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

January 3, 1992

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

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

SUBJECT: MINUTES OF CRGR MEETING NUMBER 213

The Committee to Review Generic Requirements (CRGR) met on Thursday, December 19, 1991 from 8:00 a.m. to 1:30 p.m. A list of attendees at the meeting is enclosed (Enclosure 1). The following items were discussed at the meeting:

1. The CRGR continued its discussion of a draft Supplement 4 to Generic Letter 89-10. Review of this supplement was begun at Meeting No. 212 on December 10, 1991. The supplement would relax the staff's current position regarding position changeable motor operated valves for boiling water reactors. A majority of the CRGR recommended in favor of the supplement as proposed by the staff, with a minority of CRGR dissenting. This matter is discussed in Enclosure 2.

2. M. Jamgochian of RES and R. Hasselberg of NRR presented for CRGR review a draft proposed Revision 3 to Regulatory Guide 1.101 on emergency planning. The revision, which would be published for comment, would endorse industry developed guidance on emergency action levels as an acceptable alternative to the current staff guidance in NUREG-0654, Appendix 1. The CRGR recommended in favor of the proposed revision subject to some revisions to be coordinated with the CRGR staff. This matter is discussed in Enclosure 3.

3. C. E. Rossi and L. Phillips of NRR presented for CRGR review a draft Supplement 1 to Generic Letter 89-02 on reconstituting fuel assemblies. The supplement restricts the definition of approved methods which may be used by licensees in justifying fuel assembly reconstitution. The CRGR recommended in favor of the supplement subject to some revisions and receipt of a justification as to why the staff did not propose to modify certain existing technical specifications. These items were to be coordinated with the CRGR staff. This matter is discussed in Enclosure 4.

4. The CRGR discussed a draft proposed amendment to 10 CFR Parts 72 and 73 involving proposed relaxations to current reporting requirements, primarily for invalid actuation of certain engineered safety features. The CRGR agreed with the staff's proposal to defer CRGR review until the final rule stage, after receipt of public comments. This agreement was

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subject to the CRGR being informed if the backfit analysis in the draft package is substantially changed prior to publication. The matter is discussed in Enclosure 5.

Questions concerning these meeting minutes should be referred to Dennis Allison (492-4148).

E. L. Jordan

Edward L. Jordan, Chairman
Committee to Review Generic
Requirements

Enclosures:
As stated

cc w/encl:
Commission (5)
SECY
J. Lieberman
P. Norry
D. Williams
W. Parler
Regional Administrators
CRGR Members

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Original Signed by:
E. L. Jordan

Edward L. Jordan, Chairman
Committee to Review Generic
Requirements

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CRGR Members

Distribution:
Central File w/o encl.
PDR (NRC/CRGR) w/o encl.

S. Treby	P. Kadambi
M. Taylor	J. Sniezek
J. Heltemes	G. Mizuno
J. Richardson	E. Rossi
C. Grimes	W. Minners
E. Sullivan	M. Jamgochian
L. Phillips	T. Novak
P. Baranowski	P. Campbell
D. Ross	D. Allison
C. Conran	E. Jordan
CRGR C/F	CRGR S/F

DJA	Previous concurrence	
CRGR:AEOD	DD:AEOD	C/CRGR:AEOD
DAllison:slm	DRoss	EJordan
12/ /91	12/ /91	12/3/92
1/3/92		

ENCLOSURE 1

Attendance List

CRGR Meeting No. 212

December 10, 1991

CRGR Members

E. Jordan
F. Miraglia
G. Arlotto
J. Callan
J. Moore
B. Sheron

CRGR Staff

D. Allison
J. Conran

NRC Staff

D. Barss
S. Boynton
C. Ader
M. Jamgochian
A. Mohseni
R. Hasselberg
E. Weiss
N. P. Kadambi
T. Sullivan
J. Norberg
P. Campbell
T. Scarbrough
L. Cohen
F. Kantor
W. Minners
G. Mizuno
B. Erickson
J. Minns
C. E. Rossi
L. Phillips
P. Wen
R. Tripathi
J. Crooks

Enclosure 2 to the Minutes of CRGR Meeting No. 213
Draft Supplement 4 to Generic Letter 89-10
on Relaxing Staff Position regarding Position
Changeable Valves for Boiling Water Reactors (BWRs)

December 10, 1991

TOPIC

The supplement would, for BWR's, relax a current staff position. The position essentially indicates that motor operated valves which are position changeable from the control room should be examined to ensure that, in the event they are mispositioned during an accident or transient, they are capable of being returned to the proper position. It also indicates that they should be included in other testing and maintenance programs prescribed by IE Bulletin 85-03 and Generic Letter 89-10 and their supplements, such as checking torque switch settings.

The relaxation would withdraw this position for BWR's. Position changeable valves which have an active safety function would still have to be capable of being repositioned under the differential pressure or flow conditions contemplated in the original design basis for each valve; however, for some valves the differential pressure or flow created by mispositioning could be substantially greater than specified in the design bases. Position changeable valves which have an active safety function would have to be included in other specified testing and maintenance programs such as checking torque switch settings; however, passive valves would not.

The relaxation was proposed in response to an appeal by the BWR Owner's Group (BWROG) asserting that the staff position would not provide a substantial safety enhancement and was not justified as a backfit. It was supported by a staff contractor's PRA study which examined the potential effects of various assumed operator error and valve failure rates for some of the valves involved at three plants.

BACKGROUND

The background material was described in the Minutes of Meeting No. 212.

CONCLUSIONS/RECOMMENDATIONS

A majority of the CRGR recommended in favor of the staff's proposal. Two CRGR members dissented; they would support a relaxation if it were not applied to valves covered by the original IE Bulletin 85-03 (e.g., high pressure injection system valves).

The primary reason for one dissent was that such relaxation would be counter to a lesson from the June 9, 1985 Davis Besse event in which operators did misposition auxiliary feedwater containment isolation valves during the course of the response to the event and had to manually open the valves to restore feedwater.

In addition, there was concern over using PRA to disassemble parts of a generic action.

The other dissent was also concerned about lessons of the Davis Besse event, particularly that passive position changeable valves would no longer have to be included in other maintenance and testing programs for motor-operated valves prescribed in IE Bulletin 85-03 and Generic Letter 89-10 and their supplements, such as checking torque switch settings.

BACKFITTING

The staff determined that the absence of a direct review of this single aspect in the original value-impact analysis for the entire set of MOV positions did not constitute an inadequate backfitting process for this issue.

The staff also determined that this action (relaxing the position regarding position changeable valves) did not constitute backfitting.

The CRGR accepted these determinations.

Enclosure 3 to the Minutes of CRGR Meeting No. 213
Proposed Revision 3 to Reg. Guide 1.101 (to Endorse NUMARC Guidance on Dvelopment of Emergency Action Levels)

December 19, 1991

TOPIC

M. Jamgochian (RES) and R. Hasselberg (NRR) presented for CRGR review proposed Revision 3 to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Reactors". The purpose of the Reg. Guide revision is to endorse the guidance provided in NUMARC/NESP-007, Revision 1 as an acceptable alternative method to that described in NUREG-0654/FEMA-REP-1 for developing emergency action levels required by 10 CFR Part 50, Appendix E, Section IV.B. Briefing slides used by the staff to guide their presentation and discussion with the Committee at this meeting are enclosed (Attachment 1).

BACKGROUND

The package submitted to CRGR for review in this matter was transmitted by memorandum dated November 18, 1991, E.S. Beckjord to E.L. Jordan; the review package included the following documents:

1. Draft Revision 3, dated September 1991, to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Reactors",

2. Draft Regulatory Anlysis (undated), "Revision of Regulatory Guide 1.101 to Accept the Guidance in NUMARC/NESP-007, Rev. 1 as an Alternative Methodology for the Development of Emergency Action Levels",

3. Letter, dated May 16, 1991, T.E. Murley (NRC) to T.E. Tipton (NUMARC) regarding receipt of NUMARC submittal, and attachment:

   a. NUMARC/NESP-007, Revision 1, dated February 1991, "Methodology for Development of Emergency Action Levels".

CONCLUSIONS/RECOMMENDATIONS

As a result of their review of this matter, including the discussions with the staff at this meeting, the Committee recommended in favor of issuing for comment the proposed Revision 3 to Reg. Guide 1.101, subject to several clarifications and modifications discussed with the staff at this meeting, as follows:

1. In its current form, the NUMARC guidance gives too little emphasis to shutdown risk. A brief statement should be included in the "Discussion" section of the proposed Reg. Guide revision noting the current studies of shutdown risk that are underway, and cautioning that the guidance will likely change based on the results of those studies.

