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

July 9, 1980

MEMORANDUM FOR: ACRS Members

FROM: Richard Major, Staff Engineer

SUBJECT: STATUS OF GENERIC ITEMS - INFORMATION FOR JULY 9, 1980 PROCEDURES SUBCOMMITTEE MEETING

Enclosed are several documents dealing with generic items. The purpose of the discussion will be to reconsider how unresolved generic items are addressed by the Committee. Dr. Siess has proposed that the ACRS list be combined with the Staff's now separate list of generic items. If adopted, this alternative would abolish the ACRS list allowing the Committee to monitor the Staff's progress towards resolution of generic items. It also frees the Committee from the chore of producing its own generic items reports from time to time. The Committee could still review the resolution of each generic item as it is achieved. Dr. Siess also has several suggestions if it is decided to retain the separate ACRS generic items list. His June 5, 1980 memo on this subject is Enclosure 1.

Another item worthy of discussion is Dr. Shewmon's January 30, 1980 letter on, "When Is An Unresolved Safety Issue Resolved?" The specific theme of his letter centers on reactor vessel materials toughness, but the general sense of his letter is applicable to all generic items. For reference, the definition of "resolved" used in the latest ACRS generic items report (March 21, 1979) was:

> "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 of in-service inspection of reactor pressure vessels."

Dr. Shewmon's letter on, "when is an item resolved," is included as Enclosure 2.

Priorities deserve some attention. The Staff, starting in early 1978, compiled all their generic items in documents called, "Task Action Plans For Generic Activities." This document prioritized the generic items into four categories. The Staff's prioritization scheme was:

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ATTACHMENT 3

PRIORITY CATEGORY DEFINITIONS

Category A:

Those generic technical activities judged by the staff to warrant priority attention in terms of manpower and/or funds to attain early resolution. These matters 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.

Category B:

Those generic technical activities judged by the staff to be important in assuring the continued health and safety of the public but for which early resolution is not required or for which the staff perceives a lesser safety, safeguards or environmental significance than Category A matters.

Category C:

Those generic technical activities judged by the staff to have little direct or immediate safety, safeguards or environmental significance, but which could lead to improved staff understanding of particular technical issues or refinements in the licensing process.

Category D:

Those proposed generic technical activities judged by the staff not to warrant the expenditure of manpower or funds because little or no importance to the safety, environmental or safeguards aspects of nuclear reactors or to improving the licensing process can be attributed to the activity.

Task descriptions and schedules were only produced for the Category A items. In general, B, C, and D category items were merely listed.

The ACRS, in Report No. 6, did prioritize its unresolved generic items using the Staff's classification scheme. This practice was not continued when ACRS generic item Report No. 7 was issued.

The A, B, C, and D system of priorities was pre-TMI. Since TMI, resources which sould have been allocated to Category A generic items are being reprogramed to address issues in the TMI Action Plan. Only those generic items elevated to the rank of Unresolved Safety Issue (USI) will be receiving any attention until the Action Plan response starts to wind down. The current list of USIs is given in NUREG-0649. These items are:

TASK ACTION PLANS

Number	Title
Task A-1	Water Hammer......................
Task A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant System....
Tasks A-3 A-4 A-5	Westinghouse, Combustion Engineering, and Babcock and Wilcox Steam Generator Tube Integrity..........
Task A-7	Mark I Containment Long-Term Program (LTP).....................
Task A-8	Mark II Containment Pool Dynamic Loads.....................
Task A-9	ATWS.............................
Task A-10	BWR Nozzle Cracking ....
Task A-11	[illegible] materials Toughness.........................
Task A-17	Systems Interaction in Nuclear Power Plants......................
Task A-36	Control of Heavy Loads Near Spent Fuel........................
Task A-39	Determination of Safety-Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containment......................
Task A-40	Seismic Design Criteria.........

Since the USI system began, the following issues have been considered resolved by the Staff:

Table 2. NRC Documents Providing Staff's Resolution of "Unresolved Safety Issues"

Task No.	Document No. and Title	Document Date
A-6	NUREG-0408, "Mark I Containment Short-Term Program Safety Evaluation Report"	December 1977
A-12	NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports"	October 1979
A-24	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"	December 1979
A-26	NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors"	September 1978
A-31	Regulatory Guide 1.139, "Guidance for Residual Heat Removal"	May 1978
A-42	NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping"	October 1979

The Staff's definition of an Unresolved Safety Issue from the December 13, 1977 Amendment (PL 95-209) to the Energy Reorganization Act of 1974, Section 210 is: "An unresolved safety issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants affected.

The following table is a comparison of the Unresolved Safety Issues (the original 17, before the 5 1/2 items above were resolved) to the ACRS generic items from Report No. 7. Remember that in Report No. 7, those items with a number below #53 are items that are carried as "resolved" on the ACRS list.

TABLE

"[UNRE]SOLVED SAFETY ISSUES" - NUREG-0606 AQUA BOOK		ACRS GENERIC ITEMS - REPORT NO. 7 MARCH 21, 1979
	Water Hammer	74. Water Hammer
	Asymmetric Blowdown Loads	73. Vessel Support Structures
, A-4, A-5	Steam Generator Tube Integrity	64. Steam Generator Tube Leakage
	Mark I Long Term Program	75. Behavior of BWR Mark I Containments
3	Mark II Programs	No ACRS Generic Item on BWR Mark II Containments although two generic items on BWR Containments: 67. Behavior of BWR Mark III Containments 75. Behavior of BWR Mark I Containments Fluid Dynamics Subcommittee looks at BWR containment programs.
-9	ATWS	29. Anticipated Transients Without Scram
-10	BWR Feedwater Nozzle Cracking	68. Stress Corrosion Cracking in BWR Piping
-11	Reactor Vessel Materials Toughness	15. Pressure Vessel Surveillance of Fluence and NDT Shift 16. Nil-ductility Properties of Pressure Vessel Materials 55. Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock
A-12	Fracture Toughness of Steam Generator and Reactor Coolant and Pump Supports (This item came up during the North Anna licensing process - questions were raised as to the potential for lamillar tearing and low fracture toughness of the support materials used. Similar material used at other PWRs made the issue generic.	No one-to-one relation with ACRS generic items, although some relation to 73. Vessel Support Structures, however, item 73 is basically blowdown loads. This issue is being followed the the Metal Components Subcommittee

TABLE

...LVED SAFETY ISSUES" - NUREG-0606 AQUA BOOK	ACRS GENERIC ITEMS - REPORT NO. 7 MARCH 21, 1979
17 Systems Interaction in Nuclear Power Plants	This item is related to these generic items: 58. Non-random Multiple Failures 52. Safety-Related Interfaces between Reactor Island and Balance of Plant Plant Arrangements Subcommittee is directly following this item.
-24 Qualification of Class IE Safety Related Equipment	33. Performance of Critical Components (Pumps, Valves, etc.) in Post-LOCA Environment
-36 Control of Heavy Loads Near Spent Fuel Pool	6. Fuel Storage Pool Design Bases (There is an ACRS Subcommittee on Spent Fuel Storage Pool Design which follows this issue.)
...-39 Determination of Safety Relief Valves (SRV) Dynamics Loads and Temperature Limits for BWR Containment	67. Behavior of BWR Mark III Containments 75. Behavior of BWR Mark I Containments
A-40 Seismic Design Criteria Short-term Program	77. Soil Structure Interaction 61. Advisability of Seismic Scram 22. Seismic Design Steam Line
A-42 Pipe Cracks in Boiling Water Reactors	68. Stress Corrosion Cracking in BWR Piping
A-43 Containment Emergency Sump Performance	1. NPSH FOR ECCS Pumps
A-44. Station Blackout	2. Emergency Power 35. Emergency Power for Two or More Reactors at the Same Site

During the 233rd ACRS meeting (September 1979) the Committee reviewed the items listed in its Generic Items Report No. 7. Action was recommended on a number of items in both the resolved and unresolved categories. Subcommittee assignments were made so that an appropriate subcommittee could monitor the progress on a particular generic item.

The following table is a list of ACRS generic items which includes those reoslved items on which another look was suggested and the list of ACRS unresolved generic items. Where possible, I have updated and collected information on each item to show ACRS priority (from Rpt. No. 6) and any further action on a particular item. I have attempted to correlate the ACRS generic items list to the Staff's list of generic items.

