UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 15, 1977

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS: REPORT NO. 6

Dear Dr. Hendrie:

The Advisory Committee on Reactor Safeguards has previously reported on the "Status of Generic Items Relating to Light-Water Reactors" in its letters of December 18, 1972, February 13, 1974, March 12, 1975, April 16, 1976 and February 24, 1977. Since the Committee limits its definition of generic items to those cited specifically in its letters pertaining to projects and related matters, the attached listing is not all-inclusive; the Nuclear Regulatory Commission Staff has additional generic items.

Groups I through ID of the attachments are a reiteration of the generic items considered resolved at the time the Committee issued its Report No. 5 on February 24, 1977. Group IE includes those items resolved since February 1977. Following each resolved item is a brief statement of the specific action that resulted in the resolution. Groups II through IID include items previously listed as those for which resolution on a generic basis is still pending. Group IIE includes those added in the present report. The ACRS and the NRC Staff will continue to consider the safety significance of items in Groups II through IIE on a case-by-case basis until generic resolution is reached. Formal actions, such as issuance of Regulations or Regulatory Guides, are anticipated for many of these items.

Owing to questions raised concerning the scope and intent of various generic issues, the Committee has incorporated into the attachments a brief description for all unresolved items cited in this report.

With regard to the status of generic issues, as they apply to each plant, the NRC Staff addresses the status of the pertinent issues in the applicable Safety Evaluation Report. The ACRS identifies those that it believes relevant in its reports on individual projects.

The ACRS has received requests concerning the priorities to be placed on the resolution of outstanding generic issues. Such priorities are shown in Table 1, attached.

"Resolved" as used in the Generic Items reports refers to the following: In some cases an item has been resolved in an administrative sense, recognizing that technical evaluation and satisfactory implementation are yet to be completed. Anticipated Transients Without Scram represents an example of this category. In other instances, the resolution has been accomplished in a narrow or specific sense, recognizing that further steps are desirable, as practical, or that different aspects of the problem require further investigation. Examples are the possibility of improved methods of locating leaks in the primary system, and of improved methods or augmented scope to inservice inspection of reactor pressure vessels.

Sincerely yours,

M. Bender

M. Bender
Chairman

Attachments:
(1) Group I; (2) Group IA; (3) Group IB; (4) Group IC; (5) Group ID; (6) Group IE; (7) Group II; (8) Group IIA; (9) Group IIB; (10) Group IIC; (11) Group IID; (12) Group IIE; and (13) Table 1, Priorities For Resolution of ACRS Generic Items.

GENERIC ITEMS

Group I - Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps: Covered by Regulatory Guide 1.1.

2. Emergency Power: Covered by Regulatory Guides 1.6, 1.9, and 1.32 and portions of IEEE-308 (1971).

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA): ACRS concurred in proposed Staff position, covered by NRC Standard Review Plan for Nuclear Power Plants.

4. Instrument Lines Penetrating Containment: Covered by Regulatory Guide 1.11 and Supplement.

5. Strong Motion Seismic Instrumentation: Covered by Regulatory Guide 1.12.

6. Fuel Storage Pool Design Bases: Covered by Regulatory Guide 1.13.

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles: Covered by Regulatory Guide 1.14.

8. Protection Against Industrial Sabotage: Covered by Regulatory Guide 1.17.

9. Vibration Monitoring of Reactor Internals and Primary System: Covered by Regulatory Guide 1.20.

10. Inservice Inspection of Reactor Coolant Pressure Boundary: Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI and Regulatory Guide 1.65.

11. Quality Assurance During Design, Construction and Operation: Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971, Regulatory Guides 1.28, 1.33, 1.64, 1.70.6 and Proposed Standard ANS-3.2.

12. Inspection of BWR Steam Lines Beyond Isolation Valves: Covered by ASME BPV Code, Section XI.

13. Independent Check of Primary System Stress Analysis: Covered by ASME BPV Code, Section III.

14. Operational Stability of Jet Pumps: Test and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns.

Group I Continued

15. Pressure Vessel Surveillance of Fluence and NDT Shift: Covered by 10 CFR 50, Appendix A and Appendix H; and ASTM Standard E-185.

16. Nil Ductility Properties of Pressure Vessel Materials: Covered by 10 CFR 50, Appendix A and Appendix G; ASME BPV Code, Section III; "Report on the Integrity of Reactor Vessels for Light-Water Power Reactors," (WASH-1285) by the Advisory Committee on Reactor Safeguards dated January 1974.