2. The language of the "Implementation" section of the proposed Reg. Guide revision should be revised to more clearly reflect the staff's intent

that no new positions or requirements will be imposed by this staff action, but rather implementation of the NUMARC guidance by licensees will be strictly on a voluntary basis.

3. In several places in the package, the NUMARC guidance is termed "generally" acceptable as an alternative method for development of emergency action levels by licensees. If the staff intends no exceptions to that guidance (and none are apparent in the draft package, as written), the word "generally" should be deleted.

4. The staff should note explicitly in the package issued for comment that implementation of the NUMARC guidance will result in a significant reduction in the numbers of Notifications of Unusual Events that are treated by licensees, under existing regulatory guidance, as emergency conditions and are routinely reported to NRC, and to state and local government entities as such.

5. It should be made clearer in the package that the staff is seeking comment on the new NUMARC guidance for developing EALs, not on the existing NRC guidance in this area (e.g., NUREG-0654).

6. Although the CRGR recommended in favor of issuing for comment the proposed Reg. Guide revision endorsing NUMARC/NESP-007, Rev. 1, the Committee felt that the discussion in the NUMARC document of the development process for the new guidance was self serving and did not credit the contribution of the extensive NRC staff interactions/comments in helping significantly to shape the final form of the new guidance. The CRGR does not support endorsing the introduction to the NUMARC document and recommended it be deleted.

7. In addition to the preceding, the Committee recommended changes to the package in specific locations, as indicated below:

<u>Proposed Rev.3 to Reg. Guide 1.101</u>:

a. At the bottom of p.1, revise the last sentence beginning on the page to read as follows:

> "In both cases, the NRC will make a finding after consideration of..(FEMA) findings...and the NRC assessmentment ...as to whether...capable of being implemented."

b. Delete the word "generally" (preceding the the word "acceptable") in the first sentences of both the first and second paragraph of "Section C. Regulatory Position". (See item 3. above.)

c. In the fourth line of the second paragraph in "Section C. Regulatory Position", insert "Appendix 1" after "NUREG-0654/FEMA-REP-1". Make conforming changes throughout the package as appropriate.

<u>Draft Regulatory Analysis</u>:

a. P.3, second and third paragraphs:

Delete the word "generally" immediately preceding the word "acceptable". (See item 3. above.)

b. P.3, under "2. Objectives":

Replace the existing second sentence with one that simply notes that industry perceived the need, based on experience, for an EAL development methodology to provide greater consistency in the identification and reporting of emergencies.

c. P.3, second paragraph:

See item 1. above, and note here the ongoing studies of shutdown risk. Also, add the caution that the results of the shutdown risk studies will likely necessitate revision of the new NUMARC guidance in the future.

d. P.7, third full paragraph on the page:

Add reference here to existing documentation (identified for CRGR at this meeting) of the NRC staff's point-by-point scenario comparisons between exisiting regulatory guidance on EALs and the new NUMARC guidance on EAL development. Also, place the referenced documentation in the Public Document Room.

e. Pp.8-19, Section 4.1:

Review for completeness the estimation of costs to state and local governments associated with implementation of the new NUMARC guidance, as discussed with the Committee at this meeting.

f. P.21, second bullet under "Disposition of NUREG-0654 Examples...":

Add an explicit comment as part of, or following, the last sentence noting that states will be getting significantly fewer Notifications of Unsual Events in the emergency context as a result of licensees implementing the new NUMARC guidance. (See item 4. above.)

g. P.24, second paragraph under "Regulatory Analysis":

Add a statement following the underlined words at the end of the paragraph highlighting that this will be a second big source of reduced emergency reporting to state and local government entities by licensees who implement the new NUMARC EAL guidance. (See item 4. above.)

h. P.26, second paragraph:

At the beginning of the first sentence of the paragraph, insert the word "emergency" in front of the words "onsite power capability".

i. P.27, under item 10. at the bottom of the page:

Add an explanation regarding the apparent time difference involved (i.e., 10 min. vs 15 min.) in treating this IC under the new NUMARC EAL guidance instead of existing reglatory guidance on EAL development.

j. P.29, item 13.:

Compare this item with item 17 at p.40. The escalation from Unusual Event to Alert status in the NUMARC scheme is understandable given the increase in severity of the natural phenomena involved; but what is the explanation for the restrictions to "Protected Area" (in item 13) and "Vital Area" (in item 17)? Are the natural phenomena involved so restricted in the existing NRC guidance referenced? Reexamine and clarify with NUMARC the intended distinction/categorization.

k. P.29, item 14.:

Reexamine the NUMARC rationale for not addressing subitem e. (i.e., turbine rotating components failure...) The rationale seems questionable in the light of the recent Salem event. (Similar comment regarding related item 18. at p.41.)

l. P. 36, item 8.:

See item 1. above, and consider the need to revise the wording here to reflect the current concerns regarding shutdown risk, and the likelihood that the results on ongong risk studies will necessitate revision of both existing NRC EAL guidance and the new NUMARC guidance as well.

m. P.43, item 1.:

The definition of "make-up capacity" ascribed here to NUMARC is incorrect. It is inconsistent with the conventional usage in the General Design Criteria and in many existing staff safety evaluations. This point should be clarified with NUMARC to ensure common technical understanding and to avoid unnecessary confusion on this important point. The NUMARC document should be changed in this regard, or an exception should be taken in proposed Rev.3 to Reg. Guide 1.101.

All changes made to the package in response to CRGR comments and recommendations should be closely coordinated with the CRGR staff.

BACKFITTING

The proposed Reg. Guide revision imposes no new or changed positions or requirements; implementation by licensees will be strictly on a voluntary basis. Therefore, this action does not involve backfitting.

REGULATORY GUIDE 1.101

EMERGENCY PLANNING AND PREPAREDNESS
FOR NUCLEAR REACTORS

Attachment 1
to Enclosure 3

PURPOSE

- TO PUBLISH FOR PUBLIC COMMENT REVISION 3 ENDORSING AN ALTERNATIVE EMERGENCY ACTION LEVEL (EAL) SCHEME TO APPENDIX 1 OF NUREG 0654

BACKGROUND

- APP. 1 OF NUREG 0654 PROVIDED EXAMPLE INITIATING CONDITIONS FOR EACH EMERGENCY CLASSIFICATION
- DEVELOPMENT OF ALTNERATIVE EAL SCHEME IS AN INDUSTRY INITIATIVE WHICH REFLECTS 11 YEARS OF EXPERIENCE
- DEVELOPED BY NUMARC WITH SIGNIFICANT INPUT FROM NRC. FEMA WAS ALSO INVOLVED.

IMPORTANT ASPECTS OF NUMARC EAL GUIDELINES

- NUMEROUS INDUSTRY - STAFF INTERACTIONS (1989 - 1991).
- TESTED AGAINST EXISTING BWR/PWR EAL SCHEMES AND COMPARED (INDEPTH) WITH NUREG-0654 APP. 1.
- NUMARC NESP-007 EXAMPLE INITIATING CONDITIONS PROVIDE AN ACCEPTABLE ALTERNATIVE METHODOLOGY TO THE EXAMPLE INITIATING CONDITIONS LISTED IN NUREG-0654 APP. 1.
- NUMARC PROVIDES EXAMPLE EAL'S FOR EACH INITIATING CONDITION.
- USES THE SAME EMERGENCY CLASSIFICATION LEVELS AS NUREG 0654

IMPORTANT ASPECTS OF NUMARC EAL GUIDELINES (CONTINUED)

- EACH EXAMPLE IC/EAL HAS A TECHNICAL BASIS AND MODE APPLICABILITY
- EACH EXAMPLE EAL HAS ONE OR MORE RECOMMENDED THRESHOLD VALUES FOR BWR AND/OR PWR FACILITIES
- NUMARC CLEARLY DEFINES "FISSION PRODUCT BARRIER LOSS" AND "POTENTIAL LOSS"

IMPORTANT ASPECTS OF NUMARC EAL GUIDELINES (CONTINUED)

- EAL THRESHOLDS UTILIZE MANY OF THE SAME OBSERVABLE AND QUANTIFIABLE PARAMETERS NOW USED IN PLANT EOP'S

  - TEMPERATURES
  - PRESSURES
  - VESSEL LEVELS
  - INJECTION FLOW RATES
  - FEEDWATER FLOW RATES
  - SUBCOOLING MARGIN
  - CONTAINMENT TEMPERATURE
  - CONTAINMENT PRESSURE
  - CONTAINMENT RADIATION
  - ISOLATION SYSTEM STATUS
  - ACTIVITY/RADIATION LEVELS
  - CRITICAL SAFETY FUNCTION STATUS

