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FINAL REPLY:

Denwood F. Ross, Jr.
Rockville, Maryland

TO:

Chairman Meserve

FOR SIGNATURE OF : ** GRN ** CRC NO: 00-0280

DESC:

Need for NRC-Sponsored Research in the Thermal-Hydraulic Area

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EDO --G20000215

7441 Bee Bee Drive
Rockville, Maryland 20855
April 7, 2000

The Honorable Richard Meserve
Chairman, US Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2738

Dear Chairman Meserve:

I am writing to you to express my views on the need for NRC-sponsored research in the thermal-hydraulic area. These views are based in part on my 31 years of service with the NRC, in various senior management capacities with NRR, RES, and AEOD, and in part on my consultant work with the NEA and IAEA since retirement in 1998.

I think that a systematic review of operating experience, worldwide, reveals that root causes and corrective actions can be better assessed through a relatively modest investment in confirmatory or exploratory research. In the past, such a nexus between lessons learned from operating experience, and research, has not been done systematically.

To illustrate this point, I delivered an invited paper, entitled Research Needs Based on Operational Experiences, at the recent ICONE-8 conference. A copy of this paper is enclosed. Some of the more important aspects of the paper are discussed below.

There are thermal-hydraulic challenges, as evidenced by operational occurrences, of a less dramatic nature than the traditional loss-of –coolant-accident (which was the object of very extensive and expensive research programs around the world). These thermal-hydraulic challenges, of greater frequency and some cases greater risk, are worthy of additional research. A systematic review of operational incidents in the international database revealed the following:

- Thermal fatigue continues to cause failures in piping systems that are interconnected to the main piping of reactor cooling systems;
- There have been a number of events around the world where pressurized water reactors have experienced a loss of heat removal while the reactor vessel is operating at reduced water level (so-called mid-loop operation). In some cases vortexing at the residual heat removal pump suction has contributed to the loss of forced circulation; the evolution of dissolved gases from the primary coolant has played a role in confounding both the instruments and the operational staff;
- In several instances there have been reports of power oscillations in boiling water reactors, contrary to design criteria;

- In at least two events there have been loss of primary coolant inventory through a tortuous interconnecting pipeline, from a pressurized water reactor at intermediate pressure and temperature condition;
- Water hammer events have continued to be reported.

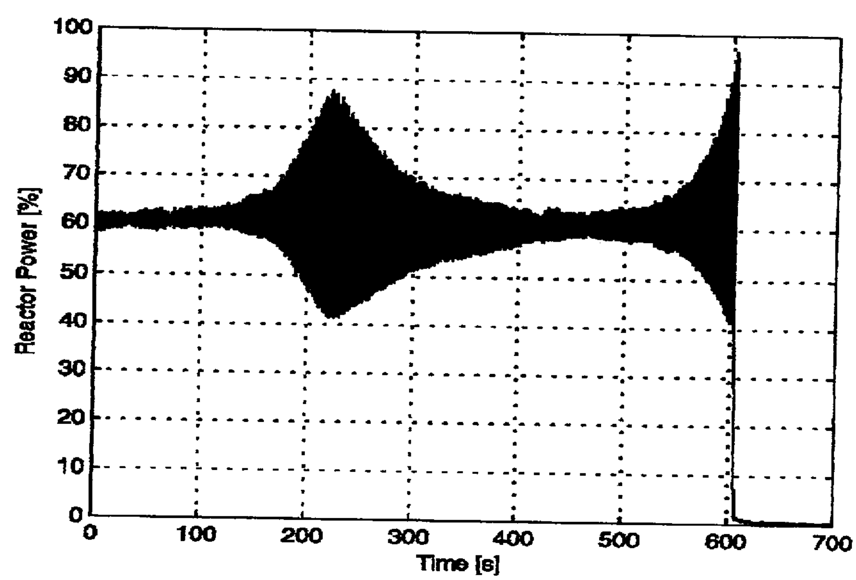
The recurrent nature of these events, some of risk significance, is a matter of current concern to the international reactor safety organizations, and will be discussed in depth at forthcoming meetings.

It is asserted, on the basis of this review of operational occurrences, that more experimental and analytical thermal-hydraulic research is indicated in these areas. In some cases the research would consist of adapting system analysis codes to the particular situation. On other cases, some experimental work would be necessary. More and better information can assist both the operator and regulator in reducing the incidence or severity of these accident precursors, and in recovering from them in an orderly manner. Expanding the core capability in these areas would also advance the expertise of the NRC technical staff.

Some technical details might suffice to illustrate some of these points.[1]

BWR Power Oscillations

There have been a least ten events in the period 1982-99, worldwide, where a BWR has had power oscillations, of varied sorts. The NRC General Design Criterion #12 states that oscillations should be precluded by design or else detected and suppressed. Most other countries with BWRs have similar requirements. An illustration of rather extreme oscillations is shown to the right, where power is oscillating over a ten-minute interval.



