giving no guidance for identifying "minor" versus "major" reactivity changes. This deficiency has been corrected. Practically, however, it is believed that subsequent highlighting of poor practice was the more important corrective influence.

ROE 68-12

LUBE OIL BLOCKAGE BY TAPE FRAGMENT

Summary

Maintenance on a reactor primary coolant pump motor involved disconnecting and reconnecting the bearing oil supply and drain lines. When the motor was started, the temperature on the inboard bearing slowly increased to above normal. The motor was shut down and it was found that a small piece of "Teflon" tape was partially blocking an orifice, thus cutting down the oil flow to the bearing. No damage resulted.

Circumstances

The "Teflon" tape in question is a thin (∿1 mil), soft, pliable material made especially for use as a sealant on threaded pipe joints in place of white lead or other pipe joint dope. Maintenance has used this type extensively throughout the plant for over three years without incident.

It is believed that during maintenance the female portion of one of the pipe joints was not properly cleaned and that a fragment of used tape remained in the fitting. When the joint was then remade and oil flow was established, the piece of tape was carried to the orifice causing the partial blockage.

During startup the temperatures of all bearings are scanned automatically, approximately every two minutes. During level operation the scan is reduced to about five minutes per cycle. The reactor operator is alerted by an alarm if any bearing temperature exceeds a predetermined limit. The alarm temperature is selected to provide sufficient time to shut down the motor before there is damage to the bearing. Instruments functioned properly, an alarm sounded, and the pump motor was promptly shut down; there was no damage.

Conclusions

a. A tape-type pipe joint sealant should not be used in applications where a piece of free tape in the system could produce a significant adverse effect on safety or operation.

b. Where pipe joint tape is used, maintenance procedures should include precautions to prevent pieces of tape breaking off and entering pipelines.

Disposition

Action through procedure revision and instruction to personnel has been taken to correct the assumed cause.

ROE 68-13

LOSS OF WATER IN THE STEAM DRUM OF AN EXPERIMENTAL POWER REACTOR

Summary

Water was lost to the steam drum and evaporators of an experimental power reactor when the water level controller-recorder failed. As the alarm circuit originated from the same source, no low-level alarm was obtained. The response of the operating crew to the emergency was adequate in terminating the incident without allowing damage to the evaporators. Approximate changes have since been made to pertinent instrumentation and to associated procedures.

Circumstances

The reactor was being operated at about 50% power when it was noted that the primary system temperature was rising. Secondary pump problems had been evidenced earlier in the day, so the temperature anomaly was initially attributed to a secondary pump failure. A shutdown was immediately initiated. While the shutdown was in progress it was detected that the water level in the steam drum had fallen below the sight glass. Feedwater flow was discontinued as a precaution against thermal shocking of the evaporators. The system was allowed to stabilize below 400°F and an evaluation of possible thermal stress damage to the evaporators was made before the evaporators and steam drum were recharged with preheated water. There has been no evidence that the loss of water damaged the system. The system has performed satisfactorily during six months of operation at varying temperatures following the incident.

Causes

The steam drum level recorder-controller alarm instrument failed because a fiber gear in the recorder had worn to the extent that the gear teeth were no longer able to engage. This failure deprived the system of the necessary signal to provide adequate feedwater flow to the steam drum, it deprived the operators of a recorder indication of the falling level, and it deprived the operators of an alarm signifying the loss of water from the steam drum.

There is a closed circuit TV system that shows the steam drum sight glass on a monitor in the control room, but because of the pool location of the monitor in the control room and the poor contrast of the picture, it has not been effectively utilized. It is interesting to note that a high-level alarm on the condensate storage tank had been removed one week prior to the incident as part of an approved system modification. Had this alarm been present, it would have signalled the dislocation of the water from the steam drum and evaporators.

