LESSONS LEARNED FROM MAJOR CORE DAMAGE ACCIDENTS

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ABSTRACT

The cumulative experience with major core damage accidents is reviewed to identify trends, recurring problems and nonproblems, and generic lessons learned. Fourteen major core damage accidents are examined. Although these accidents occurred in a wide range of reactors that were generally unique and nonprototypical of modern plants, some trends, observations, and conclusions are identified that may be relevant to current practice.

INTRODUCTION

The nuclear industry has learned much from major core damage accidents. Most of the lessons learned have been identified and implemented by responding to individual events following their occurrence. In the current work, we have reviewed the cumulative experience with major core damage accidents to identify trends, recurring problems and nonproblems, and generic lessons learned. Particular emphasis is placed on identifying the root causes of these events to assist in the prevention of similar accidents.

Fourteen major core damage accidents were identified that had occurred in experimental, developmental, and commercial electricity-producing plants, large research and test reactors, and production and propulsion reactors. Destructive tests were excluded from the data base, as were nonreactor accidents and events that did not result in severe fuel damage or melting.

DATA BASE

The 14 accidents and the plants in which they occurred are summarized in Tables I and II. Of these 14 accidents, 5 occurred in water-cooled plants, 6 in gas-cooled plants, and 3 in sodium-cooled plants. All involved severe damage or melting in one or more fuel assemblies. About one-half affected more than 10% of the core. These accidents are briefly described below in chronological order.

1. NRX (1952)[1-3]

The NRX was a 30-MWt, heavy-water-moderated, light-water-cooled research and testing reactor. During a test with the reactor initially shut down, a series of errors resulted in several control rods being withdrawn from the core, making the reactor supercritical. Manual scram was initiated, but power continued to increase because of insufficient control rod insertion. Reactor shutdown was achieved by dumping the heavy-water moderator, but not before 22 of 192 fuel assemblies had suffered extensive damage, including melting and metal-water reactions.

2. EBR-I (1955)[1,4-6]

The EBR-I was a 1.4-MWt, sodium-potassium-cooled experimental breeder reactor. It had two principal shutdown mechanisms – one fast and one slow – involving control rods and movable blankets. During a test in which the reactor was to be manually scrammed in the fast mode, a slow scram was initiated by mistake, followed 1 to 2 s later by a fast scram. About 40 to 50% of the core melted due to the delay.

3. Windscale-1 (1957)[1,7-8]

The Windscale-1 pile was an air-cooled, graphite-moderated plutonium production reactor. Its graphite had to be annealed periodically to release its stored (Wigner) energy, which resulted from irradiation at low temperatures. During one such annealing, too much heat was applied, resulting in fuel failure, uranium oxidation, and graphite burning. The fire was contained by removing surrounding unaffected fuel, and eventually it was extinguished with water. About 12% of the available iodine was released to the atmosphere over a 2-day period in this accident.

4. HTRE-3[1,9-10]

The HTRE-3 was an air-cooled, experimental aircraft-propulsion reactor. Both the control and protection systems in this reactor used as input the flux measured by a single set of uncompensated ionization chambers. During an automatic reactor startup, the ion chambers erroneously indicated decreasing flux (due to faulty design modifications) whereas actual flux was increasing. This caused the control system to withdraw control rods and at the same time compromised the plant protection system. The resulting unprotected transient lasted for about 80 s, at which time fuel melting and rearrangement and automatic scram (due to the melting of thermocouple leads) shut down the reactor.

5. SRE (1959)[1,11-15]

The SRE was a 20-MWt, sodium-cooled, graphite-moderated experimental power reactor. Its main coolant pumps used an organic coolant (Tetralin) to freeze-seal the pump shafts. This coolant leaked into the primary coolant on three separate occasions over an extended period, causing carbon buildup and plugging of the fuel elements. The seriousness of the problem was not recognized until 10 of 43 core assemblies had suffered severe melting.

6. WTR (1960)[1,16-17]

The WTR was a light-water-cooled and moderated reactor that had initially been licensed for 20 MWt. During tests to upgrade its power to 60 MWt, a fuel element melted. The cause of the accident could not be determined with certainty, but manufacturing defects were judged to be the most likely candidate.