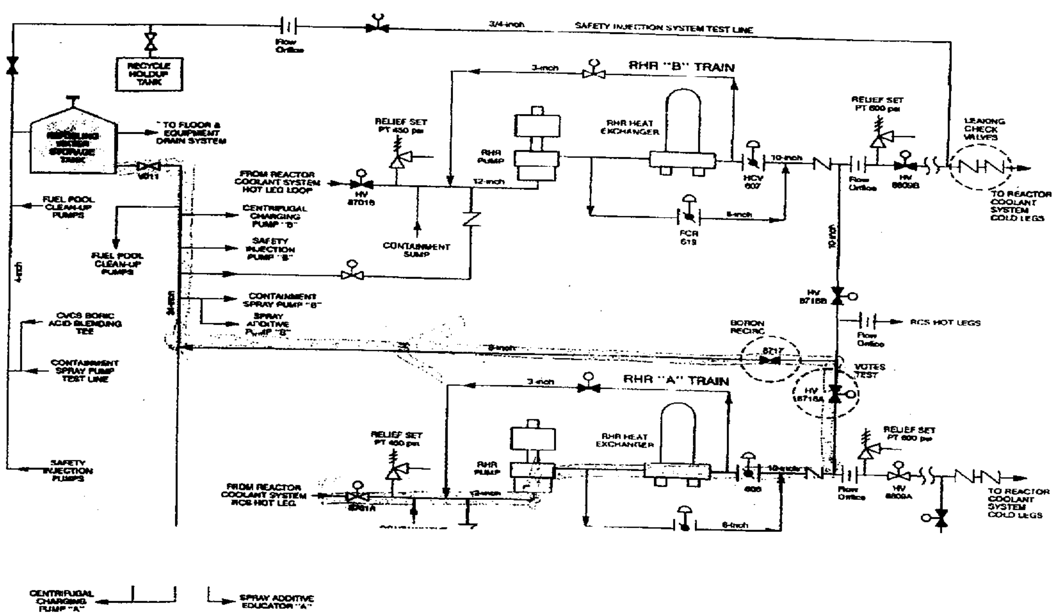
Generally, experts have concluded that while oscillations such as these are less than desirable, they may not be particularly risk-significant. Nonetheless, it seems that further knowledge and understanding that might accrue from research would assist the regulator in being more proactive in ensuring an adequate regulatory remedy.

[1] More detail, and citations, are in the enclosed paper.

Loss of Coolant During Transition to Shutdown

There have been two recent events in the US where a PWR has had an inadvertent loss of inventory from the primary system while the reactor was in a mode transition. The first such event, for which there is more public information, occurred at the Wolf Creek plant several years ago. The reactor was at intermediate temperature and pressure, and the ECCS was disabled, per procedure. Due to errors in mispositioning of two valves, there was at blowdown of the primary cooling system to the borated water storage tank, at a rate of about 10,000 gpm. No automatic systems were in service to cope with this, again per procedure. Fortunately, operator action resulted in valve closure before the residual heat removal systems became damaged[2], or before core uncovery. The blowdown path was somewhat tortuous for this event, as shown below (blowdown path highlighted in green).

Of significance to the regulator is that this event is not within the design basis for the plant, and that available tools for LOCA analysis are not readily



adaptable for sensitivity studies, such as time to uncover the core. The blowdown path is through long lines, with area changes, through pumps, valves, and heat exchangers. Modeling improvements to codes such as RELAP would enable the regulator to better appraise the outcome of other, more extreme events. This was not a singular occurrence, as a related event occurred at Waterford just a few months ago.

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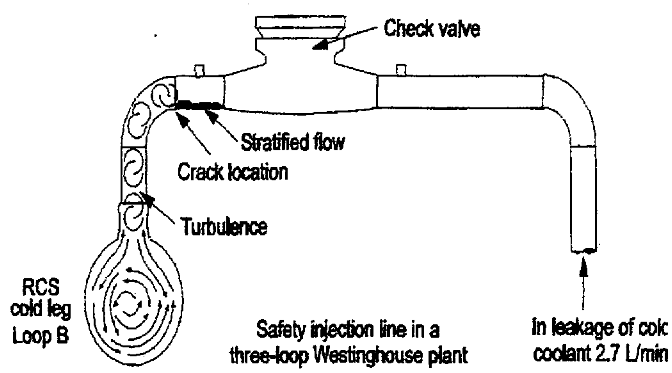
[2] Pumps could be damaged by cavitation, during emergency injection, if the suction supply becomes saturated.

Perhaps if the staff had better analysis tools they could prescribe remedies to cope with such recurrent events. Of risk significance, in addition to the need for prompt and (no doubt) *ad hoc* operator actions, is the fact that there is a direct path from the reactor coolant system to the outside of the containment.

Thermal fatigue failures in primary system piping

Thermal fatigue has caused piping degradation, including through-wall cracking, in numerous plants over the past 30 years. In some cases the failure locations have been at risk-sensitive locations, such as piping connected directly to the primary cooling system (without an intervening isolation valve). There are several contributing factors, among them the cyclic changes in thermal environment due to turbulence and valve leakage.

Some research has been done on the thermal-hydraulic driving forces. There was an EPRI program a few years ago. Finland has a test loop in service at this time. France and Japan have both conducted research.