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Corrective Action

Control, recording and alarm functions have been separated on this system, and other systems at the facility are being revised as necessary to conform with the higher standards for high-pressure steam plants as proposed by the Instrument Society of America. The steam drum sight glass TV monitor has been moved to the front of the control room, the picture image has been sharpened, and routine checks of the monitor are now required. In addition, the high-level alarm has been reinstalled in the condensate storage tank, and all plant emergency procedures are in the process of being reviewed for simplicity and clarity.

Conclusions

All safety associated instrumentation systems should be designed to separate control and alarm or control and record functions. This principle cannot be overstated and would, had it been respected on the steam drum level instrumentation, have prevented the subject incident. It is also apparent that the closed circuit TV system used on this case gave the illusion of a safeguards mechanism that was beyond its capability. The immediate, appropriate reaction of the operations crew to the loss of water gave positive proof of the value of a through training, testing and certification program that includes emergency plant procedures.

References

Argonne National Laboratory, EBR-2 Steam Generator Level Controller Malfunction, p. 94, Reactor Development Program Progress Report, November 1967, USAEC Report ANL-7399, December 1967.

ROE 68-14

ERROR IN SHIM CYLINDER INDEXING

Summary

An error in indexing outer shim control cylinders during assembly occurred at a large testing reactor. The error was not discovered until the reactor was operated and the shim position at critical was found to be appreciably different from the predicted value.

Circumstances

The reactor is light water cooled and beryllium reflected. Part of the reactivity control is provided by 16 rotating outer shim cylinders around the core periphery. Each cylinder extends the full length of the core and is made of beryllium with a hafnium plate occupying a 120° segment of the circumference. As originally assembled, the hafnium plates were positioned at their closest approach to the core in their full "in" position. The shim cylinders can be rotated through 160°.

The outer shims functioned as intended during the intial core loading. Critical facility studies had shown that the reactor power distribution could be improved by reversing the direction of rotation of all outer shim cylinders and by shifting the zero position of eight of the cylinders 35°. These changes were made in the reactor before the second core was loaded and operated.

During startup with the second core, the outer shims had to be withdrawn in a group approximately 40° beyond the predicted position to achieve criticality. In the ensuing investigation it was discovered that eight of the sixteen cylinders had been installed so that their zero positions differed from that intended by 70°. When the shim drivers were actuated to withdraw the outer shims, these eight cylinders first inserted hafnium until they reached the design zero position and then began to withdraw.

Causes

The modified shim cylinders had been installed correctly according to the approved procedure. However, as a result of the modification, the drive part used to index the drums had to be supplied in three different types, each slightly different but physically interchangeable. Two of these types were inadvertently interchanged in the procedural tabulation identifying the shim cylinders and indexing parts. Once assembled, the shim cylinder position could not be checked visually.

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Corrective Action

The shim cylinders were reassembled in the proper orientation. It is expected that these cylinders will require replacement during the life of the reactor and it is intended on the next procurement to modify the design so that the various parts required cannot be improperly interchanged.

ROE 68-15

FAILURE OF LOG COUNT RATE CHANNELS

Summary

During the initial fuel loading and startup of a large testing reactor, both of the two permanently installed log count rate channels were found to be inoperative. The chambers had pulled away from their connecting signal cables.

Circumstances

For the initial loading a temporary count rate channel had been provided. This chamber was installed near the center of the core and near the startup source. The temporary channel was required to be operable and on-scale before fuel loading could begin. The permanently installed log count rate channels were positioned outside the core and were not expected to register any activity until the core had been about half loaded into the reactor. When this point was reached, neither channel showed any activity and an investigation was made. Both channels were completely dead.

Causes

The nuclear instrumentation chambers for this reactor are located in vertical thimbles inside the reactor vessel but outside the core. The chambers and cables had been assembled and completely checked outside the reactor. Then the chambers and their carriages were lowered into the thimbles. No source check had been made after final positioning. The chambers for the level and log-N period channels were provided with a separate support cable which effectively relieved any strain on the signal cable and were still fully operational. However, the 25 pound chamber and carriage for each log count rate channel was equipped with a braided wire support. When subjected to full load, the log count rate chamber support stretched sufficiently to stress the signal cable and to pull loose a pin connection at the chamber.