7. SL-1 (1961)[1,18-21]

The SL-1 was a 3-MWt, boiling-water experimental power reactor. During maintenance that required the manual manipulation of control rods, one rod was withdrawn too far, making the reactor super-prompt critical. Since the control rods had been sticking before the incident, the accidental withdrawal may have occurred when a stuck rod suddenly came free. In the central 16 fuel assemblies, 5% of the fuel vaporized and 50% melted. Three operators died as a result of this accident.

8. ETR (1961)[1,22]

The ETR was a 90-MWt, swimming-pool research reactor. A clear plastic sight box used for viewing the core was allowed to sink, unknown to the operators, onto the top of the core. It blocked the (downward) core flow in six fuel assemblies and caused minor melting in them.

9. Fermi-1 (1966)[23-25]

The Fermi-1 was a 200-MWt, sodium-cooled fast reactor. Two zirconium liner plates from a core debris flow guide within the inlet plenum broke loose, unknown to the operators. One of these liners blocked flow to two fuel assemblies and caused melting. Fuel was molten for about 20 min before high radiation alarms prompted a manual shutdown.

10. Lucens (1969)[26-27]

Lucens was a 30-MWt, heavy-water-moderated, carbon-dioxide-cooled experimental power reactor. Water leaked into the carbon dioxide cooling system and corroded the fuel cladding. The resulting corrosion products partially blocked two fuel elements. One of these elements melted and burst, causing a steam explosion when the molten core material sprayed into the heavy water moderator.

11. St. Laurent A-1 (1969)[28]

St. Laurent A-1 is a 460-MWe, carbon-dioxide-cooled, graphite-moderated power reactor. During on-line refueling, a flow restrictor intended for a peripheral reflector was mistakenly installed in a central fuel channel. The resulting flow reduction caused five fuel elements to overheat and melt.

12. Bohunice A-1 (1977)

Bohunice A-1 is a 144-MWe, carbon-dioxide-cooled, heavy-water-moderated power reactor. A major core damage accident apparently occurred during online refueling, but we have no additional information concerning this accident.

13. TMI-2 (1979)[29-35]

The TMI-2 plant is an 880-MWe, pressurized-water reactor. A loss of coolant occurred through a stuck-open relief valve, and operators erroneously throttled the safety injection system, due largely to misleading instrumentation. A steam bubble formed in the top of the reactor vessel, resulting in severe core damage. The accident was brought under control by closing the block valve downstream from the stuck-open relief valve, refilling the system, and establishing decay heat removal. Little or no fuel melting is thought to have occurred, but the core above the water line was apparently reduced to a pile of rubble due to cladding oxidation and fragmentation.

14. St. Laurent A-2 (1980)[36-37]

St. Laurent A-2 is a 530-MWe, carbon dioxide-cooled, graphite-moderated power reactor. A thermocouple shroud came loose and blocked flow, causing damage and melting in six assemblies. The flow blockage was exacerbated when the on-line refueling machine jammed the shroud into the top of an inlet orifice.

DISCUSSION

Although these accidents occurred in a wide range of reactors that had little in common with each other, there are similarities in their root causes and consequences. Two trends apparent in the data base and our observations about the causes and consequences of these accidents are discussed below.

1. Accident Chronology

Of the 14 accidents, 4 involved overpower transients, 6 were caused by coolant blockages, 1 involved a loss of coolant, and 3 were of unknown or other causes. All four overpower transients (NRX, EBR-1, HTRE-3, and SL-1) occurred early in the development of nuclear power, between 1952 and 1961. All but one of the six coolant blockage incidents (SRE, ETR, Fermi-1, Lucens, and St. Laurent A-1 and A-2) occurred between 1959 and 1969. Apparently, the lessons learned from these accidents have helped prevent recurrences. This trend is illustrated schematically in Figure 1.

Time will tell how well the lessons learned from TMI-2 will help prevent recurrences of core damaging loss-of-coolant accidents. Given the historical trend that major accident types tend to repeat themselves several times before they are brought under control, the current emphasis in the nuclear industry on loss-of-coolant accidents seems appropriate. However, it is also important to remember the lessons learned from previous major core damage accidents, lest they be repeated, and to learn from less serious accidents, which may be precursors of new types of core damage accidents.

