Appendix H
HISTORICAL INFORMATION

GI2-94-235 CERTIFICATION PACKAGE, RESPONSE TO NRC REQUESTS, FOR ADDITIONAL INFORMATION on, EMERGENCY CLASSIFICATION

This package contains: (126 PAGES)

- The basis for Emergency Plan Table 3 values
- The QEDPS runs that computed the doses
- ENTIRE LICENSING CERTIFICATION PKG.

ISCR -1341 for WEA-RIS-14, including PPM 12.11.5 setpoint calculation (29 PAGES)

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CERTIFICATION PACKAGE

RESPONSE TO NRC REQUESTS FOR

ADDITIONAL INFORMATION ON

EMERGENCY CLASSIFICATION

CONTENTS

Response to RAI (Markup with numbered material statements)

Certification Form with Table of Exhibits and References

Certification Form Attachments

- Technical Bases for GU1, GA1, GS1 & GG1
- RASCAL Dose Projection Code Print-out
- EPA 400, Appendix B (one page excerpt)
- FSAR Table 12.3-1
- Instrument Master Data Sheet, ARM-RIS-2
- Instrument Master Data Sheet, ARM-RIS-1
- Table of Representative ARM Readings
- Instrument Master Data Sheets, 14 ARMs
- PPM 5.1.6, RPV Flooding-ATWS
- PPM 5.1.4, RPV Flooding
- NUMARC/NESP-007, page 5-21
- NUMARC/NRC Questions & Answers, page 17
- FSAR page 6.2-5
- PPM 5.2.1, Primary Containment Control
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- NUMARC/NRC Questions & Answers, page 27
- NUMARC/NESP-007, page 5-80
- NUMARC/NRC Questions & Answers, page 28

October 11, 1994

Responses to Request for Additional Information

RAI #1: General - Mode Applicability

The WNP-2 EAL scheme did not include the defueled mode as an applicable mode for EALs corresponding to NUMARC/NESP-007 EALs for which "all modes" were applicable.

Justify this deviation from the NUMARC/NESP-007 guidance.

Response to RAI #1:

For many BWRs, operating mode applicability is directly associated with Technical Specification defined operating conditions as is the case for operating conditions 1 through 5 given in the WNP-2 EAL submittal. At WNP-2, Technical Specifications do not define an operating mode for the DEFUELED condition. None the less, the DEFUELED condition will be added to WNP-2 EALs as specified in NUMARC/NESP-007. 1

RAI #2: General - Safe Shutdown Buildings

The WNP-2 EAL scheme included EALs that referred to events occurring in or affecting "safe shutdown buildings". A list of safe shutdown buildings was not included in the EAL scheme. A list was included in the basis document supporting the EAL scheme. Including the list of safe shutdown buildings within the EAL scheme would expedite the classification process.

Revise the EAL scheme to include a list of safe shutdown buildings or explain why including a list of safe shutdown buildings is not needed to efficiently classify events.

Response to RAI #2:

The EAL scheme has been revised such that where a list of equipment or structures is needed to efficiently classify events, the buildings in which they are located are given directly within the EAL or by a tabular reference in the EAL. The list of safe shutdown buildings is the following: 2

- Vital portions of the Radwaste/Control Building
- Reactor Building
- Turbine Building
- Standby Service Water Pump Houses
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Area

This list was developed from equipment location requirements identified in PPM 4.12.1.1, Control Room Evacuation and Remote Cooldown, and FSAR Appendix F, Fire Safety Evaluation.

RAI #3: WNP-2 Initiating Condition (IC) GU1: Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment That Exceeds Two Times the Radiological Technical Specifications for 60 minutes or Longer.

a. The WNP-2 EAL under initiating Condition (IC) GU1 ("Sample analysis and offsite dose calculations indicate greater than TWO times ODCM B6.2.2.1 (B6.2.1.2) limits for greater than 60 minutes") deviates from the NUMARC/NESP-007 EAL under IC AU1 ("Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates with a release duration of 60 minutes or longer in excess of two times (site-specific technical specifications)"). The WNP-2 EAL requires that sample analysis and dose calculations exceed the threshold, whereas the corresponding NUMARC/NESP-007 EAL requires only a confirmed sample analysis.

Justify this deviation.

Response to RAI #3a:

At WNP-2, the only ODCM (previously Technical Specifications) limits for gaseous effluent are stated in terms of dose and dose rate. ODCM 6.2.2.1 limits offsite doses rates to 500 mrem/yr total body and 3000 mrem/yr to the skin from noble gases and 1500 mrem/yr to any organ from I-131,I-133, tritium, and all other radionuclides in particulate form with half lives greater than 8 days. 3

Results of laboratory sample analyses are reported in units of uCi/cc. To make a meaningful comparison to the applicable limit(s), laboratory results must be converted to units of dose rate. This is most expeditiously done by converting the measured isotopic concentrations to release rate by multiplying by the flow rate and appropriate conversion factors, then in-putting the data to the dose projection system. The time required for dose projection is estimated to be on the order of 15 minutes.