- CLEARLY LIMITS "DELAY TIMES" FOR THE RECOGNITION OF FAILED MITIGATION EFFORTS

NUMARC EAL SCHEME

AN EVENT-BASED EAL CLASSIFICATION SYSTEM INCORPORATING A FISSION PRODUCT BARRIER CHALLENGE/BREACH SCHEME

PREFIX A -- ABNORMAL RADIOLOGICAL CONDITIONS

PREFIX H -- HAZARDOUS CONDITIONS

PREFIX S -- SYSTEM MALFUNCTIONS

PREFIX F -- FISSION PRODUCT BARRIER CHALLENGES/BREACHES

EXAMPLES:

AU -- UNUSUAL EVENT BASED ON ABNORMAL RADIOLOGICAL CONDITIONS
HU -- UNUSUAL EVENT BASED ON HAZARDOUS CONDITIONS
SU -- UNUSUAL EVENT BASED ON SYSTEM MALFUNCTIONS
AA -- ALERT BASED ON ABNORMAL RADIOLOGICAL CONDITIONS
HA -- ALERT BASED ON HAZARDOUS CONDITIONS
FA -- ALERT BASED ON FISSION PRODUCT BARRIER CHALLENGES/BREACHES
SS -- SITE AREA EMERGENCY BASED ON SYSTEM MALFUNCTIONS
AS -- SITE AREA EMERGENCY BASED ON ABNORMAL RADIOLOGICAL CONDITIONS
SG -- GENERAL EMERGENCY BASED ON SYSTEM MALFUNCTIONS
HG -- GENERAL EMERGENCY BASED ON HAZARDOUS CONDITIONS
FG -- GENERAL EMERGENCY BASED ON FISSION PRODUCT BARRIER FAILURES

NUMARC NESP-007 EAL GUIDELINES

BACKGROUND MATERALS

NUMARC NESP-007 EAL GUIDELINES

- NUMEROUS INDUSTRY - STAFF INTERACTIONS (1989 - 1991)
- TESTED AGAINST EXISTING BWR/PWR EAL SCHEMES AND COMPARED (IN DEPTH) WITH NUREG-0654 APP. 1
- NUMARC NESP-007 EXAMPLE INITIATING CONDITIONS PROVIDE AN ACCEPTABLE ALTERNATIVE METHODOLOGY TO THE EXAMPLE INITIATING CONDITIONS LISTED IN NUREG-0654 APP. 1
- NUMARC PROVIDES EXAMPLE EAL'S FOR EACH INITIATING CONDITION
- EACH EXAMPLE IC/EAL HAS A TECHNICAL BASES AND MODE APPLICABILITY
- EACH EXAMPLE EAL HAS ONE OR MORE RECOMMENDED THRESHOLD VALUES FOR BWR AND/OR PWR FACILITIES
- EAL THRESHOLDS UTILIZE MANY OF THE SAME OBSERVABLE AND QUANTIFIABLE PARAMETERS NOW USED IN PLANT EOP'S
  - TEMPERATURES
  - PRESSURES
  - VESSEL LEVELS
  - INJECTION FLOW RATES
  - FEEDWATER FLOW RATES
  - SUBCOOLING MARGIN
  - CONTAINMENT TEMPERATURE
  - CONTAINMENT PRESSURE
  - CONTAINMENT RADIATION
  - ISOLATION SYSTEM STATUS
  - ACTIVITY/RADIATION LEVELS
  - CRITICAL SAFETY FUNCTION STATUS
- NUMARC CLEARLY DEFINES "FISSION PRODUCT BARRIER LOSS" AND "POTENTIAL LOSS"
- CLEARLY LIMITS "DELAY TIMES" FOR THE RECOGNITION OF FAILED MITIGATION EFFORTS

NUMARC EAL SCHEME

AN EVENT-BASED EAL CLASSIFICATION SYSTEM INCORPORATING
A FISSION PRODUCT BARRIER CHALLENGE / BREACH SCHEME

PREFIX A -- ABNORMAL RADIOLOGICAL CONDITIONS

PREFIX H -- HAZARDOUS CONDITIONS

PREFIX S -- SYSTEM MALFUNCTIONS

PREFIX F -- FISSION PRODUCT BARRIER CHALLENGES / BREACHES

EXAMPLES:

AU - UNUSUAL EVENT BASED ON ABNORMAL RADIOLOGICAL CONDITIONS

HU - UNUSUAL EVENT BASED ON HAZARDOUS CONDITIONS

SU - UNUSUAL EVENT BASED ON SYSTEM MALFUNCTIONS

AA - ALERT BASED ON ABNORMAL RADIOLOGICAL CONDITIONS

HA - ALERT BASED ON HAZARDOUS CONDITIONS

FA - ALERT BASED ON FISSION PRODUCT BARRIER CHALLENGES / BREACHES

SS - SITE AREA EMERGENCY BASED ON SYSTEM MALFUNCTIONS

AS - SITE AREA EMERGENCY BASED ON ABNORMAL RADIOLOGICAL CONDITIONS

SG - GENERAL EMERGENCY BASED ON SYSTEM MALFUNCTIONS

HG - GENERAL EMERGENCY BASED ON HAZARDOUS CONDITIONS

FG - GENERAL EMERGENCY BASED ON FISSION PRODUCT BARRIER FAILURES

RECOGNITION CATEGORY A

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

INITIATING CONDITION MATRIX

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
AU1 Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Technical Specifications for 60 Minutes or Longer. Op. Modes: All	AA1 Any Unplanned Release or Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times Radiological Technical Specifications for 15 Minutes or Longer. Op. Modes: All	AS1 Site Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR Whole Body or 500 mR Child Thyroid for the Actual or Projected Duration of the Release. Op. Modes: All	AG1 Site boundary Dose Resulting from an Actual or Imminent Release or Gaseous Radioactivity that Exceeds 1000 mR Whole Body or 5000 mR Child Thyroid for the Actual or Projected Duration of the Release Using Actual Meterology. Op. Modes: All
AU2 Unexpected Increase in Plant Radiation Levels or Airborne Concentration. Op. Modes: All	AA2 Major Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel. Op. Modes: All		
	AA3 Release of Radioactive Material or Increases in Radiation Levels Within the Facility that Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown. Op. Modes: All		

RECOGNITION CATEGORY H

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

INITIATING CONDITION MATRIX

UNUSUAL EVENT

- **HU1** Natural and Destructive Phenomena Occurring Within the Protected Area. Ops. Modes: All
- **HU2** Fire Within Protected ARea Boundary Not Extinguished Within 15 Minutes of Detection. Op. Modes: All
- **HU3** Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant. Op. Modes: All
- **HU4** Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant. Op. Modes: All
- **HU5** Other Conditions Existing Which in the Judgement of the Emergency Director Warrant Declaration of an Unusual Event. Op. Modes: All

ALERT

- **HA1** Natural and Destructive Phenomena Occurring Within Plant Vital Area. Op. Modes: All
- **HA2** Fire or Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown Op. Modes: All
- **HA3** Release of Toxic or Flammable Gases Within a Facility Structure Which Jeopardizes Operation of Systems Required to maintain safe operations or to Establish or Maintain Cold Shutdown. Op. Modes: All
- **HA4** Security Event in a Plant Protected Area. Op. Modes: All
- **HA5** Control Room Evacuation Has Been Initiated. Op. Modes: All
- **HA6** Other Conditions Existing which in the Judgement of the Emergency Director Warrant Declaration of an Alert. Op. Modes: All

SITE AREA EMERGENCY

- **HS1** Security Event in Plant Vital ARea. Op. Modes: All
- **HS2** Control Room Evacuation Has Been Initiated and Plant Control Cannot be Established. Op. Modes: All
- **HS3** Other Conditions Existing Which in the Judgement of the Emergency Director Warrant Declaration of a Site Area Emergency. Op. Modes: All

GENERAL EMERGENCY

- **HG1** Security Event Resulting in Loss of Ability to Reach and Maintain Cold Shutdown. Op. Modes: All
- **HG2** Other Conditions Existing Which in the Judgement of the Emergency Director Warrant Declaration of A General Emergency. Op. Modes: All

RECOGNITION CATEGORY F

FISSION PRODUCT BARRIER DEGRADATION

INITIATING CONDITION MATRIX

See Table 3 for BWR Example EALs
See Table 4 for PWR Example EALs

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
FU1 ANY Loss or ANY Potential Loss of Containment. Op Modes: Power Operation Hot Standby/Startup (BWR) Hot Shutdown	FA1 ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS. Op. Modes: Power Operation Hot Standby/Startup (BWR) Hot Shutdown	FS1 Loss of BOTH Fuel Clad AND RCS OR Potential Loss of BOTH Fuel Clad AND RCS OR Potential Loss of EITHER Fuel Clad OR RCS, and Loss of ANY Additional Barrier. Op Modes; Power Operation Hot Standby/Startup (BWR) Hot Shutdown	FG1 Loss of ANY Two Barriers AND Potential Loss of Third Barrier. Op. Modes: Power Operation Hot Standby/Startup (BWR) Hot Shutdown

NOTES:

1. Although the logic used for these initiating conditions appears overly complex, it is necessary to reflect the following considerations:

   - The Fuel Clad barrier and the RCS barrier are weighted more heavily than the Containment barrier (see Sections 3.4 and 3.8 for more information on this point). Unusual Event ICs associated with RCS and Fuel Clad barriers are addressed under System Malfunction ICs.

