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PRC|HECG-SECT.03.2 (BASIS)|000	4	A	1	H	282221
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AX45

HOPE CREEK GENERATING STATION
EVENT CLASSIFICATION GUIDE TECHNICAL BASIS
January 17, 2007

CHANGE PAGES FOR
REVISION #35

The Table of Contents forms a general guide to the current revision of each section and attachment of the Hope Creek ECG Technical Basis. The changes that are made in this TOC Revision #35 are shown below.

1. Check that your revision packet is complete.
2. Add the revised documents.
3. Remove and recycle the outdated material listed below.

ADD			REMOVE		
Page	Description	Rev.	Page	Description	Rev.
ALL	T.O.C.	35	ALL	T.O.C.	34
ALL	Section 3.1	05	ALL	Section 3.1	04
ALL	Section 3.2	04	ALL	Section 3.2	03
ALL	Section 3.3	06	ALL	Section 3.3	05

HOPE CREEK ECG TECHNICAL BASIS
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i	**Introduction and Usage**		00	3	01/21/97
ii	**Glossary of Acronyms & Abbreviations**		00	5	01/21/97
1.0	**Fuel Clad Challenge**		02	9	12/15/04
2.0	**RCS Challenge**		01	8	04/21/04
3.0	**Fission Product Barriers (Table)**				
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HOPE CREEK ECG TECHNICAL BASIS
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REVISION SUMMARY Biennial Review Performed: Yes ____ No __X__

- During a review of ECG Technical Basis, it was identified that changes to EOP criteria were not reflected in the guidance in the ECG Technical Basis. Specifically, guidance related to changes made to EOP 206 and EOP 206A for the minimum number of SRVs required to ensure core submergence was changed to 5 SRVs and 50 psig RPV Pressure for EOP 206, and 5 SRVs and 275 psig for EOP 206A. Therefore, revisions to ECG Technical Basis guidance for EAL 3.1.1.a, 3.1.1.b, 3.2.1.b, and 3.3.1 have been made by rewording the paragraphs mentioning the number of SRVs and required pressures to if EOP 206 or EOP206A were or were not successful in implementation. These changes are considered editorial as they reflect previously reviewed and approved EOP changes. 70064753/0010

- Editorial changes made to ECG Technical Basis for EAL 3.1, 3.2, and 3.3 in the "Reference" Section. This is an update to the nomenclature and Operation Procedure numbers for Reactor Scram and various abnormal procedures that were previously made to the Operation Abnormal Procedures.

SIGNATURE PAGE

Prepared By: Dennis Kabachinski 01/08/07

Section/Attachments Revised N/A
(List Non-Editorial Only - Section/Attachments)

Reviewed By: N/A
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Reviewed By: N/A
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Reviewed By: N/A
Hope Creek Regulatory Assurance Date
(Reportable Action Level [Section 11] and associated Attachments marked by "L")

Reviewed By: [signature] 1-10-2007
EP Manager Date

Reviewed By: N/A
Nuclear Oversight Manager Date
(If Applicable)

SORC Review and Station Approvals

N/A
Mtg. No. Hope Creek Chairman

Date

N/A
Hope Creek Plant Manager

Date

Effective Date of this Revision: 01/17/07
Date

3.0 Fission Product Barriers

3.1 Fuel Clad Barrier

3.1.1 REACTOR WATER LEVEL

3.1.1.a

IC Potential Loss of Fuel Clad Barrier = 3 POINTS

EAL

Reactor Water Level REACHES **- 161" (Top of Active Fuel)**, EXCLUDING intentional lowering of Reactor Water Level during an ATWS

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Reactor Water Level reaching -161" (Top of Active Fuel - TAF), excluding intentional lowering of Reactor Water Level during an ATWS, results in an inability to maintain adequate core cooling by core submergence, causing a Potential Loss of the Fuel Clad Barrier. Without core submergence, the integrity of the fuel clad barrier is in jeopardy. Appropriate classification under this EAL is based on reaching Reactor Water Level of -161" (instead of being able to restore and maintain above -161") due to the potentially severe consequences of a loss of core submergence. Reactor Water Level reaching this threshold results from either a LOCA exceeding available makeup capacity or a Total Loss of High Pressure injection capability.

In addition, during an Anticipated Transient Without Scram (ATWS), it is possible that operator actions will be taken to intentionally lower Reactor Water Level to between -161" and -185", for Reactor Power Control purposes. For this event, classification must be made in accordance with EAL Section 5.0

Barrier Analysis

Fuel Clad Barrier has been potentially lost

ESCALATION CRITERIA

Emergency Classification will escalate based upon the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Core Submergence is the preferred method of maintaining adequate core cooling. When Reactor Water Level decreases to below TAF, the ability to effectively remove decay heat is being challenged, and as such the Fuel Clad fission product barrier can no longer be considered intact. While the Emergency Operating Procedures provide contingencies to establish adequate core cooling when Reactor Water Level drops below TAF (Steam Cooling with or without injection), these actions are designed to be an alternative method of providing adequate core cooling while actions are taken to reestablish core submergence. Sustained partial or total core uncovery can result in fuel clad damage and a significant release of fission products to the Reactor coolant. Sustained core uncovery can also result in a breach of the Reactor Vessel due to core melt material interaction with the RPV.

A Loss of Core Submergence will occur when the rate of inventory loss is greater than the rate of inventory makeup from High Pressure injection sources. This condition can occur as the result of the following events/sequences (excluding intentional lowering of Reactor Water Level during an ATWS).

> A LOCA will cause Reactor Water Level to reach the Top of Active Fuel when the LOCA is the result of a large break (momentary core uncovery is expected to occur under this condition) or when the LOCA is due to a small or intermediate break in combination with an inability of High Pressure injection sources to keep up with the leakrate.
>
> A Loss of High Pressure injection sources without the presence of a LOCA will also result in Reactor Water Level decreasing to TAF, due to continued Reactor Steam Flow without makeup.

Either of these events/sequences results in a challenge to the Fuel Clad Barrier when Reactor Water Level reaches TAF due to core uncovery, hence classification at this threshold is appropriate. However, for both these sequences, Low Pressure ECCS are designed to inject to the Reactor as Reactor Pressure decreases below the shutoff head of the pumps. Reactor Depressurization will occur either due to the LOCA or Manual initiation of Emergency Depressurization when Reactor Water Level reaches -161", provided injection systems are available. This will allow for restoration of Reactor Water Level and re-establishment of Core Submergence. Failure of these systems to restore and maintain Reactor Water Level above -185" will require escalation.

If all Reactor Level instrumentation is lost and EOP 206 or EOP 206A is entered, then a classification of a Site Area Emergency is warranted, based on EALs 3.1.1.a and 3.2.1.b (-161"). Successful implementation of EOP 206 or EOP 206A (SAG Entry is not required) assures a level at Top of Active Fuel is maintained. If EOP 206 or EOP 206A is not successful (SAG Entry is required), the process will not restore and maintain reactor level above –185" and a General Emergency is the appropriate classification based on EALs 3.1.1.b, 3.2.1.b, and 3.3.1.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC2
HC.OP-AB.ZZ-0000 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0101A (Q)-FC, ATWS – RPV Control
HC.OP-EO.ZZ-0206 (Q)-FC, RPV Flooding
HC.OP-EO.ZZ-0206A (Q)-FC, ATWS – RPV Flooding
HC.OP-EO.ZZ-0206 (Q), Conversion Document
BWR Owner's Group Emergency Procedure Guidelines, Rev. 4

3.0 Fission Product Barriers

3.1 Fuel Clad Barrier

3.1.1 REACTOR WATER LEVEL

3.1.1.b

IC Loss of Fuel Clad Barrier = 4 POINTS

EAL

RPV Level **CANNOT** be restored <u>AND</u> maintained above **-185"**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Inability to <u>restore and maintain</u> Reactor Water Level above -185, results in a loss of adequate core cooling by all mechanisms, causing a Loss of the Fuel Clad Barrier. Without adequate core cooling, the integrity of the fuel clad barrier can no longer be assured. Appropriate classification under this EAL is based on the failure of injection systems to restore and maintain Reactor Water Level above -185", following a condition that causes level to decrease below the threshold.

