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REACTOR VESSEL PRESSURE TRANSIENT PROTECTION FOR PRESSURIZED WATER REACTORS

Final Report

Gary Zech

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

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Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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APPENDICES

1.0 INTRODUCTION

During the past few years the NRC has been studying the issue of protection of the reactor pressure vessels at Pressurized Water Reactors (PWRs) from transients when the vessels are at a relatively low temperature. This effort was prompted by concerns related to the safety margins available to vessel damage as a result of such events.

Nuclear Reactor Regulation Category A Technical Activity No. A-26 was established to set forth the NRC plan for resolution of the generic aspects of this safety issue.

The purpose of this report is to document the completion of this generic technical activity. Implementation of the positions developed through the TAP are continuing.

2.0 BACKGROUND

In the latter part of 1976, a technical safety issue was identified regarding the frequency of occurrence of pressure transients in the Reactor Coolant System (RCS) of operating pressurized water reactor (PWR) facilities. The technical issue related to the safety margins-to-failure for reactor pressure vessels should they be subjected to severe pressure transients while at a relatively low temperature, as compared to normal operating temperatures. Following the highlighting of this technical issue, on November 1, 1976 the staff published a "Technical Report on Reactor Vessel Pressure Transients" which summarized the various considerations relevant to this issue, discussed operating experience with pressure transients up to that time and indicated the actions to be taken to reduce the likelihood of such events. A copy of that report is enclosed as Appendix A.

As indicated, the pressure transients of concern had occurred at PWR facilities. The majority of events were during startup or shutdown operations when the RCS was in a water solid condition (i.e., no steam bubble present in the pressurizer to act as a surge volume). During such conditions the RCS is susceptible to a rapid increase in system pressure through thermal expansion of the RCS water, or injection of water into the system without adequate relief capacity or discharge flow path to control the pressure increase. Boiling Water Reactor (BWR) facilities never operate in a water solid condition. During cold shutdown conditions, a letdown path is maintained through the reactor water cleanup system to remove water that is added to the reactor through control rod drive seals. This flow is controlled to maintain reactor water levels within a narrow range. The upper region of the reactor vessel, therefore, always contains a surge volume of either steam vapor or gas (air).

Each operating PWR facility has Technical Specification requirements to identify the allowable RCS pressure for a given temperature. These limits are based on the Fracture Toughness Requirements as contained in Appendix G to 10 CFR Part 50. The Technical Specification limits are as least as conservative and, in most cases, are more conservative than the Appendix G requirements. A violation of the Technical Specification limits, therefore, does not necessarily also involve a violation of Appendix G requirements. In determining the potential damage to a pressure vessel, as the result of a pressure transient, the effects of radiation damage and the potential for flaws in the vessel are also considered. Based on the above, the staff concluded that the pressure transients that had occurred were such that the fracture mechanics and fatigue calculations indicated that the reactor vessels were not damaged and that continued operation of these vessels was permissible.

With regard to operating PWR facilities in general, the staff concluded that adequate protection existed* to protect the health and safety of the public until any design changes that were determined

*Calculations performed using a computer code (OCTAVIA) recently developed by the Office of Nuclear Regulatory Research provide a quantitative basis in support of this general conclusion. The OCTAVIA computer code is described in NUREG-0258, "The OCTAVIA Computer Code: PWR Reactor Pressure Vessel Failure Probabilities Due to Operationally Caused Pressure Transients," March 1978.

to be necessary could be implemented. In addition, since very large margins to safety exist for unirradiated reactor vessels, new plants should be permitted to be licensed under existing safety criteria. We also concluded, however, that additional measures should be taken to reduce the likelihood of occurrence of future pressure transients. This is being accomplished through a two-phase approach which is discussed in the following Corrective Actions section.

3.0 CORRECTIVE ACTIONS

3.1 Phase I

Phase I of the program to reduce the likelihood of pressure transients in operating PWRs was an effort that began in the fall of 1976 and which was completed in early 1977. It involved a review of operating and administrative procedures at each of the facilities to improve or modify these documents such that the conditions known to cause or lead to RCS pressure transients were properly identified and, if possible, eliminated. Appendix B provides examples of the types of controls that resulted from this review.

The staff's Technical Report of November 1, 1976 summarized the operating experience with regard to overpressure events up to that time. Since then, the increased administrative measures, coupled with a greater awareness of such pressure transients, has resulted in a general decline in the frequency of occurrence of such events.

Appendix C sets forth that history of occurrence of pressure transients in PWRs in the United States since 1969. It is expected that the rate of occurrence will further decrease as Phase II of the program is fully implemented in operating PWRs.

As indicated in Appendix C, three pressure transients have occurred in operating PWRs since 1976. These events occurred while the plants were shutdown and in a water-solid condition at a relatively low reactor coolant system (RCS) temperature. They also occurred before final implementation of equipment design changes to mitigate the consequences of such transients. Such design changes would probably have terminated the transients before the Technical Specification limits were exceeded. It should be noted, however, that the Appendix G limits for each of the three facilities involved were not exceeded during the pressure transients.

3.2 Phase II

Phase II of the program is on-going and involves the implementation of system design changes, such as added pressure relief capability during low temperature conditions, to prevent any future pressure transient from exceeding the requirements of Appendix G to 10 CFR Part 50. The timing of this phase was such that it coincided with the development of the NRR Technical Activity Program. In August 1977, the staff established a Category A Technical Activity entitled

"Reactor Vessel Pressure Protection (Overpressure Protection)." The Task Action Plan (TAP) A-26, was designed to develop acceptance criteria for overpressure protection systems to resolve this issue. A copy of TAP A-26 is attached as Appendix D.

TAP A-26 identified the acceptance criteria for design changes at operating PWR facilities and those criteria that were then being developed for overpressure protection systems in CP and OL applications. These criteria dealt with the following general areas:

1) Administrative Controls
2) Single Failure
3) Testability
4) Seismic Qualifications
5) Electrical Design
6) Enabling and Operability

These criteria for CP and OL applications were also identified at that time in a staff draft Branch Technical Position (BTP) which has subsequently been approved and will be incorporated into the Standard Review Plan (SRP). A copy of the BTP is attached as Appendix E. It is intended that the BTP be implemented before start up for all OLs issued after March 14, 1979, and by the first refueling for OLs issued between March 14, 1978 and March 14, 1979.

Certain differences exist in the criteria now being applied to operating PWRs and those to be applied to CPs and OLs, as identified in the BTP. These differences are in the areas of Administrative controls, and electrical and seismic design requirements of the overpressure protection systems. These differences are not significant in terms of the degree of protection that will be provided. Also, since system design changes are more readily implemented at facilities under construction than at operating PWRs, the impacts involved in backfitting all the criteria of the BTP to operating PWRs are not justified.

All operating reactor PWR licensees have completed an evaluation of their RCS response to potential pressure transients and, where determined necessary, have submitted a description of proposed design changes at their respective facilities that are intended to mitigate the consequences of pressure transients.

The implementation of design changes to protect against pressure transients that might exceed the requirements of Appendix G to 10 CFR Part 50 is continuing. In some cases such as the generic analysis for Babcock and Wilcox-designed PWRs and the plant specific analyses of certain other facilities, the staff and licensees are working to resolve questions with regard to the extent to which these analyses meet the acceptance criteria. Each of the facilities involved, however, have what is considered to be an interim overpressure protection system installed.

The completion of the staff's review and the issuance of Safety Evaluation Reports and appropriate Technical Specification changes are expected by the end of 1978 or early 1979.

The majority of the design changes proposed by licensees involve the addition of a second, lower setpoint on the existing power operated relief valve(s) located on piping off the RCS pressurizer. The original purpose of these relief valves was, in most cases, to provide pressure relief during normal operating conditions at a setpoint slightly lower than the code safety relief valves that are also installed on piping from the RCS pressurizer. When cooling down for a refueling outage or for maintenance, the lower setpoint would be selected to provide low temperature overpressure protection for the RCS pressure vessel.

Another type of proposal being evaluated by the staff involves the use of existing spring-loaded relief valves in the Residual Heat Removal (RHR) system, or the addition of similar valves in piping off the RCS pressurizer. Isolation valves would be opened during a RCS cooldown to place the relief valves in service and thereby provide overpressure protection. Issues remaining to be resolved with this type of overpressure protection include possible modifications to the automatic closure feature on isolation valves between the RHR and RCS systems.

4.0 CONCLUSIONS

The upgraded procedural controls which were implemented at operating PWR facilities to reduce the likelihood of reactor coolant system pressure transients pending the development and implementation of long term design changes have significantly reduced the occurrence rate of such events. The relatively few events that have occurred during this interim period have not been significant and were of the type that will be precluded by design changes when the overpressure protection systems are installed.

Task A-26 has been completed with the criteria for overpressure protection systems identified. Those minor differences that exist between the criteria for operating PWR's and those for CP's and OL's are not significant in terms of protection that will be provided and are justified in terms of impact on operating facilities were the criteria of the BTP to be universally applied.

At present, most licensees have installed at least some type of overpressure protection system except in certain newly licensed facilities which must complete such design changes by their first refueling outage. The staff's review of the proposed design changes is continuing with final approval and issuance of Safety Evaluations and necessary Technical Specifications expected during 1978 or early 1979.

APPENDIX A

TECHNICAL REPORT

ON

REACTOR VESSEL PRESSURE TRANSIENTS

November 1, 1976

TECHNICAL REPORT

ON

REACTOR VESSEL PRESSURE TRANSIENTS

Prepared by

an NRR

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1.0 INTRODUCTION

The NRC staff continually reviews experience from operating reactors to assure that an adequate level of safety is maintained at each individual nuclear plant and for the total population of nuclear plants. Accordingly, as new technical information and operating experience become available, the NRC evaluates whether such information could significantly alter the previously determined levels of safety. If the staff concludes that the level of safety needs to be increased, action is taken to accomplish this objective. Under some circumstances immediate action is warranted. In other cases it is more appropriate to develop additional information and take an action in a longer time frame, consistent with maintaining an acceptably low risk to public safety.

Over the last few years, incidents identified as pressure transients* have occurred in pressurized water reactors. To date there have been about thirty such events; eight have occurred in 1976. Half of these events occurred before a plant achieved initial criticality (i.e., before initial operation of the reactor). The majority occurred during startup or shutdown operations. Because of the increasing frequency of such events, the staff for the last six months has been actively discussing the need for improving low

* The term "pressure transients," as used through this report, refers to events that have exceeded the temperature-pressure limits of the reactor vessel that are included in the facility Technical Specifications.

temperature overpressure protection of nuclear reactor vessels with licensees and their nuclear steam supply system vendors. In addition, this subject was recently highlighted by allegations from a now former NRC employee.

This report summarizes the technical considerations relevant to this matter, discusses the safety concerns and existing safety margins at operating reactors, and describes the regulatory actions being taken to resolve this issue by reducing the likelihood of future pressure transient events at operating reactors. The report has been prepared by the many NRC technical specialists involved in various aspects of overpressure protection for reactor vessels.

2.0 REACTOR VESSEL PRESSURE TRANSIENTS DURING STARTUP AND SHUTDOWN CONDITIONS

This section discusses reactor vessel pressure transients during startup and shutdown. The reactor systems considerations are presented, including a brief description of typical systems, a summary of pressure transient events that have occurred, and a discussion of the causes of these events. The reactor vessel materials considerations are also discussed, including fracture toughness, effects of radiation damage, and the potential for flaws in the vessels.

As described herein, all of the pressure transients that have occurred to date were such that fracture mechanics and fatigue calculations indicate that the reactor vessel was not damaged and therefore continued operation of these vessels is permissible. In addition, since very large safety margins to failure exist for unirradiated reactor vessels, new plants can be permitted to be licensed under existing safety criteria. Nevertheless, the staff has concluded that administrative procedures and overpressure protection devices should be upgraded in an appropriate time frame to reduce the likelihood of future pressure transient events.