   - At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from General Emergency. For example, if Fuel Clad barrier and RCS barrier "Loss" EALs existed, this would indicate to the Emergency Director that, in addition to offsite dose assessments, continual assessments of radioactive inventory and containment integrity must be focused on. If, on the other hand, both Fuel Clad barrier and RCS barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

   - The ability to escalate to higher emergency classes as an event gets worse must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

2. Fission Product Barrier ICs must be capable of addressing event dynamics. Thus, the EAL Reference Tables 3 and 4 state that IMMINENT (i.e., within 1 to 2 hours) loss or Potential Loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

TABLE 3
BWR EMERGENCY ACTION LEVEL
FISSION PRODUCT BARRIER REFERENCE TABLE
THRESHOLDS FOR LOSS OR POTENTIAL LOSS OF BARRIERS

* Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event (or multiple events could occur which result in the conclusion that exceeding the Loss or Potential Loss thresholds is IMMINENT (i.e., within 1 to 3 hours). In this IMMINENT LOSS situation use judgement and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY Loss or ANY Potential Loss of Containment	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS	Loss of BOTH Fuel Clad AND RCS OR Potential Loss of BOTH Fuel Clad AND RCS OR Potential Loss of EITHER Fuel Clad OR RCS, and Loss of ANY Additional Barrier	Loss of ANY Two Barriers AND Potential Loss of Third Barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCS BARRIER EXAMPLE EALS		PRIMARY CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
1. Primary Coolant Activity Level		**1. RCS Leak Rate**		**1. Drywell Pressure**	
Coolant activity GREATER THAN (site-specific) value	Not Applicable	(site-specific) Indication of Main Steamline Break	RCS leakage GREATER THAN 50 GPM inside the drywell OR unisolable primary sys leakage outside drywell as indicated by area temp or area rad alarm	Rapid unexplained decrease following initial increase OR Drywell pressure response not consistent with LOCA conditions	(site-specific) PSIG and increasing OR explosive mixture exists
	OR		OR		OR
2. Reactor Vessel Water Level		**2. Drywell Pressure**		**2. Containment Isolation Valve After Containment Isolation**	
Level LESS THAN (site-specific) Value	Level LESS Than (site-specific value	Pressure GREATER THAN (site-specific) psig	Not applicable	Failure of both valves in any one line to close AND downstream pathway to the environment exists	Not applicable
				OR	
				Intentional venting per EOPs	Not applicable
				OR	
				Unisolable primary sys leakage outside dry well as indicated by area temp or area rad alarm	Not applicable
	OR		OR	OR	
3. Drywell Radiation Monitoring		**3. Drywell Radiation Monitoring**		**3. Significant Radioactive Inventory in Containment**	
Drywell Rad Monitor Reading GREATER THAN (site-specific) R/hr	Not applicable	Drywell Rad Monitor Reading GREATER THAN (site-specific) R/hr	Not applicable	Not applicable	Containment radmonito reading GREATER THA (site-specific) R/h
	OR		OR	OR	
4. Other (site-specific) Indications		**4. Reactor Vessel Water Level**		**4. Reactor Vessel Water Level**	
(Site-specific) as applicable	(site-specific) as applicable	Level LESS THAN (site-specific) value	Not applicable	Not applicable	Reactor vessel water level LES THAN (site-specific) value an the maximum core uncovery tim limit is in the UNSAFE regio
	OR		OR		OR

TABLE 3

BWR EMERGENCY ACTION LEVEL

FISSION PRODUCT BARRIER REFERENCE TABLE

THRESHOLDS FOR LOSS OR POTENTIAL LOSS OF BARRIERS[a]

[a] Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event (or multiple event could occur which result in the conclusion that exceeding the Loss or Potential Loss thresholds to IMMINENT (i.e., within 1 to 3 hours). In this IMMINENT LOSS situatio use judgement and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY Loss or ANY Potential Loss of Containment	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS	Loss of BOTH Fuel Clad AND RCS OR Potential Loss of BOTH Fuel Clad AND RCS OR Potential Loss of EITHER Fuel Clad OR RCS, and Loss of ANY Additional Barrier	Loss of ANY Two Barriers AND Potential Loss of Third Barrier

<u>5. Emergency Director Judgement</u> Any condition in the judgement of the Emergency Director that indicates Loss or Potential Loss of the FUEL CLAD barrier	<u>5. Other (Site-specific) Indications</u> (Site-specific) as applicable (site-specific) as applicable OR <u>6. Emergency Director Judgement</u> Any condition in the judgement of the Emergency Director that indicate Loss or Potential Loss of the RCS barrier	<u>5. Other (site-specific) Indications</u> (site-specific) as applicable (site-specific) applicable OR <u>6. Emergency Director Judgement</u> Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the CONTAINMENT barrier

TABLE 4
PWR EMERGENCY ACTION LEVEL
FISSION PRODUCT BARRIER REFERENCE TABLE
THRESHOLDS FOR LOSS OR POTENTIAL LOSS OF BARRIERS*

* Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event (or multiple events) could occur which result in the conclusion that exceeding the Loss or Potential Loss thresholds in IMMINENT (i.e., within 1 to 2 hours). In this IMMINENT LOSS situation use judgement and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY Loss or ANY Potential Loss of Containment	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCB	Loss of BOTH Fuel Clad AND RCB OR Potential Loss of BOTH Fuel Clad AND RCB OR Potential Loss of EITHER Fuel Clad OR RCB, and Loss of ANY Additional Barrier	Loss of ANY Two Barriers AND Potential Loss of Third Barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCB BARRIER EXAMPLE EALS		CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
1. Critical Safety Function Status		1. Critical Safety Function Status		1. Critical Safety Function Status	
Core Cooling-Red	Core Cooling-Orange OR Heat Sink-Red	Not applicable	RCB Integrity-Red OR Heat Sink-Red	Not applicable	Containment-Red
OR		OR		OR	
2. Primary Coolant Activity Level		2. RCB Leak Rate		2. Containment Pressure	
Coolant Activity GREATER THAN (site-specific) value	Not applicable	GREATER THAN available makeup capacity as indicated by a loss of RCB subcooling	Unisolable leak exceeding the capacity of one charging pump in the normal charging mode	Rapid unexplained decrease following initial increase OR containment pressure or sump level response not consistent with LOCA conditions.	(Site-specific) PSIG and increasing OR Explosive mixture exists. OR Containment pressure greater that containment depressurization system setpoint with less than one full train of depressurization equipment operating.
OR		OR		OR	
3. Core Exit Thermocouple Readings		3. SG Tube Rupture		3. Containment Isolation Valves Status After Containment Isolation	
GREATER THAN (site-specific) degree F	GREATER THAN (site-specific) degree F	(Site-specific) indication that a SG is Ruptured and Has a Non-Isolable secondary line break OR (Site-specific) indication that a SG is ruptured and a prolonged release of contaminated secondary coolant is occurring from the affected SG to the environment	Site-specific indication that a SG is ruptured and the Primary-to-Secondary leak rate exceeds the capacity of one charging pump in the normal charging mode	Valve(s) not closed AND downstream pathway to the environment exists	Not applicable
OR		OR		OR	

TABLE 4
PWR EMERGENCY ACTION LEVEL
FISSION PRODUCT BARRIER REFERENCE TABLE
THRESHOLDS FOR LOSS OR POTENTIAL LOSS OF BARRIERS*