We have removed the necessity of offsite dose calculations for liquid releases by referencing alternative ODCM limit 6.2.1.1 which is stated in terms of concentration. 4

RAI #3:

b. Give the setpoint for the "HIGH Alarm" specified in the WNP-2 EAL under IC GU1 ("Valid High alarm on any of the following monitors ...") and information on the relationship between the alarm setpoint and the EAL threshold of two times the radiological technical specifications.

Response to RAI #3b:

The alarm setpoints for the building exhaust gaseous effluent monitors are calculated using annual average meteorology as specified in the WNP-2 Offsite Dose Calculation Manual (ODCM). The total allowable release limit is apportioned to the three release points as follows:

PRM-RE-1B (Rx Bldg)	40% of the ODCM limit
TEA-RIS-13 (Turb Bldg)	40% of the ODCM limit
WEA-RIS-14(RadWaste)	20% of the ODCM limit

The corresponding emergency action levels are therefore equal to the alarm setpoint multiplied by two and by the inverse of the allocated fraction of the ODCM limit. We have revised the EAL to use numerical values equal to the appropriate multiple of the alarm setpoint for gaseous monitors. 5

Alarms on monitors SW-RE-4, SW-RE-5, AND TSW-RE-5 are set at 80% of the ODCM 6.2.1.1 limit which is the 10 CFR 20, Appendix B, Table 2, Column 2 value for Cs-137. The alarm setpoint and ODCM 6.2.1.1 limit for monitor FDR-RE-6 is the 10 CFR 20 limit, after blowdown dilution, for the isotopic mixture determined by laboratory analysis prior to discharge. EAL thresholds for the TSW and SW monitors are therefore two times the alarm setpoints divided by 80%. The EAL threshold for the FDR monitor is two times the alarm setpoint. The EAL thresholds for SW and TSW have been restated as numerical values and rounded to the nearest discernible meter graduation. The alarm for FDR-RE-6 is set for each batch discharge so the EAL threshold has been expressed as a multiple of the HIGH-HIGH alarm.

RAI #3:

c. Provide information regarding the procedure that prompts dose assessments to be performed.

Response to RAI #3c:

Abnormal Condition Procedure, PPM 4.12.2.1, ABNORMAL RELEASE OF RADIOACTIVITY **(Enclosure 1)** prompts offsite dose calculation and other assessment and control activities. 6

OTHER CHANGES TO THIS EAL

Offgas flow is mixed with other effluent in the Reactor Building exhaust and monitored by PRM-RE-1A,B, and C. These monitors provide indication of the total mix of effluent. The Offgas Post Treatment monitors OG-RE-601 A,B have therefore been removed from the EAL. 7

RAI #4: WNP-2 IC GA1: Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment That Exceeds 200 Times the Radiological Technical Specifications for 15 minutes or Longer

a. The WNP-2 EAL under IC GA1 ("Sample analysis and offsite dose calculations indicate greater than 200 times ODCM limits for greater than 15 minutes") deviates from the NUMARC/NESP-007 EAL under IC AA1 ("Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates with a release duration of 15 minutes or longer in excess of 200 times (site-specific technical specifications)"). The WNP-2 EAL requires that sample analysis and dose calculations exceed the threshold, whereas the corresponding NUMARC/NESP-007 EAL requires only a confirmed sample analysis.

Justify this deviation.

Response to RAI #4a:

At WNP-2, the only ODCM (previously Technical Specifications) limits for gaseous effluent are stated in terms of dose and dose rate. As noted in the response to item 3 above, a meaningful comparison with ODCM limits requires conversion of effluent data to units of dose rate which is most efficiently done with one of the emergency dose projection codes. In the case of Initiating Condition GA1, however, the 15 minutes allowed for confirmation of Site Boundary dose rates precludes laboratory analysis. 8

With effluent monitor readings that approximate 200 times normal operating limits, it is reasonable to assume the presence of an isotopic mixture that is more representative of an accident source term mixture than a normal operating source term. (Our evaluation shows that a minimal containment bypass event with fuel gap release, as evaluated by the RASCAL emergency dose projection code, closely approximates the Alert level threshold.) We have therefore revised this EAL to use effluent monitor threshold values equivalent to 10 mrem/hr TEDE or 50 mrem/hr thyroid CDE based on emergency dose calculation methods and parameters as opposed to ODCM methodology. Where Standby Gas Treatment is applicable, it was assumed to be operating as this provides the more conservative level for beginning assessment of the release. 9 10

The effect is to more realistically maintain the gradient from Unusual Event to Alert to Site Area emergency through consideration of the change in source term isotopic mixture that is expected to occur between two and two hundred times ODCM limits. It also reduces the threshold for beginning assessment of a release without impacting the level at which declaration of an alert will occur. Assuming an accident source term allows assessment to be performed using monitor readings and an emergency dose projection code in accordance with PPM 13.8.1, Emergency Dose Projection System Operation, or 13.8.2, Backup Emergency dose Projection System Operation.