1. NPSH for ECCS Pumps -- Reactor Operations SC.
   This is covered by Reg. Guide 1.1. The Reactor Operations Subcommittee could review this with the Division of Operating Reactors to determine whether all plants are in compliance. Potential for vortex problems should be considered.

   No additional ACRS Subcommittee action yet.

   Related Staff Item: A-43, Containment Emergency Sump Performance
   B-18, Vortex Suppression Requirments for Containment Sumps

   Staff Priority: A&B

   ACRS Priority:* Not specified

2. Emergency Power -- Joint Power and Electrical Systems and Reactor Operations Subcommittees

   Reg. Guide 1.6, 1.9, and 1.32 in conjunction with portions of IEEE-308 (1971) covers this matter. However, the question concerning loss of DC power or combined loss-of-offsite- and -onsite-AC power are presently of concern from a risk standpoint. The Power and Electrical Systems Subcommittee and the Reactor Operations Subcommittee should jointly review the status of emergency power requirements. The question of grandfathering older plants should also be considered regarding emergency power.

   No additional ACRS Subcommittee action yet.

   Related Staff Item: A-30, Adequacy of Safety-Related DC Power Supplies
   A-44, Station Blackout
   B-56, Diesel Reliability

   Staff Priority: A&B

   ACRS Priority: Not specified

* ACRS priorities were specified for just unresolved items in Report No. 6. (items beyond # 53)

3. Hydrogen Control After Loss-of-Coolant Accident -- TMI-2 Implications Subcommittee - Class 9 Accident Subcommittee now addressing, Plant-specific attention (Sequoyah)

   The present hydrogen control requirements are based primarily on the concern for hydrogen build-up in containment following a LOCA where the fuel temperature rises to the level at which zirconium-water reaction proceeds rapidly, leading to hydrogen generation sufficient to cause burning or explosion. The Reg. Guide limits in 1.97 presume an oxidiation rate that is a function of surface area and a termination point related to ECCS capability. The Three Mile Island Accident displayed high hydrogen generation because the ECCS was not permitted to do its job. The TMI-2 Implication Subcommittee should recommend actions for reevaluation of this generic item.

   Related Staff Item: Action Plan, Rulemaking addressing Class 9 Accidents B-14, Study of Hydrogen Mixing Capability in Containment Post-LOCA

   Staff Priority: A&B

   ACRS Priority: Not specified

4. Instrument Lines Penetrating Containment -- No action required

   Reg. Guide 1.11 and its Supplement adequately cover this point and no further action is needed.

5. Strong Motion Seismic Instrumentation -- No action required

   This is covered in Reg. Guide 1.12 and there does not appear to be the need for further action.

6. Fuel Storage Pool Design Bases -- Safeguards & Security Subcommittee and Plant Arrangements Subcommittee have met on this item.

   This is covered by Reg. Guide 1.13, however, the committee has frequently raised questions concerning the location of the fuel storage pool because of industrial sabotage questions. The Plant Arrangements and Safeguards and Security Subcommittee should review this matter and make recommendations to the full committee concerning the need for further action, especially regarding the location of the fuel pool with respect to grade.

   No change in items status.

   Related Staff Item: A-29, Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

   Staff Priority: A

   ACRS Priority: Not specified

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles -- No action required

   This is covered by Reg. Guide 1.14 supported by knowledge developed in the Safety Research Program. Based on the staff evaluation of the R&D work, this matter appears to be adequately covered.

8. Protection Against Industrial Sabotage -- Joint Plant Arrangements and Safeguards and Security Subcommittees met on this item.

   Reg. Guide 1.17 covers this matter, but since the issuance of Reg. Guide 1.17, committee letters have continued to raise questions about the adequacy of industrial sabotage protection. This matter should be addressed by joint effort of the Plant Arrangements Subcommittee and the Safeguards and Security Subcommittee.
   No change in item status.

   Related Staff Item: A-29, Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

   Staff Priority: A

   ACRS Priority: Not specified

9. Vibration Monitoring of Reactor Internals and Primary System -- No action required

   Reg. Guide 1.20 covers these matters and the recent review of the loose parts monitoring technology indicated that current interpretations of Reg. Guide 1.20 by the NRC Staff serve the situation adequately.

   The ACRS "Review of LERs", NUREG-0572 discusses fialures due to flow-induced vibration in appendix D-19.

   Related Staff Item. B-73, Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel
   C-12, Primary System Vibration Assessment

   Staff Priority: B&C

   ACRS Priority: Not specified

10. Inservice Inspection of Reactor Coolant Pressure Boundary -- Metal Components Subcommittee

This is covered by Section XI of the ASME Boiler and Pressure Vessel Code and Reg. Guide 1.,65 along with other modifications of the Code recently evaluated by the Reg. Guide Subcommittee. Questions remain as a result of Duane Arnold piping problems and various PWR feedwater line problems. This matter is under active review by the Metal Components Subcommittee and an update of recommendations concerning this matter should be provided from that Subcommittee.

Related Staff Item: A-3, 4, 5, W, CE, B&W, Steam Generator Tube Integrity
A-10, BWR Nozzle Cracking
A-11, Reactor Vessel Materials Toughness
A-14, Flaw Detection

Staff Priority: A

ACRS Priority: Not specified

11. Quality Assurance During Design, Construction, and Operation -- Reactor Operations Subcommittee

Requirements of 10 CFR 50, Appendix B, ASME Boiler and Presure Vessel Code, Section III, ANSI-N45.2 (1971), Reg. Guides 1.28, 1.33, 1.64, 1.70.6, and proposed standard ANS-3.2, all address these matters. The NRC staff should be asked for a collective appraisal concerning the coverage in these documents. The Reactor Operations Subcommittee should then reassess the adequacy of this coverage. Recent experiences at Three Mile Island and concerns about the seismic restraints justify a determination concerning QA control adequacy.

No additional ACRS Subcommittee action yet

Related Staff Item: None

Staff Priority: None

ACRS Priority: Not specified

12. Inspection of BWR Steam Lines Beyond Isolation Valves -- No action required

This adequately covered by ASME Boiler and Pressure Vessel Code, Section XI.

13. Independent Check of Primary System Stress Analysis -- No action required

This is adequately covered by ASME Boiler and Pressure Vessel Code, Section III.

14. Operational Stability of Jet Pumps -- No action required

The work on Dresden-2 and -3 installations and other operating experiences adequately satisfy the ACRS concern.

15. Pressure Vessel Surveillance of Fluence and NDT Shift -- Metal Components Subcommittee (Review together with Item 16)

This is covered by 10 CFR 50, Appendix A and ASTM Standard E-185. The NRC staff has recently recommended and the ACRS has approved the use of surveillance specimens from multiple reactor installations as satisfying the intent of the regulatory requirements. 10 CFR 50 will be modified accordingly under rulemaking proceedings.

16. Nil-ductility Properties of Pressure Vessel Materials -- Metal Components Subcommittee has met on these issues.

This is covered by 10 CFR 50, Appendix A and Appendix G, ASME Boiler and Pressure Vessel Code, Section III and was addressed in the ACRS 1970 Report on Light Water Reactor Pressure Vessel Integrity, WASH-1285. The situation still appears to be adequate from a safety standpoint, but the ACRS Metal Components Subcommittee should reexamine the nil-ductility problem as a function of temperature for some of the older vessels nearing the end of their specified life and any new questions that have arisen concerning the upper shelf properties of materials.

Related Staff Item: A-11, Reactor Vessel Materials Toughness
B-26, Reactor Vessel Pressure Transient Protection (Overpressure Protection)

Staff Priority: A

ACRS Priority: Not specified

17. Operation of Reactor With Less Than All Loops in Service -- No action required

Standard Review Plan, Appendix 7A and Branch Technical Position EICSB-12 cover this matter adequately.

18. Criteria for Preoperational Testing -- Reactor Operations Subcommittee

This is covered by the most recent revision to Reg. Guide 1.68 but the uniformity of the preoperational testing program at various sites is unclear. The present concerns about plant operating skills suggests a need to have the Reactor Operations Subcommittee examine the nature of preoperational test programs in order to determine whether the requirements of Reg. Guide 1.68 really satisfy regulatory needs.