* Determine which combination of the three barriers are lost or have a potential loss and use the following key to classify the event. Also an event (or multiple events) could occur which result in the conclusion that exceeding the Loss or Potential Loss thresholds in IMMINENT (i.e., within 1 to 2 hours). In this IMMINENT LOSS situation use judgement and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY Loss or ANY Potential Loss of Containment	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCB	Loss of BOTH Fuel Clad AND RCB OR Potential Loss of BOTH Fuel Clad AND RCB OR Potential Loss of EITHER Fuel Clad OR RCB, and Loss of ANY Additional Barrier	Loss of ANY Two Barriers AND Potencial Loss of Third Barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCB BARRIER EXAMPLE EALS		CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
4. Reactor Vessel Water Level		4. Containment Radiation Monitoring		4. SG Secondary Side Release With Primary to Secondary Leakage	
Not applicable	Level LESS than (site-specific) value	Containment rad monitor reading GREATER THAN (site-specific) R/hr	Not applicable	Release of secondary side to atmosphere with primary to secondary leakage GREATER THAN tech spec allowable	Not applicable
OR		OR		OR	
5. Containment Radiation Monitoring		5. Other (Site-Specific) Indications		5. Significant Radioactive Inventory in Containment	
Containment rad monitor reading GREATER THAN (site-specific) R/hr	Not applicable	(Site-Specific) as applicable	(Site-Specific) as applicable	Not applicable	Containment rad monitor reading GREATER THAN (site-specific) R/hr
OR		OR		OR	
6. Other (Site-Specific) Indications		6. Emergency Director Judgement		6. Core Exit Thermocouple Readings	
(Site-Specific) as applicable	(Site-Specific) as applicable	Any condition in the opinion of the Emergency Director that indicate loss or potential loss of the RCS barrier		Not applicable	Core exit thermocouples in excess of 1200° and restoration procedures not effective within 15 minutes; or, core exit thermocouples in excess of 700° with reactor vessel level below top of active fuel and restoration procedures not effective within 15 minutes
OR		OR		OR	

TABLE 4
PWR EMERGENCY ACTION LEVEL
FISSION PRODUCT BARRIER REFERENCE TABLE
THRESHOLDS FOR LOSS OR POTENTIAL LOSS OF BARRIERS*

* Determine which combination of the three barriers are lost or have a potential loss and use the following key to cla[illegible] the event. Also an event (or multiple events) could occur which result in the conclusion that exceeding the Loss or Potential Loss thresholds is IMMINENT ([illegible], within 1 to 2 hours). In this IMMINENT LOSS situation use judgement and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY Loss or ANY Potential Loss of Containment	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCB	Loss of BOTH Fuel Clad AND RCB OR Potential Loss of BOTH Fuel Clad AND RCB OR Potential Loss of EITHER Fuel Clad OR RCB, and Loss of ANY Additional Barrier	Loss of ANY Two Barriers AND Potential Loss of Third [illegible]

7. Emergency Director Judgement

Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the FUEL CLAD barrier

7. Other (Site-Specific) Indications

(Site-Specific) as applicable

(Site-Specific) as applicable

OR

8. Emergency Director Judgement

Any condition in the opinion of the Emergency Director indicates Loss or Potential Loss of the CONTAINMENT ba[illegible]

RECOGNITION CATEGORY S
SYSTEM MALFUNCTION
INITIATING CONDITION MATRIX

UNUSUAL EVENT

SU1 Loss of All Offsite Power To Essential Busses for Greater Than 15 Minutes. Op. Modes: All

SU2 Inability to Reach Required Shutdown Within technical Specification Limits. Op. Modes: Power Operation Hot Standby Hot Shutdown

SU3 Unplanned Loss of most or all All Safety System Annunciators for Greater Than 15 Minutes. Op. Modes: Power Operation Hot Standby Hot Shutdown

SU4 Fuel Clad Degradation. Op. Modes: All

SU5 RCS Leakage. Op. Modes: Power Operation Hot Standby Hot Shutdown Cold Shutdown

SU6 Unplanned Loss of All Onsite or Offsite Communication Capabilities. Op. Modes: All

SU7 Unplanned Loss of Required DC Power During Cold Shutdown or Refueling Mode for Greater Than 15 Minutes. Op. Modes: Cold Shutdown Refueling

ALERT

SA1 Loss of All Offsite Power and Loss of All Onsite AC Power During Cold Shutdown or Refueling Mode. Op. Modes: Cold Shutdown Refueling Defueled

SA2 Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was Successful. Op. Modes: Power Operation Hot Standby

SA3 Inability to Maintain Plant in Cold Shutdown. Op. Modes: Cold Shutdown Refueling

SA4 Unplanned Loss of All Safety System Annunciators With Transient in Progress. Op. Modes: Power Operation Hot Standby Hot Shutdown

SA5 Loss of All Offsite Power to Essential Buses for Greater than 15 Minutes with Degraded Onsite Power Capabilities. Op. Modes: Power Operations Hot Standby Hot Shutdown

SITE AREA EMERGENCY

SS1 Loss of All Offsite Power and Loss of All Onsite AC Power. Op. Modes: Power Operation Hot Standby Hot Shutdown

SS2 Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once A Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was NOT Successful. Op. Modes: Power Operation

SS3 Loss of All Vital DC Power. Op. Modes: Power Operation Hot Standby Hot Shutdown

SS4 Complete Loss of Function Needed to Achieve or Maintain Hot Shutdown. Op. Modes: Power Operation Hot Standby Hot Shutdown

SS5 Loss of Water Level That Has or Will Uncover Fuel in the Reactor Vessel. Op. Modes: Cold Shutdown Refueling

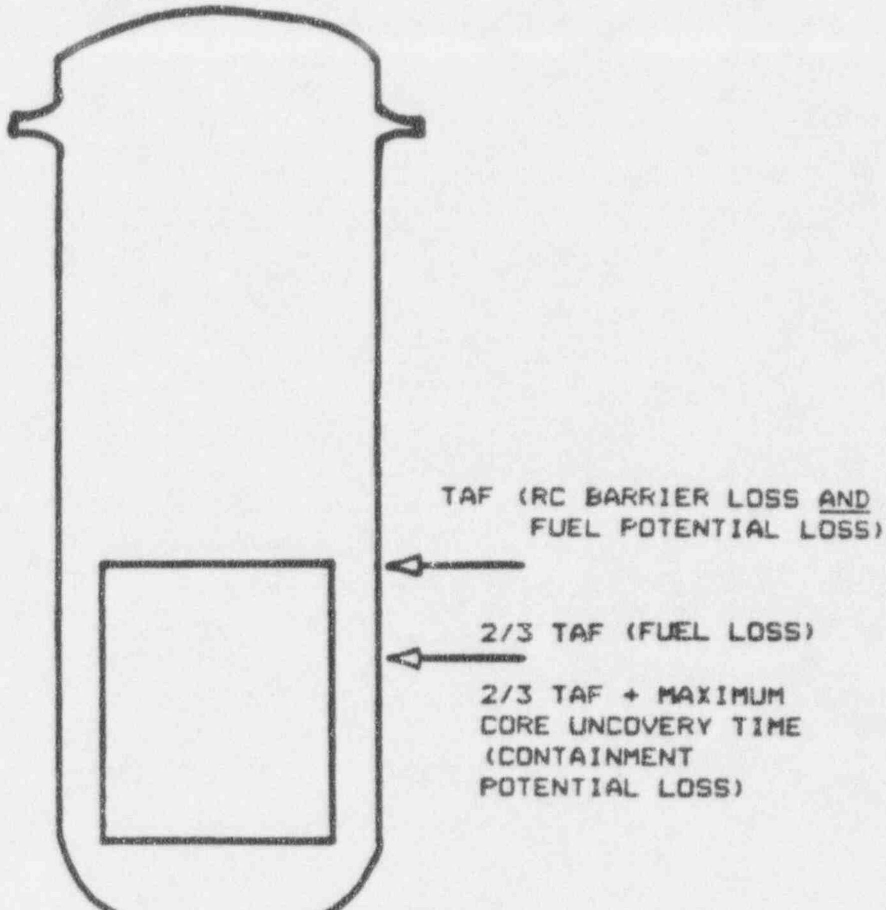
SS6 Inability to Monitor a Significant Transient in Progress Op. Modes: Power Operation Hot Standby Hot Shutdown

GENERAL EMERGENCY

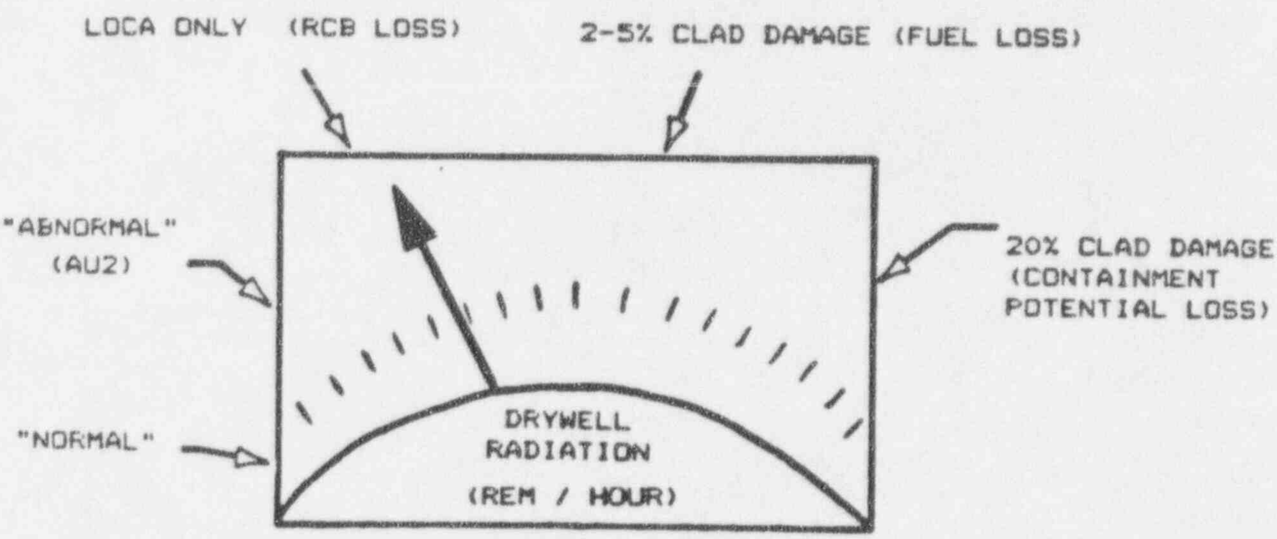
SG1 Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power. Op. Modes: Power Operation Hot Standby Hot Shutdown