As noted above, WNP-2 ODCM limits restrict Site Boundary dose rate to 500 mrem/yr TEDE and 1500 mrem/yr CDE to the thyroid. Two hundred times the thyroid dose rate equates to approximately 34 mrem/hr. Fifty mrem/hr is used in the EAL for IC GA1 where thyroid CDE is found to be most limiting, to maintain the desired gradient between Alert and Site Area EALs. 11

RAI #4:

b. Give the setpoint for the "HIGH Alarm" specified in the WNP-2 EAL under IC GA1 ("Valid HIGH alarm on any of the following monitors ...") and information on the relationship between the alarm setpoint and the EAL threshold of 200 times the radiological technical specifications.

Response to RAI #4b:

The gaseous effluent monitor thresholds have been revised to provide a specific monitor indication for entry into the EAL. The thresholds were established using a RASCAL based dose projection code with meteorology parameters that are representative of annual average conditions for the WNP-2 site. The values are equivalent to Site Boundary dose rates that would be indicative of releases that are 200 times ODCM limits, i.e., 10 mrem/hr TEDE or 50 mrem/hr thyroid CDE (which ever is most limiting). 12 13

Alarms on monitors SW-RE-4, SW-RE-5, AND TSW-RE-5 are set at 80% of the ODCM 6.2.1.1 limit which is the 10 CFR 20, Appendix B, Table 2, Column 2 value for Cs-137. The alarm setpoint and ODCM 6.2.1.1 limit for monitor FDR-RE-6 is the 10 CFR 20 limit, after blowdown dilution, for the isotopic mixture determined by laboratory analysis prior to discharge. EAL thresholds for the TSW and SW monitors are therefore two hundred times the alarm setpoints divided by 80%. The EAL threshold for the FDR monitor is two hundred times the alarm setpoint. The EAL thresholds for SW and TSW have been restated as numerical values and rounded to the nearest discernible meter graduation. The alarm for FDR-RE-6 is set for each batch discharge so the EAL threshold has been expressed as a multiple of the HIGH-HIGH alarm. 14

RAI #5: WNP-2 IC GS1: Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity That Exceeds 100 mR Total Effective Dose Equivalent or 500 mR Child Thyroid Committed Dose Equivalent for the Actual or Projected Duration of the Release

a. The WNP-2 EAL under IC GS1 ("Reading on one or more of the following monitors that exceeds or is expected to exceed the value shown: PRM-RE-1B 133,000 CPS with SGT off ... AND Is verified by Offsite Dose Calculation to indicate greater than 100 mrem TEDE or 500 mrem thyroid CDE for the actual or projected duration of the release") deviates from the corresponding NUMARC/NESP-007 EAL under IC AS1 ("A valid reading on one or more of the following monitors that exceeds or is expected to exceed the value shown indicates that the release may have exceeded the above criterion and indicates the need to assess the release with (site-specific procedure) ... Note: If the monitor reading(s) is sustained for longer than 15 minutes and the required assessments cannot be completed within this period, then the declaration must be made based on the valid reading "). The WNP-2 EAL does not require declaration on the valid reading if dose assessment cannot be completed within 15 minutes.

Justify this deviation.

Response to RAI #5a:

The WNP-2 EAL is changed to ensure that classification will be made if dose assessment cannot be completed within 15 minutes 15

RAI #5:

b. The meteorology (worst case) and default time (2 hours) used to calculate the setpoints for the WNP-2 EAL under IC GS1 differs from the meteorology (annual average) and default time (1 hour) specified in NESP.

Justify this deviation. In addition, provide information regarding the relationship of the Alert level radiological effluent monitor EAL setpoint and the Site Area Emergency level radiological effluent monitor EAL setpoint.

Response to RAI #5b:

The EAL has been revised using 5 mph wind speed and stability class E, conditions that are representative of annual average meteorology at WNP-2. Other assumptions used in the calculations are release duration of one hour, one train of standby gas treatment where SGTS is in use, and design building effluent flow rates where SGTS is not applicable. 16

Gaseous effluent monitor thresholds for entry into the revised Alert EAL were developed using the same methodology and source term isotopic mixture as was used to establish the threshold values for the Site Area EALs. The desired gradient of one order of magnitude is maintained by basing the Alert threshold on 10 mrem/hr TEDE and 50 mrem/hr CDE to the thyroid while the Site Area threshold is base on 100 mrem/hr TEDE and 500 mrem/hr thyroid CDE. 17

RAI #5c:

c. The WNP-2 IC GS1 refers to "Child Thyroid," whereas the EAL under this IC refers to "Thyroid."

Justify the difference in terminology between the IC and EAL.

Response to RAI #5c:

Reference to "Child Thyroid" has been removed. The EAL thresholds were calculated with a new RASCAL based dose projection code using adult thyroid dose conversion factors. Use of adult thyroid factors is consistent with the conclusions stated in Appendix B to EPA 400 MANUAL OF PROTECTIVE ACTIONS GUIDES AND PROTECTIVE ACTIONS FOR NUCLEAR INCIDENTS. Terminology has been made consistent throughout the revised EALs. 18 19 20 21

The Supply System has recently adopted a new Emergency Dose Projection code. For consistency between values used in the EALs and the code which will be used by the Control Room staff in assessing the EAL indicators, EALs under Initiating Conditions GA1, GS1, and GG1 have been recalculated using the new code.

