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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 26 1989

MEMORANDUM FOR: Victor Stello, Jr.
Executive Director for Operations

FROM: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING NUMBER 155

The Committee to Review Generic Requirements (CRGR) met on Wednesday, January 11. 1988 from 11:00 a.m. - 5:15 p.m. A list of attendees for this meeting is enclosed (Enclosure 1). The following items were addressed at the meeting:

1. The Committee completed at this meeting their review (begun at Meeting No. 155) of proposed modifications to Mark I containments, and related safety enhancements (e.g., accident management procedures), aimed at improving significantly the severe accident capability of Mark I containments. The Committee recommended in favor of imposing the proposed severe accident related backfits, subject to satisfactory resolution of several residual issues identified in their review and incorporation of a number of specific changes to the wording of the package (all changes to be coordinated with the CRGR staff). This matter is discussed in Enclosure 2.

2. S. Crockett (OGC) and J. Wilson (RES) presented for CRGR review the proposed final rule, 10 CFR Part 52, on standardization of advanced reactor designs, passive LWR designs, and evolutionary LWR designs. The Committee recommended in favor of sending the proposed rule forward for final consideration by the Commission, subject to several modifications (to be coordinated with CRGR staff). This matter is discussed in Enclosure 3.

In accordance with the EDO's July 18, 1983 directive concerning "Feedback and Closure on CRGR reviews," a written response is required from the cognizant office to report agreement or disagreement with the CRGR recommendations in these minutes. The response, which is required within five working days after receipt of these minutes, is to be forwarded to the CRGR Chairman and if there is disagreement with CRGR recommendations, to the EDO for decisionmaking.

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CRGR

Questions concerning these meeting minutes should be referred to Jim Conran (492-9855).

Original Signed By:
E. L. Jordan

Edward L. Jordan, Chairman
Committee to Review Generic
Requirements

Enclosures: As stated

cc/w enclosures:
Commission (5)
SECY
Office Directors
Regional Administrators
CRGR Members

Distribution: (w/o encl)
Central File
PDR (NRC/CRGR
S. Treby
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M. Lesar
P. Kadambi (w/enc.)
CRGR CF (w/enc.)
W. Houston (w/enc.)
A. Thadani (w/enc.)
S. Crockett (w/enc.)
J. Wilson (w/enc.)
CRGR SF (w/enc.)
M. Taylor (w/enc.)
E. Jordan (w/enc.)
J. Heltemes (w/enc.)
J. Conran (w/enc.)
C. Sakenas (w/enc.)

*SEE PREVIOUS CONCURRENCE

OFC	*CRGR:AEOD	DD:AEOD	C:CRGR:AEOD			
NAME	JConran	CJHeltemes	ELJordan			
DATE	1/19/89:jr	1/23/89	1/23/89			

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Enclosure 1
Attendance List
CRGR Meeting No. 155

CRGR Members

E. Jordan
J. Sniezek
B. Sheron (for D. Ross)
R. Bernero
J. Goldberg
C. Paperiello

NRC Staff

C. J. Heltemes
C. Sakenas
J. Conran
E. Beckjord
W. Houston
B. Beckner
A. Thadani
J. Ridgeley
J. Kudrick
L. Soffer
T. Collins
M. Thadani
T. Cox
A. Serkiz
M. Taylor
J. Scinto

Enclosure 2 to the Minutes of CRGR Meeting No. 155
Proposed (Severe Accident) Enhancements to Mark I Containments

TOPIC

E. Beckjord (RES), T. Murley (NRR), W. Houston (RES), B. Beckner (RES), and A. Thadani (NRR) presented for CRGR review a proposed package of modifications to Mark I containments, and related safety enhancements (e.g., accident management procedures), aimed at improving significantly the severe accident capability of the Mark I Containments. (The Committee began their review of this matter at Meeting No. 152.) Copies of the briefing slides used by the staff to guide their presentations and the discussions at this meeting are attached (Attachment 1).

BACKGROUND

1. Subsequent to Meeting No. 152, the staff revised the original proposed Mark I package in response to CRGR comments. The revised material was transmitted to CRGR by memorandum dated January 4, 1989, E.S. Beckjord to E.L. Jordan; the revised material was comprised of the following:

   a. Draft Commission Paper, "Mark I Containment Performance Improvement Program," and (revised) enclosures as follows:

      i. Enclosure 4 - Regulatory Analysis

      ii. Enclosure 7 - Draft Proposed Rule

      iii. Appendix A - Backfit Analysis

2. The January 4 version of the package was further revised on the basis of EDO office comments. Due to time constraints, that latest version of the Mark I package was not transmitted formally to CRGR, but was provided directly to Committee members at Meeting No. 155. That material is attached to these minutes for completeness of record (Attachment 2); it includes the following:

   a. Draft Commission Paper dated January 11, 1989 and (further revised) Enclosure 8 to the Commission Paper, "Draft Proposed Rule."

   b. Draft Plant-Specific Order.

CONCLUSIONS/RECOMMENDATIONS

As a result of their review of the proposed Mark I upgrades, including the discussions with the staff at Meeting No. 152 and this meeting, the Committee recommended in favor of imposing the proposed Mark I improvements, subject to resolution of the following comments/recommendations (to be coordinated with the CRGR staff):

1. The Committee recommended expedited rulemaking as the method for implementing approved Mark I upgrades. First, this is consistent with existing Commission guidance on preferred approach for addressing new severe accident requirements. Second, the concern was stated in discussions at this meeting that litigation of the issues involved in the proposed Mark I upgrades (in hearings that could be requested under the plant-specific Orders approach recommended by the staff) could delay implementation of the recommended upgrades significantly longer than expedited rulemaking. The staff did not agree that such litigative risks are an overriding concern, but did agree to further highlight that concern to the attention of the Commission in the final package.

2. Instrumentation requirements in this package (e.g., at the top of p.2 of the Draft Rule) should be sharpened and more fully defined to better assure that the improved severe accident coping functions intended can actually be successfully carried out by plant operators.

3. The extensive cost-benefit treatment of the separate elements of the overall package of Mark I fixes containe in this package (e.g., in the discussion of Alternatives ii., iii., iv. and v. in the Regulatory Analysis) detracts from the case the staff is trying to make for the synergistic, integrated set of modifications finally recommended (i.e., Alternative vi.). The package should be revised to give greater emphasis to the staff's objective of providing defense-in-depth protection (i.e., both preventive and mitigative measures) against the dominant severe accident sequences for Mark I containments, so that even if a (low probability) core melt occurs, there is reasonable expectation (i.e., comparable to that for most PWRs) that the containment will be able to mitigate the consequences.

4. A major weakness of the current package is the discrepancy (a factor of three-or-so) between the staff's estimate of licensee costs to implement the recommended backfits and the actual costs incurred by one licensee in implementing (voluntarily) a number of those same Mark I upgrades. The staff should resolve that discrepancy, and revise the package to better explain the apparent difference.

5. The Committee questioned seriously the feasibility or practicality of accelerating ATWS and Station Blackout (SBO) rule implementation, as recommended in the current package. The package should be revised to more clearly indicate (in accordance with discussions with the staff at this meeting) that (a) implementation of approved Mark I upgrades must be carefully coordinated with those ongoing rule implementation efforts, but (b) no acceleration of licensees' actions is intended.

