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REVIEW OF SAFETY-RELATED OCCURRENCES FROM

A MATERIALS PERFORMANCE POINT OF VIEW

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Safety related occurrences - failures, incidents and deficiencies - reported by light water reactor nuclear power plants are reviewed. The period covered is 1967 through 1972. The items selected for review were limited to those in which a material failure was involved. Tables were prepared for each year which indicates the facility and components involved, the cause of failure, and the area in which the deficiency originated, i.e., material selection, design, fabrication, installation, operation, or maintenance. The tables are discussed, and areas where additional efforts can be made to improve materials performance is indicated.

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Introduction

The material for this review was obtained from USAEC licensee reports of abnormal occurrences and unusual events at nuclear facilities. The requirements for reporting those safety-related occurrences are described in detail in AEC Safety Guide No. 16 (1).

All of these required reports are received by the Nuclear Safety Information Center (NSIC) at the Oak Ridge National Laboratory where they are abstracted and indexed for computer storage and retrieval. Compilations of the occurrence reports are prepared by NSIC each year and made available through the National Technical Information Service (2, 3, 4, 5).

The annual compilations were reviewed and those items which appeared to be related to the subject of materials performance were selected. The review covers those occurrences reported during 1967 through 1972 and has been limited to light-water nuclear power stations because these facilities are presently commanding the emphasis and interest.

At the beginning of 1973 there were 28 nuclear power stations in operation with a combined capacity of 14, 407 MW(e). More than half of the capacity [7910 MW(e)] was added in 1971 and 1972. Consequently, the following reviews will concentrate on those occurrences reported in 1972 and only a cursory review will be given of those occurrences reported in 1967 through 1971.

The performance of materials is dependent upon the extent of consideration given to the materials properties and capabilities for a specific application. To obtain the required performance, adequate consideration must be given to the selection of the material, the design application, fabrication, installation, operation, and maintenance. Failure or poor performance of materials can usually be traced back to inadequate consideration of one of these aspects. The experiences reported herein tend to substantiate this and should provide some guidance as to where additional efforts can be made to improve materials performance.

Review of Occurrences

The selected incidents have been tabulated according to the year in which they were reported. The tables indicate the facility and the specific item involved, the cause of the failure, and the area in which the deficiency occurred; i.e., selection, design, fabrication, etc.

1967 Occurrences

Table I lists the selected incidents which were reported in 1967.

The Bonus and Pathfinder plants reported cracking and embrittlement of boron-stainless steel control rods due to neutron irradiation damage and the control rods were replaced with different materials, such as hafnium or boron carbide.

Bonus and Dresden 1 experienced core flow reductions due to corrosion products from copper alloy feedwater heater tubes.

Stress corrosion cracking occurred on the control rod drives at Bonus and on the nozzle safe ends at Tarapur, a BWR in India.

Table I. 1967 Occurrences

Facility	Item	Cause	Deficiency
Bonus	B-SS control rods	Embrittlement - radiation damage	Selection
Pathfinder	B-SS control rods	Embrittlement - radiation damage	Selection
Bonus	Cu-alloy feedwater heater tubes	Corrosion	Selection
Dresden 1	Cu-alloy feedwater heater tubes	Corrosion	Selection
Bonus	Control rod drives	Stress corrosion cracking	Fabrication
Tarapur	Nozzle safe ends	Stress corrosion cracking	Fabrication

1968 Occurrences

Table II lists the selected incidents which were reported in 1968.

A fire occurred at San Onofre due to thermally overloaded wires which were in an overfilled cable tray.

Thermal shield and core barrel bolt fatigue failures were reported at four facilities (Yankee, Sena, Trino, and KWO), due to hydraulically induced vibration (6).

At Oyster Creek 1, stress corrosion cracking was observed on some of the nozzle safe ends, and problems occurred as the result of incomplete fusion of welds of the control rod drive tubes to the stub tubes.

Table II. 1968 Occurrences

Facility	Item	Cause	Deficiency
San Onofre	Cable trays	Overloaded (fire)	Design
Yankee	Thermal shield bolts	Flow vibration	Design
Sena	Core barrel bolts	Flow vibration	Design
Trino	Core barrel bolts	Flow vibration	Design
KWO	Core barrel bolts	Flow vibration	Design
Oyster Creek 1	Nozzle safe ends	Stress corrosion cracking	Fabrication
Oyster Creek 1	Control rod drive tubes	Weld flaws	Installation

1969 Occurrences

Table III lists the incidents reported in 1969.

Elk River reported cracking of boron-stainless steel control rods and replacement with rods of boron carbide in stainless steel tubes.

Elk River, Dresden 2, and Oyster Creek 1 reported slow response times for the control rod drives caused by plugging of the control rod drive inner screens by corrosion deposits. The 1 mil mesh screens were removed or replaced with 10 mil screens.