For example, a large break LOCA is expected to cause Reactor Water Level to momentarily decrease below -185", due to the response time of Low Pressure ECCS. As these systems initiate and commence injection to the Reactor, water level will begin to increase and should be able to be maintained above -185". In this case, classification under this EAL is not appropriate as plant systems have performed their intended design function and will eventually restore adequate core cooling by core submergence.

However, in the event that Low Pressure ECCS and alternate injection system, as defined in the EOPs are in a degraded condition (i.e., Station Blackout, ECCS Suction Strainer plugging, etc.) and Reactor Water Level can not be restored and maintained above -185", then classification under this EAL should occur due to the potential for release of energy to the containment from imminent fuel failure.

Barrier Analysis

Fuel Clad Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based upon the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Core submergence is the preferred method for maintaining adequate core cooling. The failure to reestablish Reactor Water Level above -161", the Top of Active Fuel (TAF), for an extended period of time could lead to a significant of fuel damage. With Reactor Water Level below TAF, but above -185", adequate core cooling occurs due to the cooling effects of steam generated in the covered portion of the core flowing through the uncovered portion (Steam Cooling). This method of cooling precludes any fuel clad temperature in the uncovered portion of the core from exceeding 1800°F. As Reactor Water Level drops below -185" with no injection available, this method of cooling becomes inadequate.

Prolonged lack of cooling may result in severe overheating of the fuel clad, additional release of energy from accelerated clad oxidation, and eventual fuel melting. For events starting from full power operation, the failure to promptly reflood could result in some fuel melting. Even under these conditions vessel failure and containment failure with resultant release to the public would not be expected for some time. Reactor Water Level remaining below TAF for an extended amount of time represents an early indicator that significant core damage is in progress while providing sufficient time to initiate public protective actions.

Ample time should be allowed for Low Pressure ECCS and alternate injection systems to restore Reactor Water Level prior to entry into this classification. The time basis for deciding whether or not Reactor Water can be maintained > -185" should be based on the rate of reactor depressurization, the availability of low-pressure injection sources, (ECCS and alternate injection systems), and the rate of Reactor coolant inventory loss. Indications such as Reactor Water Level trend, injection flow rates, containment parameter trends, and low pressure injection system operability should also be considered.

In the event, Reactor Water Level cannot be restored > -185", containment flooding will be required by the EOPs. This will attempt to flood the containment as a means of flooding the RPV, and use a flooded containment as a heat sink for the nuclear fuel.

If all Reactor Level instrumentation is lost and EOP 206 or EOP 206A is entered, then a classification of a Site Area Emergency is warranted, based on EALs 3.1.1.a and 3.2.1.b (-161"). Successful implementation of EOP 206 or EOP 206A (SAG Entry is not required) assures a level at Top of Active Fuel is maintained. If EOP 206 or EOP 206A is not successful (SAG Entry is required), the process will not restore and maintain reactor level above -185" and a General Emergency is the appropriate classification based on EALs 3.1.1.b, 3.2.1.b, and 3.3.1.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, FC2
HC.OP-AB.ZZ-0000 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0101A (Q)-FC, ATWS – RPV Control
HC.OP-EO.ZZ-0206 (Q)-FC, RPV Flooding
HC.OP-EO.ZZ-0206A (Q)-FC, ATWS – RPV Flooding
HC.OP-EO.ZZ-0206 (Q), Conversion Document
BWR Owners Group Emergency Procedure Guidelines, Revision 4

3.0 Fission Product Barriers

3.1 Fuel Clad Barrier

3.1.2 DRYWELL ATMOSPHERE POST ACCIDENT (DAPA) RADIATION LEVEL

IC Loss of Fuel Clad Barrier = 4 POINTS

EAL

DAPA Radiation Monitor reading $\geq$ **5,000 R/hr**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Drywell Atmosphere Post Accident (DAPA) Radiation monitors indicating 5,000 R/hr or greater corresponds to an instantaneous release of Reactor Coolant with a concentration of 300 µCi/gm Dose Equivalent Iodine-131 (DEI-131) into the Primary Containment . This value of Reactor Coolant Activity is well above the threshold that could occur as the result of Iodine Spiking, resin/chemical intrusion transients or a HWCI System malfunction. This activity level corresponds to fuel clad damage of approximately 3.8%.

In addition, there are other events that could cause Drywell Atmosphere radiation levels to increase to this threshold, without a LOCA in the Drywell. These events involve shine from the reactor core if it is uncovered. While such events would not necessarily involve the calculated fuel clad damage percentage, they would be classifiable under other EALs as a Site Area Emergency level or higher.

Barrier Analysis

Fuel Clad Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

EAL 3.1.3 provides a core damage analysis showing that a Reactor Coolant activity of 300 µCi/gm Dose Equivalent Iodine-131(DEI) is indicative of 3.8% clad damage. Using Attachment 2 of HC.EP-EP.ZZ-0205, 1% clad damage is indicated by a DAPA reading of 1.4E3 R/hr at 0.1 hrs after shutdown (the most conservative). This is shown on the Attachment as the 0.1% TID line. Extrapolating to the 3.8% clad damage point gives 5.32E3 R/hr. This is rounded to 5.0E3 R/hr. Hence, the Fuel Clad Barrier is lost.

NUMARC EAL RC3 addresses the use of DAPA to assess the status of the RCS Barrier, based on the release of Reactor Coolant into the Drywell. This EAL threshold is calculated assuming the instantaneous release and dispersal of the Reactor Coolant noble gas and iodine inventory associated with normal operating concentrations (within TS limits) into the Drywell Atmosphere. The reading would be lower than the threshold for EAL 3.1.2, thus being indicative of an RCS leak only. <u>However</u>, due to the inability of the DAPA radiation monitors to distinguish between a cloud of released RCS gases and shine from the Reactor Vessel and adjacent piping and components, this EAL is being omitted, as permitted by the NUMARC EALs, and other indications of RCS Leakage are being used. It should be recognized that DAPA exceeding 5000 R/hr would most likely occur due to core uncovery, as Reactor Water Level decreases below the Top of Active Fuel. This condition will result in appropriate escalation to a Site Area Emergency in the Fission Product Barrier Table, and hence use of DAPA exceeding 5000 R/hr is not needed to detect a Loss of the RCS Barrier.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC3
NUMARC NESP-007, RC3
HC.EP-EP.ZZ-0205 (Q), TSC - Post Accident Core Damage Assessment
HC.OP-AR.SP-0001 (Q), Radiation Monitoring System Alarm Response – RM-11

3.0 Fission Product Barriers

3.1 Fuel Clad Barrier

3.1.3 RCS IODINE CONCENTRATION

IC Loss of Fuel Clad Barrier = 4 POINTS

EAL

Reactor Coolant Sample Activity $\geq$ **300 µCi/gm Dose Equivalent I-131**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Reactor Coolant sample analysis with specific activity greater than or equal to 300 µCi/gm Dose Equivalent I-131 (DEI-131) indicates fuel clad damage due to significant clad heating or mechanical stress, causing a Loss of the Fuel Clad Barrier. This threshold is well above the activity level that could occur as the result of Iodine spiking. The use of the term "Valid" as a qualifier for event classification is not required, since Reactor Coolant Activity of this magnitude can only occur as the result of fuel clad damage. This activity level corresponds to approximately 3.8% fuel clad damage.