Corrective Action

The signal cables were reconnected and the support cable was shortened by two inches to eliminate signal cable stress. Operation of the chambers was verified with a neutron source after they had been returned to their normal position.

ROE 68-16

UNPLANNED NUCLEAR TRANSIENT

Summary

While control rods were being withdrawn at a transient test reactor with the intent of placing the reactor on a 15-20 second period, the reactor underwent an unplanned self-terminating transient. As the reactor was designed to operate in this mode under other circumstances, no personnel hazard or facility damage resulted.

Circumstances

In preparation for running a transient with a 100 msec period, the reactor was being taken critical under manual control. The intended procedure was to withdraw the control rods in steps to about 12 inches in fast speed, switch the rod drives to slow speed, adjust all rods to the same position and then withdraw cautiously to 14.7 inches, a position calculated to give a 16 to 20 second period. At a power level of 50 watts, the power would then be leveled off and the critial rod position established. This position was predicted to be 14.52 inches. In this case the operator deviated from intended practice. He pulled all control rods in fast speed until the fastest rod reached 14.5 inches. He stopped at that point to mark charts and check the audible count rate. His intention was then to switch to slow speed, withdraw rods until the fastest rod was at 14.7 inches, and then level the rods, which should have placed the reactor on the desired period. When he made the first withdrawal, about 1 or 2 seconds in duration, the audible count rate suddenly increased rapidly and then saturated. The second operator in the control room immediately scrammed the reactor, although by the time he could react, the transient had already passed its peak.

Results

Later data analysis indicated that the reactor had undergone a transient with a period between 50 and 90 msec and a peak power of about 12.5 Mw. The reactor had previously been subjected to transients in this range intentionally with no facility damage, and no damage occurred in this instance either. Because of the nature of normal reactor operations, the reactor is controlled from a center about one-half mile from the reactor building and access to the reactor building is not permitted during any reactor operation. Thus, there was no possibility of injury to personnel.

Causes

The subsequent investigation was not able to establish the cause of this occurrence definitely, but a most probable sequence of events was established as follows:

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a. When the operator selected slow speed at a rod position of 14.5 inches, he probably did not depress the "slow" pushbutton far enough to actually change speed, and he did not verify the speed change by checking the indicating lights provided or by observing the subsequent speed of withdrawal on the rod position indicators.

b. He then pulled rods for the one to two seconds normally required to reach the intended position in slow speed, but actually pulled rods farther in fast speed. The difference in rod withdrawal would have been suficient to cause the resulting transient.

c. The rod drives were in "slow" speed after the transient, so it is postulated that the operator may have subconsciously realized what had happened and depressed the "slow" button again after the transient. The operator was unable to recall much of what happened immediately after the transient.

There was no indication after extensive investigation that any physical abnormality or malfunction in the reactor or related systems had caused the transient. Operator error appears to be the most probable cause.

Corrective Action

This reactor, because of the nature of its normal operation, necessarily depends heavily on administrative controls rather than instrumented safety circuits for safe operation. In this case the administrative control failed. Corrective action was limited to re-emphasis of these controls combined with suitable disciplinary action. Some modifications of the reactor control circuitry to reduce susceptibility of this type of incident could be made, but because the reactor program was being terminated, this possibility was not pursued.

References

T. G. Taxelius (Ed.), SPERT III E-Core, pp. 1-2, Quarterly Technical Report, SPERT Project, April, May, June 1968, USAEC Report IDO-17289, Phillips Petroleum Company, January 1969.

ROE 68-17

A SUMMARY OF EVENTS INVOLVING THE TRANSIENT RODS OF TRIGA RESEARCH REACTORS

Summary

Recent unrelated events have involved the transient rods at separate TRIGA research reactors. They are presented together because they demonstrate the need for close surveillance during checkout and maintenance. In all cases, the problems were easily corrected and there were no harmful effects as a result of the malfunctions.