2. Root Causes

Almost all 14 events resulted from a combination of poor design and operational errors. Construction and manufacturing flaws were minor contributors, and external events (e.g., seismicity, floods) apparently played no role at all.

Most of the accidents were initiated by deficiencies and errors associated with maintenance and testing. These initiators were generally exacerbated by poor design, especially instrumentation deficiencies. Improved instrumentation could have prevented most of these accidents or allowed operators to terminate them without major core damage. Some of the events were aggravated by operator errors and deficiencies as well as poor design. All serious operator errors were errors of commission or inaction in the presence of deteriorating conditions rather than errors of omission in the process of following written procedures. Most of the operator errors are attributable to inadequate instrumentation; however, some of them might have been prevented by improved training.

Most of the accidents could have been prevented had there been 1) better planning and execution of maintenance and testing activities, 2) better instrumentation and data displays, and/or 3) improved operator training to prepare them to properly diagnose accidents in spite of ambiguous indications. Therefore, these areas deserve continuing attention. For operating plants, it may be particularly cost effective to identify potential instrumentation deficiencies and limitations so that they may be offset by improved training and procedures.

Of the nine accidents that occurred in electricity-producing power plants, all but one occurred in the first effective full-power year of operation. (It sometimes required several years to accumulate this much operating experience due to low capacity factors.) That most of the events occurred early in the lifetime of the plant attests to the need for an operating staff that is fully capable from the time of plant startup, when major design deficiencies are most likely to reveal themselves. It also supports the contention that old plants built to less stringent safety requirements may have demonstrated compensating safety advantages by virtue of their successful operating experience.

Several of the accidents occurred while the plants were shut down. And of those that occurred while the plant was operating, many were due to errors associated with a prior shutdown. Clearly, accident prevention is a full-time job that requires attention during both plant shutdown and operation.

3. Accident Consequences

None of the 14 accidents measurably affected public health or safety. This observation, which has already been noted by others for a subset of these accidents,[38-39] is particularly meaningful because many of the plants had no containment building. The only accident to result in significant offsite doses (Windscale) occurred in a plant that was cooled directly with atmospheric air. The modest public consequences of these accidents was largely due to normal design margins and inherent physical and chemical processes that acted to contain accidents and retard the spread of radioactivity. Although it may not be possible to argue that such strongly mitigating processes will exist for all accidents, the evidence supports the contention that even severe accidents are unlikely to cause large public consequences. At a minimum, it suggests exercising caution in imposing new accident mitigation and emergency planning requirements until research provides an adequate understanding of accident phenomenology and radiological source terms. The low observed radiological releases also support the view that the principal consequences of accidents are more likely to be economic losses to the utility, psychological trauma to the public, and loss of industry credibility than to be large public health and safety effects.

4. Learning Curve

Nine of the 14 accidents evaluated occurred in experimental, developmental, or commercial electricity-producing plants. The other 5 occurred in research, production, or propulsion reactors. To investigate the progress of the nuclear industry in learning to prevent accidents, a learning curve was developed from the major core damage accident experience in electricity-producing plants. The following procedure was used: (1) the observed number of accidents was plotted against cumulative experience on log-log paper (Figure 2); (2) the data were fit with a straight line (which is equivalent to assuming that accident frequency is proportional to cumulative experience raised to a power); and (3) the learning curve (Figure 3) was prepared based on this frequency function and the known and projected cumulative worldwide nuclear power plant experience by calendar year. Similar learning curves have been observed in other industries.[40] This learning curve is also similar to one previously developed in Ref. 41.

The learning curve in Figure 3 is thought to be indicative of past progress toward and future prospects for preventing major core damage accidents at nuclear power plants. This curve is an estimate of the average frequency of major core damage accidents for all operating reactors in a given year and is not a good estimator of the reliability of a single plant or small group of plants. Commercial LWR power plants have apparently fared somewhat better than the average of all plants as evidenced by one major core damage accident (TMI-2) in over 1500 reactor years of operation worldwide. However, even this accident frequency needs to be reduced. Because the learning curve in Figure 3 is a (smoothed) average curve, it depicts a continuous learning process, whereas actual learning can probably also produce "step" improvements in reliability. It is to be hoped that the current emphasis in the nuclear industry on accident prevention will result in such a step improvement.