SG2 Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was Not Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core Op. Mode: Power Operation

BOILING WATER REACTOR

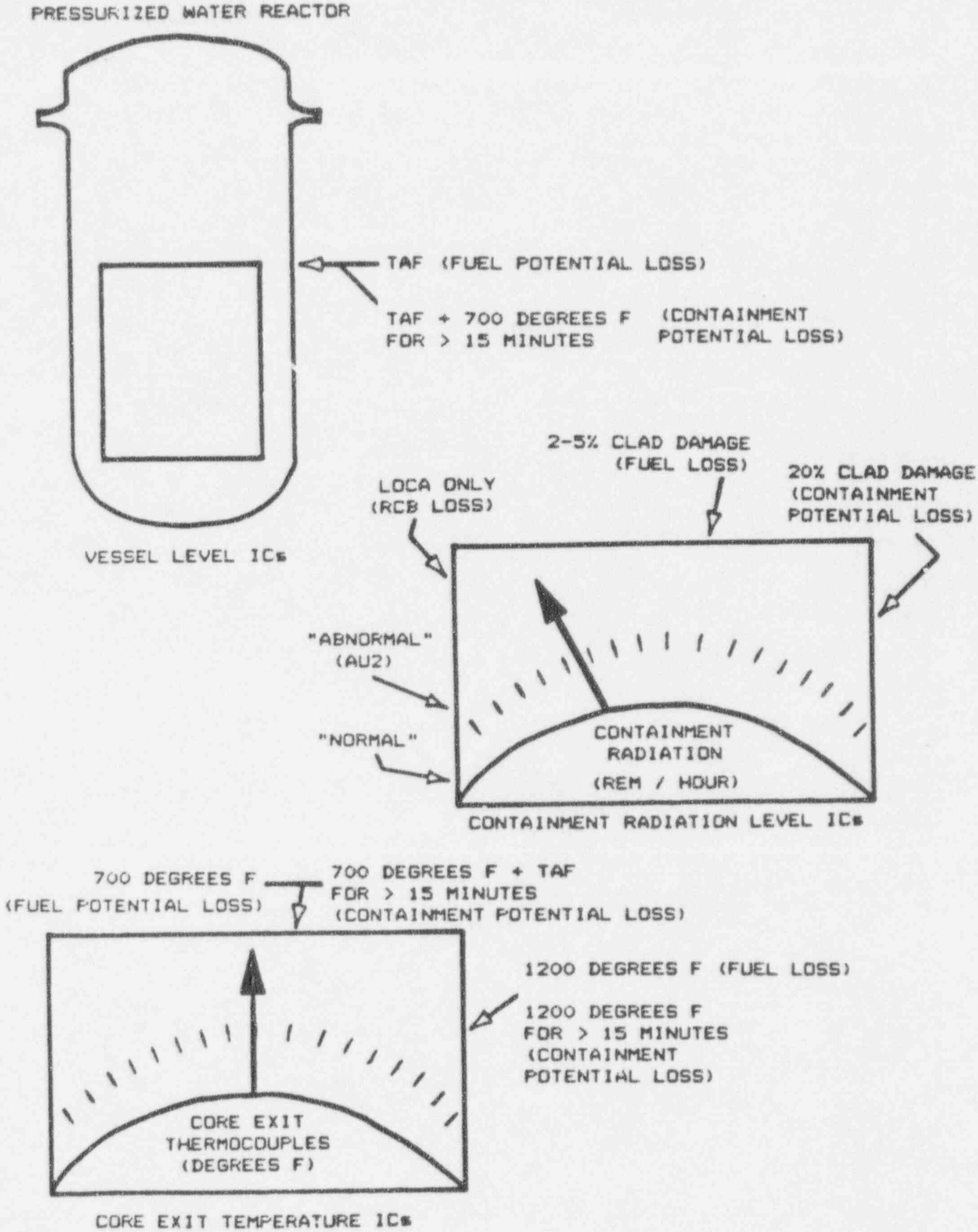


VESSEL LEVEL ICs



DRYWELL RADIATION LEVEL ICs

NUMARC INITIATING CONDITIONS FOR
BWR VESSEL LEVEL AND DRYWELL RADIATION LEVEL



NUMARC INITIATING CONDITIONS FOR PWR VESSEL LEVEL, CONTAINMENT RADIATION LEVEL AND CORE TEMPERATURE

NUREG-0654 APPENDIX 1
EXAMPLE INITIATING CONDITIONS
NOTIFICATION OF UNUSUAL EVENT

1. Emergency Core Cooling System (ECCS) initiated and discharge to vessel

2. Radiological effluent technical specification limits exceeded

3. Fuel damage indication. Examples:

   a. High offgas at BWR air ejector monitor (greater than 500,000 uci/sec; corresponding to 16 isotopes decayed to 30 minutes; <u>or</u> an increase of 100,000 uci/sec within a 30 minute time period)

   b. High coolant activity sample (e.g., exceeding coolant technical specifications for iodine spike)

   c. Failed fuel monitor (PWR) indicates increase greater than 0.1% equivalent fuel failures within 30 minutes

4. Abnormal coolant temperature and/or pressure or abnormal fuel temperatures outside of technical specification limits

5. Exceeding either primary/secondary leak rate technical specification or primary system leak rate technical specification

6. Failure of a safety or relief valve in a safety related system to close following reduction of applicable pressure

7. Loss of offsite power or loss of onsite AC power capability

8. Loss of containment integrity requiring shutdown by technical specifications

9. Loss of engineered safety feature or fire protection system function requiring shutdown by technical specifications (e.g., because of malfunction, personnel error or procedural inadequacy)

10. Fire within the plant lasting more than 10 minutes

11. Indications or alarms on process or effluent parameters not functional in control room to an extent requiring plant shutdown or other significant loss of assessment or communication capability (e.g., plant computer, Safety Parameter Display System, all meteorological instrumentation)

12. Security threat or attempted entry or attempted sabotage

13. Natural phenomenon being experienced or projected beyond usual levels

    a. Any earthquake felt in-plant or detected on station seismic instrumentation

    b. 50 year floor or low water, tsunami, hurricane surge, seiche

    c. Any tornado on site

    d. Any hurricane

14. Other hazards being experienced or projected

    a. Aircraft crash on-site or unusual aircraft activity over facility

    b. Train derailment on-site

    c. Near or onsite explosion

    d. Near or onsite toxic or flammable gas release

    e. Turbine rotating component failure causing rapid plant shutdown

15. Other plant conditions exist that warrant increased awareness on the part of a plant operating staff or State and/or local offsite authorities or require plant shutdown under technical specification requirements or involve other than normal controlled shutdown (e.g., cooldown rate exceeding technical specification limits, pipe cracking found during operation)

16. Transportation of contaminated injured individual from site to offsite hospital

17. Rapid depressurization of PWR secondary side.

NUREG-0654 APPENDIX 1
EXAMPLE INITIATING CONDITIONS
ALERT

1. Severe loss of fuel cladding

   a. High offgas at BWR air ejector monitor (greater than 5 ci/sec; corresponding to 16 isotopes decayed 30 minutes)

   b. Very high coolant activity sample (e.g., 300 uci/cc equivalent of I-131)

   c. Failed fuel monitor (PWR) indicates increase greater than 1% fuel failures within 30 minutes or 5% total fuel failures.