An illustration of the zone of interest is shown on the diagram to the right. Turbulence is induced by flow in the main pipe, and leakage from the injection line can further aggravate the thermal challenge. It is less than clear that enough work exists to specify closure on this issue, from the thermal hydraulic viewpoint.

These three examples illustrate the main conclusions of the enclosed paper, which are:

- Undesirable recurring events have been reported which have causative factors in the thermal-hydraulic arena;
- The events are sometime of moderate to high risk significance;
- In the main, NRC has not done much research in these areas;
- Closure in terms of preventing recurrence is not assured;
- The phenomena are amenable to research;
- The work would not be inordinately expensive;
- The work, if done, would lend itself to international cooperation;
- The work would establish and help to maintain additional core capability in these areas.

It is not envisioned that these programs would be inordinately costly, especially in context with the amount spent on the traditional LOCA. Further, there is some sentiment within the Nuclear Energy Agency for further international cooperation in research programs, especially since the operational events discussed herein are somewhat common amongst the member states.

I would be pleased to provide additional information if you choose. My enclosed business card states where I can be reached.

Sincerely,

Denwood F. Ross, Jr.

Denwood F. Ross, Jr.

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RESEARCH NEEDS BASED ON REACTOR OPERATIONAL EXPERIENCES

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KEYWORDS: event-based research, thermal fatigue, power oscillations, vortexing, transient behavior

ABSTRACT

Power reactor oriented thermal-hydraulic research, from the safety viewpoint, has been dominated over the past thirty years by accident considerations. The postulated loss of coolant accident, and severe accident concerns, have been the focus of world-wide research programs whose cost is undoubtedly measured in excess of a billion dollars. Various actual accidents, notably Three Mile Island and Chernobyl, have provided impetus to this desire to know more.

There are, however, additional thermal-hydraulic challenges, as evidenced by operational occurrences, of a less dramatic nature that should also be the subject of research. A systematic review of incidents revealed the following:

- Thermal fatigue continues causes failures in piping systems interconnected to the main piping of reactor cooling systems
- There have been a number of events around the world where pressurized water reactors have experienced a loss of decay heat removal while the reactor vessel is operating at reduced water level. In some cases vortexing at the inlet of the line connecting to the decay heat removal pump suction has contributed to the loss of forced circulation, and the evolution of dissolved gases has played a role;
- In several instances there have been reports of power oscillations in boiling water reactors, contrary to generally accepted design criteria.
- In at least one event there has been a loss of coolant event through a tortuous interconnecting pipeline from a pressurized water reactor at intermediate pressure and temperature condition. Restoration of cooling was somewhat fortuitous.
- Water hammer events, some with damage to systems and components, have continued to be reported.

It is asserted, on the basis of this review of operational occurrences, that more experimental and analytical thermal-hydraulic research is indicated in these areas. More and better information can assist the operator and regulator in reducing the incidence of these accident precursors, and in recovering from them in an orderly manner, as well is in allowing consideration of improvements in safety and efficiency.

INTRODUCTION

Thermal –hydraulic research in support of power reactor safety has tended in the past to focus on integral systems testing and analysis. In the decade of 1960-1970, as the modern concepts of pressurized and boiling water reactors emerged, there were scaled reactor system simulators and integral system code developments. The regulatory influence tended to support the need for such efforts. System transients of moderate frequency, such as loss of flow or separation of the turbine-generator from the electrical grid, were modeled. Events of lesser frequency or hypothetical events such as the large pipe break of the primary coolant system, were also modeled and analyzed. This focus on system behavior at the integral level was expensive and dominated research budgets.

It is now generally concluded that system analysis tools are reasonably accurate for prediction of integrated reactor performance, and that no major additional research programs are needed. Continuing efforts are mainly for adaptation of system codes such as TRAC and RELAP to newer and more efficient computers.

The question then arises as to whether there remain any significant separate effects topics in the thermal-hydraulic arena which are in the need for research. To this end, this paper provides an assessment of research needs based solely on a review of reactor operational experiences. Based on this review it is seen that there are still some needs in the following areas:

- ♣ Water hammer
- ♣ Thermal fatigue of piping
- ♣ System transients during mode transitions to shutdown conditions, and during shutdown
- ♣ Overcooling events
- ♣ Vortex formation
- ♣ Power oscillations in boiling water reactors.

These are discussed in more detail below.

Water Hammer

Water hammer events have occurred frequently at power reactors. A search by the USNRC (NRC, 1984) showed approximately 150 reports of water hammer over the time span 1969-1980, or about 14 events per year. A subsequent review (NRC, 1991) revealed 12 water events between 1986 and 1990.

Water hammer events have been categorized as condensation-induced and as flow of water slugs that interact with pipe elbows or closed valves. Significant damage to piping systems has been experienced.

In NRC (1991) there were a number of illustrations of the damage which can occur during water hammer conditions.

Example 1: During testing of a portion of the emergency core cooling system (ECCS), a slug of water was accelerated in the pipe which decelerated as it impacted a valve which was opening. Some cracking of the gate on the valve took place.

Example 2: A 2.5-cm line in the high-pressure auxiliary feedwater system of a pressurized water reactor was severed completely due to pressure pulsing when the pump discharge valve was in a throttled position.