2.1 Reactor Systems Considerations

The pressure transient events discussed in this report have affected only pressurized water reactors (PWRs). Boiling Water Reactors (BWRs) never operate in a water solid condition. During cold shutdown conditions for BWRs, a letdown path is maintained through the reactor water cleanup system to remove the water added to the reactor through control rod drive seals. This flow is controlled to maintain reactor water levels within a narrow range. Thus the upper region of the reactor vessel always contains vapor (steam) or gas (air). This provides a significant capability to accept volume surges with only small pressure changes.

The BWR reactor is pressurized for normal operation by heatup of the coolant and follows a water saturation pressure line. Thus, high pressures are not produced unless the vessel temperature is sufficient to satisfy Appendix G pressure-temperature limits.

During normal plant operation, the primary reactor coolant system of a pressurized water reactor (PWR) is maintained at a pressure of approximately 2250 psia and an average temperature of 570°F.

At this temperature, the vessel can be safely operated at high pressure with large margins of safety because of the high toughness of the materials.

Primary system pressure fluctuations in a PWR resulting from load changes during normal plant operations or other routine systems transients are controlled by the volume of steam and water maintained in the pressurizer. In the event of more severe, but anticipated, transients while operating at normal temperatures, the primary system is protected from excessive pressure by power-operated relief valves and self-actuated safety valves. These valves and the reactor trip system ensure that the system pressure will not exceed 110% of design pressure even for the most limiting anticipated transients. This is consistent with ASME Code requirements for normal and upset conditions. Higher stresses (and therefore pressures) are permitted by the ASME Code for emergency and faulted conditions.

Pressure transients affecting PWR pressure vessels are of concern during plant startup and shutdown because, at these relatively low temperatures the vessel material has less toughness than at operating temperatures. The relationship between vessel temperature and material resistance to brittle fracture is discussed in Section 2.2 of this report. The pressure-temperature curves, referred to as Technical Specification limits or "Appendix G" limits, have been calculated for operating reactors and represent limiting conditions for operation during startup and shutdown. During such operations the reactor is not critical since the Technical Specifications also require the core to be maintained in a subcritical condition until the coolant temperature is nearly at its normal operating value. Once the system piping, coolant and vessel temperatures are raised to normal operating levels in accordance with Technical Specification limits, full operational pressure can be established.

At cold shutdown conditions the reactor coolant system (RCS) pressure must be below 500 to 700 psig, depending on the specific plant and the amount of radiation the vessel has incurred, to satisfy Appendix G limits. Most reactor coolant system designs do not provide pressure relief capacity to limit the pressure in the reactor vessel to the 500 to 700 psig range in the event of certain inadvertent pressure increases that may occur due to operator errors or equipment malfunctions while in a startup or shutdown condition.

PWRs that are currently operating rely upon a variety of techniques to control system pressure during startup and shutdown. Generally, Reactor Coolant System (RCS) pressure is manually controlled by the manipulation of various flow control valves, or by intermittent operation of the charging pumps. The flow control valves used are either in the letdown line, or in the Chemical Volume and Control System (CVCS) return line on the charging pump discharge. During reactor startup and shutdown, Westinghouse and Combustion Engineering (CE) designed plants are placed in a "water solid" condition. This is accomplished by filling the pressurizer with liquid, whereas during normal operation both liquid and steam are present. A water solid system is utilized for the following reasons:

1) To avoid unnecessary introduction of air (oxygen) into the primary coolant system during cooldown.

2) To accomplish fill and venting operations of the primary system to remove all air and other gases prior to startup and heatup.

3) To pressurize the RCS such that the reactor coolant pump suction and seal pressure requirements are satisfied.

Babcock and Wilcox (B&W) designed reactors, on the other hand, control pressure during startup and shutdown operations by introducing a nitrogen gas blanket above the liquid in the pressurizer.

During normal operation of Westinghouse-designed plants, a letdown flow path is provided from the reactor coolant system through a regenerative heat exchanger, fixed pressure reducing orifices, and containment isolation valves to the CVCS which is located outside of containment. Because the fixed orifices allow only limited flow at low pressures, the additional letdown flow required when the reactor is shutdown is provided through the residual heat removal (RHR) system. Most Westinghouse designed plants have a relief valve on the RHR system to accommodate limited pressure surges prior to automatic isolation of the RHR system by its pressure protection interlock. However, the relief capacity is insufficient to accommodate large volume surges. In addition, an inadvertent isolation of the RHR system would both reduce the letdown flow and isolate the low pressure relief of the reactor coolant system via the RHR system. This would result in a RCS pressure buildup due to the continued addition of liquid from the CVCS return. Similarly, closure of the air operated valve in the line connecting the RHR system to the CVCS would have the same result. Certain Emergency Core Cooling System (ECCS) actuation signals, such as high pressure injection (HPI) initiation, also isolate the RHR system.

On CE designed reactors, normal letdown and low pressure letdown are accomplished via a single letdown path. Letdown flow is provided during normal operation or at shutdown through a flow control valve with sufficient control range to handle the normal flow rates. The flow control valve is air-operated and is in series with air-operated letdown isolation valves. Any occurrence that isolates this single path or exceeds its relief capacity could result in a pressure transient event. Instrument air is usually isolated by ECCS or HPI initiating signals which closes these air-operated valves.

Normal letdown during B&W plant operation is via a fixed orifice. Low pressure letdown during plant shutdown is through a parallel flow path having a larger capacity air-operated valve. Closing the alternate flow paths results in insufficient letdown flow to prevent overpressurization during large volume injections while the RCS is at low pressure. As mentioned previously, B&W designed plants do not routinely operate in water-solid conditions during shutdown. A nitrogen blanket in the pressurizer is used to maintain pressure during startup and shutdown operations. This procedure provides a mechanism to absorb moderate changes in reactor coolant volume without reaching high RCS pressures. However, the nitrogen volume is limited and large coolant injections that remain unchecked can defeat this protection.

2.1.1 History of Pressure Transient Events

Over the past few years, licensees have reported 30 incidents of reactor coolant system pressure transients in excess of the Technical Specification or Appendix G pressure-temperature limits. These events are shown in Table 1. The majority of cases have occurred during reactor startup or shutdown when the reactor coolant system was in a water solid condition.

Of the 30 events, 10 reached a pressure of 1000 psig or more, 4 reached a pressure of 1500 psig or more, 3 reached a pressure of 2000 or more and 1 exceeded 3000 psig. However, half of these incidents occurred prior to initial criticality of the reactor. Since there was no core decay heat or fission products, these events did not pose a potential hazard to public health and safety.

Of the 15 events that occurred after initial criticality, 6 reached a pressure of 1000 psig or more, 2 reached 1500 psig or more and 1 reached 2250 psig.

Since 1972, the frequency of occurrences has remained relatively constant, with relatively small yearly fluctuations. However, during the first half of 1976, the frequency of events was greater than average.

Comparison of the occurrence rates for each of the NSSS vendors indicates that Westinghouse designed plants, with 65% of the total PWR operating experience, have experienced 86% of the plant pressure transients.

The CE designed plants have accounted for 10% of the incidents with 12% of the total PWR operating experience.

Babcock and Wilcox designed plants, which use a nitrogen blanket in the pressurizer, have experienced only one pressure transient event although their plants represent 23% of the industry PWR operating experience.

After each event that exceeds the plant's pressure-temperature limits, the licensee is obligated to evaluate the impact of the event on plant equipment (including the pressure vessel) to assure that no damage has occurred and that continued operation would not adversely affect the public health and safety.

TABLE 1

	INCIDENT (Date)	CAUSE DESCRIPTION	PRESSURE TRANSIENT FROM (PSIG)	TO	TECH SPEC LIMIT (PSIG)	TIME TO REACH PEAK PRESSURE (minutes)
1.	Beaver Valley Unit No. 1 (2/24/76)	Operator error in transferring electrical buses caused instrument spike isolating letdown from RHR System	400	1000	440 (130 F)*	Note 1
2.	Indian Point Unit No. 2 (2/16/72)	Unknown	420	670	500 (140 F)*	2
3.	Indian Point Unit No. 2 (2/17/72)	Operator isolated letdown without verifying availability of letdown thru RHR system	420	650	500 (180 F)*	2
4.	Indian Point Unit No. 2 (3/8/72)	Reactor coolant pump starting swept cold water thru hot steam generator-pressure increase due to thermal expansion	400	640	500 (115 F)*	1
5.	Indian Point Unit No. 2 (4/6/72)	Operator inadvertently isolated letdown	420	680	500 (170 F)*	2
6.	Indian Point Unit No. 2 (5/18/73)	Closure of certain air operator valves in reactor coolant letdown system caused by freezing of moisture in air supply line.	440	575	500 (130 F)*	Note 1

	INCIDENT (Date)	CAUSE DESCRIPTION	PRESSURE TRANSIENT FROM (PSIG) to		TECH SPEC LIMIT (PSIG)	TIME TO REACH PEAK PRESSURE (min.)
7.	Indian Point Unit 2 (1/23/74)	Starting of a single reactor coolant pump caused pressure surge. A nitrogen blanket in the pressurizer to act as a surge volume had been established; however, the amount of nitrogen added to the pressurizer was insufficient.	425	525	500 (190 F)*	Note 1
8.	Indian Point Unit No. 2 (2/22/74)	An inadvertent safety injection signal was generated which, by design, caused the accumulator discharge stop valves to open.	150	560	500 (115 F)*	Note 1
9.	Oconee Nuclear Station Unit 2 (11/15/73)	During Zero Power Physics testing, test procedure instructions directed operating personnel to increase reactor coolant pressure to approximately 1860 psig violating the limits.	800	1860	1600 (300 F)*	30
10.	Palisades (9/1/74)	A procedure "CAUTION" statement was not rigorously adhered to while performing a primary coolant system leak test	---	960	Requires 160 F to pressurize above 885 (150 F)*	---

INCIDENT (Date)	CAUSE DESCRIPTION	PRESSURE TRANSIENT FROM (PSIG)	TO	TECH SPEC LIMIT (PSIG)	TIME TO REACH PEAK PRESSURE (min.)
11. Point Beach Unit No. 2 (12/10/74)	Following repair, a safety injection pump was lined up for a test run. However, safety injection pump discharge was not isolated from injecting into the reactor coolant system. Pressure transient caused by starting of SI pump.	345	1400	615 (850) (170 F)*	30 Seconds
12. Point Beach Unit No. 2 (2/28/76)	Operational reasons required the RHR system to be isolated from the reactor coolant system. Reduced letdown resulted in pressure increase	400	830	615 (168 F)*	Note 1
13. Prairie Island Unit No. 1 (10/31/73)	Reactor coolant pump starting swept cold water thru hot steam generator-pressure increase due to thermal expansion	420	1100	720 (132 F)*	Note 1
14. Prairie Island Unit No. 1 (1/16/74)	While conducting Safeguards Logic Train A monthly surveillance test, a SI signal was initiated when a step which puts Train A in TEST was inadvertently missed. The SI signal opened No. 11 accumulator outlet isolation valve. RHR System isolation occurred as designed at 600 psig.	395	840	610 (90 F)*	Note 1

INCIDENT (Date)	CAUSE DESCRIPTION	PRESSURE TRANSIENT FROM (psig)	TO	TECH SPEC LIMIT (Psig)	Time to Reach Peak Pressure (min.)
15. Prairie Island Unit No. 2 (11/27/74)	A test signal injected into the letdown controller instrument loop caused a letdown control valve to go closed. This isolated the letdown path. RHR System automatically isolated.	Note 1	900	800 (155 F)*	Note 1
16. St. Lucie Unit No. 1 (8/12/75)	Letdown isolation valve failed closed when I&C personnel removed cover from sealing relay associated with letdown isolation valve, When relay cover was removed, broken wires on relay became disconnected causing letdown valve to close.	210	600 (660)	520 (105 F)*	Note 1
17. Surry Unit No. 1 (1/28/73)	During process of filling and venting the RCS, "A" accumulator motor operated discharge isolation valve was opened to sweep any air trapped in accumulator discharge line into RCS. The opening of the valve caused the accumulator to cause the increase in RCS pressure.	450	590	500 (80 F)*	1
18. Trojan (7/22/75)	The RHR suction valve from the RCS was closed by an unknown person (i.e., this isolated letdown) while the positive displacement charging pump was operating.	400	3326	520 (between 100 and 105 F)*	10 to 12

