Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

March 20, 2003

10 CFR Part 50, App E

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop OWFN, P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, and 3 - EMERGENCY PLAN IMPLEMENTING PROCEDURE (EPIP) REVISIONS

TVA is submitting this notification in accordance with the requirements of 10 CFR Part 50, Appendix E, Section V. Specifically, portions of EPIP-1 were revised, namely, Table of Contents, Revision 34; Section II-1.0, Revision 31; Section II-3.0, Revision 30; Section III-1.0, Revision 31; and Section III-3.0, Revision 30. The revisions have an effective date of March 10, 2003.

The enclosed information is being sent by certified mail. The signed receipt signifies that you have received this information. If you have any questions, please telephone me at (256) 729-2636.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

cc: See Page 2

A045

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ENCLOSURE
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3

EMERGENCY PLAN IMPLEMENTING PROCEDURE (EPIP) REVISIONS
EPIP 1 TOC, SECTIONS II-1.0, II-3.0, III-1.0, AND III-3.0

SEE ATTACHED

GENERAL REVISIONS

FILING INSTRUCTIONS

FILE DOCUMENTS AS FOLLOWS:

PAGES TO BE REMOVED	PAGES TO BE INSERTED
TOC, Revision 33	TOC, Revision 34
Section II-1.0, Revision 30	Section II-1.0, Revision 31
Section II-3.0, Revision 29	Section II-3.0, Revision 30
Section III-1.0, Revision 30	Section III-1.0, Revision 31
Section III-3.0, Revision 29	Section III-3.0, Revision 30

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

EMERGENCY PLAN IMPLEMENTING PROCEDURE

EPIP-1

EMERGENCY CLASSIFICATION PROCEDURE

REVISION 34

PREPARED BY. T W. CORNELIUS PHONE· 2038

RESPONSIBLE ORGANIZATION EMERGENCY PREPAREDNESS

APPROVED BY: JEFF LEWIS DATE 03/05/2003

EFFECTIVE DATE 03/10/2003

LEVEL OF USE: REFERENCE USE

QUALITY-RELATED

REVISION LOG

Procedure Number: EPIP-1 Revision Number 34

Pages Affected: 1,14,15,17,30,84,86,89,118,120

Description of Change.

IC - 42 EPIP 1, rev. 31 revision is being conducted to change the Site Boundary Radiation Reading from a beta-gamma value to gamma only value. This change does not involve the numerical value. This revision is in compliance with the REP and doesn't affect the BFN EP standard emergency classification and action level scheme This revision is being conducted to ensure consistency with NUMARC/NESP-007, Reg Guide 1 101, and NEI 99-01 (Rev. 4).

IC - 43 EPIP 1, rev. 32 is being conducted to modify information that support EAL 1.1-G1, 1.1-G2, and 1.2-G The revision incorporates changes resulting from modifications to calculations that support Minimum Alternate RPV Flooding Pressures (MARFP) and Minimum Steam Cooling Reactor Water Level (MSCRWL) Revisions to these calculations were conducted in support of the EOI Program Manual Revision 21 (U3C11).

IC-44 EPIP 1, rev. 33 is being issued to modify EAL 6 7-U. The revision incorporates changes resulting from the letter from NEI to NRC (to Mr Bruce A. Boger) dated December 18, 2001 requesting confirmation for EAL basis change to include response to a Site -Specific Credible Threat. This was developed in response to NRC's October 6, 2001 Safeguards Advisor. This is additional information and does not change existing criteria in the EAL Basis.

IC-45 EPIP 1, rev. 34 is being conducted to modify information that support EAL 1.1-G1, 1.1-G2, and 3 1-S. The revision incorporates changes resulting from modifications to calculations that support Minimum Alternate RPV Flooding Pressures (MARFP), Minimum Steam Cooling Reactor Water Level (MSCRWL) and Maximum Safe Operating Area Temperature Limits. Revisions to these calculations were conducted in support of the EOI Program Manual Revision 22 (U2C13)

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REACTOR
1.0

NOTES:

1.1-U1/1.1-A1 Applicable when the Reactor Head is removed and the Reactor Cavity is flooded

1.1-S1 Applicable in Mode 5 when the Reactor Head is installed.

1.1-G2 The reactor will remain subcritical under all conditions without boron when:

- All control rods are inserted to or beyond position 02
- All control rods except one are inserted to or beyond position 00
- Determined by reactor engineering

CURVES/TABLES:

TABLE 1.1 - G2 MINIMUM ALTERNATE RPV FLOODING PRESS	
NUMBER OF OPEN MSRVs	MARFP (PSIG)
6 or More	190
5	230
4	290

WATER LEVEL

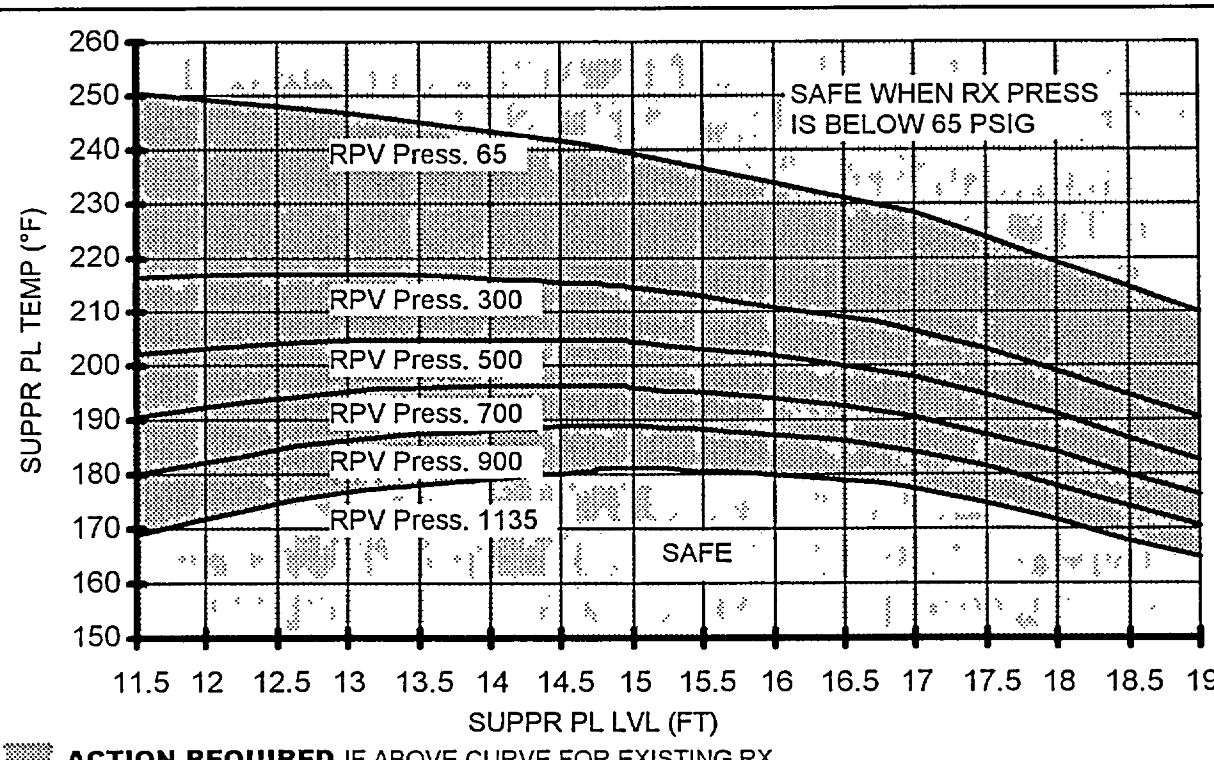
DESCRIPTION	DESCRIPTION	
1.1-U1 N Uncontrolled water level decrease in Reactor Cavity with irradiated fuel assemblies expected to remain covered by water OPERATING CONDITION - Mode 5	**1.1-U2** Uncontrolled water level decrease in Spent Fuel Pool with irradiated fuel assemblies expected to remain covered by water OPERATING CONDITION - All	UNUSUAL EVENT
1.1-A1 N Uncontrolled water level decrease in Reactor Cavity expected to result in irradiated Fuel assemblies being uncovered OPERATING CONDITION - Mode 5	**1.1-A2** Uncontrolled water level decrease in Spent Fuel Storage Pool expected to result in irradiated fuel assemblies being uncovered OPERATING CONDITION. - All	ALERT
1.1-S1 N Reactor water level CANNOT be maintained above -162 IN (TAF) OPERATING CONDITION - All	**1.1-S2** Reactor water level CANNOT be determined OPERATING CONDITION - Mode 1 - Mode 3 - Mode 2	SITE EMERGENCY
1.1-G1 Reactor water level CANNOT be restored and maintained above -185 IN OPERATING CONDITION - Mode 1 - Mode 3 - Mode 2	**1.1-G2** T N Reactor water level CANNOT be determined AND EITHER of the following conditions exists: • The reactor will remain subcritical w/o boron under all conditions and Less than 4 MSRVs can be opened, or Reactor pressure CANNOT be restored and maintained at least 65 PSI above Suppression Chamber pressure. • It has NOT been determined that the reactor will remain subcritical w/o boron under all conditions and unable to restore and maintain MARFP in Table 1 1-G2 OPERATING CONDITION - Mode 1 - Mode 3 - Mode 2	GENERAL EMERGENCY