Elk River and LaCrosse experienced stress corrosion cracking of sensitized nozzle safe ends.

Table III. 1969 Occurrences

Facility	Item	Cause	Deficiency
Elk River	B-SS control rods	Embrittlement - radiation damage	Selection
Elk River	Control rod drives	Inner filter plugging	Design
Dresden 2	Control rod drives	Inner filter plugging	Design
Oyster Creek 1	Control rod drives	Inner filter plugging	Design
Elk River	Nozzle safe ends	Stress corrosion cracking	Fabrication
LaCrosse	Nozzle safe ends	Stress corrosion cracking	Fabrication

1970 Occurrences

Table IV lists the selected incidents reported in 1970.

Connecticut Yankee and San Onofre reported thermal shield failures due to vibration caused by hydraulic forces.

Indian Point 1 reported a thermal sleeve failure in the makeup water nozzle. The weld failed due to fatigue caused by thermal shock and cycling as a result of cold water makeup to the reactor.

H. B. Robinson 2 reported the rupture of a main steam safety valve pipe nozzle as a result of overloading. The static stress analysis did not adequately consider the reaction forces for the full capacity discharge load conditions.

Nine Mile Point reported the failure of the core spray nozzle safe end due to stress corrosion cracking of the furnace sensitized 304 stainless steel.

Table IV. 1970 Occurrences

Facility	Item	Cause	Deficiency
Connecticut Yankee	Thermal shield	Flow vibration	Design
San Onofre	Thermal shield	Flow vibration	Design
Indian Point 1	Thermal sleeve	Thermal shock	Design
H. B. Robinson 2	Safety valve pipe	Reaction forces	Design
Nine Mile Point	Nozzle safe ends	Stress corrosion cracking	Fabrication

1971 Occurrences

Table V lists the selected incidents reported in 1971.

Turkey Point 3 reported the failure of two 12-in. safety valve stub headers in which 3 of 4 relief valves were catastrophically ejected due to the reaction forces generated by the discharging of the relief valves.

Dresden 2, KRB (German BWR), KWL (German BWR), KWO (German PWR), and Ginna reported zirconium fuel clad failures due to internal hydriding as a result of moisture introduced during fuel rod manufacture.

The crossmixing of fuel pellet enrichment during assembly of fuel rods affected several BWR's (Dresden 2 and 3, Monticello, Quad Cities 1 and 2, and Vermont Yankee).

The separation of Inconel cladding from the steel tube sheet in the steam generators was reported to have affected 7 units. The cladding had been attached by explosive welding. It was removed and replaced by weld overlay.

Stress corrosion cracking of stainless steel piping was reported by 4 BWR's (Dresden 1, Humbolt Bay, LaCrosse, and Nine Mile Point).

Table V. 1971 Occurrences

Facility	Item	Cause	Deficiency
Turkey Point 3	Safety valve pipe	Reaction forces	Design
3 BWR's and 2 PWR's	Zirconium fuel cladding	Internal hydriding	Fabrication
6 BWR's	Fuel rod	Pellet enrichment crossmixing	Fabrication
7 PWR's	Steam generator	Explosive welding incomplete	Fabrication
4 BWR's	Piping	Stress corrosion cracking	Installation

1972 Occurrences

Table VI lists the selected incidents reported in 1972.

During leak testing at Dresden 2 and 3, Buna-N rubber seats on containment air locks, vacuum breaker isolation valves and other valves associated with containment isolation were found to be badly worn and cracked. The valves were sent back to the manufacturer to be reseated with an improved material.

Dresden 2 also reported problems with the Phenoline 368 coating on the inside of the pressure suppression chamber. The coating above normal water level was in good condition, but below water level there was deterioration in the form of blistering and delamination. The material is to be replaced, below water level, with an inorganic zinc coating ~~because~~ which if it fails, ~~it~~ will decompose into minute particles (micron size) and not in sheets or pieces which might block the ECCS strainers.

Fermi 2 reported cracks in the concrete base slab. The cause was attributed to the high content of fines in the concrete mix for durability against chemical action of sulphurous ground water. Larger amounts of water were required which resulted in greater shrinkage during curing and cracks resulted.

At Millstone, condenser tube failures resulted in salt water entering the hot well and was transported to the vessel. One-hundred sixteen of 120 core instruments exhibited a decrease in resistance to ground and failed as a result of shallow stress corrosion attack in the crevice regions at the tips of the LPRM chambers. The cause of the condenser tube failures was attributed to corrosion, erosion, and vibration-induced abrasion. The aluminum-brass tubes were replaced with copper-nickel tubes.