INCIDENT (Date)	CAUSE DESCRIPTION	PRESSURE TRANSIENT FROM (Psig)	TO	TECH SPEC LIMIT (Psig)	Time to Reach Peak Pressure (min.)
19. Turkey Point Unit No. 3 (12/3/74)	In preparation for starting a reactor coolant pump, the operator placed the letdown control valve in automatic in order to increase reactor coolant pressure. At 465 psig the RHR system loop suction isolation valve automatically closed isolating letdown.	50	800	510 (105 F)*	Note 1
20. Zion Unit No. 1 (6/13/73)	Charging pump 1A, with suction from RWST, was started to increase reactor system pressure. Normal pressure control of continuous charging and letdown was not being used since VCT was unavailable Operator was distracted by a telephone call and left the area of the pump control switch. Unattended pump continued to pressurize system. RHR suction relief valve failed to lift and RHR system later isolated automatically at 600 psig.	110	1290	460 (105 F)*	Note 1
21. Zion Unit No. 1 (6/3/75)	Operator failed to stop the centrifugal charging pump when he secured the RHR system to replace the RHR suction relief valve. When the RHR system was secured, letdown was also secured.	100	1100	480 (115 F)*	10

INCIDENT (date)	CAUSE DESCRIPTION	PRESSURE TRANSIENT FROM (Psig)	to	TECH SPEC LIMIT (Psig)	TIME to Reach Peak (min).
22. Zion Unit No. 2 (9/18/75)	Station personnel were performing a RHR valve interlock test in which the RHR system is automatically isolated from the reactor coolant system. When the applied test signal reached the setpoint, the RHR isolation valves closed removing the letdown path.	95	1300	450 (88 F)*	15
23. Ginna (1969)	Operator inadvertently isolated letdown while charging. Safety valves relieved to terminate transient.	Note 1	2485	Note 1 (100-150 F)*	Note 1
24. Peach Bottom Unit No. 2 (3/6/74)	Following a main steam line isolation test, portions of the reactor vessel shell temperatures decayed to 125 F while reactor pressure remained at approximately 400 psig.	---	400	250 (125 F)*	
25. Beaver Valley Unit No. 1 (3/5/76)	Instrument Technician tripped wrong B/S during MSP, then OPS placed inverter in service with output breaker open, deenergizing #1 vital bus, causing SIS which isolated letdown.	400	1150	440 (150 F)*	Note 1

INCIDENT (date)	CAUSE DESCRIPTION	PRESSURE TRANSIENT FROM (Psig)	To	TECH SPEC LIMIT (Psig)	TIME TO REACH PEAK PRESSURE (Min.)
26. D. C. Cook Unit No. 1 (4/14/76)	During RPS testing, inadvertent letdown isolation was initiated.	Note 1	1040	470 (110 F)*	Note 1
27. St. Lucie Unit No. 1 (6/17/76)	With Shutdown Cooling System secured, a reactor coolant pump was started. Pressure excursion was due to a rapid heatup (95 to 130 F) of the RCS water from the reactor vessel and coolant piping when it was circulated thru the steam generators.	435	815	520 (100 F)*	1
28. Beaver Valley Unit No. 1 (3/13/76)	Inadvertent safety injection due to Solid State Protection System block failure.	425	495	470 (190 F)*	Less than one minute
29. Indian Point Unit No. 2 (9/12/76)	Instrument air header pressure was lost resulting in closure of letdown valves and opening of both charging path valves with one charging pump running.	400	515	500 (110 F)*	5
30. Indian Point Unit No. 3 (9/30/76)	Spurious closure of RHR pump suction isolation valves isolated letdown while charging.	50	2250	740 (185 F)*	7

NOTE 1 - The available abnormal occurrence report does not provide this information.

* - Temperature of reactor vessel during transient

2.1.2 Causes of Pressure Transient Events

The causes of water-solid pressure transients that exceeded Technical Specification limits can be divided into two separate categories: fluid injections and thermal expansions. Most of the events have been the result of a single operator error or equipment failure.

Fluid Injection

RCS pressure transients have resulted from coolant addition from sources such as charging pumps, safety injection pumps, and safety injection accumulators. At cold shutdown conditions a charging pump is normally kept in operation to maintain system pressure between 200 and 400 psig. Under these conditions sufficient letdown flow must be maintained to prevent an overpressure transient.

As stated above, Westinghouse designed reactors rely on an alternate letdown path through the RHR system during shutdown. Any operator action, maintenance or test procedure, or equipment failure that causes isolation of the RHR letdown path on these reactors can cause a pressure transient event. A transient can increase the RCS pressure to a value that will cause the RHR system to automatically isolate. Therefore, in some cases, the consequences of a mild pressure transient would have been mitigated if the RHR pressure-relief valves were set to open before the RHR system is isolated. To date, nine incidents have been caused by inadvertent isolation of the RHR letdown path. Reasons cited for RHR isolation have included interlock testing, operator error, and equipment failure.

If only small volumes of water are being added to the system, the normal letdown path is sufficient to control pressure during startup and shutdown conditions. To date, 7 incidents have been caused by closing of the normal letdown path while charging into the system. Reasons cited for letdown isolation include operator error and loss of valve control air supply. One incident on a CE-designed plant occurred during shutdown when a letdown isolation valve closed. Instrumentation technicians were removing a cover from a relay associated with the letdown system and inadvertently severed control wires in the relay box. This caused a short circuit which caused the isolation valve to close.

For large volume, high pressure fluid sources, such as safety injection pumps and accumulators, the letdown paths available are insufficient to accommodate the large influx. In addition, the letdown lines are generally isolated on an ECCS actuation signal and would probably be unavailable. To date, 6 incidents have resulted from inadvertent safety injection, operator error, and testing errors.

Two pressure transient incidents, one on a CE-designed plant and the other during the only B&W-designed plant, occurred during test. In the B&W case, the test procedures were incorrect and actually instructed the operator to pressurize the plant in violation of the Technical Specification limits. This procedure has been corrected. In the CE event, an operator ignored a "caution statement" in the test procedures.

Thermal Expansion

Thermal expansion of the primary coolant can result from feedback of heat from the secondary side of the steam generators, from reactor coolant pump heat generation, from decay heat and from pressurizer heaters. There have been three incidents that have occurred as a result of the startup of reactor coolant pumps which swept cold RCS coolant through hot steam generators. The resulting thermal expansion caused a rapid pressure increase. Although no incidents to date have been recorded relating to reactor coolant pump heat generation, decay heat or pressurizer heaters, these are recognized as potential causes.

2.2 Reactor Vessel Materials Considerations

Reactor vessels for pressurized water reactor plants are typically about 15 feet in diameter, 8 to 10 inches thick, and 40-50 feet high. They are constructed of high quality steel made to rigid specifications, and fabricated and inspected in accordance with the time-proven rules of the ASME Boiler and Pressure Vessel Code. Steels used for reactor vessels are particularly tough at reactor operating conditions. Since reactor vessel steels are less tough and can fail in a brittle manner at low temperatures, power reactors have always operated with restrictions on the pressure during startup and shutdown operations. Long-term neutron irradiation increases the temperature at which the steel attains maximum toughness. This effect of radiation has been considered in preparing Technical Specifications on pressure-temperature limits. Prior to 1973 these limits were developed using the available state-of-the-art information at the time of initial licensing, and the methods varied somewhat from plant-to-plant.

In 1971, the then AEC proposed rules to be used to establish these pressure-temperature limits. Industry realized their importance and developed them further using advanced fracture mechanics concepts. After many discussions, these rules were incorporated into the ASME Code, and incorporated into the Commission Regulations in 1973. These rules provide wide margins of safety for all operational conditions, and include methods to account for radiation effects. They are incorporated as Appendices G and H to 10 CFR Part 50. These rules are implemented as pressure-temperature limits in the Technical Specifications for all operating reactors.

Because the calculated limits change during the life of each plant as it becomes irradiated, and because it would be impractical to continually change these limits, they are usually calculated so as to be effective for an extended period of time. Thus, the limits in effect at a given time may be based on the properties expected in the vessel 5 or more years in the future, making them as much as 100°F more conservative than Appendix G requirements during the early portion of this period.

The methods used to determine the pressure-temperature limits are based on detailed structural-materials analytical methods known as fracture mechanics that have been developed over the past twenty or more years. These basic methods are widely used in the aerospace industry to ensure the safety of aircraft and are well-proven and accepted. Using these methods, specific margins against possible failure can be calculated and any desired degree of conservatism can be imposed in a quantitative manner, provided information on flaw size, stress, and material properties are adequately known or conservatively assumed.

In determining limits for Technical Specifications, conservative methods are used to determine the minimum toughness of the reactor vessel material taking into account radiation damage. The pressure-temperature limits are then calculated in a manner that provides a nominal factor of two against failure with a very large flaw located in the most highly irradiated area. Specifically, the rules require that it be assumed that a flaw over 2" deep and 1 foot long (1/4 of the wall thickness by 1.5 times the thickness) exists in a vessel when calculating pressure-temperature limits. It should be noted that known flaws more than about 3/4" long with a depth of about 2% of the vessel wall thickness would not be permitted during original construction of the vessel. In addition, the portion of the vessel subject to significant radiation is not subject to severe fatigue conditions that cause formation or growth of cracks.

The detailed procedure specified to conservatively estimate the fracture toughness of the vessel material at any desired temperature is described in Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. This consists of conducting two types of toughness tests (commonly known as Charpy impact and Drop Weight Nil-Ductility tests) to determine a reference temperature, RT_{NDT} (see Section 2.2.2 for definition), for the heat of steel being evaluated. The minimum toughness (K_{IR}) that must be assumed for the steel at any desired temperature is obtained from a curve giving K_{IR} as a function of temperature, using the RT_{NDT} temperature as a reference.

This curve represents the lower bound of all applicable data. At temperatures above RT_{NDT}, the range of experimental data is large, the lower bound of the data being about 25% less than the average. At lower temperatures, data scatter is less - meaning that there is less likelihood that the actual toughness is significantly higher than the K_{IR} value assumed.

To summarize, the safety margins against reactor vessel failure include:

1) At temperatures of actual power operation, reactor vessel steel is very tough and very resistant to failure in any manner even when highly overstressed and after many years of radiation.
2) Pressure-temperature limits are developed for lower temperature operation during startup, test, or shutdown conditions that have wide safety margins in terms of allowable pressure, and protection against postulated flaws.
3) Those pressure-temperature limits that are included in a plant's Technical Specifications are developed as if the reactor had already operated for a significant period of

time so the pressure-temperature limits will be even more conservative than required by Appendix G to 10 CFR Part 50 for the first several years.

As the vessel becomes more irradiated, the permitted pressure-temperature limits must shift to higher temperatures to provide a comparable margin of safety. For example, a new reactor vessel with a flaw about 1/2" deep (about 5% of the vessel wall thickness), could probably withstand pressures as high as about 6000 psi at temperatures around 100°F. A highly irradiated vessel might have to be at temperatures of 300-400°F to provide the same safety margins.

The pressure-temperature calculations for highly irradiated vessels, representative of end-of-life conditions, indicate lower safety margins at low temperatures. One reason for this is that the safety margin, due to a required design factor of two to be applied to primary stresses, may be reduced at lower pressures by the presence of otherwise unaccounted for residual stresses. Another factor is that there is less assurance that the material will have an actual toughness considerably higher than the assumed toughness.

These considerations represent the basis for the NRC position that the risk involved in exceeding the pressure-temperature limits is acceptable during initial operation of nuclear plants. Since safety margins will decrease with significant irradiation, positive steps are being taken to reduce the probability of inadvertent violation of technical specifications.

2.2.1 Fracture Toughness

Resistance to brittle fracture for a component containing flaws is described quantitatively by a material property generally denoted as fracture toughness. This resistance to fracture, or fracture toughness, has different values and characteristics depending upon the material being considered. For nuclear reactor pressure vessels steels, four considerations associated with pressure transients in the vessel's ductile-brittle transition temperature range are relevant. First, fracture toughness increases with increasing temperature. Second, fracture toughness decreases with increasing load rates. Third, fracture toughness decreases with neutron irradiation. Finally, the fracture toughness values for reactor vessel steels meet or exceed those generally available in other pressure vessel steels.