NOTES:

1 2 Subcritical is defined as Reactor power below the heating range and not trending upward.

CURVES/TABLES:

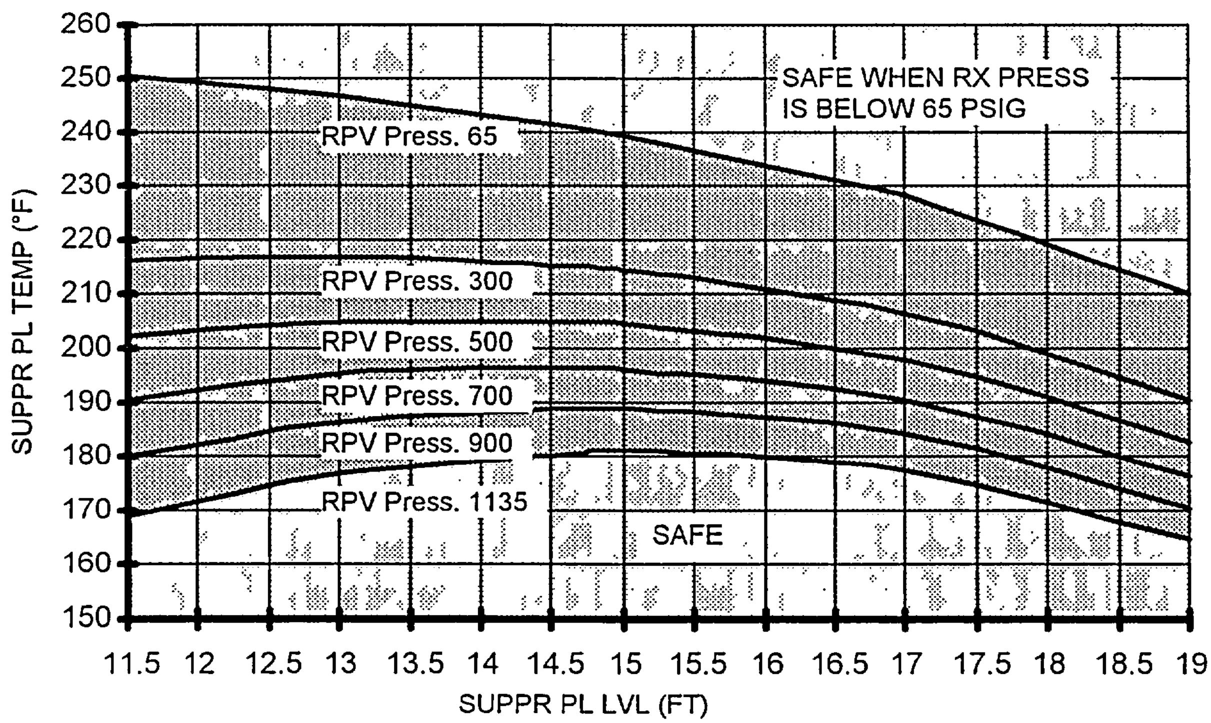
CURVE 1.2-G
HEAT CAPACITY TEMP LIMIT



SCRAM FAILURE DESCRIPTION	REACTOR COOLANT ACTIVITY DESCRIPTION	
	1.3-U Reactor coolant activity exceeds 26 µCi/gm dose equivalent I-131 (Technical Specification Limit) as determined by chemistry sample. OPERATING CONDITION - ALL	UNUSUAL EVENT
1.2-A Failure of automatic scram functions to bring the Reactor subcritical **AND** Manual scram or ARI was successful OPERATING CONDITION - Mode 1 - Mode 2	**1.3-A** Reactor coolant activity exceeds 300 µCi/gm dose equivalent Iodine-131 as determined by chemistry sample OPERATING CONDITION - Mode 1 - Mode 2 - Mode 3	ALERT
1.2-S Failure of automatic scram, manual scram, and ARI to bring the Reactor subcritical OPERATING CONDITION - Mode 1		SITE EMERGENCY
1.2-G Failure of automatic scram, manual scram, and ARI. Reactor power >3% **AND** EITHER of the following conditions exists • Suppression Pool temp exceeds HCTL Refer to Curve 1 2-G • Reactor water level CANNOT be restored and maintained at or above -185 IN OPERATING CONDITION - Mode 1 - Mode 2		GENERAL EMERGENCY

CURVES/TABLES:

CURVE 1.5-S
HEAT CAPACITY TEMP LIMIT



ACTION REQUIRED IF ABOVE CURVE FOR EXISTING RX

MSL/OFFGAS RADIATION DESCRIPTION	LOSS OF DECAY HEAT REMOVAL DESCRIPTION	
1.4-U Valid MAIN STEAM LINE RADIATION HIGH-HIGH alarm, RA-90-135C **OR** Valid OG PRETREATMENT RADIATION HIGH alarm, RA-90-157A OPERATING CONDITION - Mode 1 - Mode 2 - Mode 3		UNUSUAL EVENT
	1.5A Reactor moderator temperature CANNOT be maintained below 212° F whenever Technical Specifications require Mode 4 conditions or during operations in Mode 5 OPERATING CONDITION - Mode 4 - Mode 5	ALERT
	1.5-S C Suppression Pool temperature, level and RPV pressure CANNOT be maintained in the safe area of Curve 1 5-S OPERATING CONDITION - Mode 1 - Mode 2 - Mode 3	SITE EMERGENCY
		GENERAL EMERGENCY

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SECONDARY CONTAINMENT 3.0

NOTES:

CURVES/TABLES:

TABLE 3.1
MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS

AREA	APPLICABLE PANEL 9-21 TEMPERATURE ELEMENTS (UNLESS OTHERWISE NOTED)	MAX SAFE OPERATING VALUE °F	
		UNIT 2	UNIT 3
RHR A/C PUMP ROOM	74-95A	150	155
RHR B/D PUMP ROOM	74-95B	210	215
HPCI TURBINE AREA	73-55A	270	270
RCIC TURBINE AREA	71-41A	190	190
CS PUMP ROOM HIGH HUMIDITY OR TEMP HIGH	(XA-55-3E-29) PANEL 9-3 TI-75-69B	150	150
RCIC STEAM SUPPLY AREA	71-41B, 41C, 41D	200	250
HPCI STEAM SUPPLY AREA	73-55B, 55C, 55D	240	240
RHR A/C PUMP SUPPLY AREA	74-95H	240	240
RHR B/D PUMP SUPPLY AREA	74-95G	240	240
MAIN STEAM LINE LEAK DETECTION HIGH	(XA-55-3D-24) PANEL 9-3 TIS-1-60A	315	315
RHR VALVE ROOM	74-95E	170	175
RWCU ISOL LOGIC CHANNEL A/B TEMP HIGH	(XA-55-5B-32/33) PANEL 9-5 69-835A, B, C, D AUX INST ROOM	170	175
RWCU OUTBD ISOL VLV AREA	69-29F	220	220
RWCU HX AREA	69-29G	220	220
RWCU HX EXH DUCT	69-29H	220	220
RWCU RECIRC PUMP A AREA	69-29D	215	215
RWCU RECIRC PUMP B AREA	69-29E	215	215
RHR A/C HX ROOM	74-95C	195	200
RHR B/D HX ROOM	74-95D	195	200
FPC HX AREA	74-95F	155	155

TABLE 3.2
MAXIMUM SAFE OPERATING AREA RADIATION LIMITS

AREA	RAD MONITOR	MAX SAFE VALUE MR/HR
RHR WEST ROOM	90-25A	1000
RHR EAST ROOM	90-28A	1000
HPCI ROOM	90-24A	1000
CS/RCIC ROOM	90-26A	1000
CORE SPRAY ROOM	90-27A	1000
SUPPR POOL AREA	90-29A	1000
CRD-HCU WEST AREA	90-20A	1000
CRD-HCU EAST AREA	90-21A	1000
TIP DRIVE AREA	90-23A	1000
NORTH RWCU SYSTEM AREA	90-13A	1000
SOUTH RWCU SYSTEM AREA	90-14A	1000
RWCU SYSTEM AREA	90-9A	1000
MG SET AREA	90-4A	1000
FUEL POOL AREA	90-1A	1000
SERVICE FLR AREA	90-2A	1000
NEW FUEL STORAGE	90-1A	1000

TABLE 3.1-G/3.2-G
INDICATIONS OF POTENTIAL OR SIGNIFICANT FUEL FAILURE with the RCS Barrier Intact

DRYWELL RADIATION UNIT 2		DRYWELL RADIATION UNIT 3	
2-RE-90-272A	$\geq$ 345 R/HR	3-RE-90-272A	$\geq$ 106 R/HR
2-RE-90-273A	$\geq$ 164 R/HR	3-RE-90-273A	$\geq$ 164 R/HR
REACTOR COOLANT ACTIVITY $\geq$ 300 µCI/gm DOSE EQUILAVENT IODINE-131		REACTOR COOLANT ACTIVITY $\geq$ 300 µCI/gm DOSE EQUILAVENT IODINE-131	

	SECONDARY CONTAINMENT TEMPERATURE DESCRIPTION	SECONDARY CONTAINMENT RADIATION DESCRIPTION
UNUSUAL EVENT		
ALERT		**3.2-A** Any of the following high radiation alarms on Panel 9-3 • RA-90-1A, Fuel Pool Floor Area • RA-90-250A, Reactor, Turbine, Refuel Exhaust • RA-90-142A, Reactor Zone Exhaust • RA-90-140A, Refueling Zone Exhaust **AND** Confirmation by Refuel Floor personnel that irradiated fuel damage may have occurred OPERATING CONDITION - All
SITE EMERGENCY	**3.1-S** An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3 1 OPERATING CONDITION - Mode 1 - Mode 2 - Mode 3	**3.2-S** An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area radiation level at or above the Maximum Safe Operating area Radiation limit listed in Table 3 2 OPERATING CONDITION - Mode 1 - Mode 2 - Mode 3
GENERAL EMERGENCY	**3.1-G** An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3 1 **AND** Any indication of potential or significant fuel failure exists Refer to Table 3 1-G/3 2-G OPERATING CONDITION - Mode 1 - Mode 2 - Mode 3	**3.2-G** An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area radiation level at or above the Maximum Safe Operating area Radiation limit listed in Table 3 2 **AND** Any indication of potential or significant fuel failure exists Refer to Table 3 1-G/3 2-G OPERATING CONDITION - Mode 1 - Mode 2 - Mode 3

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REACTOR
1.0

WATER LEVEL 1.1-U1

UNUSUAL EVENT

Uncontrolled water level decrease in Reactor Cavity with irradiated fuel assemblies expected to remain covered by water.

OPERATING CONDITION - Mode 5

BASIS

This event classification only applies during Mode 5 when the Reactor Head is removed. For the purposes of this event classification the Reactor Cavity includes the cavity and the Reactor Vessel.

This event classification is anticipatory to 1.1-A1 and should only be considered if, in the opinion of the Site Emergency Director, the water level decrease is substantial enough to ultimately result in increased dose rates in the area of the Reactor Cavity due to loss of shielding by water covering irradiated fuel. Uncontrolled water level decrease during Mode 5 is indicative of valve manipulation error or failure of equipment in such a manner as to cause uncontrolled drainage of the Reactor Cavity. Uncontrolled water level decrease may be detected by the presence of the low level alarm in the spent fuel storage pool, visual observation, increased radiation levels or various other symptoms that the Site Emergency Director considers valid indicators of the event.

The degraded status of safety systems designed to makeup water to the Reactor Vessel is of particular concern during Mode 5 although plant Technical Specifications require minimum makeup systems be operable except with the spent fuel storage gates removed and water level $\geq$ 22 feet over the top of the reactor pressure vessel flange These events tend to have long lead times relative to potential for release outside the site boundary, thus impact to public health and safety is very low Classification as Unusual Event is warranted as a precursor to a more serious event

Escalation to Alert is by actual uncovery of irradiated fuel assemblies

WATER LEVEL 1.1-U1

UNUSUAL EVENT
(CONTINUED)

REFERENCES - Reg Guide 1.101 Rev. 3, (NUMARC-AU2 example -1)
Technical Specifications 3 5 2

NOTES

NOTE 1 1-U1/1 1-A1 Applicable when the Reactor Head is removed and the Reactor Cavity is flooded

WATER LEVEL 1.1-U2

UNUSUAL EVENT

Uncontrolled water level decrease in Spent Fuel Storage Pool with irradiated fuel assemblies expected to remain covered by water.

OPERATING CONDITION - All

BASIS This event classification is anticipatory to 1.1-A2 and should only be considered if, in the opinion of the Site Emergency Director, the water level decrease is substantial enough to ultimately result in increased dose rates in the area of the Spent Fuel Storage Pool due to loss of shielding by water covering irradiated fuel Uncontrolled water level decrease may be detected by the presence of the low level alarm in the spent fuel storage pool, visual observation, increased radiation levels or various other symptoms that the Site Emergency Director considers valid indicators of the event

Uncontrolled water level decrease in Spent Fuel Storage Pools is indicative of failure of equipment in such a manner as to cause uncontrolled drainage These events tend to have long lead times relative to potential for release outside the site boundary, thus impact to public health and safety is very low. Classification as Unusual Event is warranted as a precursor to a more serious event

Escalation to Alert is by actual uncovery of irradiated fuel assemblies.