Example 3: Slug flow in a 2.5-cm line occurred and resulted in a rupture in a nozzle-to-pipe weld. After repairs, the procedure was repeated, with an identical failure result.

Example 4: A recirculation system used for removal of reactor decay heat was started without proper filling and venting. The resultant slug flow caused a water hammer event.

Example 5: A 45-cm steam drain line ruptured due to condensation-induced water hammer. Water hammer events had occurred in this system before, but not to this destructive extent. Seven workers in the vicinity were burned.

Example 6: A decay heat removal line suffered damage at its pipe supports due to water hammer. Dissolved nitrogen gas had come out of solution and caused large pockets of gas. When the system was started for pump testing, the slug flow resulted in water hammer.

Example 7: A severe water hammer occurred in the containment spray line during testing. There was an interruption in flow due to a deliberate interruption in electrical power. The system drained partly; then, when power was restored, the flow started with a void in the flow line. Several pipe hangars were damaged.

Example 8: During a test of the low-pressure safety injection system, a large pressure transient was observed. Dissolved nitrogen gas had come out of solution from a higher-pressure tank (the accumulator).

Other events were also mentioned. According to NRC 1997, water hammers are continuing to occur in both high and low temperature systems and high or low-pressure systems as a result of a variety of causes.

The causes for the water hammers were listed as:

- Steam voids were formed, sometimes as a result of leakage of hotter water through valves;
- Systems were improperly filled and vented;
- Valves were stroked too fast;
- Liquid was allowed to accumulate at low points in the system;
- Dissolved nitrogen gas came out of solution and separated the flow into slugs;
- System depressurization caused steam voids.

The observation was made that there were inadequate root cause evaluations and corrective actions. The severity ranged up to catastrophic failures, with occasional personnel injury.

Inasmuch as significant water hammers have recurred with moderate frequency, and with significant damage to systems, and that there seems to be considerable uncertainty as to preventive methods, it follows that further study on corrective actions is needed.

Thermal Fatigue of Piping

Cyclic change of fluid temperature is a well-known cause of fatigue. Piping failures from thermal fatigue in nuclear power plants have been numerous, especially in piping and nozzles. (Etherington, 1989). Failures are of two types:

- High-cycle, low-stress, and
- Low-cycle, high-stress.

The difference lies in the type of thermal variations in the fluid streams. Some basic definitions of thermal stratification and thermal cycling are found in Strauch (1990). Thermal cycling is a periodic fluctuation of fluid temperature in a pipe, and may be aggravated by nearby valve leakage. Strauch noted (in 1990) that turbulent eddy currents in a branch pipe, caused by loop flow in the main pipe, and mixed with the leakage flow, may account for some thermal cycling and fatigue, but that test data did not yet support that theory.

In the past thirty years there have been numerous examples of piping damage from thermal fatigue in nuclear power plants. Some examples, not intended to represent the complete database, follow below.

Example 1. In August 1999 there was a failure in a regenerative heat exchanger due to thermal fatigue at a Japanese pressurized water reactor (Nuclear News, 1999). There were thermal stresses due to a mixture of hot and cold fluids and, over a 12-year history, produced on the order of 10^6 thermal cycles, at a frequency of 10-20 minutes. The materials of construction were SUS316 stainless steel. About 50,000 liters of primary coolant spilled out the break.

Example 2. In 1997 there was a through-wall crack in a section of piping of a United States Pressurized Water Reactor (NRC, 1997). Preliminary analysis indicated that the failure should be attributed to a combination of thermal cycling and flow-induced vibration. The location of this leak, in the high-pressure injection system, was such that the leak location could not be isolated from the primary system. Maximum leak rate was about 45 liters/minute. Similar cracks had developed at sister plants.

Example 3. In 1987 there was a report of a through-wall crack and small leak, about 3 liters/minute, at a US Pressurized Water Reactor (NRC, 1988). This leak location was in what is called an unisolable location; that is, there is no isolation provision between the leak and the primary coolant system. The crack resulted from high-cycle thermal fatigue. At times the thermal variation was as large as 40° C, with a period of thermal oscillation between 2 and 10 minutes. A contributor to this event is the combination of a leaky valve and colder fluid upstream of the valve that leaked into the vicinity of the crack.

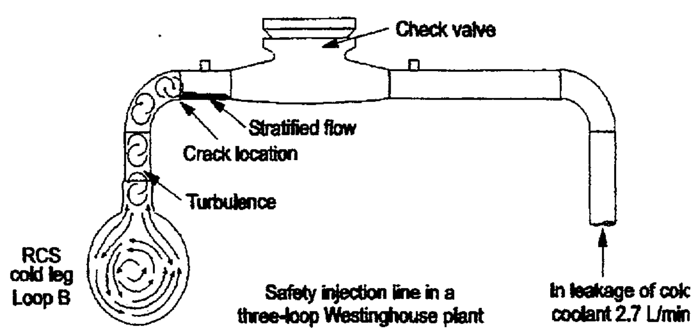

Figure 1 Connection of cold water source to main pipe.