Barrier Analysis

Fuel Clad Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

The percentage of Fuel Damage that corresponds to an RCS Activity of 300 μCi/gm DEI-131 is calculated as follows (for purposes of this calculation, cc and gm are considered equivalent):

Dose Factors (RG-1.109)

I-131 = 4.39E-3
I-132 = 5.23E-5
I-133 = 1.04E-3
I-134 = 1.37E-5
I-135 = 2.14E-4

Total core inventory (HCGS-UFSAR, table 12.2-135). This table gives 50% inventory, so table values are multiplied by 2.0.

I-131 = 8.64E7 Ci
I-132 = 1.29E8 Ci
I-133 = 1.99E8 Ci
I-134 = 2.32E8 Ci
I-135 = 1.81E8 Ci

Reactor Water Volume = 13000 cubic feet (HCGS-UFSAR, table 12.3-2)

Clad Release Fraction for iodines = 0.02 (Table 4.1, NUREG-1228)

The activity of each isotope in the clad would then be:

I-131 = 8.64E7(0.02) = 1.73E6 Ci
I-132 = 1.29E8(0.02) = 2.58E6 Ci
I-133 = 1.99E8(0.02) = 3.98E6 Ci
I-134 = 2.32E8(0.02) = 4.64E6 Ci
I-135 = 1.81E8(0.02) = 3.62E6 Ci

These activities are equivalent to 2.89E6 Ci DEI-131

$$DEI-131 = \frac{4.39E-3(1.73E6) + 5.23E-5(2.58E6) + 1.04E-3(3.98E6) + 1.37E-5(4.64E6) + 2.14E-4(3.62E6)}{4.93E-3}$$

Calculating the equivalent concentration:

$$\text{Conc} = \frac{2.89\text{E}6 \text{ Ci}(1\text{E}6\mu\text{Ci / Ci})}{13000 \text{ cf}(2.8\text{E}4 \text{ cc / cf})} = 7.94\text{E}3\mu\text{Ci/cc}$$

which represents the 100% clad damage concentration.

300 μCi/cc DEI-131 is then equivalent to:

$$\frac{300\ \mu\text{Ci / cc}}{7.94\text{E}3\ \mu\text{Ci / cc}} = 3.78\%$$

This is rounded to 3.8%.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC1
HC.OP-AB.RPV-0008 (Q), Reactor Coolant Activity
HCGS Technical Specification LCO 3.4.5
NUREG 1228 - Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents, Table 4.1
Reg. Guide 1.109, Table E-9
HCGS-UFSAR, Table 12.2-135 and Table 12.3-2
10 CFR100

3.0 Fission Product Barriers

3.1 Fuel Clad Barrier

3.1.4 EMERGENCY COORDINATOR JUDGMENT

3.1.4.a/ 3.1.4.b

IC Potential Loss (= 3 POINTS) or Loss of Fuel Clad Barrier (= 4 POINTS)

EAL

ANY condition, in the opinion of the EC, that indicates EITHER a Potential Loss OR Loss of the Fuel Clad Barrier

OPERATIONAL CONDITION - 1, 2, 3

BASIS

This EAL allows the Emergency Coordinator (EC) to address any condition that effects the integrity of the Fuel Clad Barrier that is not already covered elsewhere in the Fission Product Barrier Table. A complete loss of the ability to monitor the Fuel Clad Barrier should be considered as a "Potential Loss" of that barrier.

Barrier Analysis

Fuel Clad Barrier has been potentially lost or lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the potential loss or loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

None

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC5

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.1 REACTOR WATER LEVEL

3.2.1.a

IC Potential Loss of RCS Barrier = 3 POINTS

EAL

Reactor Water Level REACHES **-129"**, EXCLUDING intentional lowering of Reactor Water Level during an ATWS

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Reactor Water Level reaching -129", excluding intentional lowering of Reactor Water Level during an ATWS, indicates that the inventory loss from the RCS exceeds the capacity of available High Pressure injection sources. Below this threshold, a challenge to maintaining Adequate Core Cooling by core submergence exists, based on Reactor Water Level continuing to decrease, thus a Potential Loss of the RCS Barrier exists.

Without core submergence, the integrity of the Fuel Clad would be in jeopardy. Appropriate classification under this EAL is based on reaching Reactor Water Level of -129" (instead of being able to restore and maintain above -129"), due to the challenge that exists to core submergence. Reactor Water Level reaching this threshold results from either a LOCA exceeding available makeup capacity or a Total Loss of High Pressure injection capability.

In addition, during an Anticipated Transient Without Scram (ATWS), it is possible that operator action will be taken to intentionally lower Reactor Water Level to below -129" for Reactor Power Control purposes. For this event, classification must be made in accordance with EAL Section 5.0.

Barrier Analysis

RCS Barrier has been potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Core Submergence is the preferred method of maintaining adequate core cooling. When Reactor Water Level decreases to -129", a significant challenge to continued core submergence exists. The threshold for this EAL corresponds to the initiation setpoint for the low pressure Emergency Core Cooling Systems (ECCS).

Reactor Water Level reaching -129" occurs when the rate of inventory loss is greater than the rate of inventory makeup from High Pressure injection sources. This condition can occur as the result of the following events/sequences (excluding intentional lowering of Reactor Water level during an ATWS).

> A LOCA will cause Reactor Water Level to reach -129" when the LOCA is the result of a large break (momentary core uncovery is expected to occur under this condition) or when the LOCA is due to a small or intermediate break in combination with an inability of High Pressure injection sources to keep up with the leak rate.
>
> A Loss of High Pressure injection sources without the presence of a LOCA will also result in Reactor Water Level decreasing to -129", due to continued Reactor Steam Flow without makeup.

Either of these events/sequences results in a potential challenge to the RCS Barrier when Reactor Water level reaches -129", hence classification at this threshold is appropriate. However, for both these sequences, low Pressure ECCS are designed to inject to the Reactor as Reactor Pressure decreases below the shutoff head of the pumps. Reactor Depressurization will occur either due to the LOCA or Manual initiation of Emergency Depressurization when Reactor Water Level reaches -161", provided injection systems are available. This will allow for restoration of Reactor Water Level and re-establishment of Core Submergence.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, RC5
HC.OP.AB-CONT-0002 (Q), Primary Containment
HC.OP-AB.RPV-0004 (Q), Reactor Level Control
HC.OP-AB.ZZ-0000 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor/Pressure Vessel (RPV) Control
HC.OP-SO.SM-0001 (Q), Isolation Systems Operation
HCGS Technical Specifications LCO 3/4.3, Instrumentation

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.1 REACTOR WATER LEVEL

3.2.1.b

IC Loss of RCS Barrier = 4 POINTS

EAL

Reactor Water Level REACHES **-161" (Top of Active Fuel)**, EXCLUDING intentional lowering of Reactor Water Level during an ATWS

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Reactor Water Level reaching -161" (Top of Active Fuel - TAF), excluding intentional lowering of Reactor Water Level during an ATWS, results in an inability to maintain adequate core cooling by core submergence, causing a Loss of the RCS Barrier. Without core submergence, the integrity of the fuel clad barrier is in jeopardy. Appropriate classification under this EAL is based on reaching Reactor Water Level of -161" (instead of being able to restore and maintain above -161") due to the potentially severe consequences of a loss of core submergence. Reactor Water Level reaching this threshold results from either a LOCA exceeding available makeup capacity or a Total Loss of High Pressure injection capability.