REFERENCES - Reg Guide 1 101 Rev. 3, (NUMARC-AU2 example-2)

WATER LEVEL 1.1-A1

ALERT

Uncontrolled water level decrease in Reactor Cavity expected to result in irradiated fuel assemblies being uncovered.

OPERATING CONDITION - Mode 5

BASIS This event classification only applies during Mode 5 when the Reactor Head is removed For the purposes of this event classification the Reactor Cavity includes the cavity and the Reactor Vessel.

Uncontrolled water level decrease during Mode 5 is indicative of valve manipulation error or failure of equipment in such a manner as to cause uncontrolled drainage of the Reactor Cavity. The degraded status of safety systems designed to makeup water to the Reactor Vessel is of particular concern during Mode 5 although plant Technical Specifications require minimum makeup systems be operable except with the spent fuel storage gates removed and water level $\geq$ 22 feet over the top of the reactor pressure vessel flange

Uncontrolled water level decrease may be detected by visual observation, increased radiation levels or various other symptoms that the Site Emergency Director considers valid indicators of the event Expected fuel uncovery may be detected by increased radiation levels, Visual observation, RPV level instrumentation expected to drop below -162 inches, or best judgement of the Site Emergency Director based on present and past events and trends

Due to the long lead times associated with these events there is time available to take corrective actions, and there is little potential for substantial fuel damage Significant exposures to onsite personnel is likely during these events and it is probable that additional personnel will be needed onsite; therefore the Alert classification is warranted

Escalation is by Radiological Release event classifications.

WATER LEVEL 1.1-A1

ALERT
(CONTINUED)

REFERENCES - Reg Guide 1.101 Rev 3, (NUMARC-AA2 example-3)
Technical Specifications 3.5 2

NOTES

NOTE 1.1-U1/1.1-A1 Applicable when the Reactor Head is removed and the Reactor Cavity is flooded.

WATER LEVEL 1.1-A2

ALERT

Uncontrolled water level decrease in Spent Fuel Storage Pool expected to result in irradiated fuel assemblies being uncovered.

OPERATING CONDITION - All

BASIS Uncontrolled water level decrease in Spent Fuel Storage Pools is indicative of failure of equipment in such a manner as to cause uncontrolled drainage These events tend to have long lead times relative to potential for release outside the site boundary, thus impact to public health and safety is very low.

Uncontrolled water level decrease may be detected by visual observation, increased radiation levels or various other symptoms that the Site Emergency Director considers valid indicators of the event Expected fuel uncovery may be detected by increased radiation levels, Visual observation, or best judgement of the Site Emergency Director based on present and past events and trends

There is time available to take corrective actions, and there is little potential for substantial fuel damage. Offsite exposures are expected to remain below the Environmental Protection Agency's Protective Action Guidelines, however, exposures to onsite personnel is of particular concern during this event, therefore the Alert classification is warranted

Escalation is by Radiological Release event classifications

REFERENCES - Reg Guide 1 101 Rev 3, (NUMARC-AA2 example-4)

WATER LEVEL 1.1-S1

SITE AREA EMERGENCY

Reactor water level cannot be maintained above -162 in. (TAF).

OPERATING - ALL

BASIS If Reactor water level cannot be maintained above TAF the potential exist for fuel cladding damage Events most likely to result in coolant inventory loss to this extent are RCS boundary degradation events or station blackout events For this event to be declared, RPV water level must have decreased or be trending to a value that, in the opinion of the Site Emergency Director, has resulted in or will result in some actual core uncovery. Additionally, the Site Emergency Director must have evidence that Reactor level has been or can be recovered to above TAF.

This event classification also applies in Mode 5 when the Reactor Vessel head is installed. Inadvertent draining of the Reactor Vessel is possible under these conditions due to valving errors associated with the RHR system or failures associated with isolation valves during alignment changes of systems connected to the Reactor Vessel below the normal water level

The fact that the transient was severe enough to result in inability to maintain RPV level coupled with the anticipatory nature of this event classification as a precursor to more serious event warrants the Site Area Emergency event classification

For events that occur during operation, escalation to General Emergency is based on inability to assure adequate core cooling by restoring and maintaining RPV water level following transients that have resulted in extreme RPV water level decrease For events that occur during shutdown or Mode 5, escalation is by radioactive release event classifications

REFERENCES - Reg Guide 1 101 Rev. 3, (NUMARC-FS, SS5, SS4, example-1)
- EOI Program Manual Section VI-J

NOTES

NOTE 1 1-S1 Applicable in Mode 5 when the Reactor Head is installed.

WATER LEVEL 1.1-S2

SITE AREA EMERGENCY

Reactor water level cannot be determined.

OPERATING CONDITION
- Mode 1
- Mode 2
- Mode 3

BASIS

Inability to determine Reactor water level during operation may be due to boiling in the reference or variable instrument legs, instrument power failures, or conflicting information from uncontrolled indicator oscillations

This condition requires Reactor flooding following emergency depressurization. Adequate core cooling is assured by these measures Due to the severity of these actions and the uncertainty of Reactor status it is appropriate to treat this as a potential loss for Reactor Coolant System and Fuel Cladding integrity; therefore, this event is appropriate for the Site Area Emergency classification

Escalation to General Emergency is based on inability to assure adequate core cooling in this mode

REFERENCES
- Reg Guide 1 101 Rev 3, (NUMARC-FS)
- EOI Program Manual Section VI-J

WATER LEVEL 1.1-G1

GENERAL EMERGENCY

Reactor water level CANNOT be restored and maintained above -185 IN.

OPERATING CONDITION
- Mode 1
- Mode 2
- Mode 3

BASIS

If Reactor water level cannot be restored and maintained above the Minimum Steam Cooling Reactor Water Level (MSCRWL), core damage is possible due to inadequate steam generation, by the covered portion of the Reactor core, to remove decay heat and prevent cladding heatup to a point that results in clad failure

For either of the above conditions to be met, the control room operators should have progressed in the execution of the EOIs to the point that all high pressure and all low pressure systems that are available within a reasonable time frame have been attempted and are unsuccessful in reversing the adverse RPV water level trend

Events most likely to result in coolant inventory loss or loss of makeup capability to this extent are RCS boundary degradation events or events resulting from loss of multiple systems such as station blackout. During such transients or accidents the potential for Primary Containment failure increases substantially; therefore, the General Emergency classification is appropriate

REFERENCES
- Reg Guide 1.101 Rev 3, (NUMARC-FG)
- EOI Program Manual Section VI-J

WATER LEVEL 1.1-G2

GENERAL EMERGENCY

Reactor water level CANNOT be determined

AND

EITHER of the following conditions exists:

- **The reactor will remain subcritical w/o boron under all conditions.**
  and
  Less than 4 MSRVs can be opened, or Reactor pressure CANNOT be restored and maintained at least 65 PSI above Suppression Chamber pressure.