Example 4. In 1988, shortly after the event discussed in example 3, there was a similar event at a Belgium reactor (designed by Westinghouse). The leak rate appeared rather suddenly, and had a peak value of about 20 liters/minute (NRC, 1988A). The location was also unisolable. In this example, as well as the previous example 3, difficulties in flaw detection by ultrasonic testing were reported.

Lubin (1994) provided an analysis of thermal stratification in the shutdown cooling line attached to the primary coolant system. Pipe wall temperature measurements were taken to indicate the severity of turbulent penetration as it affects stratification. Large temperature variations may occur during a planned shutdown sequence.

Inasmuch as thermal fatigue failures are continuing to occur, more work may be needed. In the USA, the industry research group, EPRI, started a research program intended to shed further light on the sensitive locations and situations that might promote thermal fatigue. This effort is briefly mentioned in Roarty (1994). Experimental work is in progress in Finland (1999).

System transients during mode transitions to shutdown conditions, and during shutdown

Reactor transient analysis generally consists of using a system-level model as applied to a variety of specified transients, ranging from a simple reactor trip to a design basis loss of coolant accident. In some cases, operating experience has revealed that, due to emergence of a transient not hitherto imagined or due to incompleteness in the modeling, system performance may differ from expectations. Some examples follow below.

Example 1. Generally the loss-of –coolant accident is modeled from full power and, according to the requirements of the NRC regulations (10CFR50.46 and Appendix K to Part 50). A spectrum of break sizes is considered.

In one case in a PWR in the United States there was a substantial (*circa* 40,000 liters) loss of coolant, in about one minute, while the unit was proceeding to shutdown conditions. Initial process conditions in the primary system were 150 C° and 23 bars pressure. This loss of coolant differed from the traditional analysis in several ways:

- The reactor had been shutdown for a number of hours, so the heat production was from decay heat;
- The initial process conditions were substantially lower in temperature and pressure than that of the traditional full-power analysis; and,

- The blowdown path was quite circuitous, through a number of valves and long pipes with elbows and tees, rather than the usual simple pipe break geometry.
- The recovery from a traditional LOCA is by automatic means, such as starting a pump or using, in a passive sense, the stored water in an accumulator. That was not the case in this event, in that accumulators were valved out of service, and the ECCS function was bypassed.
- The recovery source of water, even if used in the manual sense, was also in the blowdown path. Thus, if the blowdown had persisted much longer, the suction side of the ECCS pumps would have been exposed to water at saturation temperature, making it either difficult or impossible to operate pumps (until the hot water had been vented to drain and replaced with subcooled fluid).

This sort of transient was not in the design basis and had not been modeled as part of the transient analysis.

Example 2. A complex transient at shutdown of a PWR occurred in 1990 (NRC, 1990). The PWR had been shutdown for more than 20 days, and had a decay heat production of about 2.5 Mw. The reactor vessel level had been lowered to what is known as the mid-loop point, for some maintenance work on the steam generators. There was a loss of offsite power due to an incident in the switchyard. Due to a scheduled outage of one diesel generator and the failure to start of another diesel generator, there was no onsite AC power, either. With this total loss of AC, there was no heat removal from the reactor. This lasted for about 36 minutes, until one diesel generator was started and loaded. During this time the reactor vessel water temperature increased about 25° C.

Subsequent analyses of this event for alternate heat removal processes were done. One alternative that was considered was the gravity draining of a large tank (known as the refueling water storage tank, or RWST) into the vessel and exiting at the top of the pressurizer (see Figure 1). This was considered in NRC, 1990. There were a number of uncertainties in the analysis including the gravity driving head, the efficacy of the process, the back-pressure that would be induced if the exiting flow was two-phase, instead of single-phase liquid, as well as the potential for the event occurring earlier in the shutdown, when the decay power would have been much higher. Gravity head and friction losses played a dominant role in controlling the event, in contrast to the usual system codes that might be driven by such factors as critical flow, pump behavior, and rapid depressurization.