Related Staff Item: Action Plan I.G, Preoperational & Low Power Testing

Staff Priority: 2 (can be deferred up to one year)

ACRS Priority: Not specified

19. Diesel Fuel Capacity -- No action required

Standard Review Plan 9.4 covers this matter adequately.

20. Capability of Biological Shield Withstanding Double-ended Pipe Break at Safe Ends --

Regulatory review practices cover this matter adequately. It may be appropriate to have one of the ACRS consultants examine a few examples of the design treatment to ascertain whether the approach is based on correct safety criteria.

Dr. Zudans has examined this item and concludes that "although a number of shortcomings have been identified in the analyses, I judge that the adequacy of the sacrificial shield to withstand the specified loadings has been demonstrated by the analysis."

21. Operation of One Plant While Others are Under Construction -- Have Fellows review

The coverage under Reg. Guide 1.17; 1.70; Sections 13.62; 1.101; ANSI N-18, 1.7; and Standard Review Plan 13.3, Appendix A; and 13.6 are all relevant to this question. One of the ACRS Fellows should be asked to review these documents to determine whether they treat all of the ACRS questions that have been raised and whether any other matters deserve attention. The potential for a Three Mile Island type of accident is particularly relevant to this matter. LERs should also be reviewed. Report by J. Bickel to M. Bender dtd. 10/3/79 - major problem is security background checks and maintenance procedures for the operating plants.

22. Seismic Design of Steam Line -- Combination of Dynamic Loads SC.

This is covered by Reg. Guide 1.29 but the Combination of Dynamic Loads Subcommittee is reexamining the design bases. Recommended changes to Reg. Guide 1.29 may evolve from the combination of dynamic loads review.

Related Staff Item: A-40, Seismic Design Criteria
B-24, Seismic Qualifications of Electrical and Mechanical Components

Staff Priority: A

ACRS Priority: Not specified

23. Quality Group Classification for Pressure Retaining Components --

Plant Arrangements SC (include analysis of secondary system (e.g. steam lines piping failures). Reg. Guide 1.26 covers this matter but questions arising from the interactive effect of non-safety grade equipment as seen in the Three Mile Island-2 accident may lead to changes in these classifications. The Plant Arrangement Subcommittee should review this matter.

Related Staff Item: A-17, Systems Interaction in Nuclear Power Plants
Action Plan, I.F, Quality Assurance (B)

Staff Priority: A

ACRS Priority: Not specified

24. Ultimate Heat Sink -- No action required

Reg. Guide 1.27 covers this matter satisfactorily.

25. Instrumentation to Detect Stresses in Containment Walls -- No action required

Reg. Guide 1.18 covers this matter but there are some controversial questions associated with grouted tendons. Current Staff interpretations provide adequate controls.

26. Use of Furnace Sensitized Stainless Steel --

Reg. Guide 1.44 may need an update to better define "rapid-cooling". Bring to NRC Staffs attention but do not reopen consideration of Reg. Guide.

Needs revision to better define "rapid-cooling"

27. Primary System Detection and Location of Leaks -- reassigned to Metal Components Subcommittee

Reg. Guide 1.45 addresses this matter and experiences at Duane Arnold and other plants indicate that the procedures are suitable. Exploring the use of TV cameras to find leaks could be explored.

Related Staff Item: Could not find one, since R.G. 1.45

Staff Priority: None

ACRS Priority: Not specified

28. Protection Against Pipewhip -- Combination of Dynamic Loads Subcommittee

This is covered by Reg. Guide 1.46 but the Combination of Dynamic Loads Subcommittee will be reviewing these requirements as they are being influenced by combined load considerations. The question of whether the more elaborate requirements of combined loads introduce undesirable requirements should be examined.

Related Staff Item: B-6, Loads, Load Combinations, Stress Limits
B-16, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

Staff Priority: B

ACRS Priority: Not specified

29. Anticipated Transients Without Scram -- ATWS Subcommittee

Although this matter was covered by WASH-1270, issued in September 1973, the NRC has not yet established an implementation plan nor are the technical bases fully established. The ACRS ATWS Subcommittee should continue to review this matter and recommend actions to the full Committee.

Related Staff Item: A-9, ATWS
NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," March 1980

Staff Priority: A (Unresolved Safety Issue)

ACRS Priority: Letter on ATWS dtd 4/16/80

30. ECCS Capability of Current and Older Plants (small LOCA needs attention) -- ECCS Subcommittee

The status should be updated through review by the ECCS Subcommittee, possibly with some support form the Plant Arrangements Subcommittee. Concerns about the oldest installations, e.g., Indian Point 1, have been resolved by NRC licensing action over the past several years.

Related Staff Item: B-4, ECCS Reliability
B-18, Vortex Suppression Requirements for Containment Sumps
B-61, Allowable ECCS Equipment Outage Periods
B-69, ECCS Leakage Ex-containment

Staff Priority: B

ACRS Priority: Not specified

31. Positive Moderator Coefficient -- No action required

PWR's presently follow a practice that satisfies the concerns about moderator coefficients under normal conditions. The transient questions associated with LOCA and the uncertainties associated with ATWS effects are under review.

Note: In the ACRS letter on LERs, boron addition systems received attention under D-XXIII Inadvertent Activation of Safety Injection in PWRs. Concerns identified include thermal stresses on nozzles and appropriate operator response concerning early termination of these events.

Lesson learned from TMI bear on this.

32. Fixed In-Core Detectors on High-Power PWR's -- No action required

In-core monitoring needs to be re-reviewed in the light of TMI-2 experiance, but it is unlikely that fixed in-core detector needs would change because of such a review. This item seems O.K.

33. Performance of Critical Components (Pumps, Valves, etc.) in Post LOCA Environment -- Power and Electrical Systems Ssubcommittee

The qualification requirements in Reg. Guide 1.40, 1.63, 1.73, 1.89, and IEEE Standards 382 (1972), 383 (1974), 317 (1972), and 323 (1974), all address these matters. However, the experience at Three Mile Island-2 might alter some of these requirements. The Power and Electrical Systems Subcommittee should examine the need for new requirements.

Related Staff Item: A-21, Main Steam Line Break Inside Containment-Evaluation of Environmental Conditions for Equipment Qualification

A-24, Qualification of Class IE Safety-Related Equipment

Staff Priority: A

ACRS Priority: Not specified

34. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containment -- ACRS Fellow

The NRC staff requirements for Mark II and Mark III containments address these matters adequately. A review of actual experience with Mark II design might be useful for updating our knowledge. One of the ACRS Fellows might be assigned to make such a review. LERs should also be considered. G. Young report to M. Bender 9/24/79. Most failures occurred during testing.

Related Staff Item: A-39 Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containments

Staff Priority: A

ACRS Priority: Not Specified, however, G. Young report to M. Bender on September 24, 1979

35. Emergency Power for Two or More Reactors at the Same Site -- Power and Electrical Systems Subcommittee

Reg. Guide 1.81 covers this matter. Shared diesels at older plants should be examined. Will consider all shared systems and components.

Related Staff Item: (Nothing found on shared diesels)
A-30 Adequacy of Safety-Related DC Power Supplies
A-35 Adequacy of Offsite Power Systems

Staff Priority: A

ACRS Priority: Not specified

36. Effluents from Light Water cooled Nuclear Power Reactors -- No action required

This environmental question is resolved by the requirements of Appendix I of 10 CFR 50.

37. Control Rod Ejection Accident -- No action required

This is covered adequately by the requirements of Reg. Guide 1.77.

38. Main Steam Isolation Valve Leakage of PWR -- No action required

Reg. Guide 1.96 covers this adequately.

39. Fuel Densification -- No action required

Requirements of 10 CFR 50, Appendix K and case-by-case review of vendor fuel models covers this matter satisfactorily.

40. Rod Sequence Control Systems -- No action required

The practices of the NRC staff, including those established by GE NEDO 10527 cover this matter satisfactorily.

Note: This matter received attention in the ACRS LER report. Item D-1 Separation of Control Rod from Its Drive and BWR High Rod Worth Events. Concern identified is a short-period scram less than 5 seconds.