At Surry 1, the reaction forces of the steam jet from a decay heat release system valve were sufficient to cause pipe deflection and movement of the line support hangers such that nonradioactive steam was released to a room instead of a vent pipe and two technicians received fatal burns.

Ginna, H. B. Robinson 2, Point Beach 1, and Beznau 1 (PWR in Switzerland) were affected by fuel densification effects due to unpressurized fuel rods.

At the conclusion of the first phase of the hot functional testing program, Oconee 1 reported that several internal components had failed which included incore instrument nozzles and guide tubes, and the thermal shield. The fatigue failures were due to hydraulically-induced vibrations. Redesign involved increasing the natural frequency of the incore instrument nozzles and guide tubes and providing a more rigid attachment of the thermal shield to the core barrel.

Table VI. 1972 Occurrences

Facility	Item	Cause	Deficiency
Dresden 2 and 3	Buna-N rubber seals	Cracking and wear	Selection
Dresden 2	Torus paint	Corrosion	Selection
Fermi 2	Concrete base slab	Cracking	Selection
Millstone 1	Condenser tubes	Corrosion	Selection
Surry 1	Steam release valve	Reaction forces	Design
4 PWR's	Fuel rods	Densification	Design
Oconee 1	Thermal shield and vessel internals	Flow vibration	Design
Oyster Creek 1	Safety valves	Stress corrosion cracking	Design
4 BWR's	Safety relief valves	Premature operation	Design
Indian Point 2	Steam generator	Explosive welding incomplete	Fabrication
Point Beach 1	Steam generator	Explosive welding incomplete	Fabrication
LaCrosse	Reheater level controller	Poor welds	Fabrication
Vermont Yankee	Zircoloy fuel cladding	Internal hydriding	Fabrication
Dresden 2	Flow restrictor	Inadequate weld	Installation
Quad Cities 2	Pipe hangers	Shearing	Installation
Calvert Cliffs 1	34-in. pipe	Heating and bending	Installation
Duane Arnold	Piping	Heating and bending	Installation
Connecticut Yankee	Isolation valve	Environment	Operation
6 plants	Miscellaneous	Lubrication	Maintenance
5 plants	Miscellaneous	Cleanliness	Maintenance

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At Oyster Creek 1, 7 of 16 safety valves were found with cracked seat bushings. The environment was steam and water, wet oxygen, and Cl contamination. The cause was stress corrosion cracking. All cracked bushings were replaced with the same material, i.e., 304 SS, and a drain hole was added to prevent water from standing in the valve base.

A frequently reported item is the actuation of safety or relief valves at some pressure other than at the setpoint. At Monticello, the relief valves would actuate at a pressure above the setting and then later they would actuate below the setting. The cause was attributed to the lack of a common test procedure. At Pilgrim, they found that using nitrogen in the shop to determine the setpoint results in a higher setpoint when used in a steam atmosphere (76551), so they use a lower nitrogen pressure setpoint. In March of 1973, a report was received concerning investigations into the causes of premature safety valve actuations which was initiated after several occurrences at the Dresden station in which safety valves opened prematurely. At that time 7 mechanisms for premature operation were identified. The springs were designed to operate in an environment of 150°F but the measured temperature on the spring surface was found to be 250°F. Removal of insulation from the valve reduced the spring temperature to 150°-180°F. Tests showed that at 200°F a 3% relaxation occurs and results in a 36 psi drop in setpoint. In addition to spring relaxation due to temperature effects, other possible causes included mechanical shock, pressure pulse effects, alignment, valve temperature effects on the valve body, and valve nozzle seat distortion (7).

Indian Point 2 and Point Beach 1 reported on the separation of the cladding from the steam generator tube sheet which had been attached by explosive welding. The cladding was removed and the weld material was manually deposited.

At LaCrosse, while at 97% power, a loud explosion occurred and the mezzanine floor started filling with steam. The float cage of a reheater level controller had ruptured. The cause was insufficient weld penetration at a branch connection, poor weld preparation, and inadequate weld reinforcement metal. The unit was factory made.

Vermont Yankee reported cladding failures due to internal hydriding as a result of inadequate vacuum out-gassing during a fuel rod fabrication.

At Quad Cities 2, during startup testing four failed bolts for the torus suction header hanger were found. The failures were due to shearing with no evidence of fatigue. The bolt guide holes on the torus and header brackets had been torch cut to arbitrary sizes as specified on the construction drawing. The respective bolt holes in the hanger strap pieces were punched instead of drilled. Punching had left a sharp edge and some holes were enlarged with a

to allow the bolts to pass through rather than cleanly drilled to size

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torch to achieve fit. Also fully threaded bolts had been used. The straps and hangers were replaced, the bolt holes were drilled and bolts with non-threaded bearing surfaces were installed.