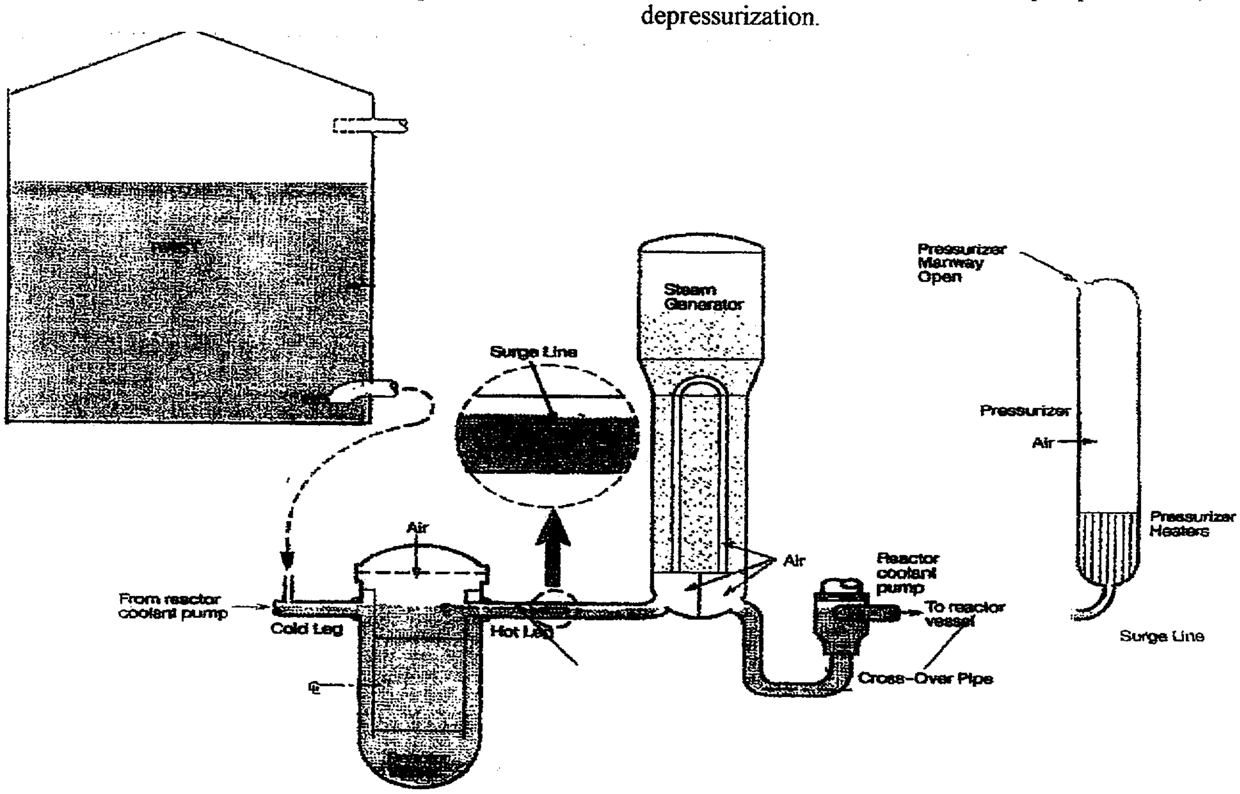

Figure 2. Relationship of Refueling Water Storage Tank to Reactor Vessel and Steam Generator (During Midloop Operation.

Generally, reactor system performance at low pressure is not done. Further, on events such as these (loss of decay heat removal at low power and pressure have occurred numerous times, around the world), there is an alternate heat removal path, perhaps, through the steam generators by what is known as reflux cooling. This too has not been subject to rigorous modeling and analysis. (There is some analytical information in Andreychek, 1988).

Some common characteristics of these complex events are that:

- The events are generally not considered in the design basis;
- The events are of modeling complexity;
- System codes used for traditional design basis accidents, such as the LOCA, might not be applicable as written;
- Experimental verification may be lacking; and
- The events are risk-significant.

Overcooling events

Overcooling, in this context, refers to a concern in pressurized water reactors wherein there might be an event that causes a pressure reduction that activates the high pressure injection system, and then there is a return to normal pressure with the primary system in a temperature-reduction mode. Such events have occurred. The generic term for this event is Pressurized Thermal Shock (PTS). The safety concern is that the reactor vessel might respond in a brittle fashion, where at the extreme there could be a rapidly-propagating flaw in the vessel, due to high tensile stresses near a flaw.

The most aggravated instance of overcooling occurred at the now-shutdown PWR, Rancho Seco (add reference) in 1978. A malfunction in the feedwater control system caused a mismatch between steam and feed flow, and resulted in an overcooling. An extreme combination of high pressure, low vessel temperature would be of most concern when the vessel had been operated for many years, when the mid-core reactor vessel had been embrittled somewhat by neutron radiation. (The Rancho Seco event was early in life.) Regulatory bodies generally responded with policies and rules governing actions to be taken in response to the PTS scenario.

In the intervening 20 years there have been additional overcooling events, although not of the Rancho Seco severity. In 1996 there was an event at the US plant Catawba-2 (NRC, 1995). A loss of offsite power and safety injection from the high-pressure pumps occurred. The main circulation pumps for the primary coolant system tripped when offsite power was lost, and core cooling was by natural convection. High steam loads and high rate of auxiliary feedwater to the steam generators produced an excess cooldown which in turn lowered reactor coolant system pressure. This, by design, isolated the letdown from the reactor and increased makeup. The reactor system pressure went up, and the system became water-solid. For about six hours, the system was in a high-pressure, water solid condition, with the potential for cold water conditions in the reactor vessel downcomer.

There have not been too many of these operating events. Overcooling, or PTS, become more important risk contributors as reactor vessel life is prolonged by license condition. PTS scenarios could be made more credible by a revisiting to the regulatory methods promulgated a number of years ago, with a view of more accurately accounting for operating experience, and other information relative to the thermal-hydraulic challenge.

Vortex formation

There have been a number of events where vortices have formed at the inlet to pumping systems, with loss of flow and, in some instances, pump damage in vital systems. A contributory factor in many of these cases is the evolution of dissolved gases, which confounds the liquid level measurement system. In general the solution and dissolution of noncondensable gases is not modeled in system codes, nor is it systematically accounted for in experiments. Several examples follow below.