To determine appropriate fracture toughness values for reactor vessel design, many experimental tests have been conducted at various temperatures and loading rates. The results of these tests indicate that at any one temperature and load rate, the material fracture toughness covers a range of values. This range is a measure of the dispersion or scatter about some expected value of material fracture toughness and is representative of variations normally encountered in properties of steels. Using these test results, a fracture toughness curve has been constructed for the evaluation of reactor pressure vessels. This design curve employs two conservative assumptions to ensure an adequate safety margin for fracture toughness values.

First, while many operational transients, including the pressurization transients, are representative of relatively low loading rates, the design curve is based on the lower fracture toughness values associated with the rapid load rate test results. Second, the design curve is constructed so that it is a lower bound for all the available and applicable experimental data. The resulting fracture toughness design curve is about 25% less than the expected toughness values for reactor vessels, except at very low temperatures representative of end-of-life design conditions where the curve is typical of the average fracture toughness data. The design curve constructed in this manner is given in Appendix G to Section III of the ASME Code.

2.2.2 Radiation Damage

Exceeding established pressure-temperature limits becomes a more important consideration as reactor vessels accumulate radiation damage in service. Neutron radiation during power operation gradually changes the strength and ductility of the vessel material. The resulting decrease in resistance to fracture is compensated for during startup by warming the vessel to a higher temperature before applying pressure. Therefore, a pressurization transient that occurs at 180°F, for example, is more significant after 10-15 years of service than it is in the first year of service.

Most of the neutron radiation damage occurs in the "beltline", that part of the cylindrical shell of the reactor vessel directly opposite the core. The beltline usually contains at least two shell courses and several welds. By design, there are no nozzles, flanges or changes in thickness of the shell of the vessel in the most highly irradiated region. Thus, stresses are more accurately determinable because there are no stress concentrations involved.

The amount of radiation damage after a given amount of service can be predicted with fair accuracy on the basis of experience and analysis, and the prediction is checked by surveillance testing as required by Appendix H of 10 CFR Part 50.

From reactor-physics principles, the neutron flux that reaches the wall of the vessel at a given power level is calculated, taking into account the spectrum of energy levels so that the number of high energy neutrons per second per unit area is predicted. Attenuation of the neutron energy as the neutrons penetrate the vessel wall is also calculated. The final result is a predicted fluence (flux times time) of high energy neutrons (normally those above 1 MeV) at any selected depth in the wall, as needed in the fracture analysis.

The effect of a given neutron fluence on the fracture toughness of the reactor vessel steel can be predicted on the basis of test data from many programs. Only one basic type of steel was in the reactors that are of concern today, but research has revealed a considerable variation in sensitivity to radiation damage from plate to plate and weld to weld, which results from variations in chemical composition. The effects of elements such as copper and phosphorus are now known with sufficient accuracy to permit the use of empirical formulas that predict sensitivity to radiation damage as a function of fluence and chemical composition.

Radiation damage is measured in two ways: by the shift in temperature that is required to achieve a pre-selected level of toughness, relative to that of the unirradiated material, and by the decrease in "upper shelf" toughness, the plateau achieved at a certain temperature when the fracture appearance is fully ductile.

The preponderence of data on radiation damage has been obtained with a test procedure known as the Charpy test*.

Exposure of the test specimens for research purposes takes place in test reactors where the neutron flux, irradiation temperature, etc. can be carefully controlled. Exposure temperature is important because the radiation damage anneals out to a significant degree at 550°F, the typical reactor service temperature.

Surveillance programs required by Appendix H to 10 CFR Part 50 for each operating reactor are designed to monitor the neutron fluence and the radiation damage to the vessel material and thus provide an experimental check on the predictions described above. Samples of the beltline materials are made into Charpy specimens and other types of tensile and fracture toughness specimens and placed in sealed capsules inside the reactor near the vessel wall opposite the core. Typically, there are 5 or more capsules per reactor. Dosimetry materials (neutron flux monitors) are also placed in the capsules to provide an experimental check on the neutron fluence and thus serve as monitors when their activity is subsequently analyzed. Surveillance data from various reactors also make up part of the data base of test results on radiation damage.

*The Charpy test is an impact test of a small, notched beam. A test consists of a set of 6-12 or more specimens, which are broken at a range of temperatures selected to cover the transition from brittle to ductile behavior.

For a typical PWR plant, the first surveillance capsule is withdrawn at the first or second refueling outage. The test results from dosimetry and from the Charpy and other mechanical tests are studied to establish the correctness of the fluence and radiation damage predictions. The NRC's Regulatory Guide 1.99* gives an acceptable basis for the predictions, but if a credible surveillance data point is available, it may be regarded as the pertinent quantity for the adjustment of reference temperature that sets the updated pressure-temperature limits for startup/shutdown of the reactor after suitable correction for fluence. The surveillance capsule leads the reactor vessel wall in fluence received, because neutron energy is attenuated by passage through the water and steel. Furthermore, the pressure-temperature limits are established for a selected operating period of several years; thus the predicted fluence some years hence is used in predicting the adjustment of reference temperature for the purposes of establishing Technical Specifications. Regulatory Guide 1.99 also describes acceptable means for extrapolation from the fluence of the surveillance data to that of the desired operation data.

Elsewhere in this report, reference is made to the term "RT_{NDT}", which literally means the "reference temperature for nil-ductility transition". (The temperature at which transition from brittle fracture to a condition of crack arrest in the drop weight test

*Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials", July, 1975 (effective January 1, 1976).

described below occurs.) It is a quantity established initially for each plate, forging, and weld in the beltline. Following details specified in the ASME Code, RT_{NDT} is a temperature derived from the Charpy test results (described above) and a drop weight test. A drop weight test is another test for resistance of the metal to propagation of a crack under impact loading. As the materials accumulate radiation damage, the values of RT_{NDT} for each material of the vessel beltline increase. As described above, the increase varies with the fluence at the location of the material in the reactor and with its chemical content, notably copper. A key feature of this part of the fracture control procedure is to select which material will be controlling after a given service history. The controlling material may change as operational time increases.

2.2.3 Reactor Vessel Flaws

All operating reactor vessels, regardless of whether they were fabricated in accordance with the ASME Section I, III, or VIII Code, received a rigorous design control, a rigorous fabrication quality control (forming, welding, heat treating, etc.), and an extensive nondestructive examination (NDE) during fabrication.

The principal volumetric examination method specified by these codes for the inspection of vessel welds at fabrication is based on a radiographic technique (RT). These codes also establish a set of acceptance standards based on the length of indications. These standards have evolved from many years of fabrication and service experiences. Specifically, any indication in the reactor vessels not exceeding 3/4" in length or 2% of the wall thickness is acceptable. In addition, these radiographic acceptance standards further specify that any type of crack or zone of incomplete fusion or penetration is unacceptable.

All operating reactor vessels, as specified in the 10 CFR Part 50.55a(g) must be subjected to inservice inspections in accordance with the requirements of Section XI of the ASME Code. Section XI volumetric inspections are typically performed using ultrasonic techniques (UT). The ultrasonic technique has the additional advantage of measuring the flaw depth, which is the crucial dimension in evaluating the structural integrity of the vessel. Further, a flaw depth of 1/10 of that postulated in Appendix G of the ASME Section III Code was selected, as a measure of conservatism, to be the allowable surface flaw depth for the acceptance standards of Section XI of the ASME Code.

In spite of the rigorous quality control and testing, nondestructive examination methods have limitations. Fabrication flaws have been detected during the baseline UT inspection after the vessel had successfully passed either the Section III RT examination of a cursory non-Code ultrasonic examination. Specifically, fabrication cracks have been detected in the Fermi-2, Hatch-1, LaSalle-1, and Shoreham reactor vessel nozzles. These cracks occurred in the welds between the nozzle and the vessel shell and generally occurred near the middle of the vessel wall. These cracks were primarily caused by a combination of setup and welding procedures. All cracks occurred in the same fabrication facility. For the Hatch-1 vessel, the cracks were detected after "N" Code stamping and during the ultrasonic baseline examination required by Section XI of the Code. These cracks were removed and repaired; a maximum depth of approximately 3/4" was observed.

Recently 10 CFR Part 50.55a(g) has been amended to require appropriate inservice inspections, per Section XI of the ASME Code, of all reactor vessels prior to their reaching ten years of service. While many newer vessels have received preservice UT inspections, operating reactor vessels have not yet received a Section XI inservice inspection for the beltline welds. From the above discussion, it is possible that flaws may exist in some of the older vessels that have not yet been subjected to ultrasonic baseline or inservice inspection. However, since fabrication of the beltline welds is simplier than in the nozzle region, discussed above, the occurrence of flaws in the beltline region would be expected to be less likely than in the nozzle region.

2.2.4 Reactor Vessel Pressure Tests

Section III of the ASME Code requires that all pressure vessels be hydrostatically tested at 1.25 times the design pressure in the presence of an authorized inspector before the ASME Code stamp is applied to the vessel. The hydrostatic test assures leak tightness and the absence of structurally significant material or manufacturing defects. The Code recommends that component hydrostatic tests (before the vessel is installed) be conducted at RT_{NDT} + 60^{o}F or higher which means that these tests are typically performed at or near ambient temperature. This demonstrates that the vessels are capable of sustaining pressure significantly higher than the established Technical Specification limits, even at relatively low temperature.

Section III of the ASME Code also requires that a system hydrostatic test be performed on the reactor primary pressure boundary after the pressure vessel is installed. The Section III system hydrostatic test is also conducted at 1.25 times the design pressure and under certain conditions may be substituted for the component hydrostatic test. The minimum temperature for this test is established by Appendix G to 10 CFR Part 50 and is typically performed in the range of 140 to 200^{o}F for new vessels.

Section XI of the ASME Code requires that pressure vessels be subjected to periodic system leakage and system hydrostatic tests after they have been placed in service. System leakage tests are conducted at the normal vessel operating pressure prior to plant startup following each refueling outage. Section XI system hydrostatic tests are conducted at 1.02 to 1.1 times the normal operating pressure (depending on the test temperature) at or near the end of each 10-year inspection interval. The temperature at which these tests are performed is established by Appendix G of 10 CFR Part 50 and is dependent on the accumulated neutron exposure of the pressure vessel. Throughout the service life of the reactor coolant system, additional unscheduled hydrostatic tests are required after repair or modification of the reactor coolant system. The tests provide assurance of continued leak tightness and structural integrity during operation for the life of the plant.

3.0 RESEARCH CONSIDERATIONS

The primary safety consideration related to exceeding pressure limitations at low temperatures is the ability of reactor vessel materials to withstand the imposed pressure stresses at temperatures near or below the RT_{NDT} of the material. Linear elastic fracture mechanics (LEFM) is used to establish reactor operating limitations and provide failure predictions. Research activities related to this consideration have been conducted as part of or in support of the Heavy Section Steel Technology (HSST) program. All of the data from the applicable research activities have substantiated the use of linear elastic fracture mechanics as the basis for Appendix G to Section III and Appendix A to Section XI of the ASME Code.

The Heavy Section Steel Technology program, initiated in 1965, is an NRC-sponsored engineering research activity devoted to extending and developing the technology for assessing both analytically and experimentally the margin of safety against fracture of thick-walled steel pressure vessels of the type used in light-water-cooled nuclear power reactors. The principal area of investigation is the behavior and structural integrity of the steel pressure vessels containing cracklike flaws. The program is administered by the Oak Ridge National Laboratory in Oak Ridge, Tennessee.

Specifically, with regard to the fracture behavior of pressure vessel materials near RT_{NDT}, tests were performed on Intermediate Test Vessels (ITV) as part of the HSST program. Each of the vessels tested had a 6-inch wall thickness, a 39 inch outside diameter, and a length of approximately 8 feet. Three tests were performed in the fracture toughness transition region, two of which had induced flaws in the cylindrical part of the vessel (ITV-2 tested at 32°F and ITV-4 tested at 75°F). The RT_{NDT} for the vessel material was 10°F. These tests demonstrated the applicability of linear elastic fracture mechanics at the temperature of interest. Reference (1) discusses these tests and describes the significance of the results.