In addition, during an Anticipated Transient Without Scram (ATWS), it is possible that operator actions will be taken to intentionally lower Reactor Water Level to between -161" and -185", for Reactor Power Control purposes. For this event, classification must be made in accordance with EAL Section 5.0

Barrier Analysis

RCS Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based upon the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Core Submergence is the preferred method of maintaining adequate core cooling. When Reactor Water Level decreases to below TAF, the ability to effectively remove decay heat is being challenged, and as such the Fuel Clad barrier can no longer be considered intact. While the Emergency Operating Procedures provide contingencies to establish adequate core cooling when Reactor Water Level drops below TAF (Steam Cooling with or without injection), these actions are designed to be an alternative method of providing adequate core cooling while actions are taken to reestablish core submergence. Sustained partial or total core uncovery can result in fuel clad damage and a significant release of fission products to the Reactor coolant. Sustained core uncovery can also result in a breach of the Reactor Vessel due to core melt material interaction with the RPV.

A Loss of Core Submergence will occur when the rate of inventory loss is greater than the rate of inventory makeup from High Pressure injection sources. This condition can occur as the result of the following events/sequences (excluding intentional lowering of Reactor Water Level during an ATWS).

A LOCA will cause Reactor Water Level to reach the Top of Active Fuel when the LOCA is the result of a large break (momentary core uncovery is expected to occur under this condition) or when the LOCA is due to a small or intermediate break in combination with an inability of High Pressure injection sources to keep up with the leak rate.

A Loss of High Pressure injection sources without the presence of a LOCA will also result in Reactor Water Level decreasing to TAF, due to continued Reactor Steam Flow without makeup.

Either of these events/sequences results in a challenge to the Fuel Clad Barrier when Reactor Water Level reaches TAF due to core uncovery, hence classification at this threshold is appropriate. However, for both these sequences, Low Pressure ECCS are designed to inject to the Reactor as Reactor Pressure decreases below the shutoff head of the pumps. Reactor Depressurization will occur either due to the LOCA or Manual initiation of Emergency Depressurization when Reactor Water Level reaches -161", provided injection systems are available. This will allow for restoration of Reactor Water Level and re-establishment of Core Submergence.

If all Reactor Level instrumentation is lost and EOP 206 or EOP 206A is entered, then a classification of a Site Area Emergency is warranted, based on EALs 3.1.1.a and 3.2.1.b (-161"). Successful implementation of EOP 206 or EOP 206A (SAG Entry is not required) assures a level at Top of Active Fuel is maintained. If EOP 206 or EOP 206A is not successful (SAG Entry is required), the process will not restore and maintain reactor level above –185" and a General Emergency is the appropriate classification based on EALs 3.1.1.b, 3.2.1.b, and 3.3.1.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, RC4
HC.OP-AB.ZZ-0000 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0101A (Q)-FC, ATWS – RPV Control
HC.OP-EO.ZZ-0206 (Q)-FC, RPV Flooding
HC.OP-EO.ZZ-0206A (Q)-FC, ATWS – RPV Flooding
HC.OP-EO.ZZ-0206 (Q), Conversion Document
BWR Owner's Group Emergency Procedure Guidelines, Rev. 4

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.2 RCS LEAK RATE/DRYWELL PRESSURE

3.2.2.a

IC Potential Loss of RCS Barrier = 3 POINTS

EAL

Unisolable RCS Leak Rate ≥ **50 GPM** INSIDE Primary Containment

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Unisolable RCS Leak Rate exceeding 50 GPM, inside Primary Containment is indicative of a potential loss of the RCS. An unisolable leak rate of this magnitude is significant due to the potential for further break propagation, resulting in a much higher loss of inventory with an inability to isolate the leak source. As such, this threshold is considered a Potential Loss of the RCS. Leakage just above the 50 GPM threshold is well within the capacity of normal and emergency injection systems and is not a significant concern for core uncovery. However, 50 GPM is the minimum leak rate that would be classified under this EAL, with the maximum being equivalent to the leak rate that would result in either Reactor Water Level reaching -129" or Drywell Pressure reaching 1.68 PSIG, since these two conditions are obviously more recognizable to Control Room personnel, than an existing leak rate.

Specifying an unisolable RCS leak as part of the threshold for this EAL precludes classifying events such as an isolable Reactor Recirculation Pump dual seal failure under this EAL.

Barrier Analysis

RCS Barrier has been potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

It is important to recognize that the unisolable RCS leak rate established in this EAL is inside the Primary Containment. The inability to isolate the leak would eventually lead to a High Drywell Pressure (> 1.68 PSIG) actuation of RPS, ECCS and PCIS. The actuation would lead to an isolation of the Drywell Floor and Equipment Drain sumps, complicating efforts to further identify and quantify any changes in the existing leak rate. In addition, monitoring of the leak rate could be limited by reaching the upper range (50 GPM) of the Drywell Leak Detection channels (9AX313 - Equipment, 9AX314- Floor Drain).

For leakage outside Containment, since quantification of the leak rate is much more difficult due to the physical size of the Reactor Building, receipt of a **Valid** isolation signal has been established as the threshold for classification of this type of leakage.

DEVIATION

None

REFERENCES

NUMARC NESP-007, RC1
NUMARC Questions and Answers, June 1993, "Fission Product Barrier Question #11"
HC.OP-AB.CONT-0001 (Q), Drywell Pressure
HC.OP-AB.CONT-0002 (Q), Primary Containment
HC.OP-AB.ZZ-0000 (Q)-FC, Reactor Scram
HC.RP-AR.SP-0001 (Q), Radiation Monitoring System Alarm Response – RM-11
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor/Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0103/4 (Q)-FC, Reactor Building and Radioactive Release Control
HC.OP-GP-ZZ-0005 (Q), Drywell Leakage Source Detection
HC.OP-SO.SM-0001 (Q), Isolation Systems Operation

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.2 RCS LEAK RATE/DRYWELL PRESSURE

3.2.2.b

IC Loss of RCS Barrier = 4 POINTS

EAL

Valid High Drywell Pressure Condition (**≥ 1.68 psig**)

OPERATIONAL CONDITION - 1, 2, 3

BASIS

A **Valid** High Drywell Pressure Condition (≥ 1.68 PSIG) is indicative of the release of high energy Reactor Coolant from the RCS into the Drywell and hence is considered a Loss of the RCS Barrier. **Valid** is defined as the High Drywell Pressure condition specifically due to RCS leakage into the Drywell, ensuring that event classification under this EAL is truly reflective of a degraded RCS Barrier. This precludes unwarranted event declaration as the result of system malfunctions, including a loss of Drywell Cooling or inadvertent Drywell makeup. Indication of an RCS leak should be positively determined by observing Primary Containment parameters, including Drywell Pressure and Temperature trends, Drywell Equipment and Floor Drain sump levels, DAPA Radiation levels, atmospheric pressure, Torus Pressure, and the status of Drywell Cooling systems.

An isolable Reactor Recirculation Pump dual seal failure should not result in Drywell Pressure reaching the threshold for this EAL, hence classification under this EAL should not occur.

Barrier Analysis

RCS Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

RCS Leakage into the Drywell exceeding 50 GPM is substantially greater than the RCS leakage thresholds established in EAL Section 2.1.1, and represents further degradation of the RCS barrier. Inability to isolate the RCS leakage would eventually result in a High Drywell Pressure (>1.68 PSIG) actuation of RPS, ECCS and PCIS. The actuation would lead to an isolation of the Drywell Floor and Equipment Drain sumps, complicating efforts to further identify and quantify any changes in the leak rate. In addition, monitoring of the leak rate could be limited by reaching the upper range (50 GPM) of the Drywell Leak Detection channels (9AX313 - Equipment, 9AX314 - Floor Drain).