- At a PWR in the USA there was an event at shutdown (San Onofre, 1987) wherein the residual heat removal system was lost for about an hour due to vortexing and cavitation of the pumps. The system was in a lowered water level condition for maintenance, and the system was open to the containment atmosphere. Due to a miscalibration of the level sensors, the water level was lower than desired. At the system flow conditions, and at the lower than desired level, there was air ingestion and vortex formation in the suction line. Due to loss of heat removal, the decay heat caused a rise in temperature, almost to saturation, which further exacerbated the problem. The plant had no formal data on the potential for vortexing at the lowered levels in the system.
- NRC (1997B) contained a description of several events involving reactor coolant system inventory control during reactor shutdown. In two cases there was an evolution of dissolved gases from the reactor coolant system water due to depressurization or due to heating of cold liquid that was saturated with nitrogen. The gas evolution was substantial; in one case it displaced more than 25,000 l of liquid. Such behavior causes erratic and misleading level indications, and has the potential of contributing to vortex formation.
- Another report (NRC, 1993) discussed inaccuracies in level instrumentation due to the release of dissolved gases during plant cooldown and depressurization. In this instance the concern was that protection systems for pumps, intended to isolate the RHR system at low reactor water level, might not have worked due to the presence of noncondensible gases in the mearurement systems. This concern was also noted in NRC, 1992.

- Additional observations have been made (NRC, 1990) about vortex formation and loss of decay heat removal. Some common deficiencies in this regard were that:
  - There was lack of knowledge about the correlation between water level and pump speed at the onset of vortexing;
  - There were deficiencies in the operator procedures with respect to vortexing and how the pump would indicate cavitation;
  - There was a general lack of appreciation about the evolution of dissolved gases and how the level measurement systems would perform during such a transient condition;
  - In at least one instance there was a model test that was used to define the range of permissible flow rates that would suffice to avoid vortexing; however, the test was deficient in that it did not simulate the range of ambient pressures that might exist in the reactor.

Bird, Stuart, and Lightfoot (1960) have commented on the difficulties of modeling vortex formation.

Recommendations for definitive tests for the vortex potential have been made in the past (NRC, 1988C).

Power Oscillations in boiling water reactors

Power oscillations have been observed at several BWRs around the world in the last 15 years. A number of communications from the regulatory bodies and the regulated industry have been issued to correct the problem. In one report (NRC, 1994) there was a summary of two such events, one in 1988 and another in 1992. (Power oscillations have occurred in other countries, as well.) Generally, oscillatory behavior should be excluded by design, or else such behavior should be readily detected and suppressed.

The reactor operating boundary that was intended to serve as a barrier to oscillations has not been well defined in the past, with the result that during transients (mostly flow transients where one or both recirculation pumps trip for some reason) unanticipated oscillations occur. As a result, some plants are considering the installation of additional protective circuits to automatically shut down the reactor.

The variables important to oscillatory tendency include power distribution (both radial and axial); inlet feedwater temperature; use of mixed fuel types; core monitoring devices, and power and flow rate conditions. The various "solutions" that have issued after each oscillatory event have not proven fully effective in all cases.

METHODS DEVELOPMENT AND EXPERIMENTAL WORK

For the most part, methods development and experimental work in the thermal-hydraulic area have focused on the less-likely accidental conditions such as the loss-of-coolant accident, and severe accidents. System codes have been developed for this purpose which model macroscopic behavior. The more likely events discussed herein, including water hammer, thermal fatigue, off-normal transients, overcooling, vortex formation, and power oscillations have not been subject to systematic modeling and experimentation by the regulatory bodies. Further, the regulated industry has not executed research programs in all of these areas, either.

Lubin (1995) observed that data on wall temperature measurements in operating plants showed the presence of thermal stratification in three different pipe lines, and suggested three different mechanisms. The authors were of the opinion that these operating conditions reflected a "new class of operating transients that need to be considered as part of the plant's design basis". As yet, it is not clear that the regulator has taken this suggested action.

One utility (Bain, 1993) has a long-term program related to thermal stratification and fatigue. This report mentioned, in addition to thermal cycling, thermal striping. It was thought that achieving leak-tightness in the valves would have an important role in leak-stoppage.

Roarty (1994) mentioned a recently-completed EPRI program that evaluated thermal-hydraulic mechanisms associated with thermal stratification, thermal cycling, and striping. The program was to determine the height of a stratification interface, the heat transfer of a stratified pipe and turbulence from the header pipe, thermal cycling and heat transfer coefficients for a thermal stratification loading. The results of the EPRI program were intended to give utilities the models, correlations, and guidelines to assist industry in evaluating mechanisms.

One drawback in achieving closure could be the lack of confirmatory research by regulatory bodies. To date the regulatory presence has generally been guidance of an informal nature and optional on the part of the utilities. Some references (for example, Lubin, 1995) have suggested that thermal fatigue could be part of a new design basis. If so, some more precise regulatory guidance might be needed. But, that might take some independent confirmatory research to assure that the regulatory prescription can be written correctly.

Current assessments of reactor risk have shown that some of these phenomena can be risk-important. For example, loss of residual heat removal during shutdown can be a dominant contributor to risk.