- **It has NOT been determined that the reactor will remain subcritical w/o boron under all conditions and unable to restore and maintain MARFP in Table 1.1-G2.**

OPERATING CONDITION
- Mode 1
- Mode 2
- Mode 3

BASIS Inability to determine Reactor water level during operation may be due to boiling in the reference or variable instrument legs, instrument power failures, or conflicting information from uncontrolled indicator oscillations. This condition requires Reactor flooding following emergency depressurization. It the reactor will remain subcritical without (w/o) boron under all conditions, adequate core cooling is assured only if at least 4 MSRVs are opened and Minimum Reactor Flooding Pressure (MRFP) is maintained with Reactor pressure at least 65 PSI above Suppression Chamber pressure If it has not been determined that the reactor will remain subcritical without (w/o) boron under all conditions, adequate core cooling can only be assured when the Minimum Alternate Reactor Flooding Pressure (MARFP) is restored and maintained If adequate core cooling is not assured core damage is probable under this scenario due to the extreme nature of the plant conditions that resulted in the inability to determine Reactor level (i e., high containment temperatures, loss of multiple power supplies, etc) Primary Containment integrity cannot be assured under all these conditions, therefore, the General Emergency classification is appropriate

REFERENCES
- Reg Guide 1 101 Rev. 3, (NUMARC-FG)
- EOI Program Manual Section VI-J

WATER LEVEL 1.1-G2

GENERAL EMERGENCY
(CONTINUED)

CURVES/TABLES

TABLE 1.1 - G2
MINIMUM ALTERNATE RPV FLOODING PRESS

NUMBER OF OPEN MSRVs	MARFP (PSIG)
6 or More	190
5	230
4	290

NOTES

NOTE 1 1-G2 The reactor will remain subcritical under all conditions w/o boron when:

- All control rods are inserted to or beyond position 02
- All control rods except one are inserted to or beyond position 00
- Determined by reactor engineering

SCRAM FAILURE 1.2-A

ALERT

Failure of automatic scram functions to bring the Reactor subcritical

AND

Manual scram or Alternate Rod Insertion (ARI) was successful.

OPERATING CONDITION - Mode 1
- Mode 2

BASIS A manual scram is any set of actions by the Reactor Operator(s) at the Reactor Control Console which causes control rods to be rapidly inserted into the core and brings the Reactor subcritical

This event classification indicates failure of the RPS to automatically scram the Reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus plant safety has been compromised, and design limits of the fuel may have been exceeded

An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS barrier Any set of actions by the Reactor Operator at Panel 9-5 that cause control rods to rapidly insert into the core and bring the Reactor subcritical is considered a manual scram

Escalation to Site Area Emergency is based on fuel clad barrier or RCS barrier event classifications

REFERENCE - Reg Guide 1.101 Rev 3, (NUMARC-SA2)

NOTES

NOTE 1.2 Subcritical is defined as Reactor power below the heating range and not trending upward

SCRAM FAILURE 1.2-S

SITE AREA EMERGENCY

Failure of automatic scram, manual scram, and ARI to bring the Reactor subcritical.

OPERATING CONDITION - Mode 1

BASIS Manual scram, and ARI are not considered successful if action away from the Reactor Control Console (Panel 9-5) was required to scram the Reactor

A failure of the automatic and manual scram systems may result in the Reactor producing more heat than the maximum decay heat load for which the safety systems are designed A Site Area Emergency classification is appropriate because conditions exist that lead to potential loss of both fuel clad and Reactor Coolant System (RCS) barriers Therefore, this event classification ensures timely emergency response to the event before actual barriers loss has taken place

Escalation to General Emergency is based upon inability to bring Reactor power within decay heat removal capability before Suppression Pool temperature reaches the Heat Capacity Temperature Limit (HCTL)

REFERENCES - Reg Guide 1 101 Rev. 3, (NUMARC-SS2, SS4 example -1)

NOTES

NOTE 1 2 Subcritical is defined as Reactor power below the heating range and not trending upward

SCRAM FAILURE 1.2-G

GENERAL EMERGENCY

Failure of automatic scram, manual scram, and ARI. Reactor power > 3%.

AND

EITHER of the following conditions exists:

- **Suppression Pool temperature exceeds HCTL. Refer to curve 1.2-G.**
- **Reactor water level CANNOT be restored and maintained at or above -185 IN.**

OPERATING CONDITION - Mode 1
- Mode 2

BASIS

Automatic scram, manual scram, and ARI are not considered successful if action away from the Reactor Control Console was required to scram the Reactor

Under these conditions all efforts, including boron injection, have been unsuccessful in bringing Reactor power within the decay heat removal capability of the Emergency Core Cooling Systems (ECCS) Additionally, an extreme challenge to the ability to cool the Reactor Core exist if Reactor Pressure Vessel (RPV) water level cannot be maintained sufficient to ensure adequate core cooling

Another consideration is the inability to remove heat using the Main Condenser or Suppression Pool. In the event that neither heat sink is effective and Reactor power remains above this level, then a core melt sequence exists In this situation, core degradation can occur rapidly, therefore, a General Emergency classification is appropriate in anticipation of degradation of multiple fission product barriers

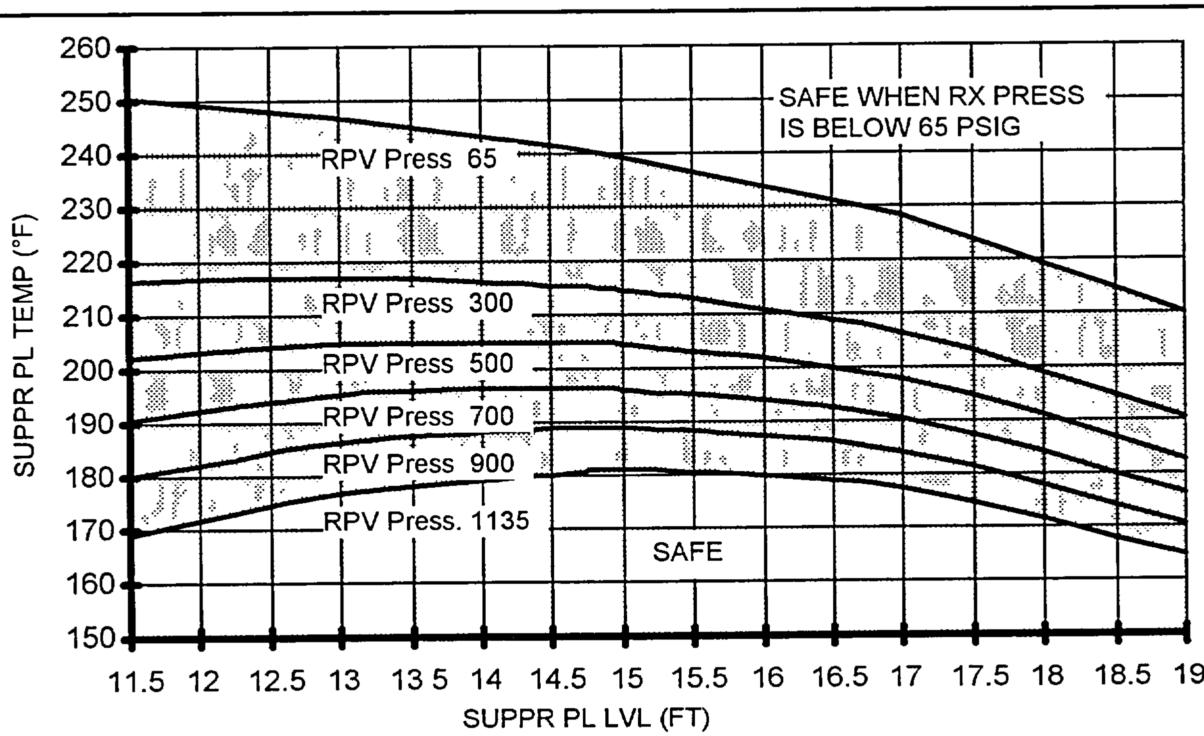
REFERENCES - Reg Guide 1.101 Rev. 3,(NUMARC-SG2)
- EOI Program Manual Section V-K and Section V-D

SCRAM FAILURE 1.2-G

GENERAL EMERGENCY
(CONTINUED)

CURVES/TABLES

CURVE 1.2-G
HEAT CAPACITY TEMP LIMIT



ACTION REQUIRED IF ABOVE CURVE FOR EXISTING RX

REACTOR COOLANT ACTIVITY 1.3-U

UNUSUAL EVENT

Reactor coolant activity exceeds 26 µCi/gm dose equivalent I-131 (Technical Specification limit) as determined by chemistry sample.