2. Rapid gross failure of one steam generator tube with loss of offsite power

3. Rapid failure of steam generator tubes (e.g., several hundred gpm primary to secondary leak rate)

4. Steam line break with significant (e.g., greater than 10 gpm) primary to secondary leak rate (PWR) or MSIV malfunction causing leakage (BWR)

5. Primary coolant leak rate greater than 50 gpm

6. Radiation levels or airborne contamination which indicate a severe degradation in the control of radioactive materials (e.g., increase of factor of 1000 in direct radiation readings within facility)

7. Loss of offsite power and loss of all onsite AC power (see Site Area Emergency for extended loss)

8. Loss of all onsite DC power (See Site Area Emergency for extended loss)

9. Coolant pump seizure leading to fuel failure

10. Complete loss of any function needed for plant cold shutdown

11. Failure of the reactor protection system to initiate and complete a scram which brings the reactor subcritical

12. Fuel damage accident with release of radioactivity to containment or fuel handling building

13. Fire potentially affecting safety systems

14. Most or all alarms (annunciators) lost

15. Radiological effluents greater than 10 times technical specification instantaneous limits (an instantaneous rate which, if continued over 2 hours, would result in about 1 mr at the site boundary under average meteorological conditions)

16. Ongoing security compromise

17. Severe natural phenomena being experienced or projected

    a. Earthquake greater than OBE levels

    b. Flood, low water, tsunami, hurricane surge, seiche near design levels

    c. Any tornado striking facility

    d. Hurricane winds near design basis level

18. Other hazards being experienced or projected

    a. Aircraft crash on facility

    b. Missile impacts from whatever source on facility

    c. Known explosion damage to facility affecting plant operation

    d. Entry into facility environs of uncontrolled toxic or flammable gases

    e. Turbine failure causing casing penetration

19. Other plant conditions exist that warrant precautionary activation of technical support center and placing near-site Emergency Operations Facility and other key emergency personnel on standby

20. Evacuation of control room anticipated or required with control of shutdown systems established from local stations

NUREG-0654 APPENDIX 1
EXAMPLE INITIATING CONDITIONS
SITE AREA EMERGENCY

1. Known loss of coolant accident greater than makeup pump capacity

2. Degraded core with possible loss of coolable geometry (indicators should include instrumentation to detect inadequate core cooling, coolant activity and/or containment radioactivity levels)

3. Rapid failure of steam generator tubes (several hundred gpm leakage) with loss of offsite power

4. BWR steam line break outside containment without isolation

5. PWR steam line break with greater than 50 gpm primary to secondary leakage and indication of fuel damage

6. Loss of offsite power <u>and</u> loss of onsite AC power for more than 15 minutes

7. Loss of all vital onsite DC power for more than 15 minutes

8. Complete loss of any function needed for plant hot shutdown

9. Transient requiring operation of shutdown systems with failure to scram (continued power generation but no core damage immediately evident)

10. Major damage to spent fuel in containment or fuel handling building (e.g., large object damages fuel or water loss below fuel level)

11. Fire compromising the functions of safety systems

12. Most or all alarms (annunciators) lost and plant transient initiated or in progress

13. a. Effluent monitors detect levels corresponding to greater than 50 mr/hr for 1/2 hour <u>or</u> greater than 500 mr/hr W.B. for two minutes (or five times these levels to the thyroid) at the site boundary <u>for adverse meteorology</u>

    b. These dose rates are projected based on other plant parameters (e.g., radiation level in containment with leak rate appropriate for existing containment pressure) or are measured in the environs

    c. EPA Protective Action Guidelines are projected to be exceeded outside the site boundary

14. Imminent loss of physical control of the plant

15. Severe natural phenomena being experienced or projected with plant not in cold shutdown

    a. Earthquake greater than SSE levels

    b. Flood, low water, tsunami, hurricane surge, seiche greater than design levels or failure of protection of vital equipment at lower levels

    c. Sustained winds or tornadoes in excess of design levels

16. Other hazards being experienced or projected with plant not in cold shutdown

    a. Aircraft crash affecting vital structures by impact or fire

    b. Severe damage to safe shutdown equipment from missiles or explosion

    c. Entry of uncontrolled flammable gases into vital areas. Entry of uncontrolled toxic gases into vital areas where lack of access to the area constitutes a safety problem

17. Other plant conditions exist that warrant activation of emergency centers and monitoring teams or a precautionary notification to the public near the site

18. Evacuation of control room and control of shutdown systems not established from local stations in 15 minutes

NUREG-0654 APPENDIX 1
EXAMPLE INITIATING CONDITIONS
GENERAL EMERGENCY

1. a. Effluent monitors detect levels corresponding to 1 rem/hr W.B. or 5 rem/hr thyroid at the site boundary under actual meteorological conditions

   b. These dose rates are projected based on other plant parameters (e.g., radiation levels in containment with leak rate appropriate for existing containment pressure with some confirmation from effluent monitors) or are measured in the environs

   Note: Consider evacuation only within about 2 miles of the site boundary unless these site boundary levels are exceeded by a factor of 10 or projected to continue for 10 hours or EPA Protective Action Guideline exposure levels are predicted to be exceeded at longer distances

2. Loss of 2 of 3 fission product barriers with a potential loss of 3rd barrier, (e.g., loss of primary coolant boundary, clad failure, and high potential for loss of containment)

3. Loss of physical control of the facility

   Note: Consider 2 mile precautionary evacuation

4. Other plant conditions exist, from whatever source, that make release of large amounts of radioactivity in a short time period possible, e.g., any core melt situation. See the specific PWR and BWR sequences below.

5. Example PWR Sequences

   a. Small and large LOCA's with failure of ECCS to perform leading to severe core degradation or melt in from minutes to hours. Ultimate failure of containment likely for melt sequences. (Several hours likely to be available to complete protective actions unless containment is not isolated)

   b. Transient initiated by loss of feedwater and condensate systems (principal heat removal system) followed by failure of emergency feedwater system for extended period. Core melting possible in several hours. Ultimate failure of containment likely if core melts.

   c. Transient requiring operation of shutdown systems with failure to scram which results in core damage or additional failure of core cooling and makeup systems (which could lead to core melt)

   d. Failure of offsite and onsite power along with total loss of emergency feedwater makeup capability for several hours. Would lead to eventual core melt and likely failure of containment.

   e. Small LOCA and initially successful ECCS. Subsequent failure of containment heat removal systems over several hours could lead to core melt and likely failure of containment.

   NOTE: Most likely containment failure mode is melt-through with release of gases only for dry containment; quicker and larger releases likely for ice condenser containment for melt sequences. Quicker releases expected for failure of containment isolation system for any PWR.

6. Example BWR Sequences

   a. Transient (e.g., loss of offsite power) plus failure of requisite core shut down systems (e.g., scram). Could lead to core melt in several hours with containment failure likely. More severe consequences if pumps trip does not function.

   b. Small or large LOCA's with failure of ECCS to perform leading to core melt degradation or melt in minutes to hours. Loss of containment integrity may be imminent.

   c. Small or large LOCA occurs and containment performance is unsuccessful affecting longer term success of the ECCS. Could lead to core degradation or melt in several hours without containment boundary.

   d. Shutdown occurs but requisite decay heat removal systems (e.g., RHR) or non-safety systems heat removal means are rendered unavailable. Core degradation or melt could occur in about ten hours with subsequent containment failure.

7. Any major internal or external events (e.g., fires, earthquakes, substantially beyond design basis) which could cause massive common damage to plant systems resulting in any of the above.

Enclosure 4 to the Minutes of CRGR Meeting No. 213
Draft Supplement 1 to Generic Letter 90-02
regarding Technical Specifications for Reconstituting
Fuel Assemblies

December 19, 1991

TOPIC

C. E. Rossi, L. Phillips and P. Wen of NRR presented the subject supplement for CRGR review. The supplement would better define "approved methods" that licensees could use to justify fuel assembly reconstruction. This was needed because Generic Letter 90-02 had indicated that any methodology referenced in the FSAR or in reload applications would be acceptable. However, many such methodologies would not be appropriate for the purpose of substituting filler rods or voids in the reconstitution of fuel assemblies.

Copies of the handouts used by the staff in its presentation are provided as Attachment 1 to this enclosure.

BACKGROUND

The review package was transmitted by a memorandum for E. Jordan from F. Miraglia dated December 6, 1991. It included:

1. Draft generic letter supplement;
2. CRGR review package (responses to CRGR Charter questions).

CONCLUSIONS/RECOMMENDATIONS

The CRGR recommended in favor of the supplement subject to some revisions and receipt of a description of what the staff has done and why the staff did not propose to modify certain existing technical specifications. These matters will be coordinated with the CRGR staff.

Specific revisions and comments discussed included the following:

1. The backfit discussion should be modified to indicate that this action is a backfit, justified as a compliance exception. With regard to adequate protection, there was only a potential for an adequate protection issue, and then only if the current position were carried to an extreme.

2. Page 3, first full paragraph, reword to clarify that not every NRC approval is a generic type approval similar to an approval of the topical report.