In addition to the HSST ITV experiments, testing was also performed on a large number of small-scale cylindrical steel models with deliberately induced flaws. Four of these tests were performed below the RT_{NDT} of the material and clearly demonstrated the applicability of LEFM, i.e., failure conditions for the vessels were accurately predicted by LEFM techniques. Reference 2 contains a list of these small-scale model tests.

Another supporting activity related to the HSST program was the testing of 6-inch-thick flawed tensile specimens at Southwest Research Institute. One of these specimens was tested at minus 40°F and provided further substantiation that LEFM techniques can accurately characterize the fracture behavior of the material at temperatures below RT_{NDT}. Reference 3 includes a discussion of this particular test, and demonstrates the applicability of LEFM techniques when the temperature is below the RT_{NDT}.

As indicated in Section 2.2, the LEFM method of analysis was developed approximately 20 years ago and has been shown by experimental data to accurately characterize fracture behavior in high strength - low toughness materials. LEFM has been used extensively in the aerospace industry for many years. Excellent papers on this subject (References 4 and 5) provide many examples of the application of LEFM to aircraft structures.

As evidenced by the test results discussed above, it has been shown that LEFM does characterize the fracture behavior of flawed reactor pressure vessel steels at temperatures near or below the RT_{NDT}. With minor modifications to the analysis procedures, still termed LEFM, the applicability of LEFM can be extended into the transition temperature regime with only a slight reduction in the accuracy of failure predictions. However, when LEFM is used to predict failure conditions further up into the transition temperature regime, or when flaws are so small that relatively large plastic zones are formed, and as long as the net section stress does not exceed the yield strength, the predictions are always conservative, i.e., the predicted failure pressure will always be less than the actual failure pressure.

In summary, linear elastic fracture mechanics methods have been used for a number of years to characterize the brittle fracture behavior of metals, especially in the aerospace industry. With regard to reactor pressure vessels, in particular, research results from the HSST program have clearly demonstrated that LEFM methods accurately characterize the fracture behavior of thick-section reactor pressure vessel steel containing flaws when the temperature is near or below the RT_{NDT} of the material and substantiate the use of LEFM as the basis for Appendix G (to Section III) and Appendix A (to Section XI) of the ASME Code.

SECTION 3.0

REFERENCES

1. An Evaluation of the HSST Program Intermediate Pressure Vessel Tests in Terms of Light-Water Reactor Pressure Vessel Safety, J. G. Merkle, G. D. Whitman, and R. H. Bryan, ORNL-TM-5090, November 1975.

2. Quarterly Progress Report on Reactor Safety Programs Sponsored by the NRC Division of Reactor Safety Research for April - June 1975, II Heavy Section Steel Technology Program, G. D. Whitman, ORNL-TM-5021, September 1975. (See specifically Table 2.3 on P 15.)

3. Test of 6-Inch Thick Pressure Vessel, ORNL-5059, November 1975. (See specifically Appendix E.)

4. Fracture Toughness Testing and Its Applications, ASTM STP 381, April 1965. (See specifically Stress Analysis of Cracks by P. C. Paris and G. D. M. Sih and Applied Fracture Mechanics by C. F. Tiffany and J. N. Masters.)

5. Fracture Mechanics Guidelines for Aircraft Structural Applications, AFFDL-TR-69-111, D. P. Wilhem, February 1970.

4.0 STATUS OF RESOLUTION FOR OPERATING REACTORS

The NRC staff originally conducted a review of the potential for pressure transient events in PWR facilities in September 1967 at which time a preliminary draft report was written. At that time, there had been no record of pressure transients events to indicate that a significant problem existed which would require specific actions to be taken.

As more experience based on cumulative operating time was gained, isolated instances of pressure transients occurred. In December 1975, a draft of a working paper on Reactor Coolant System Overpressurization Protection was prepared by the staff which proposed criteria to be applied in the review of design changes to prevent occurrences of overpressurization. In early 1976, the NRC staff conducted a review of the reported pressure transient events that had occurred at operating PWR facilities. The staff specifically was interested in those events in which either Technical Specification limits or the limits of Appendix G to 10 CFR Part 50 had been exceeded. Considering the conditions under which these events had been occurring and the apparent increase in the frequency of occurrence, the staff considered it prudent to pursue the matter further with the intent of significantly reducing the frequency of pressure transients in PWR facilities that exceeded the limits of Appendix G to 10 CFR Part 50. The staff therefore met with the PWR NSSS vendors (Westinghouse, Combustion Engineering and Babcock and Wilcox) in May and June of 1976 to exchange information relative to the pressure transient events that had occurred and to discuss those corrective actions that the NSSS vendors were considering in their plant designs to reduce the likelihood of future occurrences and/or to mitigate the consequences of pressure transients during cold, water-solid plant operation.

In July of 1976 the staff met individually with seven PWR licensees to discuss the specific pressure transient events that had occurred at their facilities and to determine what action the utilities had taken to prevent similar occurrences at their plants. The seven licensees were selected because of their record of pressure transient events and such that each of the three NSSS designs would be represented. In each of the meetings conducted, the staff discussed the administrative measures in effect and the design modifications being considered to reduce the probability of pressure transient events. The staff recognized that some of the licensees had taken significant steps to prevent future occurrences of pressure transient events. However, other licensees that had only implemented administrative corrections, such as procedural changes, were advised that this would probably not be adequate. Based on the information gained at these meetings, the staff concluded that for the majority of the plants involved, not all potential pressure transient events would be prevented by the measures that had been identified and that further study would be required on the part of the licensee to formulate and incorporate additional measures to ensure that the limits of Appendix G to 10 CFR Part 50 are not exceeded.

In August 1976, the staff sent each PWR licensee a letter which requested that they conduct an analysis of their system design to determine the susceptibility to pressure transient events. They were advised of the information gained at the July meetings with the seven PWR licensees and of the conclusions the staff had reached following those meetings. The letter to the licensees identified the criteria to be applied in determining the adequacy of protection against pressure transients as being that no single equipment failure or single operator error will result in Appendix G limits being exceeded. Should the result of their analysis show that design modifications would be necessary to meet the acceptance criteria, the licensees were advised to include those modifications in their analysis. Pending implementation of the design modifications identified, the licensees were advised that short-term measures should be incorporated to reduce the likelihood that pressure transient events will occur in the interim period until the permanent design changes can be made. The licensees were requested to notify the staff within 20 days of receipt of the letter whether they would provide the information requested within 60 days.

The 20-day responses to our letter from the licensees with Westinghouse designed plants indicated that they had formed a Task Group of utilities to examine the complexity of the pressure transient events and to identify similarities between Westinghouse plants for the purpose of determining a consistent solution to the issue. The staff was informed that the results of the Task Group meetings would be reported at the end of the 60-day period. One Westinghouse licensee that had not joined the Task Group of utilities was requested to provide its submittal by December 3, 1976. The Task Group indicates that a modification in which a pressurizer power-operated relief valve is reset to a lower relief setting while the plant is shutdown may be capable of providing protection against pressure transients during water-solid conditions. A detailed transient analysis to verify this concept is in progress and is expected to be provided to the staff by December 3, 1976.

The licensees with Babcock and Wilcox (B&W) designed plants indicated in their 20-day responses that the information requested in our August 1976 letter would be provided within the 60-day time period requested. Three of the four B&W submittals have been received. In each of these three facilities, operating procedures preclude operation in a water-solid condition (other than system hydrostatic tests) by requiring that a steam bubble or nitrogen gas bubble be present in the pressurizer. This feature provides additional time for operator action to take place in the event of a pressure transient in that the rate of pressure increase is much less than would exist in a water-solid condition. Two of the three responses received indicate that a dual setpoint power operated relief valve is currently part of their system design. The lower setpoint is selected whenever the system temperature and pressure have decreased during plant outages and therefore provides protection from pressure transients that might otherwise exceed Appendix G limits. The licensee for the third facility has committed to incorporate the dual setpoint feature by December 1, 1976. The staff is currently reviewing these proposals to determine their acceptability. The fourth B&W facility has indicated its intention to submit its proposal by December 3, 1976. Preliminary indications are that it also will employ the dual setpoint pressurizer relief valve as in the other three B&W facilities. In addition, its operating procedures presently require that either a steam bubble or nitrogen gas bubble be present in the pressurizer at all times (except during hydrostatic pressure tests).

Licensees with Combustion Engineering (CE) designed plants indicated in their 20-day response that they had formed an owners' Task Group to develop, with CE's assistance, a generic analysis of the potential for pressure transient events in the CE plants. Results of this analysis were to be provided by February 1977, however, each of the CE licensees were recently contacted regarding their schedule for the submittal of the information requested. We advised them that they should identify to us, no later than December 3, 1976, the generic design modifications planned for the CE facilities and a sufficiently detailed analysis to support their proposals.

Regarding the short term measures to reduce the likelihood of pressure transient events until the long term fixes can be implemented, each of the licensees for the facilities that have yet to incorporate a system design modification have identified the short term measures they have implemented. All of them have reviewed their operating procedures to determine what changes should be made to alert the operators whenever the potential for a primary system pressure transient exists. Other short term measures include: minimizing the time in which the plant is in a water-solid condition; revision of reactor coolant pump startup procedures to minimize temperature differentials between the steam generators and the reactor vessel; revisions to the procedures for filling and venting the Reactor Coolant System; and opening circuit breakers of the motor operators of high pressure injection valves. Short term measures that have been identified will be reviewed by the staff to determine if they are adequate.

The staff is currently reviewing those analyses and design modifications that have been received. It is anticipated that the remaining analyses will be received by December 3, 1976. It is intended that by the end of 1976, the staff will have completed its evaluation of each of the proposals and will have established a position as to what fix should be imposed or applied to each of the facilities. Priority is being given to those facilities for which the frequency of occurrence of pressure transient events and the radiation exposure of the reactor pressure vessel are highest. The schedule for implementation of any design modifications will be established with the objective of completing all changes by the end of 1977.

5.0 OPERATING REACTOR PRESSURE VESSEL INTEGRITY

The integrity of the reactor vessel can be assured during operation, including heatup, cooldown, core operation and inservice testing conditions, by compliance with Appendix G, 10 CFR Part 50. Compliance with Appendix G is required by the Technical Specifications for all plants which include pressure-temperature limits, consistent with Appendix G, for reactor operation. The NRC staff periodically reviews these limits to make certain that they conservatively account for radiation degradation to materials in the reactor beltline region. Also, any changes in these specifications proposed by the utility must be reviewed by the NRC staff and justified by a Safety Evaluation Report. Table 3 shows the status of operating PWR plants regarding compliance with Appendix G and the corresponding Technical Specifications.

For the major pressure transients that have occurred, a summary of the pressure transient, vessel temperature, pressure allowed by the Technical Specifications, and the limiting nil-ductility reference temperature is provided in Table 4. Table 2 presents the current status of operating plants in regard to initial and current nil-ductility reference temperatures. The value of the limiting reference temperature is based on the fluence at the time of the incident. The pressure allowed by the Technical Specifications is based on a fluence value sometime in the future.

Hence, it is a conservative limit and does not necessarily indicate the pressure that would be permitted by Appendix G at the time of the incident. For example, in the three Prairie Island incidents, numbers 6, 7 and 8 in Table 4, the pressures allowed by the Technical Specifications were exceeded by approximately 400, 200 and 100 psi, respectively. However, based on the limiting RT_{NDT} value at the time of the incidents, Appendix G requirements would actually allow higher pressures than those attained in these incidents, i.e. about 300, 150 and 900 psi higher, respectively.

The safety significance of these pressure transients is affected by many factors in addition to the pressure levels reached. The major factors are vessel pressure and temperature, stresses in the vessel, the neutron fluence at the time of the incident, the size and location of any flaws in the reactor vessel, and the material properties of the reactor vessel. Considering these factors, less than 10 of these events resulted in meaningful reductions in the safety margins to vessel failure during a particular event. However, none of these transients reduced the integrity of these reactor vessels for future operation.