17. Operation of Reactor With Less Than All Loops In Service: Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent. Cited in Standard Review Plan Appendix 7-A, Branch Technical Position EICSB 12.

18. Criteria for Preoperational Testing: Covered by Regulatory Guide 1.68.

19. Diesel Fuel Capacity: Covered by ACRS-Regulatory Staff position requiring 7 days fuel (Standard Review Plan 9.5.4).

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends: Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences.

21. Operating One Plant While Other(s) is/are Under Construction: Specific requirements have been established by ACRS-Regulatory Staff. Covered in Regulatory Guide 1.17, 1.70 Section 13.6.2; 1.101; ANSI N 18.17 and Standard Review Plan 13.3 Appendix A and 13.6.

22. Seismic Design of Steam Lines: Covered by Regulatory Guide 1.29.

23. Quality Group Classifications for Pressure Retaining Components: Covered by Regulatory Guide 1.26.

24. Ultimate Heat Sink: Covered by Regulatory Guide 1.27.

25. Instrumentation to Detect Stresses in Containment Walls: Covered by Regulatory Guide 1.18.

Group IA - Generic Items Resolved Since December 18, 1972

1. Use of Furnace Sensitized Stainless Steel: Covered by Regulatory Guide 1.44.

2. Primary System Detection and Location of Leaks: Covered by Regulatory Guide 1.45.

3. Protection Against Pipe Whip: Covered by Regulatory Guide 1.46.

4. Anticipated Transients Without Scram: Covered by Regulatory Position Document, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, September 1973.

5. ECCS Capability of Current and Older Plants: Covered by Rulemaking as a general policy decision, although acceptable detailed implementation remains to be developed. Docket RM-50-1, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors," December 28, 1973.

Group IB - Generic Items Resolved Since February 13, 1974

1. Positive Moderator Coefficient: PWRs presently have or expect to have zero or negative coefficients. Where some Technical Specifications allow a slightly positive coefficient, the accident and stability analyses take this into account. Burnable poison provisions have been designed into PWRs to reduce otherwise excessive positive coefficients to allowable values.

2. Fixed Incore Detectors on High Power PWRs: Fixed incore detectors are not required for PWRs since reviews of potential power distribution anomalies have not revealed a clear need for continuous incore monitoring.

3. Performance of Critical Components (pumps, cables, etc.) in post-LOCA Environment: Qualification requirements of critical components are now covered by Regulatory Guides 1.40, 1.63, 1.73 and 1.89 and IEEE Standards 382-1972, 383-1974, 317-1972, 323-1974.

4. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments: On designs prior to GE Mark III containment, resolution lies in surveillance and testing of vacuum relief valves. For Mark III containments, an additional requirement is that the design be capable of accommodating a bypass equivalent to one square foot for a given flow condition.

5. Emergency Power for Two or More Reactors at the Same Site: Resolved by issue of Regulatory Guide 1.81.

6. Effluents from Light-Water-Cooled-Nuclear Power Reactors: Resolved by issue of Appendix I to 10 CFR 50.

7. Control Rod Ejection Accident: Resolved for PWRs by Regulatory Guide 1.77.

Group IC - Generic Items Resolved Since March 12, 1975

1. Main Steam Isolation Valve Leakage of BWR's: Covered by Regulatory Guide 1.96.

2. Fuel Densification: Covered by 10 CFR 50 Appendix K plus case-by-case review of vendor fuel models.

3. Rod Sequence Control Systems: Covered by NRC Staff Review and Approval of NEDO-10527 and Presentation to ACRS.

4. Seismic Category I Requirements for Auxiliary Systems: Covered by Regulatory Guides 1.26 and 1.29.

Group ID - Generic Items Resolved Since April 16, 1976

1. Instruments to Detect (limited) Fuel Failures - NRC document, "Fuel Failure Detection in Operating Reactors," B. L. Siegel and H. H. Hagen, June, 1976 resolves issue for limited fuel failures, but not for severe failures (See II-4).

2. "Instrumentation to Follow the Course of an Accident" Regulatory Guide 1.97 Revision 1 resolves ACRS concerns.