41. Seismic Category 1 Requirements for Auxilary Systems -- Combination of Dynamic Loads Subcommittee

This is covered by Reg. Guide 1.26 and 1.29, but may be reexamined if new questions of interpretation arise out of a Combination of Dynamic Loads Subcommittee review.

The Diablo Canyon seismic interaction of non safety equipment on safety related equipment study addresses in part.

Related Staff Item: A-40, Seismic Design Criteria - Short Term Program
B-24, Seismic Qualification of Electrical and Mechanical Components

Staff Priority: A & B

ACRS Priority: Not specified

42. Instruments to Detect Limited Fuel Failures -- Joint Power and Electrical Systems and Reactor Fuel Subcommittees

Although this has been addressed in an NRC document entitled "Fuel Failure Detection in Operating Reactors" by Siegal and Hagan, June 1976, the experience of Three Mile Island warrants further review of this matter. The Power and Electrical Systems Subcommittee should evaluate this question in combination with the Reactor Fuel Subcommittee. Call to attention of NRC Staff. Resolved. Will keep under surveillance.

Related Staff Item: None, since "Fuel Failure Detection in Operating Plants"

Staff Priority: None

ACRS Priority: None

43. Instrumentation to Follow the Course of an Accident -- Power and Electrical Systems Subcommittee

Reg. Guide 1.97, Revision 1, addresses this matter but the requirements have never been recognized. The Power and Electrical Systems Subcommittee should reexamine the requirements of 1.97 to determine whether they realistically define the need and whether a more definitive Reg. Guide should be provided based on TMI-2 experience.

Related Staff Item: Reg. Guide 1.97 for final Review by ACRS in August 1980, it is receiving much attention from the Committee.

44. Pressure in Containment Following LOCA's -- TMI-2 Implications Subcommittee

TMI-2 experience suggests the need to review this matter for low pressure containment. Will be considered during review of long-term lessons learned report

Action Plan is addressing this item. IP & Zion Studies are applicable.

45. Fire Protection -- Fire Protection Subcommittee

Branch Technical Position 9.5.1 provides a satisfactory review process. Reg. Guide 1.120 whose development has been suspended because of ACRS concerns should now be reinitiated with attention being addressed to the requirements found acceptable for current Standard Plant Designs.

New fire protection rule under ACRS Subcommittee review on July 9, 1980.

46. Control Rod Drop Accidents (BWRs) -- Core Performance Subcommittee

This had been adequtely covered by NRC review practices. However, LERs have raised questions, short period scram concern raised by E. Epler. Low probability event.

In the ACRS LER report Item D-1, "Separation of Control Rod From Its Drive and BWR High Rod Worth Events" discusses BWR rod drops.

Related Staff Item: D-3, Control Rod Drop Accident (BWRs)

Staff Priority: D

ACRS Priority: Not specified

47. Rupture of High Pressure Lines Outside Containment -- No action required

Standard Review Plan Sections 3.6.1 and 3.6.2 cover this matter adequately.

48. Isolation of Low Pressure from High Pressure Systems -- Reactor Operations SC.

Standard Review Plan 5.4.7 addresses this matter. A few LERs have been identified which may have reopened concern for this question.

In the ACRS LER report Item D-1X, "Leakage Between Interconnected Fluid Systems" highlights the concern. Suggests adequacy of instrumentation to monitor the neutrol zone should be reevaluated.

Related Staff Item: B-63, Neutral Isolation of Low Pressure Systems Connected to RCPB

Staff Priority: B

ACRS Priority: Note Specified

49. Monitoring for Loose Parts Inside the Reactor Pressure Vessel -- No action required

Reg. Guide 1.133 covers this matter.

However, Related Staff Item: B-60, Loose Parts Monitoring System
B-73, Monitoring for Excessive Vibration Inside the RPV
C-12, Primary System Vibration Assessment

50. Qualification of New Fuel Geometry -- No action required

Standard Review Plan 4.2, Revision 1, satisfies ACRS interest.

51. Maintenance and Inspection of Plants -- Reactor Operations SC.

The ACRS originally accepted the postion that recent attention f the staff to these matters was adequate. The experience at TMI-2 reopens the question. The Reactor Operations Subcommittee should determine whether this matter needs additional effort.

Related Staff Item: B-36, Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature System and for Normal Ventilation Systems

B-47, Inservice Inspection Criteria for Supports and Bolting of Class 1, 2, 3 and MC Components

B-49, Inservice Inspection Criteria for Containment

B-50, Requirements for Post-OBE Inspection Maintenance and Inspection is also receiving attention through the lessons learned requirements and the Action Plan for such areas as auxiliary feedwater systems

Staff Priority: B (maybe A through Action Plan attention)

ACRS Priority: Not Specified

52. Safety Related Interfaces Between Reactor Island and Balance of Plant -- Plant Arrangements Subcommittee

Standard Review Plan 1.8 covers the matter in an administrative sense, but systems interaction questions from the TMI-2 accident experience warrent reexamination by the Plant Arrangements Subcommittee.

Related Staff Item: A-17, Systems Interaction in Nuclear Power Plants

Staff Priority: A

ACRS Priority: Not Specified

53. Turbine Missiles -- Discussed with S. H. Bush. Nothing new to update.

Particular attention given to older plants.

Related Staff Item: A-37, Turbine Missiles

Staff Priority: A

ACRS Priority: A

54. Effective Operation of containment Sprays in a LOCA -- Generic Items Subcommittee will follow at an appropriate time.

An extensive review of this subject was recently done by Peter Tam for ACRS. In addition, he co-authored a NUREG document on Containment Sprays. NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels."

Related Staff Item: C-10, Effective Operation of Containment Sprays in a LOCA

Staff Priority: C

ACRS Priority: B

55. Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock -- Metal Components Subcommittee

Reg. Guide 1.2 covers current practice satisfactorily. The situation with respect to old plants is still unclear and the LERs display some events where thermal shocks have exceeded Tech. Spec. limits. The implications of the LERs need more attention. The Metal Components Subcommittee should address this. Special concern for repressurization after or during cooldown.

LER Subcommittee gave coverage to this item.

Related Staff Item: A-11, Reactor Vessel Materials Toughness

Staff Priority: A

ACRS Priority: A

56. Instruments to Detect (Severe) Fuel Failures -- Power and Electrical Systems Subcommittee

The Three Mile Island experience justifies reexamination of this question.

No related Staff item.

57. Monitoring for Excess Vibration Inside the Reactor Pressure Vessel -- Power and Electrical Systems Subcommittee

Methodology exists to address this matter in the pressure vessel, but the quality of its sensitivity has been related to actual safety needs. The capability seems to be adequate but the matter should be kept under surveillance by the Power and Electrical Systems Subcommittee.

Related Staff Item: B-73, Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel

Staff Priority: B

ACRS Priority: B

58. Non-Random Multiple Failures -- Single Failure Criterion Subcommittee

Items 58.a, Reactor Scram Systems; 58.b, Current Sources; and 58.c, DC Sources, are matters of concern. The systems interaction work is now under active review by the Plant Arrangements Subcommittee and it should continue to assess this question. The single-failure criterion is relevant. Sandia is reviewing

Related Staff Item: C-13, Non-Random Failures
A-9, ATWS
A-35, Adequacy of Offsite Power System
B-56, Diesel Reliability
A-30, Adequacy of DC Power Supplies
A-44, Station Blackout

Staff Priority: A,B,C

ACRS Priority: A

59. Behavior of Reactor Fuel Under Abnormal Conditions -- Reactor Fuel Subcommittee

Recent experience at Three Mile Island-2 should be evaluated to determine what is needed in this area. The ACRS Research Report has suggested that the PBF program be reoriented to address the question of intermediate level fuel degradation where fuel cladding has been significantly damaged and some fuel melting may have occurred.

Related Staff Item: B-22, LWR Fuel
B-52, Fuel Assembly Seismic and LOCA Responses

Staff Priority: B

ACRS Priority: A

60. BWR and PWR Primary Coolant Pump Overspeed During LOCA -- Joint ECCS and Plant Arrangements Subcommittees

Requires review by ECCS and/or Plant Arrangements Subcommittees.