TABLE 2 - REFERENCE TEMPERATURE FOR NIL-DUCTILITY TRANSITION

PLANT	INITIAL RT_{NDT} at 1/4T	PRESENT RT_{NDT} at 1/4T
Yankee Rowe	10	210
San Onofre	20	180
Conn. Yankee	50	140
Indian Pt. 2	60	130
Indian Pt. 3	56	75
Turkey Point 3	3	125
Turkey Point 4	0	100
Palisades	20	90
Robinson 2	0	120
Point Beach 1	0	110
Point Beach 2	33	140
Oconee 1	40	100
Oconee 2	40	90
Oconee 3	40	80
Surry 1	9	100
Surry 2	0	75
Ginna	0	120
Prairie Is. 1	0	70
Prairie Is. 2	5	50
Three Mile Is. 1	20	90
Zion 1	27	90
Zion 2	32	75
Kewaunee	0	50

PLANT	INITIAL RT_{NDT} at 1/4T	PRESENT RT_{NDT} at 1/4T
Maine Yankee	-30	70
Rancho Seco	40	80
Arkansas 1	10	60
D. C. Cook 1	40	50
Calvert Cliffs 1	40	70
Beaver Valley 1	75	75
Trojan	40	50

TABLE 3 - COMPLIANCE WITH APPENDIX G, 10 CFR 50

PLANT	STATUS
Yankee Rowe	Complies. Reviewed in June 1976
San Onofre	Complies. Reviewed in November 1974
Conn. Yankee	Complies. Reviewed in April 1974
Indian Pt. 2	Complies. Proposed Amendment dated 4/22/76 reviewed September 1976.
Indian Pt. 3	Complies. Reviewed February 1976
Turkey Pt. 3 & 4	Complies. Proposed Amendment dated 5/21/76 reviewed September 1976
Palisades	Currently under review. New limits are being developed.
Robinson 2	Complies. Reviewed May 1975, and found acceptable for 4.25 EFPY
Point Beach 1	Complies. Reviewed November 1975 and found acceptable for 6 EFPY
Point Beach 2	Complies. Reviewed November 1975 and found acceptable for 2 EFPY
Oconee 1, 2, and 3	Limits are presently being reviewed.
Surry 1 & 2	Complies. Reviewed October 1975 and found acceptable for 3.8 EFPY
Ginna	Complies. Reviewed April 1974
Prairie Island 1 & 2	Complies. Reviewed October 1974
Three Mile Island 1	Complies. Reviewed September 1976 and found acceptable for 2 EFPY
Zion 1 & 2	Currently under review.

PLANT	STATUS
Kewaunee	Complies. Reviewed September 1976 and found acceptable for 8 EFPY.
Maine Yankee	Complies. Reviewed November 1975, and found acceptable for 3.5 EFPY.
Rancho Seco	Complies. Curves good for 2 EFPY.
D. C. Cook 1	Complies. Reviewed November 1973.
Calvert Cliffs 1 & 2	Complies. Reviewed Summer 1976.
Beaver Valley 1	Complies. Reviewed November 1973.
Arkansas 1	Complies. Reviewed September 1976 and found acceptable for 2 EFPY.
Trojan	Complies. Reviewed Summer 1975.

TABLE 4 - MAJOR PRESSURE TRANSIENT INCIDENTS

	INCIDENT (Date)	PRESSURE TRANSIENT FROM (PSIG)	TO	VESSEL TEMPERATURE °F	TECH SPEC PRESSURE LIMIT, PSIG	LIMITING RT_{NDT}* °F
1.	Beaver Valley Unit No. 1 (2/24/76)	400	1000	130	440	75
2.	Oconee Nuclear Station Unit 2 (11/15/73)	800	1860	300	1600	60
3.	Palisades (9/1/74)	---	960	150	---	65
4.	Point Beach Unit No. 2 (12/10/74)	345	1400	170	615	110
5.	Point Beach Unit No. 2 (2/28/76)	400	830	168	615	125
6.	Prairie Island Unit No. 1 (10/31/76)	420	1100	132	720	15
7.	Prairie Island Unit No. 1 (1/16/74)	395	840	90	610	15
8.	Prairie Island Unit No. 2 (11/27/74)	---	900	155	800	5
9.	Trojan (7/22/75)	400	3326	100	520	40
10.	Turkey Point Unit No. 3 (12/3/74)	50	800	105	510	75
11.	Zion Unit No. 1 (6/13/73)	110	1290	105	460	40
12.	Zion Unit No. 1 (6/[illegible]/75)	100	1100	115	480	75

	INCIDENT (Date)	PRESSURE TRANSIENT FROM (PSIG)	TO	VESSEL TEMPERATURE °F	TECH SPEC PRESSURE LIMIT, PSIG	LIMITING RT_{NDT}* °F
13.	Zion Unit No. 2 (9/18/75)	95	1300	88	450	60
14.	Ginna (1969)	---	2485	100 - 150	600	45
15.	Beaver Valley Unit No. 1 (3/5/76)	400	1150	150	440	75
16.	D. C. Cook Unit No. 1 (4/14/76)	---	1040	110	470	40
17.	St. Lucie Unit No. 1 (6/17/76)	435	815	100	520	20
18.	Indian Point Unit No. 3 (9/30/76)	50	2250	185	740	75

*The limiting RT_{NDT} value is based on the fluence at the time of the incident.

6.0 SUMMARY AND CONCLUSION

As described herein, thirty events have been reported in which the pressure-temperature Technical Specification limits for reactor vessels were exceeded; however, less than 10 were of significance. All of the events resulted from either an operator error or equipment malfunction, without any release of radioactivity or damage to the reactor vessel. All of the pressure transients were such that fracture mechanics and fatigue calculations indicate that the reactor vessels were not damaged and that continued operation of these vessels was acceptable.

Since very large safety margins to failure exist for unirradiated reactor vessels, new plants can be permitted to be licensed under existing safety criteria. Nevertheless, the staff has concluded that administrative procedures and overpressure protection devices should be upgraded in an appropriate time frame to reduce the likelihood of future pressure transient events for new plants.

For operating plants, action has been taken by the licensees that is expected to reduce the number of such events by upgrading administrative procedures during the period of time while design modifications to individual plants are being incorporated. Realizing the potential safety significance of such events in the future as more reactor vessels become irradiated, the staff plans to define all necessary changes for operating plants by the end of this year and require implementation of these changes by the end of 1977.

The staff action plan provides adequate protection for the health and safety of the public by immediately reducing the likelihood of future pressure transients through improved administrative procedures, and by further reducing the likelihood of such events through design changes that will be implemented over the next year. As described in this report, reactor vessels have been conservatively designed and generally have substantial margins to failure even from unanticipated pressure transient events. The continuing staff reviews will evaluate each nuclear power plant on a case-by-case basis to assure that licensee actions provide adequate safety margins for the continued protection of the public health and safety.

APPENDIX B

EXAMPLES OF ADMINISTRATIVE CONTROLS IN EFFECT

1. Revised plant operating procedures to minimize the potential for pressure transients.
2. Minimization of the time in a water-solid condition.
3. Revised reactor coolant startup procedures to minimize temperature differentials between the steam generators and the reactor pressure vessel while in a water-solid condition.
4. Incorporation of an alarm, with a setpoint below the maximum allowable pressure for existing temperature conditions, to alert the operator of a pressure transient.
5. Disabling of pressurizer heaters and unneeded high pressure injection or charging pumps during cold, water-solid conditions.

APPENDIX C

FREQUENCY OF OCCURRENCE OF REPORTED PRESSURE TRANSIENTS IN OPERATING PWRs (1969 to April 1978)

YEAR	NUMBER OF EVENTS	NUMBER OF PWRs LICENSED FOR OPERATION	AVERAGE NO. EVENTS PER UNIT/PER YEAR
1969-72	5	13	.143
1973	5	23	.217
1974	8	30	.267
1975	4	32	.125
1976	8	37	.216
1977	1	40	.025
1978	2	41	.049

APPENDIX D

Task A-26

REACTOR VESSEL PRESSURE TRANSIENT PROTECTION
(OVERPRESSURE PROTECTION)

Lead NRR Organization:	Division of Operating Reactors (DOR)
L..d NRR Supervisor.	Darrell G. Eisenhut A/D for Operational Technology, DOR
Task Manager:	Gary G. Zech, DOR
Applicability:	PWRs
Projected Completion Date:	June 1978

1. DESCRIPTION OF PROBLEM

Since 1972, there have been over 30 reported incidents of pressure transients in pressurized water reactors which have exceeded the pressure temperature limits of the reactor vessels involved. These limits were those identified in the technical specifications for each facility and were based on the requirements of Appendix G to 10 CFR Part 50. The majority of these events occurred while in a water solid condition, during startup or shutdown operations, and at relatively low reactor vessel temperatures. Since the reactor vessel material has less toughness at these lower temperatures, it is much more susceptible to failure through brittle fracture at lower temperatures than at normal operating temperatures; and therefore, the margin of safety to vessel failure under low temperature conditions is reduced.

Reactor vessel pressure transients have been initiated by a variety of causes which can be grouped into the following categories: personnel error, procedural deficiencies, component random failure and spurious valve actuation. The resultant pressure transients are of basically two types: a mass input type from charging pumps, safety injection pumps or safety injection accumulators, or a thermal expansion type caused by the feedback of heat from the secondary side of steam generators. The magnitude of the pressure transients varied from minor violations of the Appendix G limits (500 to 1000 psig peak pressure) to pressure increases up to the safety valve setpoint (2450 psig).

Although a new nuclear reactor pressure vessel could in all likelihood withstand pressures considerably greater than the safety valve setpoint, even at lower temperatures, increased neutron irradiation can cause the existing safety margins to significantly decrease due to a reduction in the toughness properties of the vessel. The immediate safety concern is, therefore, the older operating facilities.

In view of the frequency of these transients and the associated potential for pressure vessel damage, the staff has concluded that measures should be taken to minimize the number of occurrences of pressure transients in the future and to reduce the severity of such transients should they occur.

The problem addressed by this Task Action Plan is the identification of those actions that will assure that adequate overpressure protection is provided for both operating PWR facilities and those that have yet to receive their operating licenses.

2. PLAN FOR PROBLEM RESOLUTION

Due to the frequency of occurrence of pressure transients since 1972, NRR conducted a review of the safety concerns and existing safety margins at operating reactor facilities. On November 1, 1976, a Technical Report on Reactor Vessel Pressure Transients was issued which summarized the various considerations relevant to this matter. It was concluded that adequate protection exists for the health and safety of the public by immediately reducing the likelihood of future pressure transients through improved administrative measures and by further reducing the likelihood of such events through design changes that will be implemented over the next year.

At Congressional hearings held in October 1976, the NRR Office Director committed to a schedule for implementation of any design objectives by the end of 1977.

The licensees of operating PWR reactors were requested to provide an analysis of the reactor coolant system response to pressure transients that can occur during startup and shutdown and to identify the design changes determined to be necessary to preclude exceeding the Appendix G limits for their plant. In November 1976, separate meetings were held with the licensees to each of the three PWR NSSS-designed plants to discuss their planned approach to resolve the pressure transient problem. At these meetings, specific criteria were identified that the licensees should apply in the design of equipment intended to prevent pressure transients that might exceed the limits of Appendix G to 10 CFR 50. These criteria were:

A. Credit of Operator Action - No credit can be taken for operator action until 10 minutes after the operator is aware that a pressure transient is in progress.

B. Single Failure Criteria - The pressure protection system should be designed to protect the vessel given a single failure in addition to a failure that initiated the pressure transient. In this area, redundant or diverse pressure protection systems would be considered as meeting the single failure criteria.

C. Testability - The equipment design should include some provision for testing on a schedule consistent with the frequency that the system is used for pressure protection.

D. Seismic Design and IEEE 279 Criteria - Ideally, the pressure protection system should meet both Seismic Category I and IEEE 279 criteria. The basic objective, however, is that the system should not be vulnerable to an event which both causes a pressure transient and causes a failure of equipment needed to terminate the transient.