At Calvert Cliffs 1, during installation of a 34-in. pipe from one of the steam generators, the alignment at a pair of 90° elbows was incorrect. So a come-along was used while heat was applied with heating coils. The pipe was bent resulting in a bulge. The piping was replaced and installed correctly.

At Duane Arnold, at least 16 instances of unauthorized heating and bending of pipe were found. Heating of impact-tested carbon-steel pipe to unknown temperatures produces an indeterminable effect on the NDT. All of the impact-tested material that had been heated was to be replaced.

A failure is described which occurred twice at the same plant. The first time it occurred may be ascribed to inadequate design consideration, but the second occurrence was due to inadequate ~~design~~ consideration by the operating personnel.

In January of 1972, at a PWR (Connecticut Yankee), a decreasing trend in the weight of the containment air was noted. It was found that the 8 in. containment purge exhaust bypass isolation valve had a cracked yoke and the valve was open 3/4 in. A new yoke was installed and an automatic containment weight of air alarm with a main control board annunciation was added in case it happened again. The valve was located outside the containment building and the cause of the break was attributed to thermal stress due to seasonal temperature fluctuations. In December an alarm was received and again cracks were found in the yoke and the valve open 3/4 in.; a new yoke was installed and this time the bypass and main purge valves were electrically heat traced and insulated to prevent freezing. Also, a low point drain was installed downstream of each valve.

Lubrication problems were reported at 6 plants. Valves failed to open due to hardened lubricant and excessive wear on gears was reported. At Pilgrim inadequate lubrication resulted in seizure of the bearings on a 2700 gpm salt service water pump and the line shaft was broken. The lubrication program was revised.

Five plants reported problems due to inadequate cleanliness. At Connecticut Yankee, a containment isolation valve failed to close during a test due to a dirty solenoid. At Palisades, a main steam stop valve failed to operate properly during a test. Six of 8 solenoid valves that control the main steam stop valves were sticky due to dirt. They were cleaned and relubricated and the linkages covered with plastic covers to minimize dirt pickup. Also dry lubricant was applied. Also, at Palisades, dirty contacts on a relay caused erratic operation of the diesel generator governor during a test. The relays were replaced with a closed contact type relay that will not be affected by dirt.

Summary and Conclusions

Selection deficiencies have not been too numerous and do not appear to be a major problem area.

Design deficiencies were quite numerous, but this is not unusual since it is an all encompassing area and it requires con- conceptual thinking. In addition, several of the failures attributed to design deficiencies were due to the same design inadequacy which resulted in failures at more than one facility.

Fabrication and installation deficiencies were primarily associated with welds or welding and in particular the sensitization of stainless steel making it susceptible to stress corrosion cracking.

Deficiencies attributed to operation were minimal. This is because operational deficiencies were primarily manifested as improper valve or switch actuation, procedure violation, or inadequate response to non-routine situations and did not result in a material failure. The number of maintenance deficiencies can be reduced by placing more emphasis on lubrication and cleanliness; one obvious method is to increase the surveillance frequency of items susceptible to dirt accumulation and to items requiring lubrication.

The review of the occurrences indicates that many of the problems experienced by one facility have emerged at other facilities. This is particularly true if the failure is generic. This repetition of failures is apparent when we note that (a) stress corrosion cracking of nozzle safe ends were reported in 1967, 1968, 1969, 1970, and stress corrosion cracking of sensitized material was reported in 1971 and 1972; (b) thermal shield and vessel internal failures due to hydraulically-induced flow vibration were reported in 1968, 1970, and 1972; and (c) pipe ruptures due to inadequate consideration for the reaction forces of operating relief valves were reported in 1970, 1971, and 1972.

This repetition of failures can be reduced if when a deficiency is discovered it is given adequate consideration and the information is distributed throughout the nuclear community. However, it will also require recognition by the recipient of this information. The AEC has taken the initiative in this respect and in 1972 we noted a surge in the number of notification letters sent to the utilities informing them of deficiencies and requesting them to perform an analysis to determine if modifications were required. The number of facilities reporting that backfitting would be required was significant and by making the necessary modifications potential failures were eliminated or at least minimized. Although the AEC notification letters were sent as a regulatory function in the interest of the public safety and welfare, this appears to be a

Scottvaluable service to the nuclear community and the results should be observed in the form of increases in plant availabilities.

This review has concentrated on actual occurrences because it is the operating experiences that provide the proof of the adequacy of the consideration given to the selection, design, fabrication, and installation of the materials used in the nuclear power plants. Experiences in the form of failures, deficiencies, and abnormal occurrences should not be considered as a stigma but rather as a base of knowledge to be applied toward improving the performance of materials and consequently toward improving the performance of power plants in the nuclear industry.

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