6. The Committee also recommended the following specific modifications to the current package:

   a. Draft Rule, p.2, paragraph 2.b.:

      Delete the first sentence, and revise the remaining wording, if necessary, to make clearer the staff's intent (i.e., provision must be made to assure the capability to vent at design pressure; but venting at low pressure is not precluded, and no special provision need be made to preclude inadvertent venting at low pressure).

   b. Draft Rule, p.2, paragraph 2.d.:

      Delete entirely. (Also delete corresponding paragraph IV.A.1.d. at p. 7 of the Draft Order.)

   c. Draft Rule, p.2, paragraph 2.e.:

      Change "should" to "shall."

   d. Draft Rule, p.3, paragraph 2.g.:

      Change to read "..alarmed and indicating in the control room..," and make clear that the requirement for a radiation monitor in this paragraph could be met with an approved post-TMI stack radiation monitor, if already installed.

   e. Draft Rule, p.3, paragraph 3.:

      The EQ requirements in subitem 1.) should be deleted if (as indicated in the discussions at this meeting) no additional or more stringent qualification of the subject cabling, beyond that provided by compliance with 10CFR50.49 requirements, would be required for assured operation prior to vessel failure.

   f. Draft Rule, p.3, paragraph 4.:

      Change "should' to "shall" in the second sentence, and delete the last sentence.

   g. Draft Rule, p.4, first paragraph:

      Change "..30 days.." to "..60 days.." in the first sentence, and add to the end of the second sentence "..and licensees shall certify to NRC that they have met the rule."

h. Draft Rule, p.4, paragraph 2.a.:

Change to read "..by use of an alternate AC (AAC) source, as defined in Section 3.3.5 of Reg. Guide 1.155,.."

i. Draft Order, p.6, third sentence of the first full paragraph:

Delete the words "..raising the vent valve operability pressure and/or .."

j. Draft Order, p.8, last sentence of paragraph 2.:

Change "should" to "shall."

BRIEFING ON

BWR MARK I CONTAINMENT PERFORMANCE

BEFORE

CRGR

JANUARY 11, 1989

WAYNE HOUSTON
WILLIAM D. BECKNER

OFFICE OF NUCLEAR REGULATORY RESEARCH
U. S. NUCLEAR REGULATORY COMMISSION

Attachment 1 to
Enclosure 2

BACKGROUND

- JUNE 1986, STAFF PROPOSED 5 ELEMENT PROGRAM FOR MARK I CONTAINMENT PERFORMANCE ENHANCEMENT
- JUNE-JULY 1987, TWO LICENSEES INFORMED THE STAFF OF THEIR INTENTION TO INVESTIGATE CONTAINMENT AND SAFETY ENHANCEMENT
- JULY 7, 1987, STAFF BRIEFED COMMISSION ON A PLAN FOR CLOSURE OF SEVERE ACCIDENT ISSUES
- DECEMBER 1987, "MARK I CONTAINMENT PERFORMANCE PROGRAM PLAN", (SECY-87-297)
- FEBRUARY 1988, WORKSHOP ON MARK I ISSUES
- MAY 1988, "INTEGRATION PLAN FOR CLOSURE OF SEVERE ACCIDENT ISSUES", (SECY-88-147)
- JULY 1988, "STATUS OF MARK I CONTAINMENT PERFORMANCE EVALUATION", (SECY-88-206)
- DECEMBER 6, 1988, ACRS SUBCOMMITTEE ON CONTAINMENT SYSTEMS
- DECEMBER 14, 1988, CRGR REVIEW
- DECEMBER 15, 1988, ACRS FULL COMMITTEE

SUMMARY OF STAFF RECOMMENDATIONS FOR MARK I

- ACCELERATE IMPLEMENTATION OF STATION BLACKOUT RULE (ATWS IMPLEMENTATION TO BE ESSENTIALLY COMPLETE BY 1/89)

- REQUIRE ALTERNATE WATER SUPPLY FOR DRYWELL SPRAY/VESSEL INJECTION WITH PUMPING CAPABILITY INDEPENDENT OF NORMAL AND EMERGENCY AC

- REQUIRE HARDENED VENTING CAPABILITY FROM WETWELL (ABLE TO WITHSTAND SEVERE ACCIDENT PRESSURES). ISOLATION VALVES TO BE REMOTELY OPERABLE INDEPENDENT OF NORMAL AND EMERGENCY AC.

- REQUIRE ENHANCED ADS RELIABILITY. ADDITIONAL POWER AND/OR NITROGEN SUPPLY AND CABLE RELIABILITY

- REQUIRE IMPLEMETATION OF IMPROVED EPG'S (REV. 4 OF BWROG)

CHANGES TO CPI PACKAGE FOR MARK Is

- PROPOSED IMPLEMENTATION THROUGH PLANT-SPECIFIC BACKFITS

- COORDINATED REQUIREMENTS OF THE SBO RULE WITH PROPOSED CPI BACKUP POWER NEEDS

- INCREASE ESTIMATED AVERAGE REMAINING PLANT LIFE FROM 20 YEARS TO 25 YEARS

- PERFORMED COST-BENEFIT SENSITIVITY FOR INCREMENTAL ADDITION OF ADS AND BACKUP WATER SUPPLY (SECTION 4.1.7)

- REVISED COST-BENEFIT ANALYSIS TO INCLUDE EFFECT OF PROPOSED MITIGATION ENHANCEMENTS ON SOME ATWS SCENARIOS

COST-BENEFIT RESULTS

(MAN-REM AVERTED PER MILLION DOLLARS)***

LOW RISK PLANT ($TW=10^{-5}$)

LOW INDUSTRY COSTS*	1,970
HIGH INDUSTRY COSTS**	500

HIGH RISK PLANT ($TW=10^{-4}$)

LOW INDUSTRY COSTS*	29,600
HIGH INDUSTRY COSTS**	4,570

* LOW INDUSTRY COST IS ESTIMATED TO BE $48 MILLION

** HIGH INDUSTRY COST IS ESTIMATED TO BE $176 MILLION

*** INCLUDES AVERTED ON-SITE COST OF CLEANUP, REPAIR AND REPLACEMENT POWER

COST BENEFIT FOR INCREMENTAL ADDITION OF ADS AND BACKUP WATER SUPPLY

(MAN-REM AVERTED PER MILLION DOLLARS)

ASSUMPTION	MAN REM AVERTED	LOW INST. COST	HIGH INST. COST
BASE CASE*	33	630	190
AVERAGE SBO PROB.**	55	1,050	310
AVERAGE SBO PROB. HIGH LINER MELT PROB.***	93	1,780	530
AVERAGE SBO PROB. HIGH LINER MELT PROB. ATWS MITIGATION****	135	2,580	760
HIGH SBO PROB.*****	165	3,150	930
HIGH SBO PROB. HIGH LINER MELT PROB.	280	5,344	1,580
HIGH SBO PROB. HIGH LINER MELT PROB. ATWS MITIGATION	322	6,145	1,816
HIGH SBO PROB. HIGH LINER MELT PROB. ATW MITIGATION LOW POPULATION	64	1,221	361

* BASED ON SBO FREQUENCY OF 6×10^{-6}/RY AND A CONDITIONAL LINER MELT PROBABILITY OF 0.5 GIVEN A CORE MELT.