RAI #6: WNP-2 IC OU1: Unexpected Decrease in Water Covering Irradiated Fuel Assemblies

The WNP-2 EAL under IC OU1 did not include increases in area radiation monitor readings as a site-specific indication of the decrease in water level in the reactor refueling cavity or spent fuel pool.

Justify not including area radiation monitor readings as a site-specific indication of uncontrolled water level decrease in the reactor refueling cavity or spent fuel pool.

Response to RAI #6:

Area radiation monitor readings as a site-specific indication of the decrease in water level in the reactor refueling cavity or spent fuel pool are not included in WNP-2 EAL OU1 for an Unusual Event classification because the area radiation monitors that would be included in EAL OU1 are utilized under EAL FA1 for an Alert classification. EAL FA1 is representative of an indication of major damage to irradiated fuel or loss of water level that has or will result in uncovering of irradiated fuel outside the RPV. EAL OU1 is representative of an unexpected decrease in water covering irradiated fuel assemblies. 22

The high alarm setpoint was selected for the FA1 Alert classification threshold because the alarm setpoint is operationally significant. A technical basis for a threshold value above the high alarm setpoint could not be determined. A threshold value below the high alarm setpoint may not be indicative of a water level reduction leading to irradiated fuel uncovery. In addition, elevated radiation monitor readings on the refueling floor of a BWR, in and of themselves alone, may be caused by planned evolutions and have no bearing on the level of water above irradiated fuel assemblies. Therefore, a lower threshold value would be neither operationally significant nor easily recognized. As a result, the radiation monitor high alarm setpoint value is determined to be the only radiation monitor reading that may be representative of both water level decrease that "will result in uncovering of irradiated fuel" and an "unexpected decrease in water covering irradiated fuel assemblies". Given a single radiation monitor value for the threshold of both an Unusual Event initiating condition and an Alert initiating condition, the radiation monitor reading is applied to the Alert IC to ensure that the proper event classification is made for NUMARC/NESP-007 IC AA2.

In addition, after further review of refuel floor radiation monitoring capability it was determined that three of the four ARMs included in the EAL for IC FA1 should not be listed for the following reasons:

- ARM-RIS-3 and ARM-RIS-3A detect radiation in the new fuel storage pit and, therefore, would not provide adequate indication of radiation levels associated with decreasing water level above irradiated fuel. 23

- ARM-RIS-2, Fuel Pool Area Radiation Monitor, alarm setpoint is arbitrarily adjusted to a level slightly above normal background to monitor operator performance during fuel handling and would, therefore, not be indicative of a potential refueling accident. Its high alarm setpoint is typically 15 mR/hr. 24

- The high alarm on ARM-RIS-1, Fuel Pool Area Radiation Monitor, remains the threshold condition for this EAL. Its setpoint is nominally 300 mR/hr. 25 The EAL is now worded so that the alarm must be a confirmed high alarm thus avoiding an unnecessary declaration if the condition were to be caused by a spurious alarm signal. 26

RAI #7: WNP-2 IC FU1: Unexpected Increase in Plant Radiation Levels

The WNP-2 EAL under IC FU1 is "Any of the following Area Radiation Monitors exceeding 5,000 mr/hr: ...." The corresponding NUMARC/NESP-007 EAL under IC AU2 is "Valid Direct Area Radiation Monitor readings increases by a factor of 1000 over normal levels."

Show how this WNP-2 setpoint corresponds to 1000 times normal levels.

Response to RAI #7:

A partial listing of plant ARMs is given in the following table to illustrate the process that identified 5000 mR/hr as the EAL threshold value for EAL FU1. For each ARM, ARM location, normal reading with the plant at power, its alarm setpoint, indicated range, and reading that would be expected at 1000 times normal is listed. 27 28

ARM NO.	LOCATION	NORMAL	ALARM	RANGE	1,000 X NORMAL
4	East CRD	5 mR/hr	40±8 mR/hr	1-10K mR/hr	5,000 mR/hr
5	West CRD	20 mR/hr	100±20 mR/hr	1-10K mR/hr	20,000 mR/hr[1]
6	ROA	5 mR/hr	25±5 mR/hr	1-10K mR/hr	5,000 mR/hr
7	TIP Drive	10 mR/hr	25±5 mR/hr	1-10K mR/hr	10,000 mR/hr
8	SGTS	2 mR/hr	25±5 mR/hr	1-10K mR/hr	2,000 mR/hr
9	RHR Pump 2A	65 mR/hr	100±20 mR/hr	1-10K mR/hr	65,000 mR/hr[1]
10	RHR Pump 2B	40 mR/hr	150±30 mR/hr	1-10K mR/hr	40,000 mR/hr[1]
11	RHR Pump 2C	6 mR/hr	30±6 mR/hr	1-10K mR/hr	6,000 mR/hr
12	RCIC Pump	8 mR/hr	45±9 mR/hr	1-10K mR/hr	8,000 mR/hr
13	HPCS Pump	20 mR/hr	35±5 mR/hr	1-10K mR/hr	20,000 mR/hr[1]
23	CRD Pumps	15 mR/hr	45±9 mR/hr	1-10K mR/hr	15,000 mR/hr[1]
24	Rx 471 NW	4 mR/hr	35±7 mR/hr	1-10K mR/hr	4,000 mR/hr