CONCLUSIONS

Cumulative experience with major core damage accidents has been reviewed to identify trends, recurring problems and nonproblems, and generic lessons learned. Recurring problems were identified in the areas of design (especially instrumentation) and operations (especially maintenance and testing). Construction and manufacturing flaws and external events appear not to have been major problems to date. The data also suggest that the public consequences of accidents may be much less than currently perceived. A learning curve was developed that indicates that the worldwide average frequency of major core damage accidents has decreased from about 10^{-2} events per reactor year in 1960 to about 10^{-3} in 1980. Taken together, these observations indicate a need to continue the current emphasis on the design and operational aspects of accident prevention, and suggest that current concerns about accident consequences, emergency planning, construction and manufacturing quality control, and external events may be somewhat overstated.

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TABLE I

REACTORS EXPERIENCING MAJOR CORE DAMAGE ACCIDENTS

REACTOR	PURPOSE	TYPE	SIZE	COUNTRY	DATE
NRX	RESEARCH REACTOR	H_2O COOLED, D_2O MODERATED	30 MWt	CANADA	1952
EBR-I	EXPERIMENTAL POWER REACTOR	LMFBR (POOL-TYPE)	1.4 MWt 0.2 MWe	U.S.A.	1955
WINDSCALE	PLUTONIUM PRODUCTION REACTOR	AIR COOLED, GRAPHITE MODERATED	?	U.K.	1957
HTRE-3	EXPERIMENTAL PROPULSION REACTOR	AIR-COOLED AIRCRAFT ENGINE	32 MWt	U.S.A.	1958
SRE	EXPERIMENTAL POWER REACTOR	SODIUM COOLED, GRAPHITE MODERATED	20 MWt 7.5 MWe	U.S.A.	1959
WTR	RESEARCH REACTOR	H_2O COOLED AND MODERATED	20 MWt	U.S.A.	1960
SL-1	POWER REACTOR	NATURAL CIRCULATION BOILING WATER REACTOR	3 MWt 0.2 MWe	U.S.A.	1961
ETR	RESEARCH REACTOR	SWIMMING POOL	90 MWt	U.S.A.	1961
FERMI-1	DEVELOPMENTAL POWER REACTOR	LMFBR (LOOP-TYPE)	200 MWt 22 MWe	U.S.A.	1966
LUCENS	EXPERIMENTAL POWER REACTOR	CO_2 COOLED, DO_2 MODERATED	30 MWt 6 MWe	SWITZERLAND	1969
ST. LAURENT A-1	POWER REACTOR	CO_2 COOLED, GRAPHITE MODERATED	1500 MWt 460 MWe	FRANCE	1969
BOHUNICE A-1	DEVELOPMENTAL POWER REACTOR	CO_2 COOLED, DO_2 MODERATED	560 MWt 144 MWe	CZECHOSLOVAKIA	1977
TMI-2	POWER REACTOR	PWR	2772 MWt 906 MWe	U.S.A.	1979
ST. LAURENT A-2	POWER REACTOR	CO_2 COOLED, GRAPHITE MODERATED	1700 MWt 530 MWe	FRANCE	1980