** BASED ON A SBO FREQUENCY OF 1×10^{-5}/RY WHICH IS AN AVERAGE FOR MARK I PLANTS.

*** CONDITIONAL LINER MELT PROBABILITY OF 1.0 GIVEN A CORE MELT.

**** ASSUMES ATWS MITIGATION OF 42 MAN-REM PER RY.

***** BASED ON A SBO FREQUENCY OF 3.5×10^{-5}/RY.

CO-ORDINATION WITH SBO RULE

(10 CFR 50.63)

COMPLIANCE WITH 10 CFR 50.63 BY:

- ALTERNATE AC (AAC) POWER SOURCE
  - DOES NOT NEED ADDITIONAL POWER SOURCE IF ACC SATISFIES POWER NEEDS FOR PROPOSED MARK I ENHANCEMENTS
- COPING ANALYSIS
  - MUST PROVIDE ADDITIONAL POWER SOURCE TO SATISFY POWER NEEDS FOR PROPOSED MARK I ENHANCEMENTS

PROPOSED IMPLEMENTATION

- HAVE EXAMINED TWO OPTIONS:
  - RULEMAKING
  - PLANT SPECIFIC BACKFITS (ORDERS)
- PREPARE ENVIRONMENTAL ASSESSMENT OF VENTING
- SCHEDULE
  - LICENSEES TO SUBMIT PLANS AND ANTICIPATED SCHEDULE WITHIN 60 DAYS OF BACKFIT REQUEST
  - IMPLEMENTATION TO BE COMPLETED WITHIN 30 MONTHS OF BACKFIT REQUEST

CONCLUSIONS AND RECOMMENDATIONS

- PROPOSED ENHANCEMENTS PROVIDED SUBSTANTIAL INCREASE IN OVERALL PROTECTION OF PUBLIC HEALTH AND SAFETY
- PROPOSED ENHANCEMENTS ARE GENERALLY COST BENEFICIAL
- PROPOSE TO IMPLEMENT VIA ORDERS
- CONTINUE CONFIRMATORY RESEARCH ON PHENOMENA RELEVANT TO IN-VESSEL AND EX-VESSEL ACCIDENT PROGRESSION, THE EFFECT OF WATER ON THE PROBABILITY OF LINER MELT-THROUGH, AND ASSOCIATED SOURCE TERMS

A Thadani Slides
Meeting 155

BWRs

Dominant Accident Sequences

ATWS

Station Blackout	SBO
Long Term Heat Removal	
Cont. Failure Before Core Melt	TW
Others	Decay Heat

Key Containment Failure Modes

Overpressure

Liner Melt-thru (Uncertain)

Proposed Fixes Reduce

° Core Melt Frequency	(TW, SBO, Decay Heat)
° Cont. Failure Probability	(Reduces Potential for Key Failure Modes)
° Source Term	(Provides Water)

Qualitative Value of Mark I Enhancements

	Aid In Operation	Aid in Preventing Loss of In-vessel Coolability	Aid in Maintaining Debris In Vessel	Aid in Maintaining Containment Integrity After Vessel Breach	Aid in Limiting Release After Containment Breach
(A) Improved Venting	No	Yes (TW)	Yes	Yes (Over Pressure)	Yes, If small size breach
(B) Backup Water Supply	No	Yes (Decay Heat)	Yes	Yes (Liner Melt)	Yes (DF)
(C) Enhanced ADS	No	No	No	Yes (Dir Cont. Heating)	No
(D) Improved EOPs	Yes	Yes	Yes	Yes	Possible
A+B	No	Yes (TW, Decay Heat)	Yes	Yes (Over Press. Including SBO, Liner Melt)	Yes (DF)
A+C	No	Yes (TW)	Yes	Yes (Over Pressure, Dir. Cont. Heating)	No
B+C	No	Yes (SBO, Decay Heat)	Yes	Yes (Liner Melt, Dir. Cont. Heat)	Yes (DF)
A+B+C	No	Yes (TW, SBO, Decay Heat)	Yes	Yes (Over Press. Including SBO, Liner Melt Dir. Cont. Heating)	Yes (DF)

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BWR / Mark I Improvements

	Improved EOP	Enhanced ADS	Backup Water Supply	Improved Venting
Aid in Routine Operation?	Yes	No	No	No
Aid in Preventing Accidents from Exceeding Design Basis?	Yes	No	Yes	Yes
Aid in Recovery to Stable State After DBA Exceeded?	Yes		Yes	Yes
Aid in Maintaining Containment Integrity After Vessel Break?	Yes	Yes	Yes	Yes
Aid in Limiting Release After Containment Breach?	Possible	No	Yes	Yes

T. Murley
Meeting 155

January 11, 1989

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: MARK I CONTAINMENT PERFORMANCE IMPROVEMENT PROGRAM

Purpose: To present staff recommendations on Mark I containment performance improvements and other safety enhancements.

Category: This paper covers a major policy question.

Summary: As noted in the Integration Plan for Closure of Severe Accident Issues (SECY 88-147) and in interim reports to the Commission (SECY 87-297 and SECY 88-206), the staff has undertaken a program to determine what actions, if any, should be taken to reduce the vulnerability of containments to severe accident challenges. The containment performance improvement effort is one main element of the integrated approach to closure of severe accident issues. Staff efforts have focused initially on BWR plants with a Mark I containment. The staff has now completed its assessment of generic severe accident challenges and failure modes as well as potential improvements for plants with the Mark I containment. The review of Mark II, Mark III, and other containment types are the subject of parallel but separate program efforts, as discussed in SECY 88-147.

Probabilistic Risk Assessment (PRA) studies have been performed for a number of BWRs with Mark I containments. These studies indicate that BWR Mark I risks are dominated by Loss of Decay Heat Removal, Station Blackout (SBO), and Anticipated Transient Without Scram (ATWS) sequences. Although these studies do not show the BWR Mark I plants to be risk outliers as a class relative to other plant designs, they do suggest that the Mark I containment integrity could be challenged by a large scale core melt accident, principally due to its smaller size. However, estimates of containment failure likelihood under such conditions are based on analysis of complex accident conditions, where there remains a broad band of uncertainty.

Contact: W. Beckner, RES
492-3975
L. Soffer, RES
492-3916

Attachment 2 to
Enclosure 2

The staff has concluded that the optimum way to reduce overall risk in BWR Mark I plants is to pursue a balanced approach utilizing accident prevention and mitigation.

Based on our assessment including the above described balanced approach, the staff recommends five specific improvements for Mark I containment plants: 1) an improved hardened vent capability, 2) improved reactor pressure vessel (RPV) depressurization system reliability, 3) an alternate water supply to the reactor vessel and drywell sprays, 4) extended emergency procedures and training and 5) accelerated implementation of the existing ATWS and SBO rules. These improvements, when fully implemented, will substantially enhance the safety of Mark I plants, including improvement to containment performance. The staff has evaluated them and found them to be cost effective. The staff proposes that orders be issued to all licensees with Mark I containments to implement these improvements.

Background: The Reactor Safety Study (WASH-1400) found that, for the Peach Bottom BWR Mark I nuclear plant, even though the core melt probability was relatively low, the containment could be severely challenged if a large core melt occurred. Based on this conclusion and reinforced by the anticipation of similar findings (subsequently confirmed) in the draft Reactor Risk Reference Document (NUREG-1150, February 1987) a five element program was proposed in June 1986 to enhance the performance of the BWR Mark I containment. Elements of this proposal included 1) hydrogen control, 2) containment drywell spray, 3) containment venting, 4) core debris control, and 5) emergency procedures and training. After the initial proposal, the staff held two separate meetings in early 1987 with researchers representing NRC contractors and industry. There was a wide range of views expressed regarding accident phenomenology as well as the efficacy of the various improvements. In view of the lack of technical consensus on the effectiveness of the proposed improvements, the staff decided to undertake additional efforts. In July 1987, the staff informed the Commission of its intention to examine the Mark I issue in the context of an integrated approach to the closure of severe accident issues.