There are multiple Control Room indicators and alarms that can be used to determine the presence of a High Drywell Pressure condition. Overhead Annunciators will alarm at 1.5 PSIG and 1.68 PSIG. Plant automatic response to a High Drywell Pressure condition includes: a reactor scram, ECCS initiation, trip of the drywell cooling fans and isolation of the cooling water to the drywell. These actuations may mask the trend in drywell pressure. For example, the scram will result in less heat being added to the containment and the cooling water isolation will result in no heat being removed.

Actions initiated as part of increasing drywell pressure condition include investigation of the source of the increased leakage into the drywell maximizing drywell cooling and venting the Drywell (if release criteria can be satisfied). These actions are designed to control and relieve increasing drywell pressure.

DEVIATION

None

REFERENCES

NUMARC NESP-0007, RC2
NUMARC Questions and Answers, June 1993, "Fission Product Barrier Question #11"
HC.OP-AB.CONT-0001 (Q), Drywell Pressure
HC.OP-AB.CONT-0002 (Q), Primary Containment
HC.OP-AB.ZZ-0000 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor/Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-GP.ZZ-0005 (Q), Drywell Leak Source Detection
HC.OP-SO.SM-0001 (Q), Isolation Systems Operation
Hope Creek Appendix A based on NEDO-2121, Supplement A to BWR Owners Group Emergency Procedure Guidelines, Revision 4
HCGS Technical Specifications LCO 3/4.3, Instrumentation

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.3 RCS LINE BREAK/CONTAINMENT BYPASS

3.2.3.a

IC Potential Loss of RCS Barrier = 3 POINTS

EAL

> **RCS Line Break** OUTSIDE Primary Containment, resulting in a **Valid** Containment Isolation Signal for ANY one of the following systems:
>
> - NSSSS
> - HPCI
> - RCIC
>
> **AND**
>
> **UNISOLABLE** leakage OUTSIDE Primary Containment (AFTER ISOLATION from the Main Control Room has been attempted) as indication by one of the following:
> - Downstream pathway to the environment exists
> - Radiation monitors, area temperature or flow

OPERATIONAL CONDITION - 1, 2, 3

BASIS

An RCS Line Break outside Primary Containment that results in a **Valid** Isolation Signal for any of the systems listed in the EAL requires closure of the associated Primary Containment Isolation valves to maintain RCS and Primary Containment integrity under abnormal conditions. A failure of these isolation valves to isolate directly allows Reactor Coolant to be released outside the Primary Containment (Containment Bypass), resulting in a Loss of RCS and Loss of Containment. An RCS Line is ANY line that communicates directly with the Reactor. An RCS Line Break with indication of continuing flow is classified under this EAL, due to the continuing discharge of Reactor Coolant outside the Primary Containment. This is the only condition that warrants classification under this EAL.

Valid is defined as the isolation signal specifically being the result of an RCS Line Break, thus ensuring that the RCS discharge is of significant magnitude to pose a threat to the integrity of the

RCS Barrier. This precludes unwarranted Event Classification as the result of a condition that results in limited leakage with no potential for "break propagation", including valve packing leaks outside Primary Containment and RWCU Pump Seal Leaks. In addition, isolation signal generated from known failures in other systems, that do not result in Reactor Coolant discharging outside the Primary Containment do not warrant Event Classification under this EAL either. Examples of such failures include a high temperature isolation resulting from a loss of ventilation or cooling water, spurious actuation during I&C surveillance testing or a low Reactor Water Level Condition due to a Loss of High Pressure injection capability.

UNISOLABLE means all valves in a penetration cannot be immediately closed from the Control Room. This EAL ALLOWS for valve closure from the Main Control Room to isolate any systems not completely isolated, prior to event classification. Isolation is defined as the closure of ANY valve from the Main Control Room in the system(s) not completely isolated. For example, if the isolation logic fails to cause valve closure, but operator actions implemented in the Main Control Room successfully isolates the containment breach path, then classification under this EAL is not warranted. This includes Motor Operated Valves not controlled by the isolation logic, but are manually controlled from the Main Control Room.

Barrier Analysis

RCS Barrier has been potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

NSSSS isolations, as well as HPCI and RCIC steam line isolations, are associated with systems that are part of the RCS boundary and penetrate the Primary Containment. Isolation requirements for these lines are covered in 10CFR50, Appendix A, General Design Criteria 55. These systems form a closed loop outside the Primary Containment, and are not open or potentially open to the environment. They are included in this EAL since they represent an extension of the RCS boundary beyond the Primary Containment, and a potential release path from the RCS to the environment. Without a completed isolation, continuing flow/leakage represents a situation where Reactor Coolant is discharging outside the Primary Containment, including areas in the Reactor Building addressed in the EOPs.

Indication of an unisolable leak includes: flow indication through isolated lines, increasing Reactor Building area temperatures, area radiation levels, also increases in sump levels, or room levels in spaces associated with affected lines, as well as increases in Plant Vent Effluent levels.

The isolation valve status of all isolation groups is monitored for quick reference on SPDS, to be backed up by operator observation of valve status.

DEVIATION

NONE

REFERENCES

NUMARC NESP-007, RC1
10 CFR50, App. A, GDC 55
10 CFR 100
HC.OP-AB.CONT-0002 (Q), Primary Containment
HC.OP-AB.RPV-0008 (Q), Reactor Coolant Activity
HC.OP-AB.ZZ-0000 (Q)-FC, Reactor Scram
HC.OP-AR.SP-0001 (Q), Radiation Monitoring System Alarm Response – RM-11
HC.OP-AR.ZZ-0011 (Q), Annunciator Response Procedures, Window C6
HC.OP-AR.ZZ-0012 (Q), Annunciator Response Procedures, Window C8
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor/Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0103/4 (Q)-FC, Reactor Building and Radioactive Release Control
HC.OP-SO.SM-0001 (Q), Isolation Systems Operation
HCGS Technical Specifications LCO 3/4.3, Instrumentation
HCGS UFSAR, Section 6.2.4.3.1

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.3 RCS LINE BREAK/CONTAINMENT BYPASS

3.2.3.b

IC Loss of RCS Barrier = 4 POINTS

EAL

Main Steam Line Break OUTSIDE Primary Containment, resulting in an AUTOMATIC MSIV Isolation Signal **AND** **UNISOLABLE** leakage OUTSIDE Primary Containment (AFTER ISOLATION from the Main Control Room has been attempted) as indication by one of the following: • Downstream pathway to the environment exists • Radiation monitors, area temperature or flow

OPERATIONAL CONDITION - 1, 2, 3

BASIS

This EAL is specific to a break outside the Primary Containment, since a break outside represents a potential challenge to Primary Containment Integrity due to the Containment Bypass condition that would exist until MSIV closure occurred. Failure to completely isolate the effected Main Steam Line(s) as determined by valve position and indication of continuing leakage would result in an additional Loss of the Primary Containment Barrier.

UNISOLABLE means all valves in a penetration cannot be immediately closed from the Control Room. This EAL ALLOWS for valve closure from the Main Control Room to isolate any Main Steam Line not completely isolated, prior to event classification. Isolation is defined as the closure of ANY valve from the Main Control Room in the system(s) not completely isolated. For example, if the isolation logic fails to cause valve closure, but operator actions implemented in the Main Control Room successfully isolates the effected Main Steam Line(s), then event classification under this EAL is not warranted. This includes Motor Operated Valves not controlled by the isolation logic, but are manually controlled from the Main Control Room (i.e. Main Steam Stop Valves 1ABHV-3631 A/B/C/D).