TABLE II

SUMMARY OF MAJOR CORE DAMAGE ACCIDENTS

REACTOR	REACTOR CONDITION	NATURE OF ACCIDENT
NRX	SHUTDOWN – PLANNED TEST WITH REDUCED COOLING IN SEVERAL SUBASSEMBLIES	INADVERTENT CONTROL ROD WITHDRAWAL WITH FAILURE TO SCRAM
EBR-I	POWER OPERATION – PLANNED TEST INVOLVING FAST POWER RISE	PLANNED MANUAL SCRAM TO TERMINATE TEST WAS PERFORMED TOO LATE
WINDSCALE	SHUTDOWN – THERMALLY ANNEAL-ING GRAPHITE	EXCESSIVE HEATING LED TO GRAPHITE AND FUEL BURNING
HTRE-3	POWER OPERATION – TESTING NEW CONTROL SYSTEM	CONTROL SYSTEM FAULT LED TO EXCESSIVE ROD WITHDRAWAL
SRE	POWER OPERATION – NUMEROUS POWER CHANGES	BUILDUP OF FOREIGN MATERIAL ON FUEL ELEMENTS LED TO CORE UNDERCOOLING
WTR	POWER OPERATION – TESTING AT HIGHER THAN RATED POWER	FLAWED FUEL ELEMENT OVERHEATED AND MELTED
SL-1	SHUTDOWN – MAINTENANCE INVOL-VING CONTROL ROD DRIVE MECHANISMS	MANUAL REMOVAL OF CONTROL ROD LED TO PROMPT SUPERCRITICALITY
ETR	SHUTDOWN/POWER OPERATION – BLOCKAGE OCCURRED DURING SHUTDOWN, AND MELTING OCCURRED AT POWER	CLEAR PLASTIC VIEWING BOX BLOCKED FUEL CHANNEL INLET
FERMI-1	POWER OPERATION – CHANGING POWER LEVEL	CORE INLET ORIFICES BLOCKED BY LOOSE FLOW GUIDES FROM INLET PLENUM
LUCENS	SHUTDOWN/POWER OPERATION – CORROSION OCCURRED DURING SHUTDOWN, WHICH LED TO MELTING AT POWER	CORROSION PRODUCTS BLOCKED FUEL ELEMENT FLOW CHANNELS
ST. LAURENT A-1	POWER OPERATION – ON-LINE REFUELING	ERRONEOUS INSTALLATION OF PLUG IN FUEL ELEMENT INLET BY ON-LINE REFUELING MACHINE
BOHUNICE A-1	POWER OPERATION – ON-LINE REFUELING	?
TMI-2	POWER OPERATION – CONSTANT HIGH-POWER OPERATION	LOSS OF COOLANT THROUGH STUCK-OPEN PRESSURIZER RELIEF VALVE AND ISOLATION OF SAFETY INJECTION LED TO CORE UNCOVERING
ST. LAURENT A-2	POWER OPERATION – ON-LINE REFUELING	LOOSE THERMOCOUPLE SHROUD JAMMED INTO FUEL INLET ORIFICE BY ON-LINE REFUELING MACHINE

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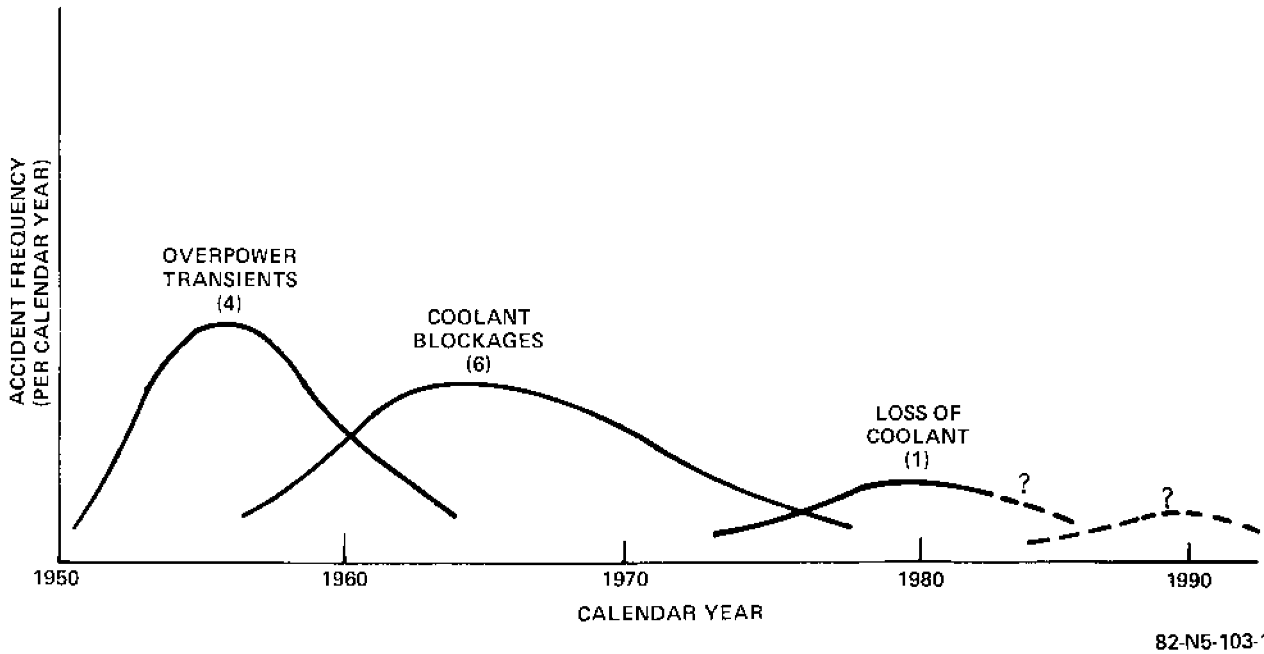