On December 18, 1987, the staff issued a plan (SECY 87-297) for resolving generic severe accident containment performance issues for Mark I and other containment types. As part of the plan, a workshop was held on February 24-26, 1988 to discuss a number of issues associated with Mark I containment challenges, failure modes and potential containment improvements with researchers, industry representatives and interested members of the public. A major topic at the workshop was the phenomena associated with containment shell meltthrough as discussed in Enclosure 6. The Integration Plan for Closure of Severe Accident Issues, (SECY 88-147) characterizes the containment performance improvement effort as being one of the main elements of the integrated approach to closure of severe accident issues. Other main elements include a) Individual Plant Examinations (IPEs), b) improved plant operations, c) the severe accident research program, d) examination of external events, and e) a program on accident management. The containment performance improvement program is related to the IPE effort, and is considered complementary to it, since this effort is primarily focused on the potential generic vulnerabilities of specific containment classes, whereas the IPE effort is focused on plant unique vulnerabilities.

A Commission paper (SECY 88-206) dated July 15, 1988 provided a status report on the staff's efforts regarding the Mark I containment. This paper reaffirmed that the risk from BWR Mark Is is low. Nevertheless, the staff proposed a program intended to further reduce overall risk in BWR Mark I plants by pursuing a balanced approach involving accident prevention and mitigation. A number of safety enhancements were identified which appeared attractive in terms of their potential risk reduction capability as well as implementation costs.

Following that meeting the Commission requested additional information via a staff requirements memorandum dated August 1, 1988. Responses to these questions are included as Enclosure 1.

Discussion: Probabilistic Risk Assessment (PRA) studies for BWRs indicate that accidents initiated by transients rather than Loss-Of-Coolant-Accidents (LOCAs) dominate the total core damage frequency estimates. The principal accident sequences for BWRs consist of Long-term Loss of Decay Heat Removal (TW), Station Blackout (SBO), and Anticipated Transient Without Scram (ATWS). WASH-1400 indicated that TW is the dominant core damage accident sequence for Peach Bottom. Draft NUREG-1150, however, indicated that the dominant contribution to core melt frequency at Peach Bottom is due to Station Blackout, and estimated that TW has been greatly reduced at Peach Bottom by implementation of containment venting procedures with the assumption that said venting actions can be successfully accomplished. For those plants in which TW has been eliminated as the dominant contributor, the residual risk is largely due to ATWS and SBO sequences. These studies also indicate that the estimated likelihood of core damaging accidents for existing Mark I plants is predicted to vary widely over two orders of magnitude or more. The primary containment challenges and potential failure modes for BWR Mark I containments are shown in Enclosure 2.

The staff has examined potential Mark I containment and plant improvements in the following six areas: (1) hydrogen control, (2) alternate water supply for reactor vessel injection or containment drywell sprays, (3) containment pressure relief capability (venting), (4) enhanced RPV depressurization system reliability, (5) core debris controls, and (6) procedures and training. Each of these was evaluated to determine their potential benefits in terms of reducing the (1) core melt frequency, (2) containment failure probability, and (3) offsite consequences.

Hydrogen Control:

Although BWR Mark Is are required to be operated with an inerted containment atmosphere, plant Technical Specifications permit de-inerting to commence 24 hours prior to plant shutdown, and do not require inerting to be completed until 24 hours after plant startup, in order to permit plant personnel access. In the event of a severe accident, such as a long-term station blackout, a concern was expressed that loss of control of the valves and containment leakage could eventually lead to containment de-inerting.

Two potential improvements with regard to hydrogen control were evaluated. These were: (1) elimination of the two 24 hour de-inerted periods and (2) providing a backup supply of nitrogen. Since the probability of a severe accident occurring during either of the two 24 hour de-inerted periods is small compared to the probability of accident occurrence during normal operations, eliminating this time of de-inerting would not significantly reduce risk.

During a severe accident, reactor pressure is anticipated to increase, releasing steam and non-condensable gases into the containment. This will increase containment pressure, preventing ingress of air. Therefore, the containment atmosphere would not become de-inerted for an extended period of time. Since offsite supplies of nitrogen could readily be obtained during this period, an onsite backup supply of nitrogen would not significantly reduce risk.

Therefore, the staff concludes that additional Mark I improvements to control hydrogen beyond the existing hydrogen control rule and the procedures in Revision 4 of the Emergency Procedure Guidelines would have no significant benefit and are not warranted.

Alternate Water Supply for Drywell Spray/Vessel Injection

An important proposed improvement would be to employ a backup or alternate supply of water and a pumping capability that is independent of normal and emergency AC power. By connecting this source to the low pressure residual heat removal (RHR) system as well as to the existing drywell sprays, water could be delivered either into the reactor vessel or to the drywell, by use of an appropriate valving arrangement.

An alternate source of water injection into the reactor vessel would greatly reduce the likelihood of core melt due to station blackout or loss of long-term decay heat removal, as well as provide significant accident management capability.

Water for the drywell sprays would also provide significant mitigative capability to cool core debris, to cool the containment liner to delay or prevent failure, and to scrub air borne particulate fission products from the atmosphere.

A review of some BWR Mark I facilities indicates that most plants have one or more diesel driven pumps which could be used to provide an alternate water supply. The flow rate using this backup water system may be significantly less than the design flow rate for the drywell sprays. The potential benefits of modifying the spray headers to assure a spray were compared to having the water run out of the spray nozzles. Fission product removal in the small crowded volume in which the sprays would be effective was judged to be small compared to the benefit of having a water pool on top of the corium. Therefore, modifications to the spray nozzles are not considered warranted.

Containment Pressure Relief Capability (Venting):

Venting of the containment is currently included in BWR emergency operating procedures. The vent path external to existing containment penetrations typically consists of a ductwork system which has a low design pressure of only a few psi. Venting under high pressure severe accident conditions would fail the ductwork, release the containment atmosphere into the reactor building, and potentially contaminate or damage equipment needed for accident recovery. In addition, with the existing hardware and procedures at some plants, it may not be possible to open or to close the vent valves for some severe accident scenarios. The staff has concluded that venting, if properly implemented, can have a significant benefit on plant risk. However, venting via a sheet metal ductwork path, as currently implemented at some Mark I plants, is likely to greatly hamper or complicate post-accident recovery activities, and is therefore viewed by the staff as yielding reduced improvements in safety. The capability to vent has long been recognized as important in reducing risk from operation of BWR Mark I facilities for loss of long term decay heat removal events. Controlled venting can prevent the failure of ECCS pumps from inadequate NPSH and re-closure of the ADS valves. The staff agrees with this view as long as the potential downsides of using the existing hardware are corrected.

A hard pipe vent capable of withstanding the anticipated severe accident pressure loadings would eliminate these disadvantages. The vent isolation valves should also be remotely operable from the control room and should be provided with a power supply independent of normal or emergency AC power. Other changes, such as raising the vent valve operability pressure and/or raising the RCIC turbine back pressure trip setpoint, may also be desirable and should be considered as part of the IPE. This capability, in conjunction with proper operating procedures and other improvements discussed in this paper, would result in greatly reducing the probability of core melt due to the TW and SBO sequences.

Given a core melt accident, venting of the wetwell would provide a scrubbed venting path to reduce releases of particulate fission products to the environment. Venting has been estimated to reduce the likelihood of late containment over-pressure failure and to reduce offsite consequences for severe accident scenarios in which the containment shell does not fail for other reasons. Failure of the shell due to corium attack (shell meltthrough) would reduce the benefits from venting in that it would release fission products directly into the reactor building.

Inadvertent venting could result in the release of normal coolant radioactivity to the environment even when core degradation is averted or vessel integrity maintained. Measures to reduce the probability of inadvertent venting, such as a rupture disk, should be considered in the vent design.