3. Pressure in Containment Following LOCA - NRC document, "Containment Subcompartment Analysis" September 1976.

4. Fire Protection. Resolved by Branch Technical Position 9.5.1, and Regulatory Guide 1.120.

Group IE - Generic Items Resolved Since February 24, 1977

1. Control Rod Drop Accident (BWRs): Resolved through NRC review and documentation establishing such an event as not having severe consequences (Memorandum for M. Bender, Chairman ACRS, from Denwood F. Ross, Jr., Assistant Director for Reactor Safety, DSS, dated February 11, 1977.)

2. Rupture of High Pressure Lines Outside Containment: Resolved by positions in Standard Review Plan 3.6.1 and 3.6.2.

3. Isolation of Low Pressure from High Pressure Systems: Resolved by positions in Standard Review Plan 5.4.7.

Group II - Resolution Pending

1. Turbine Missiles: Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discusses the problems.*

2. Effective Operation of Containment Sprays in a LOCA: Extensive documentation in topical reports. Review and evaluation are required.

3. Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock: Regulatory Guide 1.2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data from the Heavy Section Steel Technology Program.

**4. Instruments to detect (severe) fuel failures - NRC document, "Fuel Failure Detection in Operating Reactors," B. L. Siegel and H. H. Hagen. Item ID covers limited failures. More work is required for the severe failure case to establish instrumentation criteria.

#5A. Monitoring for Loose Parts Inside the Reactor Pressure Vessel: State-of-the-Art results appear promising and some equipment has been installed.

#5B. Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel: Neutron Noise Analysis has been successful in detecting vibration of some components, however, additional work may be required concerning systems for detecting vibration in other components within the Reactor Pressure Vessel.

#6. Common Mode Failures: This heading covers a multiplicity of diverse components for which requirements should be established. Due to their diversity the ACRS feels that specific items should be separated into subsets under the general heading of common mode failures;

   6A - Reactor Scram Systems
   6B - Alternating Current Sources onsite and offsite
   6C - Direct Current Systems

   The above items are easily identified, other specific items may be added to this listing in the future.

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*Regulatory Guide is in preparation.

**Identified in the Committee's Report of April 16, 1976 as "Instruments to Detect Fuel Failures."

#These are a separation of items included under the same numbers in previous reports.

Group II Continued

7. Behavior of Reactor Fuel Under Abnormal Conditions: This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

8. BWR Recirculation Pump Overspeed During LOCA: Decision required by ACRS-NRC Staff.

9. The Advisability of Seismic Scram: Further studies required to establish need.

10. Emergency Core Cooling System Capability for Future Plants: Partially resolved by amendments to 10 CFR 50 [50.34(a)(4), 50.34(b)(4), 50.46, and Appendix K]. LOCA evaluation model complete. ACRS feels new cooling approaches should be explored.

II-1 - Turbine Missiles

Turbine failures for the past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discuses the problem.

Three issues require answers to resolve the turbine missile problem: (1) The first relates to the appropriate failure probability value; based on historical failures the probability is about 10^{-4}. Industry predicts a much lower failure probability based on improvements in materials and design. To date the ACRS has accepted the more conservative value; (2) The second issue is strongly dependent on turbine orientation with respect to critical safety structures. Strike probabilities from high angle missiles are acceptably low for single units and may be acceptable for multi-unit plants, depending on plant layout; however, lower angle missiles with non-optimum (tangential) turbine orientation have unacceptably high strike probabilities; (3) The third issue is one of penetration and damage of structures housed in the containment. The limited experimental data pertaining to penetration of large irregularly shaped missiles are not sufficient to determine structural response to impingement of turbine disc segments. Most missile penetration formulas are not relevant to this case. Some experiments with irregular missiles might resolve this issue, particularly for older plants with non-optimum turbine orientations.

II-2 - Effective Operation Of Containment Sprays In a LOCA

Review and evaluation are required of the variety of experiments which have been conducted on the effectiveness of various containment sprays on the removal and retention of airborne radioactive materials anticipated to be present within containment following a LOCA. Such review should consider adequacy of definition of the physical and chemical forms of the anticipated airborne radionuclides, and quality of evaluative tests of the removal efficiencies of various sprays under the conditions of temperature, pressure, and radiation doses expected to exist under LOCA conditions. A desirable extension might be analyses of the use of sprays containing chemicals (such as NaOH) which have the potential for damaging equipment within containment. Studies using other spray additives, such as hydrazine, have been conducted. If compounds, such as this, have distinct advantages, insofar as minimizing equipment damage in the event of inadvertent actuation, action should be taken to encourage their use.