[1]Offscale High

ARM NO.	LOCATION	NORMAL	ALARM	RANGE	1,000 X NORMAL
32	Rx 471 NE	100 mR/hr[2]	50±0.5 R/hr	0.1-10K R/hr	100 R/hr
33	Rx 501 NW	100 mR/hr[2]	50±0.5 R/hr	0.1-10K R/hr	100 R/hr

From this data, it can be seen that 5000 mR/hr is a value which is well above the normal reading and would always be interpreted as a loss of control of radioactive material (which is the basis for WNP-2 IC FU1). However, 1000 times the normal reading yields values that would, in many cases, be in excess of that impeding access for plant personnel (which is the criteria for the Alert declaration under WNP-2 IC FA2). NUMARC/NESP-007 requires that the Unusual Event IC escalate through the Alert IC. To maintain proper IC escalation for these areas, 1000 times the normal reading cannot form the basis of this EAL. Therefore, a single value of 5000 mR/hr has been selected to permit proper escalation for those areas whose normal readings are relatively high, while maintaining a multiple of the normal readings for those areas whose normal readings are relatively low.

RAI #8: WNP-2 IC FA1: Major damage to irradiated Fuel or Loss of Water Level That Has or Will Result in the Uncovering of Irradiated Fuel Outside the RPV

a. One of the EALs specified under IC FA1 is "A valid HIGH alarm on one or more of the following radiation or airborne monitors ..."

Give the setpoint for the "HIGH" alarm specified in this EAL.

Response to RAI #8a:

As discussed in the response to RAI #6, further review of refuel floor radiation monitoring capability has determined that three of the four ARMs included in this EAL should not be listed for the following reasons:

- ARM-RIS-3 and ARM-RIS-3A detect radiation in the new fuel storage pit and, therefore, would not provide adequate indication of radiation levels associated with decreasing water level above irradiated fuel.

[2]Live Zero

- ARM-RIS-2, Fuel Pool Area Radiation Monitor, alarm setpoint is arbitrarily adjusted to a level slightly above normal background to monitor operator performance during fuel handling and would, therefore, not be indicative of a potential refueling accident. Its high alarm setpoint is typically 15 mR/hr. 29

  The high alarm on ARM-RIS-1, Fuel Pool Area Radiation Monitor, remains the threshold condition for this EAL. Its setpoint is nominally 300 mR/hr. The EAL is now worded so that the alarm must be a confirmed high alarm thus avoiding an unnecessary declaration if the condition were to be caused by a spurious alarm signal. 30

  It is common practice to adjust the alarm setpoint of radiation monitors to account for gradual changes in background radiation levels, shielding alterations, or detector geometry alterations that impact radiation monitor sensitivity.

  If a specific value of a radiation monitor setpoint were to be included in the EAL and the monitor setpoint value subsequently altered, the EAL value may no longer be representative of an abnormal radiation level. Alternatively, the EAL value may be representative of an abnormal radiation level that is in excess of the desired threshold value. As a result, it is possible that emergency classification would not be made when fuel uncovery is being approached. It is also possible that the classification would be made when fuel uncovery is not threatened. To avoid these possible undesirable outcomes, it is appropriate to classify the event based on a setpoint which is indicative of an abnormal radiation level for the current state of the plant.

<u>RAI #8: WNP-2 IC FA1: Major damage to irradiated Fuel or Loss of Water Level That Has or Will Result in the Uncovering of Irradiated Fuel Outside the RPV</u>

b. WNP-2 IC OA2 ("Loss of Water Level That Has or Will Result in Uncovering of Irradiated Fuel Outside the RPV") is very similar to the IC for FA1.

   Justify the inclusion of both of these similar ICs.

<u>Response to RAI #8b:</u>

The source for both IC OA2 and FA1 is NUMARC/NESP-007 AA2. IC OA2 pertains to water level with respect to irradiated fuel uncovery; FA1 to radiation monitor readings. During the development of PPM 13.1.1 and PPM 13.1.1A, it was determined that the portion of IC AA2 pertaining to low water level fit most appropriately in the WNP-2 EAL category that pertains to system malfunctions (Category 0 - Miscellaneous Initiating Conditions/System Malfunctions). Therefore, it is appropriate to list the EALs under different categories.

Note that the WNP-2 categories specified in PPM 13.1.1 and PPM 13.1.1A and the EALs assigned to these categories are being reviewed as part of the WNP-2 EAL Upgrade Project to ensure that the EAL end users can quickly and consistently associate degraded or potentially degraded plant conditions to the appropriate event classifications.

RAI #9: Fuel Clad Barrier Loss - Reactor Vessel Water Level

The WNP-2 EAL for the RPV level indication of the loss of the fuel clad barrier and the potential loss of the containment barrier is "Entry into PPM 5.1.7.

Justify the use of this procedure entry condition as an EAL for the loss of the fuel clad barrier and the potential loss of the containment barrier. In addition, provide a copy of relevant parts of PPM 5.1.7.