Barrier Analysis

RCS Barrier has been lost

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional barriers per EAL section 3.0.

DISCUSSION

The Main Steam System is associated with systems that are part of the RCS boundary and penetrate the Primary Containment. Isolation requirements for these lines are covered in 10CFR50, Appendix A, General Design Criteria 55. These systems form a closed loop outside the Primary Containment and are not open or potentially open to the environment. These systems represent an extension of the RCS Barrier beyond the Primary Containment.

Positive identification of a Main Steam Line Break outside the Primary Containment can be based on receipt of the following Overhead Annunciators:

NSSSS ISLN SIG - STM TNL TEMP HI	(C8-C4)
NSSSS ISLN SIG - MN STM FLOW HI	(C8-B4)
MSIV CLOSURE	(C5-B3)

as well as the following indications:

MSIV TRIP LOGIC TRIPPED
Rapid changes in Main Steam Line Flow and Steam Tunnel Temperatures

DEVIATION

NONE

REFERENCES

NUMARC NESP-007, RC1
NUMARC Question and Answer, June 1983, "Fission Product Barrier- BWR" Question #4
10 CFR50, App. A, GDC 55
10 CFR 100
HC.OP-AB.CONT-0002 (Q), Primary Containment
HC.OP-AB.RPV-0008 (Q), Reactor Coolant Activity
HC.OP-AB.ZZ-0000 (Q)-FC, Reactor Scram
HC.OP-AR.SP-0001 (Q), Radiation Monitoring System Alarm Response – RM-11
HC.OP-AR.ZZ-0011 (Q), Annunciator Response Procedures, Window C6
HC.OP-AR.ZZ-0012 (Q), Annunciator Response Procedures, Window C8
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor/Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0103/4 (Q)-FC, Reactor Building and Radioactive Release Control
HC.OP-SO.SM-0001 (Q), Isolation Systems Operation
HCGS Technical Specifications, LCO 3/4.3
HCGS UFSAR, Section 6.2.4.3.1

3.0 Fission Product Barriers

3.2 RCS Barrier

3.2.4 EMERGENCY COORDINATOR JUDGMENT

3.2.4.a/ 3.2.4.b

IC Potential Loss (= 3 POINTS) or Loss of RCS Barrier (= 4 POINTS)

EAL

ANY condition, in the opinion of the EC, that indicates EITHER a Potential Loss OR Loss of the RCS Barrier

OPERATIONAL CONDITION - 1, 2, 3

BASIS

This EAL allows the Emergency Coordinator (EC) to address any condition that affects the integrity of the RCS Barrier that is not already covered elsewhere in the Fission Product Barrier Table. A complete loss of the ability to monitor the RCS barrier should be considered as a "Potential Loss" of that barrier.

Barrier Analysis

RCS Barrier has been potentially lost or lost.

ESCALATION CRITERIA

Emergency Classification will be escalate based on the Potential Loss or Loss of additional barriers per EAL section 3.0.

DISCUSSION

None

DEVIATION

None

REFERENCES

NUMARC NESP-007, RC6

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.1 REACTOR WATER LEVEL

IC Potential Loss of Containment Barrier = 1 POINT

EAL

RPV Level **CANNOT** be restored AND maintained above **-185"**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Inability to restore and maintain Reactor Water Level above -185", results in a loss of adequate core cooling by all mechanisms, causing a Potential Loss of the Fuel Clad Barrier. Without adequate core cooling, the integrity of the Containment is being challenged and can no longer be assured. Appropriate classification under this EAL is based on the failure of injection systems to restore and maintain Reactor Water Level above -185", following a condition that causes level to decrease below the threshold.

For example, a large break LOCA is expected to cause Reactor Water Level to momentarily decrease below -185", due to the response time of Low Pressure ECCS. As these systems initiate and commence injection to the Reactor, water level will begin to increase and should be able to be maintained above -185". In this case, classification under this EAL is not appropriate as plant systems have performed their intended design function and will eventually restore adequate core cooling by core submergence. However, in the event that Low Pressure ECCS and alternate injection system, as defined in the EOPs are in a degraded condition (i.e., Station Blackout, ECCS Suction Strainer plugging, etc.) and Reactor Water Level can not be restored and maintained above -185", then classification under this EAL should occur due to the Potential Loss of Containment from the release of energy to the containment from imminent fuel failure.

Barrier Analysis

Primary Containment Barrier has been potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based upon the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Core submergence is the preferred method for maintaining adequate core cooling. The failure to reestablish Reactor Water Level above -161", the Top of Active Fuel (TAF), for an extended period of time could lead to significant fuel damage. With Reactor Water Level below TAF, but above -185", adequate core cooling occurs due to the cooling effects of steam generated in the covered portion of the core flowing through the uncovered portion (Steam Cooling).

This method of cooling precludes any fuel clad temperature in the uncovered portion of the core from exceeding 1800°F. As Reactor Water Level drops below -185" with no injection available, this method of cooling becomes inadequate. Prolonged lack of cooling may result in severe overheating of the fuel clad, additional release of energy from accelerated clad oxidation, and eventual fuel melting.

For events starting from full power operation, the failure to promptly reflood could result in some fuel melting. Even under these conditions vessel failure and containment failure with resultant release to the public would not be expected for some time. Reactor Water Level remaining below TAF for an extended amount of time represents an early indicator that significant core damage is in progress while providing sufficient time to initiate public protective actions.

Ample time should be provided for Low Pressure ECCS and alternate injection systems restore Reactor Water Level prior to entry into this classification. The time basis for deciding whether or not Reactor Water can be maintained >-185" should be based on the rate of reactor depressurization, the availability of low-pressure injection sources, (ECCS and alternate injection systems), and the rate of Reactor coolant inventory loss. Indications such as Reactor Water Level trend, injection flow rates, containment parameter trends, and low pressure injection system operability should also be considered.

In the event Reactor Water Level cannot be restored >-185", Severe Accident Guidelines entry is required (containment flooding) by the EOPs. This will attempt to flood the containment as a means of flooding the RPV, and use a flooded containment as a heat sink for the nuclear fuel.

If all Reactor Level instrumentation is lost and EOP 206 or EOP 206A is entered, then a classification of a Site Area Emergency is warranted, based on EALs 3.1.1.a and 3.2.1.b (-161"). Successful implementation of EOP 206 or EOP 206A (SAG Entry is not required) assures a level at Top of Active Fuel is maintained. If EOP 206 or EOP 206A is not successful (SAG Entry is required), the process will not restore and maintain reactor level above –185" and a General Emergency is the appropriate classification based on EALs 3.1.1.b, 3.2.1.b, and 3.3.1.