Subsequent discussions with licensees and between NRR divisions has caused a reconsideration of certain aspects of the above criteria, particularly as they apply to the instrumentation, control and power areas of the proposed design changes.

Because of the large safety margins to vessel failure that exist in unirradiated reactor pressure vessels, it has been determined that new plants can continue to be licensed under existing safety criteria. However, administrative procedures and overpressure protection devices to reduce the likelihood of future pressure transients in a new plant are being required on a timely basis (prior to second cycle). The Reactor Systems Branch (DSS) is developing a Branch Technical Position on Reactor Coolant System Overpressure Protection. This Branch Position will apply to all CP and OL applications, with certain qualifications, and will provide the guidance for continued DSS, DPM and DSE review of the adequacy of the design of the overpressure protection system. Comments have been received on the draft Branch Position which are being evaluated prior to incorporation. The major aspects of the proposed Branch Position are as follows:

A. A system shall be designed and installed which will prevent the exceeding of the applicable Technical Specifications and Appendix G limits for the reactor pressure vessel during plant cooldown or startup. The system shall be capable of relieving pressure during all potential overpressurization events at a rate sufficient to satisfy the Technical Specification limits, particularly while the Reactor Coolant System is in a water solid condition.

B. The system must be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must be provided which demonstrate that the system will provide the required pressure relief capacity assuming the most limiting single failure. The cause for initiation of the event, i.e., operator error, component malfunction, etc., will not be considered as the single active failure. The analysis should assume the most limiting allowable operating conditions (e.g., one RHR train operating or available for letdown, other components in normal operation when the system is water solid such as pressurizer heaters and charging pumps). All potential overpressurization events must be considered when establishing the worst case event.

C. The system must operate automatically, providing a completely independent backup protective feature for the operator. The design must not include manual actions to enable or "turn on" the system or to mitigate the consequences of a potential overpressure event.

D. To assure operational readiness, the overpressure protection system must be tested in the following manner:

(1) A test must be performed to assure operability of the system electronics prior to each shutdown.

(2) A test for valve operability must be conducted as specified in the ASME Code Section XI.

(3) Subsequent to system, valve, or electronics maintenance, a test on that portion(s) of the system must be performed prior to declaring the system operational.

E. The system must meet the design requirements of IEEE 279. The design must be of at least the same quality as those system(s) to which it is connected, such that no portion of the plant design is compromised. The requirements of Regulatory Guide 1.26 must be satisified.

F. The protection system does not have to meet Seismic Category I requirements if it can be shown that an earthquake would not initiate an overpressure transient. The postulated earthquake should be of magnitude equivalent to the SSE. If the earthquake can initiate an overpressure transient, then it should be assumed that loss of offsite power is an expected consequence of the event and the protection system should be designed to Seismic Category I requirements and not require the availability of offsite power to perform its function. Should the applicant show that a postulated earthquake could not cause an overpressure event, the overpressure protection system design must not compromise the design criteria of any other safety-grade system with which it would interface. The requirements of Regulatory Guide 1.29 must be satisfied.

G. The loss of offsite power shall be considered as an anticipated transient which could occur while in a shutdown condition. If this event can initiate an overpressure transient, the overpressure protection system must be independent of offsite power, in addition to performing its function assuming any single active failure.

H. Plant designs which take credit for an active component(s) to mitigate the consequences of an overpressurization event must include an additional analysis considering inadvertent initiation or provide justification to show that existing analyses bound such an event.

The proposed implementation of the Branch Position would be that it should apply to all CP and OL applications, with the exception of the requirement for the system to meet IEEE 279. OL applicants would be allowed to justify reasonable deviations from the requirements of IEEE 279. For those applicants expected to receive an operating license this year, installation of all equipment would occur no later than the first refueling outage. For any plant receiving an operating license in 1978 or later, installation of equipment should be made prior to plant startup.

The basic suggested differences in the criteria that would be applied by DOR to design changes in operating reactors and by DSS, DSE and DPM to applications for a CP or OL are as follows:

A. System Alignment for Operation (Enabling)

DOR: Operator action to align the system for operation is sufficient when accompanied by alarms and procedural verification.

DSS: Fully automatic operation.

B. Administrative Controls

DOR: Administrative controls may be used to eliminate from consideration transients from certain specific sources. Technical Specification controls will be allowed on accumulators, maximum temperature difference between steam generators and the Reactor Coolant System, and one of two trains of high pressure safety injection.

DSS: Administrative controls are not specifically identified as an acceptable means for protection.

C. Seismic Design

DOR: The system should meet Seismic Category I requirements to the extent that an event which causes a pressure transient does not also cause a failure of equipment needed to terminate the transient. The staff, however, will evaluate a licensee's rationale for not fully meeting the Seismic Category I criteria.

DSS: The system does not have to meet Seismic Category I requirements if it can be shown that an earthquake would not initiate an overpressure transient.

D. Electrical Criteria

DOR: IEEE 279 equipment required at an interface with existing safety systems. The balance of the system must be of good quality, have redundancy in actuation channels and function with a loss of offsite power.

DSS: The system must meet the design requirements of IEEE 279.

The task items that require accomplishment for completion of the generic solution are:

A. A finalization of the criteria to be applied in the review of design changes to operating PWR reactors. This will include the resolution of, or justification for, the differences that exist between the DOR criteria and those contained in the Branch Position, as discussed above.

B. The submittal of the Branch Position for approval.

C. Approval of the Branch Position by the Regulatory Requirements Review Committee.

3. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION OF TASK

As indicated in Section 1, there have been over 30 reported incidents of pressure transients in pressurized water reactors which have exceeded the pressure temperature limits of the reactor vessels involved. However, each of these reported events were such that fracture mechanics and fatigue calculations indicated that the reactor vessels were not damaged and that continued operation with these vessels was acceptable. It was recognized, however, that with increased irradiation of reactor pressure vessels, the potential safety significance of such events warrants further action.

The staff initiated a two step approach to the resolution of this problem. First, the likelihood of future pressure transients at operating facilities was immediately reduced by requiring improved administrative measures and secondly, the likelihood of exceeding the reactor vessel pressure limits during such a transient will be further reduced by requiring facility modifications specifically designed to provide overpressure protection.

For construction permit applications, a commitment to provide an overpressure protection system is being required. The system to be installed will be required to meet the criteria that will be delineated in the Branch Technical Position that will result from this task.

The proposed Branch Technical Position is described in some detail in Section 2.0. The final position is not expected to be significantly different than the proposed position in Section 2. These criteria will result in protection systems that reduce the severity of any transients that may occur such that Appendix G limits are not exceeded. Accordingly, we find that such criteria will provide reasonable assurance of no undue risk to the health and safety of the public. Design changes to implement the criteria are expected to include such measures as the installation or modification of power-operated relief valves or spring-loaded relief valves and associated control and alarm circuity. These changes are state-of-the-art and are relatively easy to implement on existing designs after construction has begun. In addition, since completion of this task, i.e., finalizing of the Branch Technical Position, is imminent, there is reasonable assurance that there will be satisfactory resolution of the outstanding safety question prior to operation of any facility receiving a construction permit in the future.

Plants in the operating license stage will also be required to have installed overpressure protection devices. Depending on the timing of the issuance of the operating license, these systems will either be in place prior to commencing operation or installation will be required prior to the first refueling outage.

In the interim, for this latter group of facilities, administrative procedures to reduce the likelihood of pressure transients will be required prior to issuance of the operating license. In addition, because the new vessels are unirradiated, large safety margins to vessel failure exist. Based on the foregoing, there is reasonable assurance that the continued licensing of plants for operation will not present an undue risk to the health and safety of the public.

The licensee of each operating PWR facility has identified short term procedural measures that have already been implemented to reduce the likelihood of reactor coolant system pressure transients, pending design and installation of the final, long term design changes that meet our criteria. These short term measures are, for the most part, administrative in nature and include the following: (1) Upgrading of operating procedures to alert plant operators to the potential for pressure transients, (2) minimization of the time during which the plant is in a water-solid condition, (3) deenergization of high head pumps not required during cold shut down, and (4) installation of an alarm to alert the operator when the system pressure approaches the limits of Appendix G to 10 CFR 50.

In addition, the licensees have identified design changes, utilizing the design criteria described in Section 2, to futher reduce the likelihood of a pressure transient that might exceed Appendix G limits. These design changes have been implemented on most operating PWR facilities and will be implemented on the remaining facilities during forthcoming scheduled plant outages.

In light of the measures that have been or will be taken at operating PWR facilities, we have concluded that the continued operation of facilities until all measures are fully implemented will not present an undue risk to the health and safety of the public.

4. NRR TECHNICAL ORGANIZATIONS INVOLVED

A. Reactor Safety Branch, Division of Operating Reactors

Has overall lead responsibility for the finalization of the design criteria to be applied to overpressure protection systems in operating reactors.

Manpower Estimate: .08 man-year FY 1977

B. Plant Systems Branch, Division of Operating Reactors

Has lead responsibility for the criteria and design requirements to which the instrumentation, control and power aspects of the proposed overpressure protection system in operating reactors must conform.

Manpower Estimate: .04 man-year FY 1977

C. Reactor Systems Branch, Division of Systems Safety

Has lead responsibility for DSS in justifying (or resolving) the differences that exist between the criteria in the Branch Position and those that have been used by DOR. Has lead responsibility for development of the Branch Technical Position identifying the review criteria for overpressure protection systems by applicants for CPs and OLs, and for initiating subsequent changes to Standard Review Plans.

Manpower Estimate: .04 man-year FY 1977

D. Task Manager

Has overall responsibility for the coordination between NRR Branches in the accomplishment of Task Items to complete hte generic solution as identified in this Task Action Plan.

Manpower Estimates: .02 man-year FY 1977
.02 man-year FY 1978

5. TECHNICAL ASSISTANCE REQUIREMENTS

Significant work is in progress under an interagency agreement between the NRC and NRL to evaluate the radiation effects, analytical techniques and advanced testing methods for the analysis of radiation damage of materials in operating reactor vessels. Although not required for this Task Action Plan, information from this program could ultimately affect the acceptance criteria applied to the design of future systems to provide overpressure protection. Engineering Branch of DOR has management responsibility for this program which is described in Category A Technical Activity No. A-11.

6. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

A. Westinghouse Owners' Group

Most of the licensees with Westinghouse-designed operating PWR facilities have formed an ad hoc owners' group to evaluate the problems of reactor vessel overpressurization. These licensees have engaged Westinghouse to perform a transient analysis to include consideration of both mass input and heat input induced overpressurizations. The range of system and component physical parameters, performance characteristics and operating limits applicable to Westinghouse-designed plants are to be used to bound the analysis. The final results of this analysis were submitted in late July 1977.

B. Combustion Engineering Owners' Group

Five of the six operating Combustion Engineering-designed PWR facilities have also formed an owners' group to evaluate the generic aspects of the overpressurization problem. Combustion Engineering has performed an analysis similar to that conducted by Westinghouse and has been submitted by the licensees for the staff's review.

7. ASSISTANCE REQUIREMENTS FROM OTHER NRC OFFICES

A. Office of Nuclear Regulatory Research, Division of Reactor Safety Research

RES, at their own initiative, has developed the OCTAVIA computer code capability of determining the reactor vessel failure probability of operating PWRs relative to the pressure transient events that have occurred. The Engineering Branch (DOR) has used OCTAVIA to compile a listing of these probabilities based on information currently available. This listing has been used by the staff in ordering its review schedules. Ongoing efforts in this area will provide additional analyses with RES continuing to perform in an advisory capacity.

8. POTENTIAL PROBLEMS

A delay in the finalization of the criteria to be applied to design changes in operating reactors would delay the review of the proposed system modifications from PWR licensees.