OPERATING CONDITION - All

BASIS Reactor coolant activity samples exceeding Technical Specification limits for Iodine spikes are representative of fuel clad degradation An Unusual Event is declared because of potential degradation in the level of safety of the plant Iodine levels exceeding Technical Specification limits are a potential precursor of more serious problems

Escalation to Alert would be based on higher Reactor coolant activity values indicative of significant fuel failure

REFERENCES
- Reg Guide 1 101 Rev. 3, (NUMARC-SU4 example-2)
- Technical Specification 3 4 6

REACTOR COOLANT ACTIVITY 1.3-A

ALERT

Reactor coolant activity exceeds 300 μCi/gm dose equivalent Iodine-131 as determined by Chemistry sample.

OPERATING CONDITION
- Mode 1
- Mode 2
- Mode 3

BASIS Reactor coolant activity samples exceeding 300 μCi/gm dose equivalent Iodine-131 are well above those expected for Iodine spikes and represent a significant loss of the fuel clad barrier Any loss or potential loss of the fuel clad barrier warrants the declaration of an Alert

Escalation to Site Area Emergency would be based on the conditions given above coupled with a loss or potential loss of either the Primary Containment or Reactor Coolant System barrier or Radiological Releases

REFERENCE
- Reg Guide 1 101 Rev. 3, (NUMARC-FA)
- RIMS L36 921201 806

MSL/OFFGAS RADIATION 1.4-U

UNUSUAL EVENT

Valid MAIN STEAM LINE RADIATION HIGH-HIGH alarm, RA-90-135C
OR
Valid OG PRETREATMENT RADIATION HIGH alarm, RA-90-157A.

OPERATING CONDITION
- Mode 1
- Mode 2
- Mode 3

BASIS Main Steam Line radiation high high or offgas radiation high is indicative of fuel cladding leakage

The Main Steam Line radiation high high alarm setpoint is normally set at 3 times normal full power background. 3 times normal full power background is in excess of any spikes expected from operational transients that do not result in cladding failure This alarm setpoint is substantially above that which would be indicative of fuel cladding damage above Technical Specification allowable limits, however, the presence of a valid alarm warrants declaration of an Unusual Event and consideration of other symptoms and event classifications for possible upgrade of the event based on fission product barrier loss.

The offgas pretreatment radiation high alarm setpoint is set at a value that is indicative of the ODCM allowable limits for radiation release

Either of these conditions is considered a potential degradation in the level of safety of the plant and a potential precursor of a more serious problem

Escalation to the Alert is based on either Reactor coolant samples exceeding 300 μCi/gm or Drywell radiation levels indicative of loss of the fuel cladding barrier.

REFERENCES - Reg Guide 1 101 Rev. 3, (NUMARC-SU4 example-1)

LOSS OF DECAY HEAT REMOVAL 1.5-A

ALERT

Reactor moderator temperature CANNOT be maintained below 212°F whenever Technical Specifications require Mode 4 conditions or during operations in Mode 5.

OPERATING CONDITION - Mode 4
- Mode 5

BASIS This event classification addresses loss of decay heat removal functions when Mode 4 is required or during Mode 5 Loss of decay heat removal capability can result in more serious consequences depending upon whether Primary Containment is in tact and Emergency Core Cooling System (ECCS) equipment status. In any condition where Mode 4 is required, loss of decay heat removal capability represents a significant degradation in plant conditions that can lead to fuel cladding damage or RCS degradation In order to maintain anticipatory philosophy the Alert classification is appropriate for this event

Escalation to Site Area Emergency or General Emergency is by loss of Reactor water level that has or will uncover the fuel or Radiological Release Event classification

REFERENCES - Reg Guide 1 101 Rev. 3, (NUMARC-SA3)

LOSS OF DECAY HEAT REMOVAL 1.5-S

SITE AREA EMERGENCY

Suppression Pool temperature, level and RPV pressure CANNOT be maintained in the safe area of Curve 1.5-S (Heat Capacity Temperature Limit)

OPERATING CONDITION
- Mode 1
- Mode 2
- Mode 3

BASIS

Suppression Pool temperature is limited by Curve 1 5-S as a function of suppression pool level and reactor pressure in order to preclude failure of Primary Containment or equipment necessary for the safe shutdown of the plant following emergency depressurization When Suppression Pool temperature cannot be maintained below the limits of the curve corresponding to existing suppression pool level and reactor pressure, emergency depressurization is required and continued decay heat removal at operating temperature and pressure is no longer permissible

Suppression Pool level is limited by Curve 1 5-S to the range of 11 5 feet to 19 feet in order to preclude failure of Primary Containment or equipment necessary for the safe shutdown of the plant and preserve the pressure suppression function of the containment for possible future emergency depressurization When Suppression Pool level cannot be maintained within the limits of the curve, continued decay heat removal at operating pressures and temperatures is no longer permissible and emergency depressurization is required

Exceeding the limits of Curve 1.5-S represents a loss of heat sink for decay heat removal and inability to maintain Mode 3 Under these conditions there is an actual failure of systems intended for protection of the public, therefore, Site Area Emergency is warranted Escalation to General Emergency is by Abnormal Rad levels, Radiological Release or Primary Containment failure events

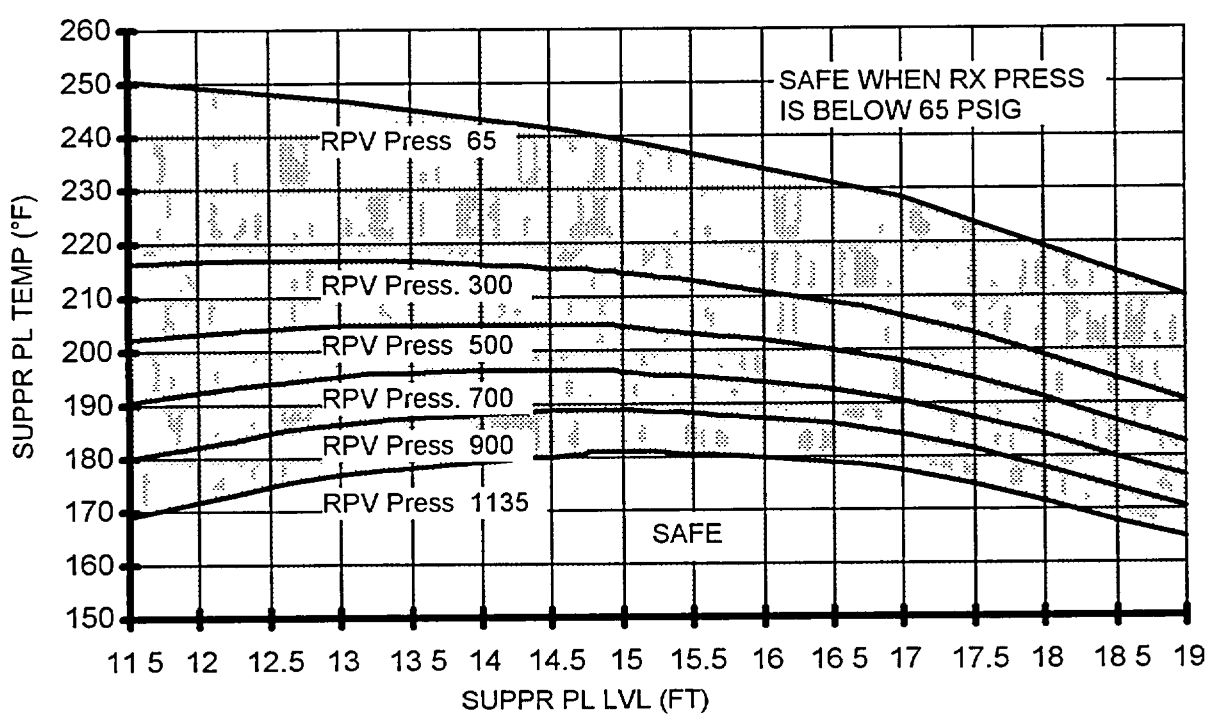
REFERENCES
- Reg Guide 1 101 Rev 3, (NUMARC-SS4)
- EOI Program Manual Sections VI-C and VI-F