Related Staff Item: B-68, Pump Overspeed During a LOCA

Staff Priority: B

ACRS Priority: B

61. Advisability of Seismic Scram -- Extreme External Phenomena Subcommittee

Information is available from the Japanese and from the Canadians with respect to seismic scram . The Extreme External Phenomena Subcommittee should evaluate whether this new information provides sufficient background to make a judgment about when seismic scrams may be desirable in nuclear plants.

Related Staff Item: D-1, Advisability of a Seismic Scram

Staff Priority: D

ACRS Priority: C

62. Emergency Core Cooling System Capability for Future Plants --Joint ECCS and Plant Arrangements Subcommittee

The requirements of 10 CFR 50, Section 50.3.4 (a)(4), 50.3.4 (b)(4), 50.4 6, and Appendix K, establish fuel performance requirements that have enhanced the emergency core cooling system capability of plants since this generic item was identified. All of the LOCA evaluation models have now been completed. The need for other cooling approaches to improved ECCS capability needs to be reviewed by the ACRS. The ECCS and Plant Arrangements Subcommittees should jointly attempt to determine whether this generic matter is adequately resolved, and if not, what actions are needed.

Related Staff Item: D-2, Emergency Core Cooling System Capability for Future Plants

Staff Priority: D

ACRS Priority: A

63. Ice Condenser Containment -- ECCS Subcommittee

The ECCS Subcommittee should determine whether adequate design margin exists during LOCA for ice condenser containments. If design margins are of importance, the action required to establish design margins should be identified.

Related Staff Item: B-54, Ice Condenser Containments

Staff Priority: B

ACRS Priority: B

64. Steam Generator Tube Leakage -- Metal Components Subcommittee

Regulatory Guide 1.83 establishes a safe operating mode, but the leakage frequency is still of concern. The Metal Components Subcommittee should review this matter and establish the path of action for generic resolution. Reg. Guide handles plugging. Question is how to present SC tube failure.

Related Staff Item: A-3, W Steam Generator Tube Integrity
A-4, CE Steam Generator Tube Integrity
A-5, B&W Steam Generator Tube Integrity

Staff Priority: A

ACRS Priority: A

65. ACRS/NRC Periodic Ten-Year Review of All Power Reactors -- Reactor Operations Subcommittee

The Three Mile Island accident reemphasizes the need to establish a policy concerning this matter. The NRC Staff presently has a program to review the older licensed reactor systems as a basis for defining periodic review policy. The ACRS Reactor Operations Subcommittee should evaluate this activity on a continuing basis until the NRC has established an acceptable policy.

The SEP is as close as the Staff has come.

66. Computer Reactor Protection System -- Power and Electrical Systems Subcommittee

This system continues to be reviewed by the Power and Electrical Systems Subcommittee and a periodic status report on the progress represents adequate action for the present.

Related Staff Item: A-19, Digital Computer Protection System

Staff Priority: A

ACRS Priority: B

67. Behavior of BWR Mark III Containments -- Fluid Dynamics Subcommittee

The experimental programs to verify Mark III containment behavior are in progress and the Fluid Dynamics Subcommittee is maintaining an overview of this work and reporting regularly to the full Committee. These actions seem appropriate.

Related Staff Item: A-39, Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containments
B-10, Behavior of BWR Mark III Containment

Staff Priority: A&B

ACRS Priority: B

68. Stress Corrosion Cracking in BWR Piping -- Metal Components Subcommittee

This matter is under active review by the ACRS Subcommittee on Metal Components. R&D work is underway under Industry sponsorship as well as by DOE and NRC. The problem is still of concern but the actions underway meet the present need.

Priority: Policy

69. Locking Out of ECCS Power Operated Valves -- Reactor Operations Subcommittee

This matter should be examined by the Reactor Operations Subcommittee and appropriate action suggested.

Related Staff Item: B-8, Locking Out of ECCS Power Operated Valves

Staff Priority: B

ACRS Priority: B

70. Design Features to Control Sabotage -- Joint Safeguards and Security and Plant Arrangements Subcommittees

This applies only to newly designed plants. The Committee's intent is unclear. The Safeguards and Security Subcommittee should reexamine this question in conjunction with the Plant Arrangements Subcommittee for the purpose of establishing a direction for resolution.

Related Staff Item: A-29, Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Staff Priority: A

ACRS Priority: A

71. Decontamination of Reactors -- Joint Metal Components and Reactor Radiologicalf Effects Subcommittees

The Three Mile Island accident shows the importance of this question but the original intent was primarily to address the decontamination of reactors to reduce operator exposure during in-service inspection and other circumstances. The status of the experimental work sponsored by Industry needs to be reviewed by either the Reactor Operations Subcommittee or the Metals Components Subcommittee. NOTE: Reactor Radiological Efffects Subcommitteee will consider occupational exposure aspects, and Waste Management Subcommittee will consider waste disposal.

Related Staff Item: A-15, Primary Coolant System Decontamination and Steam Generator Chemical Cleaning

Staff Priority: A

ACRS Priority: B

72. Decommissioning of Reactors -- Reactor Radiological Effects Subcommittee

This is an active NRC program of long duration and the status should be reported periodically by the Waste Management Subcommittee.

Related Staff Item: B-64, Decommissioning of Reactors

Staff Priority: B

ACRS Priority: B

73. Vessel Support Structures -- Combination of Dynamic Loads Subcommittee

The problem here is primarily asymmetric load questions and load combinations. This matter should probably be addressed on a probaba-

Related Staff Item: A-2, Asymmetric Blowdown Loads on PWR Primary Coolant Systems and Temperature Limits for BWR Containments
B-10, Behavior of BWR Mark III Containment

Staff Priority: A

ACRS Priority: B

74. Water Hammer -- Fluid Dynamics Subcommittee

The NRC staff is actively studying this matter but the problem should be addressed on a case-by-case basis. An ACRS Subcommittee with competent personnel to address the fluid mechanics questions should be assigned to review the status. Will review NRC Staff report.

This subject received attention in the ACRS LER report, Item D-V, Water Hammer.

Related Staff Item: A-1, Water Hammer

Staff Priority: A

ACRS Priority: A

75. Behavior of BWR Mark I Containment -- Fluid Dynamics Subcommittee

This matter is being addressed through R&D programs by the Mark I owners group and all of the open questions are nearing resolution. The ACRS needs an update of the status of this work. The Fluid Dynamics Subcommittee should be requested to summarize current status and establish the actions ultimately needed to resolve open questions.

Related Staff Item: A-6, Mark I Short Term Program
A-7, Mark I Long Term Program

Staff Priority: A

ACRS Priority: A

76. Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment -- Power and Electrical Systems Subcommittee

The TMI-2 accident reemphasizes the importance of this type of question and perhaps related ones. The Power and Electrical Systems Subcommittee should review this matter with the Regulatory Staff and Industry representatives to establish whether current practice is satisfactory, and if not, what actions might be appropriate to improve current practice.

Related Staff Item: C-1, Assurance of Continuous Long-Term Integrity of Seals on Instrumentation and Electric Equipment

Staff Priority: C

ACRS Priority: C

77. Soil Structure Interaction -- Extreme External Phenomena Subcommittee

The technology for evaluating soil structure interactions is developing rapidly. The ACRS should request one or more of its consultants who are not actively pursuing personal interest in this question to summarize the current status of technology in order to determine whether the current situation satisfies the generic concerns. The Extreme External Phenomena Subcommittee could undertake to sponsor such a review.