Response to RAI #9:

Entry to PPM 5.1.7, Primary Containment Flooding, is required when the reactor core can not be adequately cooled. The conditions that direct entry to PPM 5.1.7 are indications of a loss of adequate core cooling. These conditions are listed in paragraph 1.b on page 17 of PPM 13.1.1.A, and are discussed in detail below:

1) RPV water level cannot be restored and maintained above -161 in. (top of active fuel). This condition and the direction to enter PPM 5.1.7 is given in PPM 5.1.1, RPV Control, at step L-21. Core submergence is the mechanism of core cooling whereby each fuel element is completely covered with water. Indicated RPV water level at or anywhere above the elevation corresponding to the top of active fuel (TAF) constitutes the principal means of confirming the adequacy of core cooling achieved through this mechanism. Assurance of continued adequate core cooling through core submergence is achieved when RPV water level can be maintained at or anywhere above TAF. 31

2) For ATWS conditions, RPV water level cannot be maintained above -192 in. (31 in. below the top of active fuel). This condition and the direction to enter PPM 5.1.7 is given in PPM 5.1.2, RPV Control - ATWS, at steps L-15 and L-2. The RPV water level value of -192 in. is termed the Minimum Steam Cooling RPV Water Level (MSCRWL), The MSCRWL was determined in accordance with BWROG Emergency Procedure Guideline (EPG), Revision 4. With injection into the RPV established, adequate core cooling exists when steam flow through the core is sufficient to preclude the peak clad temperature of the hottest fuel rod from exceeding 1500 F, the threshold temperature for fuel rod perforation. This mechanism of core cooling is employed during the level/power control evolution (PPM 5.1.2) when RPV water level is controlled below the top of active fuel to reduce reactor power. RPV water level provides the means of confirming the adequacy of core cooling achieved via this mechanism. Assurance of continued adequate core cooling is achieved when RPV water level can be maintained at or above the MSCRWL 32

3) When RPV water level cannot be determined, RPV flooding for ATWS or non-ATWS conditions cannot be established and maintained. 33

RPV flooding for ATWS conditions is specified in PPM 5.1.6, RPV Flooding - ATWS, at step F-10. The direction to enter PPM 5.1.7 is given in PPM 5.1.6 steps F-11 and F-12. With injection into the RPV established, adequate core cooling exists when steam flow through the core is sufficient to preclude the peak clad temperature of the hottest fuel rod from exceeding 1500 F, the threshold temperature for fuel rod perforation. This mechanism of core cooling is employed during the RPV flooding evolution when the reactor may not be shutdown. RPV pressure and the number of open SRVs provide the means of confirming the adequacy of core cooling achieved via this mechanism. Assurance of continued adequate core cooling is achieved when RPV pressure can be maintained at or above the Minimum Alternate RPV Flooding Pressure. The Minimum Alternate RPV Flooding Pressure is a list of RPV pressures and open number of SRVs (PPM 5.1.6 Table 15). 34

RPV flooding for non-ATWS conditions is specified in PPM 5.1.4, RPV Flooding, at step F-6. The direction to enter PPM 5.1.7 is given in PPM 5.1.4 steps F-7 and F-8. When the reactor is shutdown, three conditions must be satisfied to verify RPV flooding without direct indication of RPV water level: 35

a. At least five SRVs must be open. This ensures that adequate steam flow will exist for cooling any unsubmerged portion of the core when RPV pressure reaches 60 psig. (See item c, below .)

b. RPV pressure must not be decreasing. This ensures that the required steam flow will be maintained.

c. RPV pressure must be greater than wetwell pressure by at least 60 psig, the Minimum RPV Flooding Pressure (MRFP).

The MRFP is defined to be the lowest differential pressure between the RPV and the wetwell at which steam flow through five SRVs is sufficient to remove decay heat. The decay heat generation rate is ten minutes after shutdown from full power. Since ten minutes is the earliest that RPV flooding could reasonably be expected to be needed, establishing and maintaining RPV pressure above the MRFP assures that more than enough steam flows through the SRVs to carry away all core decay heat.

This requires that a sufficient quantity of water reach the core to carry away decay heat by boiling, which in turn requires that RPV water level increase. Maintaining this minimum pressure (60 psig) assures that the RPV will ultimately flood to the main steam lines. Since the primary containment could be pressurized when this step is performed, the MRFP is expressed as a differential pressure across the SRVs (i. e., RPV pressure minus wetwell pressure).

When adequate core cooling cannot be assured by one of the three mechanisms described above, it directly follows that the loss of the fuel clad barrier must be assumed.

When failure of assuring one of the three mechanisms described above requires entry to PPM 5.1.7, the primary containment barrier may be threatened because the actions of PPM 5.1.7 raise water level in the primary containment by injection of water from sources external to the containment. If RPV water level reaches or exceeds the Pressure Suppression Pressure before the RPV is depressurized, subsequent rapid depressurization of the RPV without the operability of the pressure suppression function of the containment could lead to failure of the primary containment barrier. If primary containment flooding causes water level in the containment to reach the elevation of the lowest recirculation system piping, the operator is directed to deliberately vent the RPV to areas outside the primary containment. Should this occur, a loss of primary containment barrier must be assumed. Therefore, steps that lead to this action must be considered a potential loss of the primary containment barrier.