It should be noted that the Branch Technical Position uses Appendix G to 10 CFR 50 as the limit for all postulated initiating events, regardless of the probability of that event or combination of events occurring. It is recognized that the probability of a safe shutdown earthquake and a resultant overpressurization event is less probable than an anticipated operational occurrence and, therefore, the application of the upset criteria of Appendix G to these less probable events may represent an excessive degree of conservatism. The same is true for a loss of offsite power if it is shown to result in the initiation of an overpressure transient. A less conservative pressure vessel brittle fracture limit may prove to be appropriate for such events. There is no effort presently planned to develop other limits and, therefore, no attempt will be made in the near future to define any additional criteria less conservative than Appendix G. The Branch Technical Position and the criteria to be utilized by DOR for Operating Reactors require that Appendix G be applied to all possible overpressure events including those associated with the safe shutdown earthquake or loss of offsite power. Some licensees have expressed viewpoints which question the validity of the Appendix G limits, similar to the discussion above, and may challenge the criteria we have identified.

APPENDIX E

BRANCH TECHNICAL POSITION RSB 5-2

OVERPRESSURIZATION PROTECTION OF PRESSURIZED WATER REACTORS

WHILE OPERATING AT LOW TEMPERATURES

A. Background

General Design Criterion 15 of Appendix A. 10 CFR 50, requires that "the Reactor Coolant System and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences".

Anticipated operational occurrences, as defined in Appendix A of 10 CFR 50, are "those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power".

Appendix G of 10 CFR 50 provides the fracture toughness requirements for reactor pressure vessels under all conditions. To assure that the Appendix G limits of the reactor coolant pressure boundary are not exceeded during any anticipated operational occurrences, Technical Specification pressure-temperature limits are provided for operating the plant.

The primary concern of this position is that during startup and shutdown conditions at low temperature, especially in a water-solid condition, the reactor coolant system pressure might exceed the reactor vessel pressure-temperature limitations in the Technical Specifications established for protection against brittle fracture. This inadvertent overpressurization could be generated by any one of a variety of malfunctions or operator errors. Many incidents have occurred in operating plants as described in Reference 1.

Additional discussion on the background of this position is contained in Reference 1.

B. Branch Position

1. A system should be designed and installed which will prevent exceeding the applicable Technical Specifications and Appendix G limits for the reactor coolant system while operation at low temperatures. The system should be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to satisfy the Technical Specification limits, particularly while the reactor coolant system is in a water-solid condition.

2. The system must be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must be provided which demonstrate that the sytem will provide the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event, e.g., operator error, component malfunction, will not be considered as the single active failure. The analysis should assume the most limiting allowable operating conditions and systems configuration at the time of the postulated cause of the overpressure event. All potential overpressurization events must be considered when establishing the worst case event. Some events may be prevented by protective interlocks or by locking out power. These events should be reviewed on an individual basis. If the interlock/power lockout is acceptable, it can be excluded from the analyses provided the controls to prevent the event are in the plant Technical Specifications.

3. The system must meet the design requirements of IEEE 279 (see implementation). The system may be manually enable, however, the electrical instrumentation and control system must provide alarms to alert the operator to:

    a. properly enable the system at the correct plant condition during cooldown;

    b. indicate if a pressure transient is occurring.

4. To assure operational readiness, the overpressure protection system must be tested in the following manner:

    a. A test must be performed to assure operability of the system electronics prior to each shutdown.

    b. A test for valve operability must, as a minimum be conducted as specified in the ASME Code Section XI.

c. Subsequent to system, valve, or electronics maintenance, a test on that portion(s) of the system must be performed prior to declaring the system operational.

5. The system must meet the requirements of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants" and Section III of the ASME Code.

6. The overpressure protection system must be designed to function during an Operating Basis Earthquake. It must not compromise the design criteria of any other safety-grade system with which it would interface, such that the requirements of Regulatory Guide 1.29, "Seismic Design Classification" are met.

7. The overpressure protection system must not depend on the availability of offsite power to perform its function.

8. Overpressure protection systems which take credit for an active component(s) to mitigate the consequences of an overpressurization event must include additional analyses considering inadvertent system initiation/actuation or provide justification to show that existing analyses bound such an event.

C. Implementation

The Branch Technical Position, as specified in Section B, will be used in the review of all Preliminary Design Approval (PDA), Final Design Approval (FDA), Manufacturing License (ML), Operating License (OL), and Construction Permit (CP) applications involving plant designs incorporating pressurized water reactors. All aspects of the position will be applicable to all applications, including CP applications utilizing the replication option of the Commission's standardization program, that are docketed after March 14, 1978. All aspects of the position, with the exception of reasonable and justified deviations from IEEE 279 requirements, will be applicable to CP, OL, ML, PDA, and FDA applications docketed prior to March 14, 1978 but for which the licensing action has not been completed as of March 14, 1978. Holders of appropriate PDA's will be informed by letter that all aspects of the position with the exception of IEEE 279 will be applicable to their approved standard designs and that such designs should be modified, as necessary, to conform to the position. Staff approval of proposed modifications can be applied for either by application by the PDA-holder on the PDA-docket or by each CP applicant referencing the standard design on its docket.

The following guidelines may be used, if necessary, to alleviate impacts on licensing schedules for plants involved in licensing proceedings nearing completion on March 14, 1978:

1. Those applicants issued an OL during the period between March 14, 1978 and a date 12 months thereafter may merely commit to meeting the position prior to OL issuance but shall, by license condition, be required to install all required staff-approved modifications prior to plant startup following the first scheduled refueling outage.

2. Those applicants issued an OL beyond March 14, 1979 shall install all required staff-approved modifications prior to initial plant startup.

3. Those applicants issued a CP, PDA, or ML during the period between March 14, 1978 and a date 6 months thereafter may merely commit to meeting the position but shall, by license condition, be required to amend the application, within 6 months of the date of issuance of the CP, PDA, or ML, to include a description of the proposed modifications and the bases for their design, and a request for staff approval.

4. Those applicants issued a CP, PDA, or ML after September 14, 1978 shall have staff approval of proposed modifications prior to issuance of the CP, PDA, or ML.

D. References

1. NUREG-0138, Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR, to NRR Staff.

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC): NUREG-0224

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7. AUTHOR(S): Gary G. Zech

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16. ABSTRACT (200 words or less)

In 1976, a technical issue was identified involving the inadvertent pressure transients in Pressurized Water Reactors (PWRs) during period of reactor shutdown and relatively low reactor temperatures. This report addresses the safety concerns involved, i.e., the margins to failure of the reactor pressure vessels from a fracture toughness standpoint, the steps taken to reduce the likelihood of pressure transients pending staff review of appropriate corrective action and the final design changes that licensees have proposed to preclude pressure transients that could exceed the Appendix G limits for the pressure vessels at operating PWR facilities. In addition, this report describes the Branch Technical Position (BTP) that has been developed as guidance for the design of plants undergoing a CP or OL review and the implementation of that BTP.

17. KEY WORDS AND DOCUMENT ANALYSIS

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17b. IDENTIFIERS/OPEN-ENDED TERMS

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D. Lawham

INDIANA & MICHIGAN POWER COMPANY
DONALD C. COOK NUCLEAR PLANT
P.O. Box 458, Bridgman, Michigan 49106

November 8, 1978

Mr. J.G. Keppler, Regional Director
Office of Inspection and Enforcement
United States Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Operating License DPR-74
Docket No. 50-316

Dear Mr. Keppler:

Pursuant to the requirements of the Appendix A Technical Specifications the following reports are submitted:

RO 78-034/03X-2
RO 78-078/03L-0
RO 78-079/03L-0.

Sincerely,

D.V. Shaller
Plant Manager

/bab

Dup

cc: J.E. Dolan
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UPDATE REPORT PREVIOUS REPORT DATE 7-11-78

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1 LICENSEE CODE (2)	M I D C C 2
LICENSE NUMBER (3)	0 0 0 0 0 0 0 0 0 0 0
LICENSE TYPE (4)	4 1 1 1 1
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CON'T 0 1 REPORT SOURCE (6)	L
DOCKET NUMBER (7)	0 5 0 0 0 3 1 6
EVENT DATE (8)	0 5 1 5 7 8
REPORT DATE (9)	1 1 0 8 7 8

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 DURING SURVEILLANCE TESTING ON 5-15-78 AND 6-20-78, THE HYDROGEN SKIMMER

0 3 RECIRCULATION FANS STARTING TIMES WERE FOUND TO BE 10 MINUTES-6 SECONDS FOR

0 4 2 HV-CEQ2 AND 10 MINUTES-5 SECONDS FOR 2-HV-CEQ1. THESE TIMES EXCEEDED THE

0 5 TIMES SPECIFIED BY T. S. 4.6.5.6.a AND T.S. TABLE 3.3-5 ITEM 7.d. PREVIOUS

0 6 OCCURRENCES INCLUDE: RO-050-315/75-15.

0 7

0 8

SYSTEM CODE (11)	CAUSE CODE (12)	CAUSE SUBCODE (13)	COMPONENT CODE (14)	COMP. SUBCODE (15)	VALVE SUBCODE (16)
S C	B	A	R E L A Y S	M	Z

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	7 8	—	0 3 4	/	0 3	X	—	2

ACTION TAKEN (18)	FUTURE ACTION (19)	EFFECT ON PLANT (20)	SHUTDOWN METHOD (21)	HOURS (22)	ATTACHMENT SUBMITTED (23)	NPRD-4 FORM SUB. (24)	PRIME COMP. SUPPLIER (25)	COMPONENT MANUFACTURER (26)
A	X	Z	Z	0 0 0 0	Y	N	A	A 1 0 9

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 INVESTIGATION REVEALED THAT THERE WERE TWO ITEMS AFFECTING THIS EVENT: THE 1

1 1 MINUTE T.S. TOLERANCE FOR UNIT 2, AND THE AGASTATS INSTALLED ARE OF A 0-60 MINUTE

1 2 RANGE, WHICH IS AN IMPROPER RANGE FOR THE 9-10 MINUTE SETTING. PER THE MANUFACTURER,

1 3 AGASTATS WITH A TIME SETTING OF GREATER THAN 200 SECONDS HAVE A REPEAT ACCURACY OF

1 4 APPROXIMATELY 10 PER CENT, WITH THE FIRST DELAY BEING APPROXIMATELY 20 PER CENT

1 5	FACILITY STATUS (28): E	% POWER (29): 0 7 0	OTHER STATUS (30): NA	METHOD OF DISCOVERY (31): B	DISCOVERY DESCRIPTION (32): SURVEILLANCE TEST
1 6	ACTIVITY RELEASED (33): Z	CONTENT OF RELEASE (34): Z	AMOUNT OF ACTIVITY (35): NA		LOCATION OF RELEASE (36): NA
1 7	PERSONNEL EXPOSURES NUMBER (37): 0 0 0	TYPE (38): Z	DESCRIPTION (39): NA		
1 8	PERSONNEL INJURIES NUMBER (40): 0 0 0	DESCRIPTION (41): NA			
1 9	LOSS OF OR DAMAGE TO FACILITY TYPE (42): Z	DESCRIPTION (43): NA			
2 0	PUBLICITY ISSUED (44): N	DESCRIPTION (45): NA			NRC USE ONLY

Dup

7811240218

NAME OF PREPARER Jack Rischling PHONE 616-465-5901

ATTACHMENT TO LER # 78-034/03X-2

SUPPLEMENT TO CAUSE DESCRIPTION

LONGER THAN SUBSEQUENT TIME DELAYS DUE TO COIL WARM-UP. CORRECTIVE ACTIONS INCLUDE: 1) A T.S. CHANGE HAS BEEN SUBMITTED TO INCREASE THE PRESENT 1 MINUTE T.S. SPAN TO A 2 MINUTE SPAN, AND 2) A DESIGN CHANGE (12-1578) HAS BEEN INITIATED TO REPLACE THE 0-60 MINUTE AGASTATS WITH OTHERS HAVING A 0-10 MINUTE RANGE. THE DESIGN CHANGE WILL BE COMPLETED ON BOTH UNITS. THE COMBINATION OF THESE ACTIONS SHOULD PREVENT RECURRENCE.

A REVIEW OF THE OTHER AGASTATS USED TO MEET T.S. REQUIRED TIME DELAYS HAS BEEN MADE. THE REPEAT ACCURACY OF THE OTHER AGASTATS IS CONSISTENT WITH THE T.S. ALLOWANCE IN THAT THESE HAVE A SETTING OF LESS THAN 200 SECONDS.