<u>Enhanced Reactor Pressure Vessel (RPV) Depressurization System Reliability:</u>

The Automatic Depressurization System (ADS) consists of relief valves which can be operated to depressurize the reactor coolant system. Actuation of the ADS valves requires DC power. In an extended station blackout after station batteries have been depleted, the ADS would not be available and the reactor would re-pressurize. With enhanced RPV depressurization system reliability, depressurization of the reactor coolant system would have a greater degree of assurance. Together with a low pressure alternate source of water injection into the reactor vessel, the major benefit of enhanced RPV depressurization reliability would be to provide an additional source of core cooling which could significantly reduce the likelihood of high pressure severe accidents, such as from the short-term station blackout.

Another important benefit is in the area of accident mitigation. Reduced reactor pressure would greatly reduce the possibility of core debris being expelled under high pressure, given a core melt and failure of the reactor pressure vessel. Enhanced RPV depressurization system reliability would also delay containment failure and reduce the quantity and type of fission products ultimately released to the environment. In order to increase reliability of the RPV depressurization system, assurance of electrical power beyond the requirements of existing regulations may be necessary as discussed later in this paper. In addition, performance of the cables needs to be reviewed for temperature capability during a severe accident.

Core Debris Controls:

Core debris controls, in the form of curbs in the drywell and/or curbs or weir walls in the torus room under the wetwell have been proposed in the past to prevent containment shell meltthrough and to retain sufficient water to permit fission product scrubbing. However, as noted in SECY 88-206, the technical feasibility for such controls has not been established, and the design and installation costs as well as the occupational exposure during installation could be significant. The staff intends to pursue research programs to evaluate the need for and feasibility of core debris controls. There is a growing consensus that water in the containment (from an alternate supply to the drywell sprays) may help mitigate risk either by fission product scrubbing or by preventing or delaying shell melt by core debris. Research is continuing in order to confirm and help quantify these initial conclusions.

A discussion of Mark I shell melt phenomena and the current state of knowledge is included in Enclosure 6.

Emergency Procedures and Training:

A major element of the Mark I containment performance improvement evaluation involves emergency procedures and training. Current emergency operating procedures (EOPs) are symptom-based procedures that originated from requirements of TMI Task Action Plan item I.C.1. Plant-specific EOPs are generally implemented based on generic Emergency Procedure Guidelines (EPGs) developed by the BWR Owners Group. As part of the balanced approach to examining potential BWR Mark I plant improvements, both the generic EPGs and the plant-specific implementation of EOPs and training have been examined.

NRC has recently reviewed and approved Revision 4 of the BWR Owners Group EPGs (General Electric Topical Report NEDO-31331, BWR Owner's Group "Emergency Procedure Guidelines, Revision 4," March 1987). Revision 4 to the BWR Owners Group EPG is a significant improvement over earlier versions in that they continue to be based on symptoms, they have been simplified, and all open items from previous versions have been closed. The BWR EPGs extend well beyond the design bases and include many actions appropriate for severe accident management.

The improvement to EPGs is only as good as the plant-specific EOP implementation and the training that operators receive on use of the improved procedures. A recent staff safety evaluation report (Ltr. Thadani to Grace, "Safety Evaluation of 'BWR Owners' Group - Emergency Procedure Guidelines, Revision 4,' NEDO-31331, March 1987," dated September 12, 1988) encouraged licensees to implement Revision 4 of the EPGs and reiterated the need for proper implementation and training of operators. Implementation of the guidelines has been voluntary, but is strongly recommended in the SER.

Impact of Existing Requirements:

As part of the balanced approach, for completeness, and to provide a more accurate picture of Mark I plant risk, the staff has also evaluated the impact on Mark I risk of several recent rules that have been imposed on light water reactors - the Station Blackout Rule and the ATWS Rule. As discussed earlier, PRAs typically indicate that Mark I reactor risks are dominated by TW, SBO and ATWS sequences. Upon implementation of these two rules at all Mark I plants, risk from SBO and ATWS sequences would be expected to be reduced to a low level. The response to Question #2 in Enclosure 1 provides a discussion of expected risk reductions from changes to Mark I plants as a result of these rules.

Assuring the operability of the proposed improvements under severe accident conditions, including an extended period of station blackout, may require assurance of electrical power beyond the requirements of the recent Station Blackout (SBO) rule, 10 CFR 50.63. The proposed improvements have been coordinated with the requirements of this rule in order not to cause an undue proliferation of power supplies, which could be counter-productive to safety. The staff proposes that licensees intending to implement the SBO rule by use of an alternate AC (AAC) source, need not provide additional electric power supplies for the proposed Mark I improvements, provided that the capacity of the AAC is sufficient for the requirements of both the SBO rule and the improvements proposed here. Further details are given in Enclosures 7 and 8.

Benefit of Improvements:

The improvements that the staff is recommending include: (1) an improved hardened venting capability, (2) improved RPV depressurization systems reliability, (3) an alternative water supply to the reactor vessel and drywell sprays, and (4) emergency procedures and training. Accelerated implementation of the existing station blackout and ATWS rules is also planned. These improvements are unchanged from those indicated in the interim report (SECY 88-206) to the Commission.

A major benefit of these improvements is that they can provide a reduction in core melt frequency of about a factor of five to ten. With the proposed enhancements, the core melt frequency would be expected to be reduced to about 1 to 2×10^{-5} per reactor-year. It should be noted that these estimates apply to internal events only.

For plants with a high TW probability, a large fraction of the reduction in core melt frequency is attributable to improved venting which, by allowing the removal of long-term decay heat from the containment, greatly reduces the likelihood of core melt from the TW sequence. Another reduction in core melt frequency from station blackout is attributable to the enhancements taken together. In the event of station blackout, enhanced RPV depressurization reliability would permit depressurization of the reactor, availability of a low pressure backup source of water injection into the vessel would permit core cooling, while venting would allow decay heat removal from the containment. It is important to note that under these circumstances, venting would prevent core damage and not result in releases of fission products of any significance.

Accident mitigation benefits are also considered to be significant. Mitigation of fission product releases would be realized for all accident sequences, including ATWS. Venting would be effective in preventing containment failure arising from slow over-pressurization. Venting via the suppression pool would provide significant scrubbing of non-noble gas fission products by about a factor of 10 to 100 if no containment shell failure occurs. Water in the drywell may be effective in preventing or at least delaying failure of the shell by molten core debris. Finally, even if shell failure should occur, the presence of a water layer atop the core debris combined with the drywell spray would reduce any source term releases to the environment by a factor judged to range from 2 to 10.

Because of the combination of reduced core melt likelihood, reduced fission product releases due to mitigation, and possible reduction or elimination of a significant containment failure mode, the staff concludes that the overall risk reduction of the proposed improvements is in excess of one order of magnitude.

The benefits of the proposed enhancements in terms of their reduction in offsite risk can be calculated in terms of person-rem. Depending upon the probability of core melt due to the TW sequence the estimated reduction in risk, expressed in person-rem, for the proposed enhancements ranged from about 145 person-rem per reactor-year to about 1330 person-rem per reactor-year, for plants having a probability of core melt due to TW of $1x10^{-5}$ per reactor-year and $1x10^{-4}$ per reactor-year, respectively. Of this total value, the risk reduction produced by lowering the likelihood of core melt due to station blackout and mitigation of ATWS accounts for a reduction of about 33 to 210 person-rem per reactor-year. For plants whose probability of core melt due to the TW sequence is high (about 10^{-4} per reactor-year), the bulk of the risk reduction can be attributed to the large reduction in the TW sequence brought about by improved venting. Additional details are provided in Enclosure 4.

Finally, as noted earlier, the recommended improvements form a package in the sense that they complement one another in prevention or mitigation. This results in the maximum risk reduction when all are taken together.