Related Staff Item: A-40, Seismic Design Criteria - Short Term Program

Staff Priority: A

ACRS Priority: C

Recommended ACRS Action Concerning Generic Items Agreed at 235th ACRS Meeting

Resolved Items	Follow-up by
1. NPSH for ECCS Pumps — Reactor Operations SC. This is covered by Reg. Guide 1.1. The Reactor Operations Subcommittee could review this with the Division of Operating Reactors to determine whether all plants are in compliance. Potential for vortex problems should be considered.	HE/RKM MTG. Dec 3
2. Emergency Power — <u>Joint</u> Power and Electrical Systems <u>and</u> Reactor Operations SCs Reg. Guide 1.6, 1.9, and 1.32 in conjunction with portions of IEEE-308 (1971) covers this matter. However, the question concerning loss of DC power or combined loss-of-offsite- and -onsite-AC power are presently of concern from a risk standpoint. The Power and Electrical Systems Subcommittee and the Reactor Operations Subcommittee should jointly review the status of emergency power requirements. The question of grandfathering older plants should also be considered regarding emergency power.	WK/GRO has lead Mtg Dec. 13
3. Hydrogen Control After Loss-of-Cooling Accident — TMI-2 Implications SC. The present hydrogen control requirements are based primarily on the concern for hydrogen build-up in containment following a LOCA where the fuel temperature rises to the level at which zirconium-water reaction proceeds rapidly, leading to hydrogen generation sufficient to cause burning or explosion. The Reg. Guide limits in 1.97 presume an oxidiation rate that is a function of surface area and a termination point related to ECCS capability. The Three Mile Island Accident displayed high hydrogen generation because the ECCS was not permitted to do its job. The TMI-2 Implication Subcommittee should recommend actions for reevaluation of this generic item.	DO/RKM Mtg. Dec. 4 to Review TMI-2 Lessons Learned
4. Instrument Lines Penetrating Containment — No action required Reg. Guide 1.11 and its Supplement adequately cover this point and no further action is needed.	NA*
5. Strong Motion Seismic Instrumentation — No action required This is covered in Reg. Guide 1.12 and there does not appear to be the need for further action.	NA
6. Fuel Storage Pool Design Bases — <u>Joint</u> Plant Arrangements <u>and</u> Safeguards and Security SCs. This is covered by Reg. Guide 1.13, however, the committee has frequently raised questions concerning the location of the fuel storage pool because of industrial sabotage questions. The Plant Arrangements and Safeguards and Security Subcommittee should review this matter and make recommendations to the full committee concerning the need for further action, especially regarding the location of the fuel pool with respect to grade.	MS/RKM and JCM/RKM Future Joint Meeting Planned

*NA = no action

	Follow-Up By
Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles — No action required This is covered by Reg. Guide 1.14 supported by knowledge developed in the Safety Research Program. Based on the staff evaluation of the R&D work, this matter appears to be adequately covered.	NA
Protection Against Industrial Sabotage — <u>Joint</u> Plant Arrangements <u>and</u> Safeguards and Security SCs. Reg. Guide 1.17 covers this matter, but since the issuance of Reg. Guide 1.17, committee letters have continued to raise questions about the adequacy of industrial sabotage protection. This matter should be addressed by joint effort of the Plant Arrangements Subcommittee and the Safeguards and Security Subcommittee.	MB/RKM and JCM/RKM Same as item #5
Vibration Monitoring of Reactor Internals and Primary System — No action required Reg. Guide 1.20 covers these matters and the recent review of the loose parts monitoring technology indicated that current interpretations of Reg. Guide 1.20 by the NRC Staff serve the situation adequately.	NA
10. In-Service Inspection of Reactor Coolant Pressure Boundary — Metal Components SC. This is covered by Section XI of the ASME Boiler and Pressure Vessel Code and Reg. Guide 1.,65 along with other modifications of the Code recently evaluated by the Reg. Guide Subcommittee. Questions remain as a result of Duane Arnold piping problems and various PWR feedwater line problems. This matter is under active review by the Metal Componnts Subcommittee and an update of recommendations concerning this matter should be provided from that Subcommittee.	PGS/EGI On-going review will keep under surveillance
11. Quality Assurance During Design, Construction, and Operation — Reactor Operations SC. Requirements of 10 CFR 50, Appendix B, ASME Boiler and Presure Vessel Code, Section III, ANSI-N45.2 (1971), Reg. Guides 1.28, 1.33, 1.64, 1.70.6, and proposed standard ANS-3.2, all address these matters. The NRC staff should be asked for a collective appraisal concerning the coverage in these documents. The Reactor Operations Subcommittee should then reassess the adequacy of this coverage. Recent experiences at Three Mile Island and concerns about the seismic restraints justify a determination concerning QA control adequacy.	HE/RKM Mtg. Dec 3
12. Inspection of BWR Steam Lines Beyond Isolation Valves — No action required This adequately covered by ASME Boiler and Pressure Vessel Code, Section XI.	NA
13. Independent Check of Primary System Stress Analysis — No action required This is adequately covered by ASME Boiler and Pressure Vessel Code, Section III.	NA
14. Operational Stability of Jet Pumps — No action required The work on Dresden-2 and -3 installations and other operating experiences adequately satisfy the ACRS concern.	NA

Follow-Up by

ressure Vessel Surveillance of Fluence and NDT Shift - Metal Components SC (Review together with Item 16)
This is covered by 10 CFR 50, Appendix A and ASTM Standard E-185. The NRC staff has recently recommended and the ACRS has approved the use of surveillance specimens from multiple reactor installations as satisfying the intent of the regulatory requirements. 10 CFR 50 will be modified accordingly under rulemaking proceedings.

PGS/EGI
Mtg. Jan 16

Nil-ductility Properties of Pressure Vessel Materials — Metal Components SC.
This is covered by 10 CFR 50, Appendix A and Appendix G, ASME Boiler and Pressure Vessel Code, Section III and was addressed in the ACRS 1970 Report on Light Water Reactor Pressure Vessel Integrity, WASH-1285. The situation still appears to be adequate from a safety standpoint, but the ACRS Metal Components Subcommittee should reexamine the nil-ductility problem as a function of temperature for some of the older vessels nearing the end of their specified life and any new questions that have arisen concerning the upper shelf properties of materials.

PGS/EGI
Same as item #15

. Operation of Reactor with Less Than All Loops in Service — No action required
Standard Review Plan, Ap[illegible]dix 7A and Branch Technical Position EICSB-12 cover this matter adequa[illegible]y.

NA

Criteria for Preoperaitonal Testing — Reactor Operations SC.
This is covered by the most recent revision to Reg. Guide 1.68 but the uniformity of the preoperational testing program at various sites is unclear. The present concerns about plant operating skills suggests a need to have the Reactor Operations Subcommittee examine the nature of preoperational test programs in order to determine whether the requirements of Reg. Guide 1.68 really satisfy regulatory needs.

HE/RKM
Mtg. Dec. 3

9. Diesel Fuel Capacity — No action required
Standard Review Plan 9.4 covers this matter adequately.

NA

0. Capability of biological shield withstanding double-ended pipe break at safe ends. Regulatory review practices cover this matter adequately. It may be appropriate to have one of the ACRS consultants examine a few examples of the design treatment to ascertain whether the approach is based on correct safety criteria. (Reports by Zudans rcd 6/80 - HA distributed to Bender et al w/memo.)

(H.Alderman)
MCG/Zudans review by Mar.

21. Operation of One Plant While Others are Under Construction — Have Fellows review

The coverage under Reg. Guide 1.17; 1.70; Sections 13.62; 1.101; ANSI N-18, 1.7; and Standard Review Plan 13.3, Appendix A; and 13.6 are all relevant to this question. One of the ACRS Fellows should be asked to review these documents to determine whether they treat all of the ACRS questions that have been raised and whether any other matters deserve attention. The potential for a Three Mile Island type of accident is particularly relevant to this matter. LERs should also be reviewed. Report by J. Bickel to M. Bender dtd. 10/3/79=major problem is security background checks and maintenance procedures for the operating plants.