LOSS OF DECAY HEAT REMOVAL 1.5-S

SITE AREA EMERGENCY
(CONTINUED)

CURVES/TABLES

CURVE 1.5-S
HEAT CAPACITY TEMP LIMIT



ACTION REQUIRED IF ABOVE CURVE FOR EXISTING RX

SECONDARY CONTAINMENT 3.0

SECONDARY CONTAINMENT TEMPERATURE 3.1-S

SITE AREA EMERGENCY

An unisolable primary system leak is discharging into Secondary Containment

AND

Any area temperature exceeds the Maximum Safe Operating temperature limit listed in Table 3.1.

OPERATING CONDITION
- Mode 1
- Mode 2
- Mode 3

BASIS The Maximum Safe Operating Temperatures of table 3.1 are based on the Browns Ferry Environmental Qualification (EQ) program for safety related equipment. EQ program assumptions include single failure criteria for pipe break that isolates as required Temperatures in excess of those in Table 3 1 are indicative of pipe breaks that fail to isolate as required. Secondary Containment temperatures of this magnitude resulting from primary system leakage are indicative of significant loss of both the RCS pressure boundary and the Primary Containment pressure boundary. The Site Area Emergency classification is appropriate based upon loss of any two barriers

Escalation to General Emergency is based on loss or potential loss of the fuel cladding barrier or Radioactivity Release event classifications

REFERENCES
- Reg Guide 1 101 Rev. 3, (NUMARC-FS)
- EOI Program Manual, Section V-E

SECONDARY CONTAINMENT TEMPERATURE 3.1-S

SITE AREA EMERGENCY
(CONTINUED)

CURVES/TABLES

TABLE 3.1
MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS

AREA	APPLICABLE PANEL 9-21 TEMPERATURE ELEMENTS (UNLESS OTHERWISE NOTED)	MAX SAFE OPERATING VALUE °F	
		UNIT 2	UNIT 3
RHR A/C PUMP ROOM	74-95A	150	155
RHR B/D PUMP ROOM	74-95B	210	215
HPCI TURBINE AREA	73-55A	270	270
RCIC TURBINE AREA	71-41A	190	190
CS PUMP ROOM HIGH HUMIDITY OR TEMP HIGH	(XA-55-3E-29) PANEL 9-3 TI-75-69B	150	150
RCIC STEAM SUPPLY AREA	71-41B. 41C. 41D	200	250
HPCI STEAM SUPPLY AREA	73-55B. 55C 55D	240	240
RHR A/C PUMP SUPPLY AREA	74-95H	240	240
RHR B/D PUMP SUPPLY AREA	74-95G	240	240
MAIN STEAM LINE LEAK DETECTION HIGH	(XA-55-3D-24) PANEL 9-3 TIS-1-60A	315	315
RHR VALVE ROOM	74-95E	170	175
RWCU ISOL LOGIC CHANNEL A/B TEMP HIGH	(XA-55-5B-32/33) PANEL 9-5 69-835A. B. C. D AUX INST ROOM	170	175
RWCU OUTBD ISOL VLV AREA	69-29F	220	220
RWCU HX AREA	69-29G	220	220
RWCU HX EXH DUCT	69-29H	220	220
RWCU RECIRC PUMP A AREA	69-29D	215	215
RWCU RECIRC PUMP B AREA	69-29E	215	215
RHR A/C HX ROOM	74-95C	195	200
RHR B/D HX ROOM	74-95D	195	200
FPC HX AREA	74-95F	155	155

SECONDARY CONTAINMENT TEMPERATURE 3.1-G

GENERAL EMERGENCY

An unisolable primary system leak is discharging into Secondary Containment

AND

Any area temperature exceeds the Maximum Safe Operating temperature limit listed in Table 3.1

AND

Any indication of potential or significant fuel failure exists. Refer to Table 3.1-G/3.2-G.

OPERATING CONDITION
- Mode 1
- Mode 2
- Mode 3

BASIS

The Maximum Safe Operating Temperatures of table 3.1 are based on the Browns Ferry Environmental Qualification (EQ) program for safety related equipment. EQ program assumptions include single failure criteria for pipe break that isolates as required. Temperatures in excess of those in Table 3 1 are indicative of pipe breaks that fail to isolate as required Secondary Containment temperatures of this magnitude resulting from primary system leakage are indicative of significant loss of both the RCS pressure boundary and the Primary Containment pressure boundary Table 3 1-G/3.2-G provides guidance for determining if significant fuel failure should be assumed

This event classification represents loss of all three barriers designed to contain fission products during accidents; therefore, the General Emergency classification is appropriate

SECONDARY CONTAINMENT TEMPERATURE 3.1-G

GENERAL EMERGENCY
(CONTINUED)

REFERENCES - Reg Guide 1 101 Rev. 3, (NUMARC-FG)
EOI Program Manual, Section V-E
Calculation ND-N0090-930055 R3 (Unit 2/3)
Calculation ND-N0090-930050 R2 (Unit 2/3)

CURVES/TABLES

TABLE 3.1
MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS

AREA	APPLICABLE PANEL 9-21 TEMPERATURE ELEMENTS (UNLESS OTHERWISE NOTED)	MAX SAFE OPERATING VALUE °F UNIT 2	UNIT 3
RHR A/C PUMP ROOM	74-95A	150	155
RHR B D PUMP ROOM	74-95B	210	215
HPCI TURBINE AREA	73-55A	270	270
RCIC TURBINE AREA	71-41A	190	190
CS PUMP ROOM HIGH HUMIDITY OR TEMP HIGH	(XA-55-3E-29) PANEL 9-3 TI-75-69B	150	150
RCIC STEAM SUPPLY AREA	71-41B, 41C, 41D	200	250
HPCI STEAM SUPPLY AREA	73-55B, 55C, 55D	240	240
RHR A/C PUMP SUPPLY AREA	74-95H	240	240
RHR B/D PUMP SUPPLY AREA	74-95G	240	240
MAIN STEAM LINE LEAK DETECTION HIGH	(XA-55-3D-24) PANEL 9-3 TIS-1-60A	315	315
RHR VALVE ROOM	74-95E	170	175
RWCU ISOL LOGIC CHANNEL A/B TEMP HIGH	(XA-55-5B-32/33) PANEL 9-5 69-835A, B, C, D AUX INST ROOM	170	175
RWCU OUTBD ISOL VLV AREA	69-29F	220	220
RWCU HX AREA	69-29G	220	220
RWCU HX EXH DUCT	69-29H	220	220
RWCU RECIRC PUMP A AREA	69-29D	215	215
RWCU RECIRC PUMP B AREA	69-29E	215	215
RHR A/C HX ROOM	74-95C	195	200
RHR B/D HX ROOM	74-95D	195	200
FPC HX AREA	74-95F	155	155