II-3 - Possible Failure Of Pressure Vessel Post-LOCA By Thermal Shock

Earlier nuclear reactor pressure vessels subjected to fluences of $1\text{-}4 \times 10^{19}$ nvt, which are anticipated in the last 20 years of a 40-year life, may suffer severe radiation damage denoted by a pronounced shift in impact transition temperature at the inner surface. There will be a damage gradient which decreases sharply, so that the properties halfway through the wall are essentially those of the as-fabricated material. If a LOCA occurs near end-of-life, the injection of cold water on the region of degraded properties may initiate and propagate a crack because of high local stresses near the surface. Analytic procedures indicate the stresses drop rapidly with distance through the wall so the flaw should not propagate beyond some limiting point. The lack of experimental evidence and the relative width of the error band in the analytic results are such that some experiments are required to validate the analytic model. These are planned under the HSST program.

II-4 - Instruments To Detect (Severe) Fuel Failures

In the event of substantial fuel failure, including the possibility of fuel melt, large amounts of fission products could be rapidly released to the reactor coolant and possibly to the environment. Instrumentation capable of early warning and timely response may avert an incident becoming an accident.

Instrumentation related to such diagnostic purposes for limited fuel failure is being used on most power reactors. (See Item ID-1.) Further work is required to establish criteria for similar instrumentation for severe fuel failures.

II-5A - Monitoring For Loose Parts Inside The Pressure Vessel

Loose parts monitoring can provide early warning of potential mechanical problems or failures within the pressure vessel and throughout the primary coolant circuit. Reactor vendors have developed monitoring systems; however, requirements remain to be established.

II-5B - Monitoring For Excessive Vibration Inside The Reactor Pressure Vessel

Neutron noise analysis can detect vibration within specific components such as the core barrel. The detection of vibration in other reactor pressure vessel components is less well established.

II-6 - Non-Random Multiple Failures (Formerly "Common Mode Failure")

The term "common mode failures" has, in many instances, come to mean multiple failures of identical components exposed to identical or nearly identical conditions or environments, and the use of diversity in components has been proposed or required to avoid such failures. The concern of the ACRS is better expressed by the term "non-random multiple failures," which is intended to include not only the type of "common mode failure" discussed above but other types of multiple failures for which the consequences and probabilities cannot be predicted by application of the single-failure criterion. Examples include the use of the same sensors or components for both control and protection systems (a resolved matter); sequential multiple failures due to a "domino effect," and simultaneous multiple failures due to a single fault. Since designs usually do not knowingly incorporate features susceptible to such failures, techniques and criteria need to be developed to detect and avoid them in all systems important to safety. The following is a partial listing of systems whose common mode failure has been cited by the ACRS as a matter of safety concern:

II-6A - Scram Systems
II-6B - Alternating Current Sources
II-6C - Direct Current Sources

Other items may be added to this listing in the future.

II-7 - Behavior Of Reactor Fuel Under Abnormal Conditions

The behavior of reactor fuel under abnormal conditions is still considered unresolved due to the limited experimental data available. Partial melting of fuel assemblies due to flow blockage might lead to autocatalytic effects leading to more extensive fuel failure, pressure pulses, etc. Similar behavior might occur in the case of reactivity transients. The ACRS encourages analytic modeling but believes appropriate experimental data are necessary. It is anticipated that tests in the Power Burst Facility (PBF) should supply much of the required data.

II-8 - BWR Pump Overspeed During A LOCA

It is possible for a BWR recirculation pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components, with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliability of such protective measures as the use of decouplers between pump and motor is under study.

II-9 - The Advisability Of Seismic Scram

The ACRS has recommended that studies be made of techniques for seismic scram and of the potential safety advantages and potential disadvantages of prompt reactor scram in the event of strong seismic motion, say more than one-half the safe shutdown earthquake. Various suitable techniques have been identified and exist, but thus far only limited studies have been reported on the pros and cons of seismic scram. The principal potential advantage identified arises from the greatly improved coolability of a core in the unlikely event of a seismically induced LOCA, should scram precede the LOCA by several seconds. A principal reason given in opposition to seismic scram relates to a stated interest in keeping power stations on the line to provide power offsite should a severe earthquake occur.