Summary of Costs of Improvements:

Cost estimates were made of the proposed improvements. These are given in Enclosure 3 which provides a summary for all improvements that includes high and low estimates ranging from \$3.1 to \$1.6 million dollars. For purposes of the regulatory analysis included in Enclosure 4, a best estimate cost of \$2.0M has been used. Estimates of cost as high as \$7.3M were obtained based on actual costs of similar improvements at an existing Mark I plant. Actual costs at many plants may be less since, as shown in Enclosure 5, some plants already have many features of the proposed improvements.

Conclusions: Many of the proposed enhancements would require plant backfits. The staff has examined these in light of the backfit rule, 10 CFR 50.109. Section (a) 3 of that regulation indicates that the Commission shall require backfitting only when "there is a substantial increase in the overall protection of the public health and safety" and "that the direct and indirect costs ... are justified in view of this increased protection".

In reaching a conclusion with respect to the first test indicated above, the staff considered the effect of the proposed enhancements upon reductions in core melt frequency and improved containment performance. A major benefit of these enhancements is in their ability to reduce the likelihood of core melt. Core melt frequencies for BWR Mark I plants prior to any of the enhancements considered would be expected to range from about $1x10^{-4}$ to $2x10^{-5}$ per reactor year. With the combined enhancements, core melt frequency would be reduced by about a factor of five to ten. Thus, the proposed enhancements clearly offer a substantial reduction in core melt frequency. The core melt frequency reductions do not give credit for existing venting capability assumed in NUREG-1150 since the current venting capability at plants has significant uncertainty regarding its overall effectiveness.

The increased ability to cool core debris and to remove excess heat from the containment by venting, given the occurrence of an accident, is also expected to reduce the likelihood of containment failure, although this is not as readily quantifiable because of the uncertainty in core melt progression and shell meltthrough phenomenology which is discussed in Enclosure 6. In addition, the ability to scrub particulate fission products by use of venting through the suppression pool and by the use of a water layer atop any core debris also adds significant mitigative capability.

Since the proposed enhancements would be expected to reduce the likelihood of core melt by about a factor of five to ten, and provide significant additional accident mitigation capability as well, the staff concludes that the proposed enhancements do provide a substantial increase in the overall protection of the public health and safety.

With regard to the second or cost-benefit test required by the backfit rule, the discussion given earlier has shown that the costs of the enhancements are estimated to range from 1.6 to 3.1 million dollars per plant, although similar improvements at an existing Mark I plant may have cost about 7.3 million dollars. Both the estimated cost and the cost associated with an existing Mark I plant were used in the cost-benefit analysis. Based on the survey results for nine Mark I plants, the staff believes that many plants have some of these improvements already in place. Since the estimated benefits ranged from 3.6 to 33 million dollars per reactor based upon 1000 dollars per person-rem and an average remaining plant life of 25 years for Mark I plants, the staff concludes that the proposed enhancements are generally cost beneficial.

For the reasons stated above, the staff concludes that backfit of these proposed enhancements is warranted for all Mark I plants.

<u>Options:</u> 1. <u>Take no action.</u> Pro: No further resources would be

required. Con: This option would result in a situation where a number of enhancements to safety that the staff believes to be cost effective would not be implemented and closure of severe accident issues would not be obtained for Mark I plants.

2. Issue a generic letter. Pro: This option would be the quickest way to inform licensees of the staff's views and would require the least resources. Con: The generic letter can inform industry of the staff's finding, but can only request, not require, licensees to make changes to their facilities.

3. Issue an order. Pro: This option could be accomplished quickly and provide a regulatory requirement to implement the improvements. Con: This option could result in requests for hearings from both licensees and intervenors contesting the orders. A draft proposed order is included as Enclosure 7.

4. Initiate Rulemaking. Pro: This option would provide a regulatory basis for requiring the improvements. It is generally preferable to impose generic requirements by rule. Con: This option would require some staff resources and cause a delay in implementing the proposed improvements. A draft proposed rule is attached as Enclosure 8.

Recommendations: The proposed improvements could be implemented as a regulatory requirement either by use of orders or through rulemaking. Of these two viable options, although the staff considers that it is generally preferable to impose generic requirements by rule, the improvements could be carried out more quickly via orders and for this reason the staff recommends that orders be issued to require the improvements. The staff would also prepare an Environmental Assessment of venting of the containment using the improved hardware and procedures.

Coordination: OGC has no legal objections. The ACRS has reviewed these recommendations and will provide their comments separately.

Victor Stello, Jr.
Executive Director
for Operations

Enclosures:
1. Response to Commission Questions
2. Mark I Challenges and Relative Likelihood of Failure Modes
3. Summary of Costs
4. Regulatory Analysis
5. Results of Survey of Mark I Plants
6. Mark I Liner Melt Status
7. Draft Proposed Order
8. Draft Proposed Rule

Enclosure 7

In the Matter of)
)
) License No.
) Docket No.
)
)

ORDER MODIFYING LICENSE

I.

(Name of Licensee), (Licensee) is the holder of Operating License No. _____ issued by the Nuclear Regulatory commission (NRC/ Commission) on ____________________________. The license authorizes the licensee to operate (Name of Facility). The facility is a Boiling Water Reactor (BWR) located at the Licensee's site in ______________________________ which utilizes a Mark I containment.

II.

Probabilistic Risk Assessment (PRA) studies have been performed for a number of Boiling Water Reactors with Mark I containments. These studies indicate that, although the risk from the BWR Mark I is low, containment integrity could be challenged if a large scale core melt accident were to occur, principally due to the smaller size of the containment. The studies which have been performed indicate that BWR Mark I plant risks are dominated by Loss of Decay Heat Removal (TW), Station Blackout (SBO) and Anticipated Transient Without Scram (ATWS) sequences. The staff has concluded that the optimum way to reduce overall risk in BWR Mark I plants is to pursue a balanced approach utilizing accident prevention and mitigation. Based on this assessment, potential improvements have been identified in the following specific areas which, when implemented, will substantially enhance the overall safety of Mark I plants:

(1) Containment Pressure Relief Capability (Venting).

The capability to vent has long been recognized as important in reducing risk from operation of BWR Mark I facilities for loss of long term decay heat removal events. However, the vent path external to most of the existing containment penetrations typically consists of a ductwork system which has a low design pressure of only a few pounds per square inch (psi.) Venting under high pressure severe accident conditions would fail the ductwork, release the containment atmosphere into the reactor building, and potentially contaminate or damage equipment needed for accident recovery. Furthermore, with the existing hardware and procedures at some plants, it may not be possible to open or close the vent valves for some severe accident scenarios. A hard pipe vent capable of withstanding the anticipated severe accident pressure loadings would eliminate or minimize the consequences of these disadvantages. Other changes, in conjunction with proper operating procedures, would result in greatly reducing the probability of core melt due to the Loss of Decay of Heat Removal (TW) and Station Blackout (SBO) sequences.

(2) Reactor Pressure Vessel Depressurization System Reliability.

The reactor pressure vessel (RPV) depressurization system consists of relief valves which can be operated to depressurize the reactor coolant system. Actuation of these valves requires DC power. In an extended station blackout after station batteries have been depleted, the RPV depressurization system valves would not be available and the reactor would re-pressurize. With enhanced RPV depressurization system reliability, depressurization of the reactor coolant system would have a greater degree of assurance. Together with a low pressure alternate source of water injection into the reactor vessel, the major benefit of enhanced RPV depressurization reliability would be to provide an additional source of core cooling which could significantly reduce the likelihood of high pressure severe accidents, such as from the short-term station blackout.