3. Page 3, third full paragraph, reword to indicate that "Where filler rods are used, the NRC encourages..."

4. Page 4, delete the last sentence before the backfit discussion.

BACKFITTING

As discussed above, this action was considered to be a backfit, justified as a compliance exception.

PRESENTATION TO CRGR

ON PROPOSED

SUPPLEMENT 1 TO GENERIC LETTER 90-02

"Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications"

by

LARRY PHILLIPS

December 19, 1991

Attachment 1
to Enclosure 4

PROBLEMS WITH GL 90-02

- THERE IS NO BASIS OR NO EXISTING APPROVED METHODOLOGY TO SUPPORT FUEL DESIGN CHANGES (10 RODS PER ASSEMBLY) SUGGESTED BY THE GL AND ITS MODEL TS.

- CORE ALTERATIONS PERMITTED BY THE GL 90-02 AND ITS MODEL TS ARE UNLIMITED EXCEPT THAT A SPECIAL REPORT IS REQUIRED.

- THE LATITUDE OF FUEL DESIGN CHANGES PERMITTED HAS ENCOURAGED INDUSTRY INTERPRETATION THAT USE OF APPROVED METHODOLOGY BASED ON TEST DATA NOT APPLICABLE TO PROPOSED DUMMY ROD AND VACANCY CONFIGURATIONS IS ACCEPTABLE.

PROBLEMS WITH GL 90-02(continued)

- INDUSTRY PROPOSALS HAVE REVEALED IMPROPER OR INCOMPLETE EVALUATION OF FUEL DESIGN CHANGES BY RECONSTITUTION TO ASSURE COMPLIANCE WITH GDC 10

- EXTREME FUEL DESIGN CHANGES BY RECONSTITUTION MAY BE ACCOMPLISHED WITHOUT PRIOR NRC KNOWLEDGE AFTER TS CHANGES ARE COMPLETED

- RECONSTITUTION OF THE CORE TO EXTREMES PERMITTED BY GL 90-02 COULD INVALIDATE ANALYSES WHICH ASSURE THAT COOLABLE GEOMETRY IS MAINTAINED DURING DESIGN BASIS ACCIDENTS

DUMMY ROD AND VACANCY SAFETY CONCERNS

- STRUCTURAL/MECHANICAL DESIGN

  - SEISMIC/LOCA DESIGN LOADING:
    PREVENT STRUCTURAL DEFORMATION LEADING TO LOSS OF COOLABLE GEOMETRY OR RESISTANCE TO CONTROL ROD INSERTION

  - DIFFERENTIAL THERMAL EXPANSION:
    PROPER SEATING OF FUEL RODS
    SPACER GRID SPRING RELAXATION

  - RESISTANCE TO HYDRAULIC LOADS

- THERMAL-HYDRAULIC ANALYSES

  - CHF CORRELATIONS ARE EMPIRICAL AND APPLICABLE ONLY TO FLOW GEOMETRIES AND ROD-TO-ROD POWER DISTRIBUTIONS REPRESENTED IN THE TEST DATA BASE

  - 95/95 CORRELATION LIMIT VALUE IS A FUNCTION OF THE NUMBER OF TEST POINTS AND THE SCATTER IN THE MEASURED VS PREDICTED DATA

  - EXTENSIVE RECONSTITUTION MAY INTRODUCE SIGNIFICANT ERROR IN CORE WIDE ANALYSES

RESOLUTION TO GL 90-02 ISSUES

- CLARIFY THAT APPROVED METHODS MUST BE APPLICABLE TO THE PROPOSED RECONSTITUTED FUEL CONFIGURATION

- ENCOURAGE GENERIC TOPICAL REPORTS WHICH JUSTIFY SPECIFIED FUEL CONFIGURATIONS AND THE ANALYTICAL METHODS FOR CORE ANALYSIS

- REVISE THE MODEL TS

PRESENT STATUS

- VENDORS AND INDUSTRY ARE AWARE OF STAFF POSITION

- VENDORS ARE AGREEABLE IN PRINCIPLE

- SEVERAL RECONSTITUTION APPLICATIONS HAVE BEEN DELAYED AND OTHERS HAVE REQUIRED CYCLE SPECIFIC FIXES TO AVOID RELOAD DELAYS

- THREE GENERIC RECONSTITUTION METHODOLOGY REPORTS HAVE BEEN SUBMITTED FOR STAFF REVIEW

Enclosure 5 to the Minutes of CRGR Meeting No. 213
Proposed Amendment to 10 CFR 50.72 and 50.73 Reporting Requirements

December 19, 1991

TOPIC

The Committee discussed a staff proposal that formal CRGR review of this item be deferred until the final rule stage after receipt and evaluation of public comments. The proposed action involves some relaxation of current reporting requirements primarily related to invalid actuations of engineered safety features such as reactor water cleanup system and control room emergency ventilation system, where previous reporting has identified no safety concerns and provided little useful information. Implementation of the proposed relaxations by licensees would be on a purely voluntary basis, so no backfitting is involved in this proposed action.

BACKGROUND

The package submitted for consideration in this matter was transmitted by memorandum dated December 10, 1991, T. Novak to E.L. Jordan; the package included the following documents:

1. Draft Commission Paper, (undated), "Proposed Minor Rulemaking to Modify Operating Power Reactor Event Reporting Requirements - 10 CFR 50.72 and 10 CFR 50.73", and attachments as follows:

    a. Enclosure 1 - Draft Federal Register Notice (ubdated),

    b. Enclosure 2 - Draft Regulatory Analysis (undated)

CONCLUSIONS/RECOMMENDATIONS

As a result of their discussion of this matter, the Committee agreed with the staff proposal to defer formal review of this item to the final rule stage. This agreement was subject to provision that the Committee would be informed if the backfit analysis (included in the draft Federal Register Notice in the package) is changed substantially prior to final approval for publication. (The Committee believes that the backfit evaluation for the proposed action in its current form provides an appropriate discussion of backfitting considerations in connection with the proposed action; but there was some discussion at the meeting of possible revisions in that area based on OGC's review comments.)

BACKFITTING

As discussed above, this action was not considered to involve backfitting.

ROUTING AND TRANSMITTAL SLIP

Date 5-23-94

TO: (Name, office symbol, room number, building, Agency/Post)	Initials	Date
BCS P1 37		
2. PDR		
3.		
4.		
5.		

	Action		File		Note and Return
	Approval		For Clearance		Per Conversation
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REMARKS

This previous Central File material can now be made publicly available.

MATERIAL RELATED TO CACR
MEETING NO. 213

CC (LIST ONLY) JEAN RATAJE,
PDR L STREET

DO NOT use this form as a RECORD of approvals, concurrences, disposals, clearances, and similar actions

FROM: (Name, org. symbol, Agency/Post)	Room No.—Bldg.
DENNIS ALLISON	Phone No. 24148

5041-102

OPTIONAL FORM 41 (Rev. 7-76)
Prescribed by GSA
FPMR (41 CFR) 101-11.206

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MATERIAL RELATED TO CRGR MEETING NO. 213
TO BE MADE PUBLICLY AVAILABLE

1. MEMO FOR J. TAYLOR FROM E. JORDAN DATED 1-3-92
SUBJECT: MINUTES OF CRGR MEETING NUMBER 213
INCLUDING THE FOLLOWING ENCLOSURES WHICH WERE NOT
PREVIOUSLY RELEASED:

22-141 50 SHEETS
22-142 100 SHEETS
23-144 200 SHEETS

a. ENCLOSURE 2
A SUMMARY OF DISCUSSIONS OF A PROPOSED Supp. 4 to GL 89-10 on Relaxing Staff Position Regarding Position Changeable Valves for BWRs

b. ENCLOSURE 3
A SUMMARY OF DISCUSSIONS OF A PROPOSED Revision 3 to Reg. Guide 1.101 (to Endorse NUMARC Guidance on Development of Emergency Action Levels)

c. ENCLOSURE ______
A SUMMARY OF DISCUSSIONS OF A PROPOSED Supp. 1 to GL 90-02 Regarding TS for Reconstituting Fuel Assemblies

2. MEMO FOR E. JORDAN FROM E. Beckford DATED 11-16-91
FORWARDING REVIEW MATERIALS ON A PROPOSED Rev. 3 to RG 1.101, Emergency Planning + Preparedness for Nuclear Reactors

3. MEMO FOR E JORDAN FROM F. Miraglia DATED 12-6-91
FORWARDING REVIEW MATERIALS ON A PROPOSED GL 90-02, Supp. 1, Alternative Requirements for Fuel Assemblies in Design Features Section of TS

4. MEMO FOR E JORDAN FROM ______ DATED ______
FORWARDING REVIEW MATERIALS ON A PROPOSED

______