II-10 - ECCS Capability For Future Plants

The ACRS has placed considerable emphasis on ECCS safety R&D so that the extent of the conservatism in the ECCS licensing requirements could be made more precise. With more experimental data a realistic and quantitative appraisal of ECC systems would lead to valid judgments on the changes in licensing which could be put on a firm basis.

Parallel approaches that seek to improve the reliability of ECC systems, to improve the monitoring of low power peaking, and to improve those fuel assembly designs which lower peaking factors, are encouraged. Further, changes in plant design which improve the reflooding of the reactor core should be sought and evaluated.

R&D efforts on analysis of core blowdown and reflood should be increased and combined with the results of the standard problems and the associated experiments. Improved analytical methods would provide a basis for optimized ECCS.

Group IIA - Resolution Pending - Items Since December 18, 1972

1. Ice Condenser Containments: Additional analyses are required to establish response during a LOCA, and to establish design margins.

2. PWR Pump Overspeed During a LOCA: Problem arises in similar manner to that of BWRs (Item 8 Group II).

3. Steam Generator Tube Leakage: Partially resolved by issuance of Regulatory Guide 1.83 which addresses the concern from a preventative point of view.

4. ACRS/NRC Periodic 10-Year Review of all Power Reactors: A more effective, continuous alternative approach to periodic reviews is being proposed. Pending ACRS review, this item is still considered unresolved.

IIA-1 - Ice Condenser Containments

The ice condenser containments have substantially smaller volume on the assumption that the ice will condense the steam during a LOCA, thus preventing system overpressurization. The rate of condensation is critical in the initial stages of the blowdown and is influenced by interaction of vapor with the ice. If the current analyses prove that the condensation model is suitably conservative, the problem may be resolved.

IIA-2 - PWR Pump Overspeed During a LOCA

It is possible for a PWR primary coolant pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components such as flywheels with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliability of such protective measures as electrical braking of the pump motor is under study.

IIA-3 - Steam Generator Tube Leakage

Normally the steam generator is not a critical component during a LOCA-ECCS. However, a special case exists where the steam generator tubes have been degraded due to corrosion, wastage, etc. If the shock loads imposed by the LOCA cause a critical number of tubes to fail, say by a double-ended (guillotine) break, the inflow from the secondary side can cause choking of flow during ECCS, preventing adequate cooling of the core. The critical number of tubes is relatively small. A position, such as one specifying a statistically significant level of nondestructive examination (NDE), might resolve this issue. The purpose of NDE would be to confirm that damage is not excessive; such examinations should minimize the possibility of catastrophic failure of a significant number of tubes.

IIA-4 - Periodic (10-Year) Review Of All Power Reactors

In its report of June 14, 1966, the ACRS recommended that periodic comprehensive reviews be conducted of operating licensed power reactors by the NRC Staff. These reviews would be preceded by a comprehensive report by the operator which evaluated the past experience and the safety of future operation of the plant.

The NRC Staff has maintained a continuing review of the safety of operating plants. In particular, as generic matters of potential safety significance arise, the appropriate operating reactors are asked to assess the relevance of the matter to each particular reactor. This is a necessary but different aspect of the continuing surveillance and review of the safety of operating reactors than was envisaged by the ACRS in its recommendation of June 1966.

The Committee continues to believe both approaches are desirable and awaits the development of a program of periodic comprehensive reviews.

Group IIB - Resolution Pending - Items Added Since February 13, 1974

*1. Computer Reactor Protection System: Systems should be qualified for reliability, particularly through in situ tests and under various environmental conditions, prior to use in reactor system.

2. Qualification of new fuel geometries: The 16x16 and 17x17 PWR, and 8x8 BWR fuels should undergo testing to meet Item 2 in Group IC and Item 7 in Group II.

3. Behavior of BWR Mark III Containments: Various aspects, including vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads and flow bypass should be resolved. This is an extension of Item 3 in Group ID.

4. Stress Corrosion Cracking in BWR Piping: Several failures have occurred in operating BWRs. The ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution and extensive programs are underway by industry, ERDA, and NRC.

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* Identified in the ACRS Report of April 16, 1976 as "Hybrid Reactor Protection System."

IIB-1 - Computer Reactor Protection Systems*

The proposed systems would contain some types of components and subsystems not previously used for reactor protection. It is necessary that the required system reliability, both during normal operation and under postulated abnormal conditions, be established through an appropriate combination of tests and analyses. While the issue originated with the B&W Hybrid concept it is equally applicable to the proposed CE and W computer reactor protection systems.