TABLE 3.1-G/3.2-G
INDICATIONS OF POTENTIAL OR SIGNIFICANT FUEL FAILURE with RCS Barrier Intact

DRYWELL RADIATION UNIT 2		DRYWELL RADIATION UNIT 3	
2-RE-90-272A	≥ 345 R/HR	3-RE-90-272A	≥ 106 R/HR
2-RE-90-273A	≥ 164 R/HR	3-RE-90-273A	≥ 164 R/HR
REACTOR COOLANT ACTIVITY ≥ 300 μCI gm DOSE EQUILAVENT IODINE-131		REACTOR COOLANT ACTIVITY ≥ 300 μCI/gm DOSE EQUILAVENT IODINE-131	

SECONDARY CONTAINMENT RADIATION 3.2-A

ALERT

Any of the following high radiation alarms on Panel 9-3:

- **RA-90-1A, Fuel Pool Floor Area**
- **RA-90-250A, Reactor, Turbine, Refuel Exhaust**
- **RA-90-142A, Reactor Zone Exhaust**
- **RA-90-140A, Refueling Zone Exhaust**

AND

Confirmation by Refuel Floor personnel that irradiated fuel damage may have occurred.

OPERATING CONDITION - All

BASIS This event is indicative of irradiated fuel damage caused by a dropped fuel bundle or other heavy solid objects into the Reactor Cavity or Spent Fuel Storage Pools The second part of this event classification associates the listed alarms with events that could result in actual irradiated fuel damage versus increased background from other possible sources Compared to core damage that can occur from full power operating conditions, there is little potential for substantial fuel damage; however, Significant exposures to onsite personnel are possible and protective actions for site personnel may be necessary. For these reasons the Alert classification is warranted

Escalation is by Radiological Release event classifications.

REFERENCES - Reg Guide 1.101 Rev 3, (NUMARC-AA2-example-1)

SECONDARY CONTAINMENT RADIATION 3.2-S

SITE AREA EMERGENCY

An unisolable primary system leak is discharging into Secondary Containment

AND

Any area radiation level at or above the Maximum Safe Operating Area Radiation limit listed in Table 3.2.

OPERATING CONDITION
- Mode 1
- Mode 2
- Mode 3

BASIS

Secondary Containment radiation levels of this magnitude are indicative of significant inability to contain or control radioactive materials within the Primary System and Primary Containment. If the Primary System is the source then Site Area Emergency is warranted based upon loss of any two fission product barriers

Escalation to General Emergency is based on loss or potential loss of the fuel cladding barrier or Radioactive Release event classifications

REFERENCES
- Reg Guide 1 101 Rev. 3, (NUMARC-FS)
- EOI Program Manual, Section V-E

SECONDARY CONTAINMENT RADIATION 3.2-S

SITE AREA EMERGENCY
(CONTINUED))

CURVES/TABLES

TABLE 3.2
MAXIMUM SAFE OPERATING AREA RADIATION LIMITS

AREA	RAD MONITOR	MAX SAFE VALUE MR/HR
RHR WEST ROOM	90-25A	1000
RHR EAST ROOM	90-28A	1000
HPCI ROOM	90-24A	1000
CS/RCIC ROOM	90-26A	1000
CORE SPRAY ROOM	90-27A	1000
SUPPR POOL AREA	90-29A	1000
CRD-HCU WEST AREA	90-20A	1000
CRD-HCU EAST AREA	90-21A	1000
TIP DRIVE AREA	90-23A	1000
NORTH RWCU SYSTEM AREA	90-13A	1000
SOUTH RWCU SYSTEM AREA	90-14A	1000
RWCU SYSTEM AREA	90-9A	1000
MG SET AREA	90-4A	1000
FUEL POOL AREA	90-1A	1000
SERVICE FLR AREA	90-2A	1000
NEW FUEL STORAGE	90-1A	1000

SECONDARY CONTAINMENT RADIATION 3.2-G

GENERAL EMERGENCY

An unisolable primary system leak is discharging into Secondary Containment

AND

Any area radiation level at or above the Maximum Safe Operating Area Radiation limit listed in Table 3.2

AND

Any indication of potential or significant fuel failure exists. Refer to Table 3.1-G/3.2-G.

OPERATING CONDITION -
- Mode 1
- Mode 2
- Mode 3

BASIS

Secondary Containment radiation levels of this magnitude are indicative of significant inability to contain or control radioactive materials within the primary system and Primary Containment If the primary system is the source then these indications represent loss of RCS pressure boundary and Primary Containment pressure boundary Table 3.1-G/3.2-G provides guidance for determining if significant fuel failure should be assumed

This event classification represents loss or potential loss of all three barriers designed to contain fission products during accidents; therefore, the General Emergency classification is appropriate

REFERENCES -

Reg Guide 1 101 Rev 3, (NUMARC-FG)
EOI Program Manual, Section V-E
Calculation ND-N0090-930055 R3 (Unit 2/3)
Calculation ND-N0090-930050 R2 (Unit 2/3)

SECONDARY CONTAINMENT RADIATION 3.2-G

GENERAL EMERGENCY
(CONTINUED)

CURVES/TABLES

TABLE 3.2
MAXIMUM SAFE OPERATING AREA RADIATION LIMITS

AREA	RAD MONITOR	MAX SAFE VALUE MR/HR
RHR WEST ROOM	90-25A	1000
RHR EAST ROOM	90-28A	1000
HPCI ROOM	90-24A	1000
CS/RCIC ROOM	90-26A	1000
CORE SPRAY ROOM	90-27A	1000
SUPPR POOL AREA	90-29A	1000
CRD-HCU WEST AREA	90-20A	1000
CRD-HCU EAST AREA	90-21A	1000
TIP DRIVE AREA	90-23A	1000
NORTH RWCU SYSTEM AREA	90-13A	1000
SOUTH RWCU SYSTEM AREA	90-14A	1000
RWCU SYSTEM AREA	90-9A	1000
MG SET AREA	90-4A	1000
FUEL POOL AREA	90-1A	1000
SERVICE FLR AREA	90-2A	1000
NEW FUEL STORAGE	90-1A	1000

TABLE 3.1-G/3.2-G
INDICATIONS OF POTENTIAL OR SIGNIFICANT FUEL FAILURE
with RCS Barrier Intact

DRYWELL RADIATION UNIT 2		DRYWELL RADIATION UNIT 3	
2-RE-90-272A	$\geq$ 345 R/HR	3-RE-90-272A	$\geq$ 106 R/HR
2-RE-90-273A	$\geq$ 164 R/HR	3-RE-90-273A	$\geq$ 164 R/HR
REACTOR COOLANT ACTIVITY $\geq$ 300 μCI/gm DOSE EQUILAVENT IODINE-131		REACTOR COOLANT ACTIVITY $\geq$ 300 μCI/gm DOSE EQUILAVENT IODINE-131	