DEVIATION

None

REFERENCES

NUMARC NESP-007, PC4
HC.OP.EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP.EO.ZZ-0101A (Q)-FC, ATWS – RPV control
HC.OP.EO.ZZ-206 (Q)-FC, RPV Flooding
HC.OP.EO.ZZ-0206A (Q)-FC, ATWS RPV Flooding
HC.OP-EO.ZZ-0206 (Q), Conversion Document
BWR Owners Group Emergency Procedure and Severe Accident Guidelines, Revision 1

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.2 DRYWELL PRESSURE/H_2

3.3.2.a/ 3.3.2.c

IC Potential Loss of Containment Barrier = 1 POINT

EAL

Suppression Chamber pressure <u>CANNOT BE MAINTAINED</u> below 65 psig
<u>OR</u>
Primary Containment H_2 concentration > 4% and O_2 concentration > 5%

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Containment venting required by the EOPs indicates a degrading condition in containment and is implemented in an effort to preclude containment failure. Venting is required before Suppression Chamber pressure reaches **65 PSIG** or Hydrogen concentration reaches the Lower Explosive Limit (LEL = **4%**) and Oxygen concentration reaches **5%**. Exceeding these parameters creates the potential for an unisolable breach of the primary containment, which could result in an uncontrolled, unmonitored, and untreated release of radioactivity to the environment. This EAL represents a Potential Loss of Containment, since containment venting is required due to Containment parameters potentially exceeding their design limits. The magnitude of any radiological release is dependent upon events leading to the requirement for emergency venting, including a loss of the RCS and a loss of the Fuel Clad Barriers.

A Downcomer failure, by itself, <u>does not</u> represent a Loss of the Primary Containment Barrier. This failure does, however, render the Primary Containment inoperable per the Technical Specification, as Primary Containment integrity has been compromised. A Downcomer failure combined with a large break LOCA will likely result in a Potential Loss of Primary Containment under this EAL if Containment pressure cannot be maintained below 65 PSIG and Containment Venting is required.

Barrier Analysis

Primary Containment Barrier has been potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Venting of the Primary Containment is initiated to preserve containment integrity under accident conditions. Primary Containment venting is required when Suppression Chamber cannot be maintained below 65 psig, which is well above the maximum pressure expected to be present in the Primary Containment during a design basis Loss of Coolant Accident (LOCA).

Primary Containment venting is also required based on hydrogen concentrations exceeding 4%. H_2 concentration in excess of 6.0 % requires Emergency Depressurization and subsequent containment venting. Venting is continued until either H_2 concentration has been reduced to <6.0% or O_2 levels have been reduced to <5.0%. Venting with elevated hydrogen concentration conditions ensures that containment failure resulting from a hydrogen detonation or deflagration does not occur.

The elevated hydrogen in the containment may result from excessive zircaloy-water reaction occurring following a LOCA. Additionally, hydrogen and oxygen gas may be introduced into the containment environment from long term disassociation of water in the Suppression Chamber.

EOP procedural guidance in these cases is provided to vent the Primary Containment regardless of off-site dose consequences. Although radiological releases resulting from venting containment may exceed EPA limits, a controlled, monitored, and isolable release is preferred to a potential uncontrolled, unmonitored radiological release that would result from a failure of containment.

DEVIATION

None

REFERENCES

NUMARC NESP-007, PC1, PC2
HC.OP-AB.CONT-0001 (Q), Drywell Pressure
HC.OP-AB.ZZ-0201 (Q), Drywell High Pressure/Loss of Drywell Cooling
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0318 (Q), Containment Venting
BWR Owners Group Emergency Procedure and Severe Accident Guidelines, Revision 1

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.2 DRYWELL PRESSURE/H_2

3.3.2.b/ 3.3.2.d/ 3.3.2.e

IC Loss of Containment Barrier = 2 POINTS

EAL

Containment Failure as indicated by a rapid drop in Drywell pressure following a rise in pressure above **1.68 psig**

OR

Drywell pressure response not consistent with **LOCA** conditions

OR

Containment is Vented by the Emergency Operating Procedures (EOPs)

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Containment failure indicated by a rapid decrease in Drywell pressure following a significant rise in Drywell pressure is indicative of a Loss of the Containment barrier. This EAL specifically represents a Loss of Containment, whereby an unisolable breach of the Containment structure has occurred. Conditions that result in a drop in Drywell pressure following a pressure rise that are not the direct result of a Containment failure do not warrant classification under this EAL. These events include the initiation of Drywell Sprays, the re-establishment of Drywell Cooling, Containment Venting as required by the EOPs, and anticipated Drywell pressure drop due to ambient losses.

Drywell pressure response not increasing under **LOCA** conditions indicates a loss of containment integrity. This indicator relies on the operators recognition of an unexpected response for the condition and therefore does not have a specific value associated. The unexpected response is important because it is the indicator for a containment bypass condition.

Containment Venting is a controlled loss of containment. This venting is performed for the purpose of preventing an unisolable, unmonitored radiological release of containment gases.

A Downcomer failure, by itself, does not represent a Loss of the Primary Containment Barrier. This failure does, however, render the Primary Containment inoperable per the Technical Specification, as Primary Containment integrity has been compromised. A Downcomer failure combined with a large break LOCA will likely result in a Potential Loss of Primary Containment under EAL 3.3.2.a if Containment pressure cannot be maintained below 65 PSIG and Containment Venting is required.

Barrier Analysis

Primary Containment Barrier has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

Appropriate classification under this EAL occurs as the result of a Containment failure. Drywell pressure reaching 1.68 psig indicates that there is a significant release of reactor coolant to the Containment. Unless this source of leakage is isolated or the Reactor is depressurized, Drywell pressure would not be expected to drop in a rapid manner.

Other indications such as Reactor Building Area Radiation Monitors (ARMs) radiation levels, Reactor Building area temperatures, Reactor Building floor and sump levels, Plant Effluent radiation levels, and containment isolation status should be used to confirm the loss of containment integrity if possible. Reactor Building to Torus vacuum breaker status should be monitored to ensure that this pathway does not result in a loss of containment integrity.

DEVIATION

None

REFERENCES

NUMARC NESP-007, PC1
HC.OP-AB.CONT-0001 (Q), Drywell Pressure
HC.OP-AB.CONT-0002 (Q), Primary Containment
HC.OP-AB.ZZ-0000 (Q)-FC, Reactor Scram
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor/Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control

HC.OP-EO.ZZ-0103/4 (Q)-FC, Reactor Building and Radioactive Release Control
BWR Owners Group Emergency Procedure and Severe Accident Guidelines, Revision 1

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.3 DRYWELL ATMOSPHERE POST ACCIDENT (DAPA) RADIATION LEVEL

IC Potential Loss of Containment Barrier = 1 POINT

EAL

DAPA Radiation Monitor reading $\geq$ **28,000 R/hr**

OPERATIONAL CONDITION - 1, 2, 3

BASIS

Drywell Atmosphere Post Accident (DAPA) monitor reading $\geq$ 28,000 R/hr indicates significant fuel damage, well in excess of the level corresponding to the loss of the RCS and Fuel Clad barriers. This threshold corresponds to approximately 20% fuel clad damage. Regardless of whether or not containment is challenged, this amount of activity in containment, if released, could have severe consequences and it is prudent to treat this condition as a Potential Loss of containment.

Barrier Analysis

Primary Containment Barrier is potentially lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

NUREG-1228, "Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents"., states that releases of severe magnitude are not possible if plant systems function as designed, and any accident with a release of 20% or greater of the gap region must be considered severe.

Using attachment 2 of HC.EP-EP.ZZ-0205, 10% clad damage is represented by a DAPA reading of 1.4E4 R/hr at 0.1 hrs after shutdown (the most conservative). This is shown on the attachment as the 1% TID line. Extrapolating to 20% clad damage gives a reading of 2.8E4 R/hr.

Exceeding a DAPA reading of 28,000 R/hr should meet the criteria for declaration of a General Emergency.