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* Identified in the ACRS Report of April 16, 1976 as "Hybrid Reactor Protection System."

IIB-2 - Qualification Of New Fuel Geometries

New fuels proposed for both BWRs and PWRs include the 8x8 (BWR), and 16x16 and 17x17 PWR fuels. The Committee recognizes that these fuels are intended to operate at power densities lower than earlier fuel designs. However, testing programs are considered necessary to establish their densification behavior (IC-2) as well as their behavior under abnormal conditions (II-7). Appropriate experimental programs should be developed dealing with flow blockage, behavior of fuel after partial melting, and fuel response under transient conditions. It is anticipated that the solution of this item will include a synthesis of Power Burst Facility data, experiments on earlier fuel types, behavior of fuel in commercial reactors, and confirmatory experiments on these fuel designs.

IIB-3 - Behavior Of BWR Mark III Containment

The BWR Mark III Containment differs in many respects from the Mark I and II designs. Various aspects such as vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads, and flow bypass must be evaluated and approved; ongoing experimental tests should develop much of the necessary data to confirm the conservatism in design.

IIB-4 - Stress Corrosion Cracking In BWR Piping

Several failures have occurred in operating BWRs. An ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution, and extensive programs are underway by Industry, ERDA and NRC.

The austenitic stainless steels are commonly used as piping material in many of the smaller BWR lines. A combination of weld sensitization, residual stresses, superposed loads, and oxygen equal to or greater than 0.2 ppm in the BWR coolant can lead to cracking, initiating on the inner surface and propagating through the wall. In most cases there will be a leak well before pipe failure so there is adequate warning; however, one can postulate a LOCA caused by a guillotine break with minimal prior warning. Current efforts are to minimize stress corrosion by using other materials.

Group IIC - Resolution Pending - Items Added Since March 12, 1975

1. Locking Out of ECCS Power Operated Valves: The Committee suggests that further attention be given to procedures involving locking out electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS.

2. Design Features to Control Sabotage: Attention should be given to aspects of design that could improve plant security.

*3A. Decontamination of Reactors: As experience is gained in reactor decontamination it should be factored into future plants to optimize control of radioactivity levels.

*3B. Decommissioning of Reactors: Specific plans should be developed, including definitive codes and standards to cover the ultimate decommissioning of plants.

4. Vessel Support Structures: Questions that have arisen concerning the loads on pressure vessel support structures due to certain postulated loss-of-coolant accidents should be resolved.

5. Water Hammer: Several cases of water slugging or water hammer have occurred in both PWRs and BWRs. Corrective measures should be taken to minimize such events.

6. Maintenance and Inspection of Plants: Provisions should be included in the design of future plants which anticipate the maintenance, inspection and operational needs of the plant throughout its service life.

7. Behavior of BWR Mark I Containments: Various aspects relevant to the BWR Mark I Containment should be resolved. Included are such items as relief valve restraint, control of local dynamic loads in the torus, vent clearing and establishment of torus water temperature limits during a LOCA. This is an extension of Item 3 in Group ID.

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*This is a further separation of the issues identified as IIC-3 in previous reports.

IIC-1 - Locking Out Of ECCS Power-Operated Valves

The physical locking out of electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS has been required, based on the assumption that a spurious electrical signal at an inopportune time could activate the valves to the adverse position; e.g., closed rather than open, or open rather than closed. While such an event has a finite probability another probability exists that the valves might be adversely positioned due to operator error.

The ACRS believes the matter should be studied using a systems approach, and considering such items as: (1) the evaluation of the probability of a spurious signal; (2) time required to reactivate the valve operator; (3) status of signal lights when the circuit breaker is open; (4) the possibility of locking out in an improper position due to a faulty indicator; (5) other designs with improved reliability without lock-out; (6) the advantages and disadvantages of corrective action by an alert operator in case of incorrect positioning vis-a-vis a system with power locked out.

IIC-2 - Design Features To Control Sabotage

Considerable attention has been devoted to control of industrial sabotage of nuclear power plants, particularly with regard to control of unauthorized access, and potential modes of sabotage by individuals or groups external to the operating organization. The ACRS believes that deliberate attention should be given to aspects of design that could improve plant security. With the emphasis being placed on standardized plant designs, it becomes especially important to introduce design measures that could protect against industrial sabotage, or mitigate the consequences thereof.