Figure 1. Accident Chronology Suggests that Learning Can Help Prevent Recurrence of Similar Events

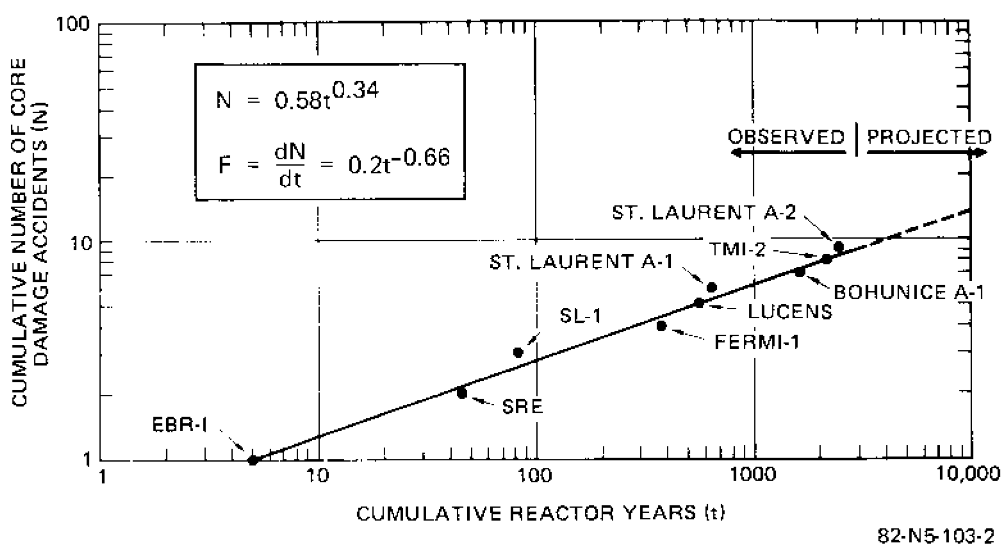

Figure 2. Major Core Damage Accidents as a Function of Cumulative Operating Experience

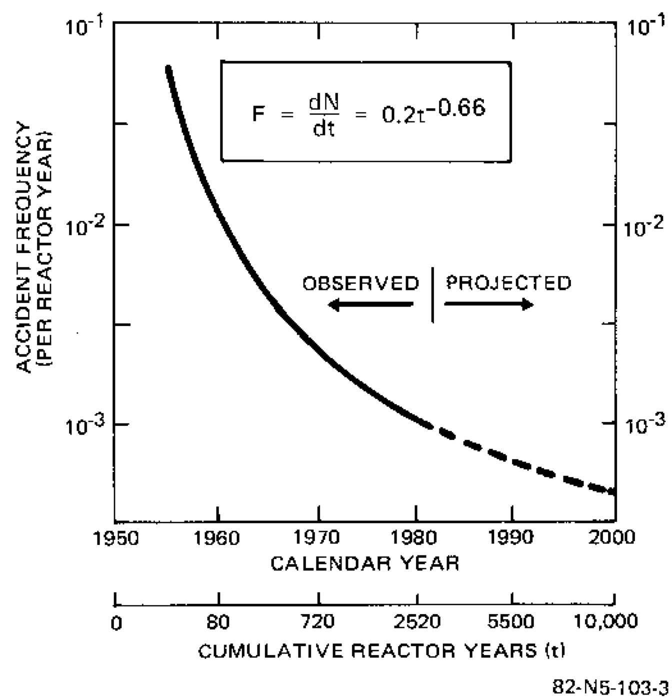

Figure 3. Accident Prevention Learning Curve