Another important benefit is in the area of accident mitigation. Reduced reactor pressure would greatly reduce the possibility of core debris being expelled under high pressure, given a core melt and failure of the reactor pressure vessel. Use of the RPV depressurization would also delay containment failure and reduce the quantity and type of fission products ultimately released to the environment. In order to increase reliability of the RPV depressurization system assurance of electrical power beyond the requirements of existing regulations may be necessary. In addition, performance of the depressurization system valves needs to be reviewed for temperature capability during a severe accident.

(3) Alternate Water Supply for Drywell Spray/Vessel Injection.

An important proposed improvement would be to employ a backup or alternate supply of water and a pumping capability that is independent of normal and emergency AC power. By connecting this source to the low pressure residual heat removal (RHR) systems as well as to the existing drywell sprays, water could be delivered either into the reactor vessel or to the drywell, by use of an appropriate valving arrangement.

An alternate source of water injection into the reactor vessel would greatly reduce the likelihood of core melt due to station blackout or loss of long-term decay heat removal, as well as provide significant accident management capability.

Water for the drywell sprays would also provide significant mitigative capability to cool core debris, to cool the containment liner to delay or prevent failure, and to scrub airborne particulate fission products from the atmosphere.

(4) Emergency Procedures and Training.

A major element of the Mark I containment performance improvement evaluation involves emergency procedures and training. Proper operator actions can preclude milder events from progressing to core damage or core meltdow accidents, and can greatly mitigate the consequences of even severe accidents. Since a variety of unusual conditions can be present particularly for beyond design bases events, emergency training and symptom based procedures are essential for guiding the operation to those actions which provide the greatest measure of protection to the public.

NRC has recently reviewed and approved Revision 4 of the BWR Owners Group EPGs (General Electric Topical Report NEDO-31331, BWR Owner's Group "Emergency Procedure Guidelines, Revision 4," March 1987). Revision 4 to the BWR Owners Group EPG is a significant improvement over earlier versions in that they continue to be based on symptoms, they have been simplified, and all open items from previous versions have been closed. The BWR EPGs extend well beyond the design bases and include many actions appropriate for severe accident management. Since operator actions affect the risk for all severe accident scenarios, implementation of procedures based upon Revision 4 to the EPGs is important for maximizing overall risk reduction.

III.

Improvements in the above mentioned areas can provide a reduction in core melt frequency of about a factor of five to ten. Furthermore, accident mitigation benefits are also considered to be significant. Mitigation of fission product releases would be realized for all accident sequences, including ATWS. Venting would be effective in preventing containment failure arising from slow over-pressurization. Venting via the suppression pool would provide significant scrubbing of non-noble gas fission products by about a factor of 10 to 100 if no containment shell failure occurs. Water in the drywell may be effective in preventing or at least delaying failure of the shell by molten core debris. Finally, even if shell failure should occur, the presence of a water layer atop the core debris combined with the drywell spray would reduce any source term releases to the environment by a factor judged to range from 2 to 10. In sum, improvements in these areas would result in reduced core melt likelihood, reduced fission product releases due to mitigation, and possible reduction or elimination of a significant containment failure mode, and provide a substantial increase in the overall protection of the public health and safety. [STAFF: ADDRESS § 50.109]

IV.

Accordingly, in view of the foregoing, and pursuant to sections 103, 161b.,161i., 161o. and 182 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 C.F.R. § 2.204 and 10 C.F.R. Part 50, it is hereby ordered that the Licensee

A. Within 30 months of the date of this Order:

1. Provide its BWR Mark I containment with an exhaust line from the wetwell vapor space to a suitable release point (e.g., plant stack). The basic design objective shall be to provide sufficient venting capacity to

prevent long-term overpressure failure of containment. This "hard vent" system shall meet the following criteria:

a) The vent shall be sized such that under conditions of 1) constant heat input at a rate equal to 1% of rated thermal power, and 2) containment pressure equal to the vent setpoint pressure, the exhaust flow through the vent is sufficient to prevent the containment pressure from increasing.

b) The venting setpoint shall be set at or above containment design pressure. Capability of RPV depressurization system valves, torus vent valves, or other equipment should not limit the venting setpoint to less than containment design pressure.

c) The venting capability shall be available during severe accident conditions and for a period up to 24 hours beyond the onset of a station blackout.

d) The hardened vent path should include a means to prevent premature or inadvertent actuation.

e) The vent path up to and including the second containment isolation barrier should be designated safety Class 2.

f) The hard vent path shall be capable of withstanding, without loss of functional capability, experted venting conditions and the effects of potential combustion phenomena.

g) The hardened vent path shall have a radiation monitor, alarmed in control room and functional during extended station blackout.

2. Examine the reactor pressure vessel (RPV) depressurization system and make modifications to ensure its functional capability during severe accidents and during extended station blackout conditions. As a minimum, the following shall be provided: 1) the capability of the RPV depressurization system cables and components to withstand, without loss of functional capability, the environment in the containment during a severe accident prior to vessel failure, and 2) an alternate power supply system capable of opening and maintaining open at least one RPV depressurization system valve for up to 24 hours beyond the onset of station blackout. Any sources of electrical power required to assure the operability of the backup water supply, containment venting system, and RPV depressurization system during an extended station blackout should be coordinated with the requirements of 10 C.F.R. § 50.63, as follows: X

a) Those licensees who choose to implement the requirements of 10 C.F.R. § 50.63 by the use of an alternate AC (AAC) source need not provide any additional power supplies to comply with the provisions of this section, provided that the capacity, capability, and duration of the AAC can be shown to meet the requirements of both 10 C.F.R. § 50.63 and this section.

b) Those licensees who choose to implement the requirements of 10 C.F.R. § 50.63 solely by means of a coping analysis must provide additional power supplies of sufficient capacity and reliability to assure the operability of the backup water supply, containment venting system and RPV depressurization systems during an extended station blackout.

[STAFF: FIX ABOVE TO BE SPECIFIC FOR THE LICENSEE TO WHOM THIS IS ISSUED]

3. Provide at least one water supply system for the containment drywell spray which shall be functional during an extended station blackout. An extended station blackout is defined as loss of all normal and emergency AC power and loss of DC power due to depletion of station batteries. Operability of controls and valves during such an event may require an independent source of power such as a dedicated battery set or a means to recharge the station batteries. Water to the spray system from this supply shall be available by remote manual operation or by simple procedures for connection and startup which can be implemented during severe accident conditions.

The water supply system shall also be capable of delivering water to the reactor vessel once the vessel has been depressurized. The mass flow rate shall be equal to or greater than the boiling rate which would occur under depressurized, saturated conditions with a constant heat input rate equal to 1% of rated thermal power.

All valve realignments or other actions necessary to realize this capability shall be reasonably achievable during an extended station blackout. Instrumentation needed to realize these capabilities shall be functional in the expected accident conditions and should, as a minimum, include [to be determined].

4. Implement procedures based on Emergency Procedure Guidelines (EPGs) developed by the BWR Owners' Group. Initially, Revision 4 to the EPGs 1/ as modified by the staff safety evaluation 2/ should be used as the basis for the procedures. Subsequent revisions to the EPGs as developed by the BWR Owners Group (or equivalent) should be used to update the procedures in a timely fashion.

1/ BWR Owners' Group Emergency Procedure Guidelines, Revision 4, NEDO-31331, March, 1987.

2/ Letter from A. Thadani to D. Grace "Safety Evaluation at BWR Owners' Group Emergency Procedure Guidelines" dated 9/12/88.

B. Within 60 days, submit to the Director of Nuclear Reactor Regulation plans for implementation of the above improvements and a schedule for implementation.