IIC-3A - Decontamination Of Reactors

The Committee believes that well developed plans, confirmed by appropriate experiments when necessary, should be available for the decontamination of primary reactor systems. At this time the information on full scale decontamination is limited. Examples of potential problems include such items as handling of decontamination solutions, potential hideout of radioactive products, enhanced corrosion and crud formation following decontamination, and the possible incompatibility of the different alloys in the pressure boundary with the decontamination solutions

IIC-3B - Decommissioning Of Reactors

Experience is limited with regard to decommissioning operations, and particularly with rules for dismantling and for mothballing. Definitive plans and standards should be developed covering such items as adequacy of action, problems in restitution of site, mutual responsibility of State and Federal Government, etc.

IIC-4 - Vessel Support Structures

A possible consequence of the instantaneous double-ended pipe break postulated to occur in certain large pipes of PWRs is the asymmetric loading of the reactor pressure vessel support structures. The magnitude and effects of such loads on the pressure vessel should be determined to establish if such loads adversely affect the predicted course of a LOCA. If analysis indicates that the results are unacceptable, appropriate corrective action should be taken. A potential effect is pressure vessel movement due to blowdown jet forces at the location of the rupture, transient differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

IIC-6 - Maintenance And Inspection Of Plants

Experience with older plants has verified that appropriate modifications in piping layout, with respect to walls and structures, type of insulation used, and weld joint design, to cite some obvious items, lead to improved maintenance, more reliable inservice inspections, and a better meeting of the operational needs of the plant throughout its service life, including decontamination and eventual decommissioning. An additional benefit is the reduction in personnel exposures in plants, making them more amenable to maintenance and inspection. Appropriate changes should be considered in future designs to meet these criteria.

IIC-7 - Behavior Of BWR Mark I Containments

Recent tests on the BWR Mark I Containment design revealed phenomena not anticipated on the basis of earlier tests where pressure loads were imposed by insertion of air. Specific problems somewhat comparable to those under review for the Mark III Containment, include relief valve discharge, pipe restraints in the torus, local dynamic loads on the torus, vent clearing, and influence of torus temperature on the LOCA.

Ongoing experiments are expected to develop the necessary data to confirm the adequacy of the existing design or to permit necessary modifications.

Group IID - Resolution Pending - Items added since April 16, 1976

1. Safety related interfaces between reactor island and balance-of-plant: The nuclear steam suppliers and some architect-engineers have submitted standardized plant designs. The Committee wishes to be sure that adequate attention is devoted to the interface between the reactor island and balance-of-plant to minimize problems during design and construction. The development and use of interdisciplinary system analyses is an aspect of this problem.

2. Assurance of continuous long-term capability of hermetic seals on instrumentation and electrical equipment: The integrity of seals during post-accident conditions may be critical in controlling such an accident. The Committee believes appropriate test and maintenance procedures should be developed to assure long-term reliability.

IID-1 - Safety Related Interfaces Between Reactor Island And Balance-Of-Plant

Questions have been raised concerning both standardized balance-of-plant and nuclear steam supply systems on the one hand and custom-designed site-related structures and components on the other hand. The depth of detail required at the stage of Preliminary Design Approval may not be adequate for construction approval. Procedures for instituting quality assurance programs covering design, procurement, construction, and startup with emphasis on timely and appropriate interdisciplinary system analyses to assure functional compatibility across the interfaces as well as for other systems, are necessary to assure functional compatibility for the postulated design basis accident conditions.

IID-2 - Assurance Of Continuous Long-Term Capability Of Hermetic Seals On Instrumentation And Electrical Equipment

Certain classes of instrumentation incorporate hermetic seals. When safety related components within containment must function during post-LOCA accident conditions, their operability is sensitive to the ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The damage processes may fall within Item IB-3, "Performance of Critical Components in Post-LOCA Environment"; however, a special case requiring evaluation has to do with personnel errors in the maintenance of such equipment since such errors could lead to the loss of effective hermetic seals.

Group IIE - Resolution Pending - Items Added Since February 24, 1977

1. Soil-Structure Interactions: Several matters related to soil-structure interaction and the appropriate seismic response spectrum for use at foundation levels of nuclear plants are under review and reevaluation.