Bickel report completed

MB has follow-up

Seismic Design of Steam Line — Combination of Dynamic Loads SC. This is covered by Reg. Guide 1.29 but the Combination of Dynamic Loads Subcommittee is reexamining the design bases. Recommended changes to Reg. Guide 1.29 may evolve from the combination of dynamic loads review.	MB/EGI Mtg held in Sept. Plan another for Feb/ Mar
Quality Group Classification for Pressure Retaining Components — Plant Arrangements SC (include analysis of secondary system (ec steam lines piping failures). Reg. Guide 1.26 covers this matter but questions arising from the interactive effect of non-safety grade equipment as seen in the Three Mile Island-2 accident may lead to changes in these classifications. The Plant Arrangement Subcommittee should review this matter.	MB/RKM Dec 5 SC Mtg (Deferred)
. Ultimate Heat Sink — No action required Reg. Guide 1.27 covers this matter satisfactorily.	NA
5. Instrumentation to Detect Stresses in Containment Walls — No action required Reg. Guide 1.18 covers this matter but there are some controversial questions associated with grouted tendons. Current Staff interpretations provide adequate controls.	NA
6. Use of Furnace Sensitized Stainless Steel — Reg. Guide 1.44 may need an update to better define "rapid-cooling". Bring to NRC Staffs attention but do not reopen consideration of Reg. Guide.	RFF will inform NRC Staff
27. Primary System Detection and Location of Leaks — reassign to Metal Components SC Reg. Guide 1.45 addresses this matter and experiences at Duane Arnold and other plants indicate that the procedures are suitable. Exploring the use of TV cameras to find leaks could be explored.	PGS/EGI Jan 9 ACRS Staff review EPRI program
28. Protection Against Pipewhip — Combination of Dynamic Loads SC. This is covered by Reg. Guide 1.46 but the Combination of Dynamic Loads Subcommittee will be reviewing these requirements as they are being influenced by combined load considerations. The question of whether the more elaborate requirements of combined loads introduce undesirable requirements should be examined.	MB/EGI Mtg. in Feb or Mar.
29. Anticipated Transients Without Scram — ATWS SC Although this matter was covered by WASH-1270, issued in September 1973, the NRC has not yet established an implementation plan nor are the technical bases fully established. The ACRS ATWS Subcommittee should continue to review this matter and recommend actions to the full Committee.	WK/PAB Committee concurred with plan proposed by S.H. Hanauer in NUREG-0600
30. ECCS Capability of Current and Older Plants (small LOCA needs attention) — ECCS The status should be updated through review by the ECCS Subcommittee, possibly with some support form the Plant Arrangements Subcommittee. Concerns about the oldest installations, e.g., Indian Point 1, have been resolved by NRC licensing action over the past several years.	MSP/ALB

POOR ORIGINAL

ositive Moderator Coefficient — No action required
PWR's presently follow a practice that satisfies the concerns about moderator coefficients under normal conditions. The transient questions associated with LOCA and the uncertainties associated with ATWS effects are under review.

NA

. Fixed In-Core Detectors on High-Power PWR's — No action required
In-core monitoring needs to be re-reviewed in the light of TMI-2 experience, but it is unlikely that fixed in-core detector needs would change because of such a review. This item seems O.K.

NA

. Performance of Critical Components (Pumps, Valves, etc.) in Post LOCA Environment — Power and Electrical Systems SC.
The qualification requirements in Reg. Guide 1.40, 1.63, 1.73, 1.89, and IEEE Standards 382 (1972), 383 (1974), 317 (1972), and 323 (1974), all address these matters. However, the experience at Three Mile Island-2 might alter some of these requirements. The Power and Electrical Systems Subcommittee should examine the need for new requirements.

WK/GRO
Keeping Under Surveillance

4. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containment — ACRS Fellow
The NRC staff requirements for Mark II and Mark III containments address these matters adequately. A review of actual experience with Mark II design might be useful for updating our knowledge. One of the ACRS Fellows might be assigned to make such a review. LERs should also be considered. G. Young report to M. Bender 9/24/79. Most failures occurred during testing.

G. Young report =problem resolved

35. Emergency Power for Two or More Reactors at the Same Site — Power and Electrical Systems SC.
Reg. Guide 1.81 covers this matter. Shared diesels at older plants should be examined. Will consider all shared systems and components.

WK/GRO
Future Mtg

36. Effluents from Light Water cooled Nuclear Power Reactors — No action required
This environmental question is resolved by the requirements of Appendix I of 10 CFR 50.

NA

37. Control Rod Ejection Accident — No action required
This is covered adequately by the requirements of Reg. Guide 1.77.

NA

38. Main Steam Isolation Valve Leakage of PWR — No action required
Reg. Guide 1.96 covers this adequately.

NA

39. Fuel Densification — No action required
Requirements of 10 CFR 50, Appendix K and case-by-case review of vendor fuel models covers this matter satisfactorily.

NA

40. Rod Sequence Control Systems — No action required
The practices of the NRC staff, including those established by GE NEDO 10527 cover this matter satisfactorily.

NA

Seismic Category 1 Requirements for Auxilary Systems — Combination of Dynamic Loads SC. | MB/EGI

This is covered by Reg. Guide 1.26 and 1.29, but may be reexamined if new questions of interpretation arise out of a Combination of Dynamic Loads Subcommittee review. | Mtg. Feb.

Instruments to Detect Limited Fuel Failures — Joint Power and Electrical Systems and Reactor Fuel SCs. | WK/GRO and FGS/PAB

Although this has been addressed in an NRC document entitled "Fuel Failure Detection in Operating Reactors" by Siegal and Hagan, June 1976, the experience of Three Mile Island warrants further review of this matter. The Power and Electrical Systems Subcommittee should evaluate this question in combination with the Reactor Fuel Subcommittee. Call to attention of NRC Staff. Resolved. Will keep under surveillance.

Instrumentation to Follow the Course of an Accident — Power and Electrical Systems SC. | CPS/SD

Reg. Guide 1.97, Revision 1, addresses this matter but the requirements have never been recognized. The Power and Electrical Systems Subcommittee should reexamine the requirements of 1.97 to determine whether they realistically define the need and whether a more definitive Reg. Guide should be provided based on TMI-2 experience. | Reg. Guide out for public comment

4. Pressure in Containment Following LOCA's — TMI-2 Implications SC. | DO/RKM

TMI-2 experience suggests the need to review this matter for low pressure containment. Will be considered during review of long-term lessons learned report

5. Fire Protection — Fire Protection SC. | MB/PST

Branch Technical Position 9.5.1 provides a staisfactory review process. Reg. Guide 1.120 whose development has been suspended because of ACRS concerns should now be reinitiated with attention being addressed to the requirements found acceptable for current Standard Plant Designs. | Mtg. Dec 5

46. Control Rod Drop Accidents (BWRs) — Core Performance SC. | WK/PAB Will follow up

This had been adequtely covered by NRC review practices. However, LERs have raised questions, short period scram concern raised by E. Epler. Low probability event

47. Rupture of High Pressure Lines Outside Containment — No action required | NA

Standard Review Plan Sections 3.6.1 and 3.6.2 cover this matter adequately.

48. Isolation of Low Pressure from High Pressure Systems — Reactor Operations SC. | HE/RKM

Standard Review Plan 5.4.7 addresses this matter. A few LERs have been identified which may have reopened concern for this question. | Mtg. Dec. 3

49. Monitoring for Loose Parts Inside the Reactor Pressure Vessel — No action required | NA

Reg. Guide 1.133 covers this matter.

50. Qualification of New Fuel Geometry — No action required | NA

Standard Review Plan 4.2, Revision 1, satisfies ACRS interest.

	Follow-Up By
Maintenance and Inspection of Plants — Reactor Operations SC. The ACRS originally accepted the postion that recent attention f the staff to these matters was adequate. The experience at TMI-2 reopens the question. The Reactor Operations Subcommittee should determine whether this matter needs additional effort.	HE/RKM Mtg/ Dec. 3
Safety Related Interfaces Between Reactor Island and Balance of Plant — Plant Arrangements SC. Standard Review Plan 1.8 covers the matter in an administrative sense, but systems interaction questions from the TMI-2 accident experience warrent reexamination by the Plant Arrangements Subcommittee.	MB/RKM Will address at next SC Mtg
Resolution of Pending Items	
. Turbine Missiles — Get update from S. H. Bush.Nothing new to update. Particular attention given to older plants.	MWL/SHB
. Effective Operation of containment Sprays in a LOCA — Generic Items SC will follow at an appropriate time. This matter should be reexamined by the Generic Items Subcommittee. The selection of chemical additives is still under review by the NRC Staff.	MB/PST Waiting for NRC Staff Report
5. Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock — Metal Components SC. Reg. Guide 1.2 covers current practice satisfactorily. The situation with respect to old plants is still unclear and the LERs display some events where thermal shocks have exceeded Tech. Spec. limits. The implications of the LERs need more attention. The Metal Components Subcommittee should address this. Special concern for repressurization after or during cooldown.	PGS/EGI Mtg. Jan 9
56. Instruments to Detect (Severe) Fuel FAilures — Power and Electrical Systems SC. The Three Mile Island experience justifies reexamination of this question.	WK/GRQ Keeping Under Surveillance
57. Monitoring for Excess Vibration Inside the Reactor Pressure Vessel — Power and Electrical Systems SC. Methodology exists to address this matter in the pressure vessel, but the quality of its sensitivity has been related to actual safety needs. The capability seems to be adequate but the matter should be kept under surveillance by the Power and Electrical Systems Subcommittee.	WK/GRQ Keeping Under Surveillance Have ACRS Fellow review
58. Non-Random Multiple Failures — Single Failure Criterion SC. Items 58.a, Reactor Scram Systems; 58.b, Current Sources; and 58.c, DC Sources, are matters of concern. The systems interaction work is now under active review by the Plant Arrangements Subcommittee and it should continue to assess this question. The single-failure criterion is relevant. Sandia is reviewing	MB/RKM Keeping Under Surveillance