DEVIATION

None

REFERENCES

NUMARC NESP-007, PC3
NUREG-1228 - Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents
HC.EP-EP.ZZ-0205 (Q), TSC – Post Accident Core Damage Assessment
EPIP 205H, TSC - Post Accident Core Damage Assessment

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.4 RCS LINE BREAK/CONTAINMENT BYPASS

3.3.4

IC Loss of Containment Barrier = 2 POINTS

EAL

UNISOLABLE leakage OUTSIDE Primary Containment as indication by one of the following:

- Downstream pathway to the environment exists
- Radiation monitors, area temperature, flow or sump level

AND
Containment Isolation is required as indicated by a signal for ANY one of the following systems:

- NSSSS
- PCIS
- HPCI
- RCIC

AND
Cannot be ISOLATED from the Main Control room

OPERATIONAL CONDITION - 1, 2, 3

BASIS

This EAL is intended to cover inability to isolate the containment when containment isolation is required. This EAL addresses two conditions where RCS is being transported OUTSIDE the Primary Containment. The first condition is associated with an Isolation signal being generated as the result of an RCS Line Break with a failure of the isolation valves to close (Downstream pathway to the environment) or a failure of both Inboard and Outboard Isolation valves to FULLY close following an Isolation signal (no reduction in area or effluent radiation monitors, area temperature, process flow or sump or room water level OUTSIDE Primary Containment). The second condition is associated with Motor Operated Valves not controlled by the isolation logic, but are manually controlled from the Main Control Room that fail to close or fully close.

UNISOLABLE means all valves in a penetration cannot be immediately closed from the Control Room. This EAL ALLOWS for valve closure from the Main Control Room to isolate any systems not completely isolated, prior to event classification. Isolation is defined as the closure of ANY valve from the Main Control Room in the system(s) not completely isolated. For example, if the isolation logic fails to cause valve closure, but operator actions implemented in the Main Control Room successfully isolates the containment breach path, then classification under this EAL is not warranted. This includes Motor Operated Valves not controlled by the isolation logic, but are manually controlled from the Main Control Room.

The term "**to the environment**" is intended to include, ANY **UNISOLABLE** leakage to the environment either directly or via systems that exhaust to the Plant Vent (e.g.; leakage to the FRVS system) or directly to any other area outside the secondary containment.

Radiation monitor indications are those that exceed normal release rate indications without a reason to expect another release source, such as a gas decay tank, spill, or fuel handling problem, and indicate a loss of the containment.

Area temperatures, system flow indications or rising sump level indications outside the primary containment may also indicate a loss of the containment. If the containment barrier is lost without a loss of the fuel barrier, effluent radiation readings may not increase significantly, however, unexpected area temperatures, flow rates, or sump increases outside of the containment may provide the indications that the containment atmosphere is no longer isolated.

A Containment Isolation Signal for any of the systems listed in the EAL requires closure of the associated Primary Containment Isolation valves to maintain RCS and Primary Containment integrity under abnormal conditions. A failure of these isolation valves to isolate a penetration or a failure of a manually controlled valve directly allows the transport of Reactor Coolant or containment atmosphere to outside the Primary Containment (Containment Breach or Bypass), resulting in a Loss of Containment.

Barrier Analysis

Primary Containment has been lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional Fission Product Barriers per EAL Section 3.0.

DISCUSSION

PCIS Isolations are associated with systems having lines that either: 1) connect directly to the Primary Containment atmosphere and penetrate the Primary Containment; or 2) penetrate the Primary Containment and are neither part of the RCS boundary nor connected directly to the

Primary Containment atmosphere (e.g. RACS, Chilled Water). Isolation requirements for these lines are covered in 10CFR50, App. A, General Design Criteria 56 and 57, respectively. Therefore, this event may potentially connect the RCS or the Primary Containment atmosphere to the environment. Without a completed isolation, continuing flow/leakage represents a release path from the RCS or Primary containment to the environment.

NSSSS isolations, as well as HPCI and RCIC steam line isolations, are associated with systems that are part of the RCS boundary and penetrate the Primary Containment. Isolation requirements for these lines are covered in 10CFR50, App. A , General Design Criteria 55. These systems form a closed loop outside the Primary Containment, and are not open to the environment. They are included in this EAL because they represent an extension of the RCS boundary beyond the Primary Containment, and are a potential release path from the RCS to the environment. Without a completed isolation, continuing leakage represents a Primary System discharging outside the Primary Containment (Containment Bypass), including areas in the Reactor Building addressed in the EOPs.

Indication of an unisolable leak includes: flow indication through isolated lines, increasing Reactor Building area temperatures, area or effluent radiation levels, also increases in sump levels, or room levels in spaces associated with affected lines, as well as increases in Plant Vent Effluent levels.

The isolation valve status of all isolation groups is monitored for quick reference on SPDS, to be backed up by operator observation of valve status.

DEVIATION

NONE

REFERENCES

NUMARC NESP-007, PC2
10CFR50, App. A, GDC 55, 56, 57
10 CFR 100
HC.OP-AB.CONT-0002 (Q), Primary Containment
HC.OP-AB.RPV-0008 (Q), Reactor Coolant Activity
HC.OP-AB.ZZ-0000 (Q)-FC, Reactor Scram
HC.OP-AR.SP-0001 (Q), Radiation Monitoring System Alarm Response – RM-11
HC.OP-AR.ZZ-0011 (Q), Annunciator Response Procedures, Window C6
HC.OP-AR.ZZ-0012 (Q), Annunciator Response Procedures, Window C8
HC.OP-EO.ZZ-0101 (Q)-FC, Reactor/Pressure Vessel (RPV) Control
HC.OP-EO.ZZ-0102 (Q)-FC, Primary Containment Control
HC.OP-EO.ZZ-0103/4 (Q)-FC, Reactor Building and Radioactive Release Control
HC.OP-SO.SM-0001 (Q), Isolation Systems Operation
HCGS Technical Specifications LCO 3/4.3, Instrumentation
HCGS UFSAR Sections 6.2.4.3.1, 6.2.4.3.2, 6.2.4.3.3

3.0 Fission Product Barriers

3.3 Containment Barrier

3.3.5 EMERGENCY COORDINATOR JUDGMENT

3.3.5.a/ 3.3.5.b

IC Potential Loss or Loss of Containment Barrier = 2 POINTS

EAL

ANY condition, in the opinion of the EC, that indicates EITHER a Potential Loss OR Loss of the Containment Barrier

OPERATIONAL CONDITION - 1, 2, 3

BASIS

This EAL allows the Emergency Coordinator (EC) to address any condition that effects the integrity of the Containment Barrier that is not already covered elsewhere in the Fission Product Barrier Table. A complete loss of the ability to monitor the Containment Barrier should be considered as a "Potential Loss" of that barrier. Conditions that result in a drop in Drywell pressure NOT preceded by challenge (LOCA) does not warrant classification under this EAL, but are covered in Technical Specifications action for Primary Containment (3/4.6.1).

Barrier Analysis

Containment Barrier has been potentially lost or lost.

ESCALATION CRITERIA

Emergency Classification will escalate based on the Potential Loss or Loss of additional barriers per EAL section 3.0.

DISCUSSION

Challenges (LOCA) with the loss of integrity should have indications such as Reactor Building Area Radiation Monitors (ARMs) radiation levels, Reactor Building area temperatures, Reactor Building floor and sump levels, Plant Effluent radiation levels, and containment isolation to confirm status of the loss of containment integrity if possible. Reactor Building to Torus vacuum breaker status should be monitored to ensure that this pathway does not result in a loss of containment integrity.

A loss of containment pressure without a challenge is covered in Technical Specifications action for Primary Containment (3/4.6.1), shutdown and cool down the plant. This is not an emergency. The basis for primary containment is to limit the release dose rate at the site boundary to be within 10CFR100 levels, without a LOCA (challenge) those levels will not be exceeded and classification under this EAL is not warranted.

DEVIATION

None

REFERENCES

NUMARC NESP-007, PC6