IIE-1 - Soil Structure Interactions

Ongoing studies by the NRC and the industry are reviewing and re-evaluating matters related to soil-structure interaction and to the appropriate seismic response spectrum to be used at the foundation level of a nuclear power plant. These reviews may lead to a modification of current criteria used in the seismic design of foundation structures.

Table 1 - Priorities For Resolution Of ACRS Generic Items

GENERIC ITEMS		RELEVANT TO PWR	RELEVANT TO BWR	PRIORITY FOR RESOLUTION 1/ ACRS	PRIORITY FOR RESOLUTION 1/ NRC
II-1	Turbine Missiles	X	X	A	A-37*
II-2	Containment Sprays	X		B	C-10
II-3	Pressure Vessel Failure By Thermal Shock	X		A	A-11
II-4	Instruments to Detect (Severe) Fuel Failure	X	X	C	-
II-5A B	Excessive Vibration	X	X	B	-
II-5B A	Loose Parts Monitoring	X	X	B	B-60

1/ The ACRS has adopted and uses the categorizations developed by the NRC Staff and paraphrased below:

A. Those items judged to warrant priority attention in terms of manpower and/or funds to attain early resolution. These items include those, the resolution of which could (1) provide a significant increase in assurance of the health and safety of the public, or (2) have a significant impact upon the reactor licensing process.

B. Those items judged to be important in assuring the health and safety of the public but for which early resolution is not required or which have a lesser safety significance than Category A matters.
(Continued on Page 2)

*The numerals have no significance regarding the priority for resolution but are included to identify the NRC program plans related to each item.

GENERIC ITEMS		RELEVANT TO PWR	RELEVANT TO BWR	PRIORITY FOR RESOLUTION 1/ ACRS	PRIORITY FOR RESOLUTION 1/ NRC
II-6	Non-Random Multiple Failures	X	X	A	C-13
6A	Reactor Scram Systems	X	X	A	A-9
6B	Alternating Current Sources Onsite & Offsite	X	X	A	A-35 B-56 B-57
6C	Direct Current Systems	X	X	A	A-30
II-7	Behavior of Reactor Fuels Under Abnormal Conditions	X	X	A	B-22
II-8	BWR Recirculation Pump Overspeed During LOCA		X	B	B-68
II-9	Seismic Scram	X	X	C	D-1
II-10	ECCS Capability for Future Plants	X	X	A	D-2

Footnote 1 Continued:

C. Those items judged to have little direct or immediate safety significance but which could lead to improved understanding of particular technical issues or refinements in the licensing process.

D. Those items judged not to warrant the expenditure of manpower or funds because little or no importance to safety or improvements to the licensing process can be attributed to the item.

GENERIC ITEMS		RELEVANT TO PWR	BWR	PRIORITY FOR RESOLUTION 1/ ACRS	NRC
IIA-1	Ice Condenser Containments	X		B	B-54
IIA-2	PWR Pump Overspeed During a LOCA	X		B	B-68
IIA-3	Steam Generator Tube Leakage	X		A	A-3 A-4 A-5
IIA-4	ACRS/NRC Periodic 10-Year Review	X	X	C	Policy*
IIB-1	Computer Reactor Protection System	X		B	A-19
IIB-2	Qualification of New Fuel Geometry	X	X	C	B-22
IIB-3	BWR Mark III Containments		X	B	A-39 B-10
IIB-4	Stress Corrosion Cracking in BWR Piping		X	B	Policy*
IIC-1	Locking Out of ECCS Power Operated Valves	X	X	B	B-8

*The resolution of this item is to be effected through administrative means rather than by a specific technical activity.

GENERIC ITEMS		RELEVANT TO PWR	RELEVANT TO BWR	PRIORITY FOR RESOLUTION 1/ ACRS	PRIORITY FOR RESOLUTION 1/ NRC
IIC-2	Design Features to Control Sabotage	X	X	A	A-29
IIC-3A	Decontamination	X	X	B	A-15
IIC-3B	Decommissioning	X	X	B	B-64
IIC-4	Vessel Support Structures	X		B	A-2
IIC-5	Water Hammer	X	X	A	A-1
IIC-6	Maintenance and Inspection	X	X	B	B-34
IIC-7	BWR Mark I Containments		X	A	A-6 A-7 A-39
IID-1	Interfaces	X	X	A	Policy A-17
IID-2	Capability of Hermetic Seals	X	X	C	C-1
IIE-1	Soil-structure Interaction	X	X	C	A-40 A-41