Circumstances

A TRIGA reactor uses a transient rod to introduce a step reactivity change (pulse) up to a few dollars of reactivity. It is pulsed from low power (∿ 1 kw) up to a few thousand megawatts. The pulse releases only a few megawatt seconds of energy (about 23 megawatt seconds for a $3 pulse) and is turned around by a strong prompt negative temperature coefficient. The reactivity pulse is generated by introducing compressed air through a solenoid valve into a cylinder which contains a piston that is attached to the neutron absorber section of the rod. The following describes difficulties encountered with the transient rod operation:

A. Solenoid Valve Failure

During checkout, a failure occurred in the pressure release mechanism of the solenoid valve that controls the air pressure to the transient rod. The rod was in the withdraw position and failed to drop when the solenoid was de-energized. After the failure, the air pressure was released by venting the pressure line, and the rod dropped. This indicated the release mechanism was at fault and the rod was not sticking. Until a replacement valve was obtained, the air pressure supply to the solenoid valve was reduced from 75 to 40 psi and operating performance was normal.

Examination showed that the rubber seats in the defective solenoid were worn. The worn seats allowed pressure to be applied to the reverse side of the release diaphragm in the valve, thus preventing the diaphragm from opening and releasing the pressure on the rod when the solenoid was electrically de-energized. The system performed as expected after the defective solenoid valve was replaced.

B. Solenoid Valve Failure

During a routine pre-startup checkout, the transient rod scram time was noted to be two to three seconds instead of the normal one-third second. In repeated tests, it was determined that the solenoid valve was not releasing the air from the cylinder. Examination of

the valve internals disclosed excessive rubbing of the needle assembly on the valve body. A new solenoid valve was installed and there has been no recurrence of this problem.

C. Interlock Failure

With the mode switch in the steady-state position, a failure was noted during a routine check of the interlocks. The failure allowed the transient rod to be raised despite the fact that the shim rod was above its fully inserted position. The purpose of the interlock is to prevent an operational error that would allow a pulse to occur with the reactor in the steady-state mode of operation. The defective relay was replaced and the interlocks checked out satisfactorily.

D. Switch Malfunction

During a routine shutdown, a malfunction occurred in the de-energizing circuit of the solenoid valve, failing to release the air pressure and preventing the transient rod from dropping. The air pressure was released by venting the pressure line; the rod dropped indicating the rod was not sticking. However, the rod moved out again as air pressure was reapplied to the solenoid. When the electrical power that controls the solenoid was turned off, the rod dropped.

Although the actual cause of the malfunction was not pinpointed, the microswitches were removed, checked, cleaned and remounted. Repeated checks of the system failed to indicate any recurrence of the problem.

E. Rust in the Carriage

During a routine checkout prior to startup and in the steady-state mode, the operator noticed the ready light was lit under the transient rod fire button with the transient rod carriage raised. This implied that the transient rod could be fired out at an unacceptably high operating power. Because of interlocks in the system, the transient rod cannot normally be fired out in the steady state mode. The position indicating microswitches were checked and found to be operating satisfactorily.

During further surveillance of the system, the transient rod carriage was raised and the trouble was immediately apparent. With no air applied, the carriage was picking up the transient rod. Closer inspection revealed that rust was forming around the inside top section of the carriage. (Even with the rust inside the carriage, the force binding the rod was very slight and when scrammed, the rod would fall freely.) The rust was removed, the transient rod carriage was cleaned and the system functioned normally. Routine

lubrication of the carriage has been instituted as part of the monthly checkout.

Conclusion

Repeated use, fatigue, wear or simply component failure are the contributing factors in these malfunctions. In no instance can the cause of the events described be attributed to operator error. Routine checks and increased preventive maintenance programs, particularly as the installation ages, will considerably decrease the probability of occurrences of these types of failures.

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