A copy of PPM 5.1.7, Primary Containment Flooding, is attached to these responses **(Enclosure 2).**

RAI #10: Fuel Clad Barrier Loss - Drywell Radiation Monitor

Give the calculations for the drywell monitor setpoints used as an indication of the loss of the fuel clad barrier, loss of reactor coolant system barrier, and potential loss of the primary containment barrier.

Response to RAI #10:

The original drywell monitor setpoints were established using our core damage assessment procedure which in turn was based on General Electric procedures for determination of core damage. We have now performed calculations of drywell monitor responses specific to the NESP-007 Initiating Conditions. We find that the detectors have a lower level of response generally consistent with 0.1% clad failure. Because this response is significantly below the threshold for indication of loss of Fuel Clad, it is used as an indicator of loss of RCS only. 36

WNP-2 Calculation No. NE-02-94-57 **(Enclosure 3)** determines the drywell monitor setpoints used as an indication of the loss of the fuel clad barrier, loss of reactor coolant system barrier, and potential loss of the primary containment barrier.

RAI #11: Fuel Clad Barrier Loss - Main Steam Line Radiation Monitor

Main steam line radiation level HI HI is included in the WNP-2 EAL scheme as an indication of the loss of the fuel clad barrier. The basis for this EAL specifies that a main steam line radiation HI HI is an indication of the potential loss of fuel clad.

Justify including the main steam line radiation monitor as an indication of the loss or potential loss of fuel clad barrier and revise the EAL or basis document as needed to achieve consistency.

Response to RAI #11:

The main steam line radiation HI-HI level is removed from the WNP-2 EAL classification table. No basis exists to correlate main steam line radiation levels at the HI-HI setpoint to a fraction of fuel clad damage. Exceeding this monitor setpoint may not be an indication of the loss or potential loss of the fuel clad barrier because other events such as resin intrusion or oil contamination can cause elevated main steam line radiation readings that are not the result of fuel clad degradation. For events in which fuel clad degradation is the cause of elevated main steam line radiation, other indicators such as offgas activity provide threshold values that are indicative of loss or potential loss of the fuel clad. 37

RAI #12: Reactor Coolant System - Drywell Pressure

The WNP-2 EAL for the drywell/containment pressure indication of the loss of the reactor coolant system (RCS) barrier is "Drywell pressure greater than 1.68 psig WITH indications of a leak inside drywell."

Justify including the condition "indication of a leak inside drywell." In particular, describe which indicators operators will use to determine whether a leak exists inside the drywell.

Response to RAI #12:

As stated on page 18 of PPM 13.1.1.A, "... the qualifier of 'indication of RCS leakage' is included as an indicator of RCS boundary degradation and eliminates a drywell pressure increase due to a loss of drywell ventilation." Although the basis for the high drywell pressure alarm signal is the loss of coolant accident, high drywell pressure may occur due to other non-LOCA related events for which it would not be appropriate to classify as an emergency event. Inerting errors, or loss of drywell cooling and other drywell ventilation degradation that increase drywell temperatures and pressure are such examples.

The source of leakage in the drywell must be discerned from the trends of a number of indications that are either present or not present. Indication of RCS leakage in the drywell is most directly provided by the trends of drywell identified and unidentified leak rate as well as the total amount of drywell leakage observed on drywell flow totalizer instrumentation. Specific indications include:

- Annunciators activated:

  H13-P601.A3-6.6, LEAK DET DRYWELL EQUIP DRAIN FLOW HIGH

  H13-P601.A3-5.8, LEAK DET REACTOR BLDG EQUIP SUMP LEAKAGE HIGH

  H13-P602.A13-3.1, REACTOR BLDG SUMP HIGH LEVEL

  H13-P602.A13-4.1, REACTOR BLDG EQUIP SUMP TEMP HIGH

  H13-P825.N-8.6, REFUEL BELLOWS LEAK DET FLOW HIGH

  H13-P602.A6-2.3, RECIRC A PUMP SEAL STAGING FLOW HIGH/LOW

  H13-P602.A6-2.7, RECIRC B PUMP SEAL STAGING FLOW HIGH/LOW

- Increasing flow into EDR-SUMP-R5, Drywell/Reactor Building Equipment Drain Sump, as indicated on EDR-FRS-623 (H13-P632).

- Excessive increase in total flow into EDR-SUMP-R5, Drywell/Reactor Building Equipment Drain Sump, as indicated on FDR-FQ-38 (bottom unit)(H13-682).

- EDR-TM-602, DW/Rx BLDG Sump EDR-SUMP-R5 Pump Out Timer on H13-P632 timed out.

- EDR-TM-603, DW/Rx BLDG Sump EDR-SUMP-R5 Pump Out Timer on H13-P632 timed out.

- EDR-P-5A, Drywell/Reactor Building Equipment Drain Sump Pump run time indicating excessive run time on EDR-P-5A Elapsed Time meter at H13-P602.

- EDR-TIS-601, Reactor Building EDR Sump Temp. increasing or excessive (H13-P602).

- EDR-P-5A, Reactor Building EDR Sump Pump auto starts on high sump level and will not realign for recirc while high level exists.