Behavior of Reactor Fuel Under Abnormal Conditions — Reactor Fuel SC.
Recent experience at Three Mile Island-2 should be evaluated to determine what is needed in this area. The ACRS Research Report has suggested that the PBF program be reoriented to address the question of intermediate level fuel degradation where fuel cladding has been significantly damaged and some fuel melting may have occurred.

PGS/PAB
Study TMI-2 core performance when available

BWR and PWR Primary Coolant Pump Overspeed During LOCA — Joint ECCS and Plant Arrangements SC.
Requires review by ECCS and/or Plant Arrangements Subcommittees.

MSP/ALB and MB/RKM
will reexamine problem

. Advisability of Seismic Scram — Extreme External Phenomena SC.
Information is available from the Japanese and from the Canadians with respect to seismic scram . The Extreme External Phenomena Subcommittee should evaluate whether this new information provides sufficient background to make a judgment about when seismic scrams may be desirable in nuclear plants.

DO/RPS
Will develop proposed Committee position

2. Emergency Core Cooling System Capability for Future Plants — Joint ECCS and Plant Arrangements SC.
The requirements of 10 CFR 50, Section 50.3.4 (a)(4), 50.3.4 (b)(4), 50.4.6, and Appendix K, establish fuel performance requirements that have enhanced the emergency core cooling system capability of plants since this generic item was identified. All of the LOCA evaluation models have now been completed. The need for other cooling approaches to improved ECCS capability needs to be reviewed by the ACRS. The ECCS and Plant Arrangements Subcommittees should jointly attempt to determine whether this generic matter is adequately resolved, and if not, what actions are needed.

MSP/ALB and MB/RKM

Will reexamine problem

3. Ice Condenser Containment — Reassign to TMI-2 Implications
The ECCS Subcommittee should determine whether adequate design margin exists during LOCA for ice condenser containments. If design margins are of importance, the action required to establish design margins should be identified.

DO/RKM
Review effects of large H_2 generation

64. Steam Generator Tube Leakage — Metal Components SC.
Regulatory Guide 1.83 establishes a safe operating mode, but the leakage frequency is still of concern. The Metal Components Subcommittee should review this matter and establish the path of action for generic resolution. Reg. Guide handles plugging. Question is how to prevent SG tube failure

PGS/EGI
Mtg. Jan 16

65. ACRS/NRC Periodic Ten-Year Review of All Power Reactors — Reactor Operations SC.
The Three Mile Island accident reemphasizes the need to establish a policy concerning this matter. The NRC Staff presently has a program to review the older licensed reactor systems as a basis for defining periodic review policy. The ACRS Reactor Operations Subcommittee should evaluate this activity on a continuing basis until the NRC has established an acceptable policy.

HE/RKM
Mtg. Dec 3

mputer Reactor Protection System — Power and Electrical Systems SC.
his system continues to be reviewed by the Power and Electrical Systems Subcommittee and a periodic status report on the progress represents adequate action for the present.

WK/GRQ
Keeping under surveillance

Behavior of BWR Mark III Containments — Fluid Dynamics SC.
The experimental programs to verify Mark III containment behavior are in progress and the Fluid Dynamics Subcommittee is maintaining an overview of this work and reporting regularly to the full Committee. These actions seem appropriate.

MSP/ALB
Keeping under surveillance

Stress Corrosion Cracking in BWR Piping — Metal Components SC.
This matter is under active review by the ACRS Subcommittee on Metal Components. R&D work is underway under Industry sponsorship as well as by DOE and NRC. The problem is still of concern but the actions underway meet the present need. Will report to Committee periodically.

PGS/EGI
Keeping under surveillance

Locking Out of ECCS Power Operated Valves — Reactor Operations SC.
This matter should be examined by the Reactor Operations Subcommittee and appropriate action suggested.

HE/RKM
Mtg. Dec. 3

Design Features to Control Sabotage — <u>Joint</u> Safeguards and Security <u>and</u> Plant Arrangements SCs.
This applies only to newly designed plants. The Committee's intent is unclear. The Safeguards and Security Subcommittee should reexamine this question in conjunction with the Plant Arrangements Subcommittee for the purpose of establishing a direction for resolution.

JCM/RKM
and MB/RKM
Future SC mtg planned

Decontamination of Reactors — Joint Metal Components <u>and</u> Reactor Radiological Effects SCs.
The Three Mile Island accident shows the importance of this question but the original intent was primarily to address the decontamination of reactors to reduce operator exposure during in-service inspection and other circumstances. The status of the experimental work sponsored by Industry needs to be reviewed by either the Reactor Operations Subcommittee or the Metals Components Subcommittee. NOTE: Reactor Radiological Effects Subcommittee will consider occupational exposure aspects, and Waste Management Subcommittee will consider waste disposal.

DWM/RM has lead
Future SC mtg. planned

72. Decommissioning of Reactors — Reactor Radiological Effects Subcommittee.
This is an active NRC program of long duration and the status should be reported periodically by the Waste Management Subcommittee.

DWM/RM
Keeping under surveillance

73. Vessel Support Structures — Combination of Dynamic Loads SC.
The problem here is primarily asymmetric load questions and load combinations. This matter should probably be addressed on a probabalistic basis and should be reviewed by the Combination of Dynamic Loads Subcommittee. BNWL is studying.

MB/EGI
Keeping under surveillance

Water Hammer — Fluid Dynamics SC. The NRC staff is actively studying this matter but the problem should be addressed on a case-by-case basis. An ACRS Subcommittee with competent personnel to address the fluid mechanics questions should be assigned to review the status. Will review NRC Staff report.	MSP/ALB
. Behavior of BWR Mark I Containment — Fluid Dynamics SC. This matter is being addressed through R&D programs by the Mark I owners group and all of the open questions are nearing resolution. The ACRS needs an update of the status of this work. The Fluid Dynamics Subcommittee should be requested to summarize current status and establish the actions ultimately needed to resolve open questions.	MSP/ALB Will report to Committee at Dec. Mtg
6. Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment — Power and Electrical Systems SC. The TMI-2 accident reemphasizes the importance of this type of question and perhaps related ones. The Power and Electrical Systems Subcommittee should review this matter with the Regulatory Staff and Industry representatives to establish whether current practice is satisfactory, and if not, what actions might be appropriate to improve current practice.	WK/GRQ Future SC mtg planned
7. Soil Structure Interaction — Extreme External Phenomena SC. The technology for evaluating soil structure interactions is developing rapidly. The ACRS should request one or more of its consultants who are not actively pursuing personal interest in this question to summarize the current status of technology in order to determine whether the current situation satisfies the generic concerns. The Extreme External Phenomena Subcommittee could undertake to sponsor such a review.	DO/RPS ACRS Consultants are reviewing

POOR ORIGINAL

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

October 9, 1980

R. F. Fraley
Executive Director, ACRS

CERTIFICATION OF MINTUES OF THE AUGUST 6, 1980 MEETING OF THE PROCEDURES SUBCOMMITTEE MEETING

I certify that, to the best of my knowledge and belief the minutes of the August 6, 1980 meeting of the Procedures Subcommittee are an accurate record of the proceedings of that meeting.

Milton S. Plesset

Milton S. Plesset, Chairman

Oct. 9, 1980

(Date)