The licensee or any person adversely affected by this Order may request a hearing within 30 days of the date of this Order. A request for hearing should be clearly marked as a "Request for Hearing" and shall be addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with copies to the Assistant General Counsel for Enforcement at the same address, the Regional Administrator, Region ___, and the NRC Resident Inspector, at (Plants affected). If a person other than the licensee requests a hearing, that person shall set forth with particularity the manner in which the petitioner's interest is adversely affected by this Order and should address the criteria set forth in 10 C.F.R. § 2.714(d).

If a hearing is requested, the Commission will issue an Order designating the time and place of the hearing. If a hearing is held, the issue to be considered shall be whether this Order should be sustained. Upon the failure to answer or request a hearing within the specified time, this Order shall be final without further proceedings.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Dated this __ day of
________, 198_

DRAFT PROPOSED RULE

SECTION 50.XX - SEVERE ACCIDENT REQUIREMENTS FOR BOILING-WATER REACTORS HAVING MARK I CONTAINMENTS

a) Applicability The requirements of this section apply to all boiling water reactors (BWR) having Mark I containments.

b) Requirements

1. Backup Water Suppy for Drywell Spray/Core Injection

All BWRs with a Mark I containment shall provide at least one water supply system for the containment drywell spray which shall be functional during an extended station blackout.[1] Water to the spray system from this supply shall be available by remote manual operation or by simple procedures for connection and startup which can be implemented during severe accident conditions.

The water supply system shall also be capable of delivering water to the reactor vessel once the vessel has been depressurized. The mass flow rate shall be equal to or greater than the boiling rate which would occur under depressurized, saturated conditions with a constant heat input rate equal to 1% of rated thermal power. All valve realignments or other actions necessary to realize this capability shall be achievable during an extended station blackout.

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1 An extended station blackout is defined as loss of all normal and emergency AC power and loss of DC power due to depletion of station batteries. Operability of controls and valves during such an event may require an independent source of power such as a dedicated battery set or a means to recharge the station batteries.

Instrumentation needed to realize this capability shall be functional in the expected accident conditions and should, as a minimum, include [to be determined].

2. Containment Venting

For BWR plants with a Mark I containment an exhaust line which is capable of withstanding expected venting conditions shall be provided from the wetwell vapor space to a suitable release point (e.g., plant stack). The basic design objective shall be to provide sufficient venting capacity to prevent long-term overpressure failure of containment. This "hard vent" system shall meet the following criteria:

a) The vent shall be sized such that under conditions of 1) constant heat input at a rate equal to 1% of rated thermal power, and 2) containment pressure equal to the vent setpoint pressure, the exhaust flow through the vent is sufficient to prevent the containment pressure from increasing.

b) The venting setpoint shall be set at or above containment design pressure. Capability of RPV depressurization system valves, torus vent valves, or other equipment should not limit the venting setpoint to less than containment design pressure.

c) The venting capability shall be available during severe accident conditions and for a period up to 24 hours beyond the onset of a station blackout.

d) The hardened vent path should include a means to prevent premature or inadvertent actuation.

e) The vent path up to and including the second containment isolation barrier should be designated safety Class 2.

f) The hard vent path shall accommodate effects of potential combustion phenomena and remain functional.

g) The hardened vent path shall have a radiation monitor, alarmed in control room and functional during extended station blackout.

3. Reactor Pressure Vessel Depressurization Capability

All licensees having BWR Mark I containments shall examine the reactor pressure vessel (RPV) depressurization system and make modifications to ensure its functional capability during severe accidents and during extended station blackout conditions. As a minimum, the following shall be provided: 1) the capability of the RPV depressurization system cables and components to withstand, without loss of functional capability, the environment in the containment during a severe accident prior to vessel failure; and 2) an alternate power supply system capable of opening and maintaining open at least one RPV depressurization system valve for up to 24 hours beyond the onset of station blackout. Coordination of this requirement with the Station Blackout Rule (10 CFR 50.63) is discussed in Section 4.C.2 below.

4. Procedures and Training

All BWRs with Mark I containments shall implement procedures based on Emergency Procedures Guidelines (EPGs) developed by the BWR Owner's Group. Initially, Revision 4 to the EPG's[2] as modified by the staff safety evaluation[3] should be used as the basis for the procedures. Subsequent revisions to the EPGs as developed by the BWR Owners Group (or equivalent) should be used to update the procedures in a timely fashion.

c) Implementation

1) Schedule

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2 BWR Owner's Group Emergency Procedure Guidelines, Revision 4, NEDO-31331, March, 1987.

3 Letter from A. Thadani to D. Grace "Safety Evaluation of BWR Owners Group Emergency Procedure Guidelines" dated September 12, 1988.

All licensees to whom this section applies shall submit their plans and anticipated schedule within 60 days after a final rule is issued which identifies any actions taken and those needed to be taken to comply with the requirements of this section. The requirements of this section shall be fully implemented within 30 months after a final rule is issued.

2) Co-ordination with requirements of the Station Blackout Rule (10 CFR 50.63)

Any sources of electrical power required to assure the operability of the backup water supply, containment venting system, and RPV depressurization system during an extended station blackout, as required in part (b) above, should be coordinated with the requirements of 10 CFR 50.63, as follows:

a) Those licensees who choose to implement the requirements of 10 CFR 50.63 by the use of an alternate AC (AAC) source need not provide any additional power supplies to comply with the provisions of this section, provided that the capacity, capability, and duration of the AAC can be shown to meet the requirements of both 10 CFR 50.63 and this section.

b) Those licensees who choose to implement the requirements of 10 CFR 50.63 solely by means of a coping analysis, must provide additional power supplies of sufficient capacity and reliability to assure the operability of the backup water supply, containment venting system and RPV depressurization systems during an extended station blackout.

Enclosure 3 to the Minutes of CRGR Meeting No. 155
Draft Final Rule on Standardization and Licensing Reform

Topic

S. Crockett (OGC) and J. Wilson (RES) presented for CRGR review a draft final rule (Part 52) on standardization and licensing reform. This package reflects resolution of public comments.

Background

The package submitted for review by CRGR in this matter was transmitted by memorandum dated January 6, 1989, S. Crockett to E. Jordan. The review package included the redrafted rule.

Conclusions/Recommendations

As a result of their review of this matter, including discussions with the staff at this meeting, the Committee made the following recommendations:

1. The rule specifies that an early site permit is valid for twenty years. The rule should allow for shorter permit times since many factors can change over a 20 year period.

2. As presently written (see p. 23 & p. 29), the rule places emphasis on testing a full-size prototype prior to certification. The rule should be neutral on this issue to permit testing and analysis in lieu of building a full-size prototype, since this can be an acceptable method for certifying a design.

3. The rule (p. 25) requires completion of a design-specific PRA and inclusion of the PRA in the application for design certification. A phrase should be added, such as "including an estimate of the uncertainties." PRA has too much uncertainty to compare to the safety goals without an inclusion of the uncertainty analysis.

4. Under section 52.47, Contents of Applications, delete ix(3) because it is redundant. The Atomic Energy Act already provides for acquisition of this information.

5. Under 52.47 (b)(2), the current language for prototype testing includes very specific conditions (i.e., normal, transient, and accident). This should be revised to read "over a suitable range of conditions" to avoid excessive testing, such as, beyond the design basis.

6. In section 52.63 (a)(2), clarify the language discussing design certification modifications by either switching the two sentences or modifying the words. As presently stated the meaning is ambiguous.

The Committee recommended in favor of forwarding the draft final rule to the Commission. The staff was requested to forward the Statement of Considerations to the Committee for review. The Committee will identify issues but will not meet again on this topic.