REGULATORY POLICIES

There are a variety of regulatory policies, such as binding rules and optional guides, which are used to regulate reactor safety. In some of the rules and guides there are proscriptive factors that would constitute required and acceptable features for safety analysis. However, in many instances the microbehavior (such as quantity of dissolved gases and conditions under which such gases might be released) is not generally modeled.

In some instances there is but general guidance, such as avoid oscillations or else detect and suppress them, with little in the way of prescriptive guidance.

In yet other instances, notably in the modeling of locations likely to be the zone of thermal fatigue, there is no guidance at all.

One benefit of additional research in these areas is that the regulatory body and the utility can each base corrective action on something other than empiricism or anecdotal experience.

CONCLUSIONS

Reactor operating experience review has shown that a number of events have occurred with overtones of thermal-hydraulic contributions. For the most part there has not been sufficient research to contribute to the long-term solution of these problems. The problems discussed herein are long standing, and are well known to the industry and the regulator. In many cases the events are of moderate to high-risk significance.

The necessary research should not be prohibitive or unnecessarily expensive, especially in comparison with other projects in the thermal-hydraulic area of less significance to risk. Since the problems are global in nature, there is an excellent opportunity to cooperate internationally for consensus solutions.

Some specific conclusions, by research topic, are:

- **Water Hammer** Water hammer is scarcely an unknown phenomenon. There has been ample research in the past, and the mechanisms are well known. Yet, damage is still occurring at nuclear power plants from this phenomenon. Damage has ranged from moderate to severe. In one instance the damage led to the premature plant decommissioning and in another instance led to a large loss of coolant from the primary system. Most events cause damage to pipe supports and valves. One potential area of research would be in comprehensive survey of all water hammer events in order to develop characteristics of systems with a significant potential for the onset of water hammer. For example, pipe lines with a large number of changes in direction and not always in a liquid-filled condition (such as containment spray lines). Possibly a global look at the events would reveal some common factors that could be used as a target for further research recommendations. Attention to design can assist in preventing or mitigating the effects of water hammer.
- **Thermal Fatigue** As noted herein, some work has been done on predictive methods for thermal fatigue in reactor systems (By, *inter alia*, EPRI and Finland). Regulatory work on the thermal-hydraulic aspects of thermal fatigue, for confirmatory purposes, seems to be lacking. Absent reliable information on the causes and cures for the various sorts of thermal challenges, it seems unlikely that the regulator will be able to participate meaningfully in the cure.
- **System transients** Research on reactor behavior to off-normal transients in the low-power and shutdown mode is noticeably absent. Computer codes should be able to model more aspects, including the effects of dissolved gases (as they become liberated during temperature and pressure changes) on control and instrumentation, specifically level instruments. Flow paths of an unusual nature, which become modes of egress for primary fluid should be modeled in plant system LOCA codes. Experimental verification will be needed.
- **Overcooling** Overcooling as a precursor to a Pressurized Thermal Shock condition may be a concern in the future, if it is desired to more accurately model the temperature and pressure conditions that might represent a PTS challenge. Many reactor-years of experience have taken place since regulatory guidance was issued for PWRs. It seems useful to revisit the subject, from the thermal-hydraulic standpoint, given the large amount of research that is now available, and the few operational events that have occurred as PTS precursors. This work could be modeled as special routines in existing system codes.
- **Vortex Formation** The general theory of vortex formation is not hardly new. However, the number of losses in heat removal due to the formation of vortices in the suction flow path, mostly in PWRs while at a lowered water level condition, is persuasive that not enough is known about the complex relationship between level, temperature, flow rate, inlet geometry, and dissolved gases to adequately guard against flow loss. It is significant here that recent (December 2, 1999) oral statements at the US NRC Advisory Committee on Reactor Safeguards was that the risk of operations at low power and shutdown conditions could be several times the full-power risk of core damage. Regulatory research in this area is not done, at least in the USA, and perhaps nowhere else either, even though vortices have interfered with shutdown cooling in other countries. This would be an excellent topic for international cooperation.

- **Power Oscillations** It is difficult to envision a comprehensive research project for BWR power oscillations. Such oscillations seem to happen every few years, in spite of lessons learned in previous instances. The restrictions on operation seem to be somewhat empirical, and have had some success, although seemingly not 100%. Many experts believe that this is not a risk-significant topic, but nonetheless worrisome. Some analytic methods exist, but do not seem to have contributed to operations restrictions. Some additional attention to analytic methods, perhaps at a modest level in comparison with the other topics previously discussed, is a good idea.

These research suggestions have been formulated from observations of reports on operating experience. The events are all of a recurring nature, and have an event frequency higher than the design basis events (LOCA, for example) that get a much higher degree of attention and funding. The risk of some the events varies, but can be risk important. They should receive more research attention.

NOMENCLATURE

ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
LOCA	Loss of Coolant Accident
NRC	Nuclear Regulatory Commission
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RELAP	Reactor Loss of Coolant Analysis Package
RHR	Residual Heat Removal System
RWST	Refueling Water Storage Tank
TRAC	Transient Reactor Analysis Code

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