- In the event of a high temperature (140 °F) in the Reactor Building drain sump, the equipment drain sump system shifts into the recirc mode and EDR-P-5A auto starts if level is above the low level pump cutoff.

- Isotopic analysis of the drywell atmosphere radioactivity to determine if the leak is from a steam or water system.

An increased drywell leakage may not necessarily originate from the RCS. For example, leakage in the reactor closed cooling water to the drywell coolers (RCC) can cause elevated drywell floor drain and equipment drain readings. Because such leakage would also exhibit a decrease in RCC head tank level the operator would be able to discern this leakage from RCS leakage. In the absence of a degradation in drywell ventilation, RCS leakage may also be inferred from an unexplained increase in local drywell temperatures and bulk drywell temperature.

Justification for the added phrase comes from the basis discussion of drywell pressure as an RCS barrier loss on page 5-21 of NUMARC/NESP-007: "... drywell pressure is based on the drywell high pressure alarm setpoint and indicates a LOCA." 38
Further, NUMARC and the NRC specifically addressed this question in response to question number 6 of the BWR fission product barriers (June 1993). The response 39
states that: "Individual plants may clarify the EAL conditions to ensure that non-accident conditions do not require emergency classification." For this fission product barrier loss condition, the Supply System believes it is necessary to clarify the wording of the NUMARC/NESP-007 EAL. Since the high drywell pressure alarm setpoint may be reached due to non-LOCA events, the added phrase "as a result of RCs leakage" is required to ensure that the intent of NUMARC/NESP-007 is maintained.

Note that the wording of this fission product barrier RCS loss condition is changed 40
to state: "1.68 psig with indications of RCS leak inside drywell", thereby emphasizing the need to associate the high drywell pressure condition with one or more indications of RCS leakage in the drywell.

RAI #13: Reactor Coolant System Potential Loss - RCS Leak Rate

One of the WNP-2 EALs for the RCS indication of the loss of the RCS barrier is "Primary System discharging outside Primary Containment as indicated by Area Temperature Alarm(s) or Radiation Alarm(s) AND The affected containment penetration cannot be isolated."

Justify including the condition "The affected containment penetration cannot be isolated" in this EAL.

Response to RAI #13:

The condition "The affected containment penetration cannot be isolated" has been deleted from this RCS barrier loss and the qualifier "unisolable" added in accordance with guidance of NESP-007. In addition, threshold area temperature and radiation levels have been added to the EAL. The threshold is that indication corresponding to the Maximum Safe Operating (MSO) values listed in PPM 5.3.1, Secondary Containment Control. When MSO values are exceeded, indication of RCS loss and potential loss is given. These values of secondary containment parameters are utilized by the operator to determine when it is necessary to shut down or scram the reactor, and reduce RPV pressure to limit discharge outside primary containment. Thus, they are operationally significant and easily recognizable. 41

RAI #14: Containment Barrier Loss - Containment RCS Leak Rate

The WNP-2 EAL for the Containment Valves indication of the loss of Containment barrier is, "Primary System discharging outside Primary Containment as indicated by Area Temperature Alarm(s) or Radiation Alarm(s) AND The affected containment penetration cannot be isolated."

Justify including the condition "The affected containment penetration cannot be isolated" in this EAL.

Response to RAI #14:

The condition "The affected containment penetration cannot be isolated" has been deleted from this RCS barrier loss and the qualifier "unisolable" added in accordance with NESP-007 guidance. Maximum Safe Operating values have been added as threshold area temperature and radiation levels. See response to RAI #13. 42

RAI #15: Containment Barrier Potential Loss - Drywell Pressure

The WNP-2 EAL for the Drywell/Containment Pressure indication of potential loss of the containment barrier is "Containment pressure greater than 39 psig .... (or) ... Wetwell pressure exceeds PSP." Discuss why the design pressure of the containment was not used as the basis for this EAL as is specified in the basis for the corresponding EAL in NUMARC/NESP-007.

Response to RAI #15:

The design pressure of the containment was not used as the basis for a potential loss of the containment barrier because Accident Analyses in the WNP-2 licensing basis documentation shows that the resultant peak drywell pressure remains below 39 psig. The WNP-2 primary containment design pressure is greater than 39 psig. However, since a credible accident sequence within the WNP-2 design basis does not exist for exceeding 39 psig, it is appropriate to base the containment potential loss condition on the accident analysis peak pressure. 43

This pressure (39 psig) is utilized in the WNP-2 EOPs as the containment pressure beyond which venting of the primary containment irrespective of the offsite radioactivity release is permitted. When venting of the primary containment is directed due to elevated containment pressure, a loss of the containment barrier occurs. Intentional venting of the primary containment is included in the WNP-2 EAL matrix as a loss of the containment barrier. If design pressure of the primary containment were utilized as the containment barrier potential loss, the WNP-2 EOPs could require venting of the primary containment before containment pressure reaches the design pressure. As a result, the containment barrier loss condition would be reached before the potential loss condition. 44

RAI #16: Containment Barrier Potential Loss - Reactor Vessel Level

The WNP-2 EAL for the potential loss of the containment barrier did not include an EAL corresponding to the NUMARC/NESP-007 EAL: "Reactor vessel water level LESS THAN (site-specific) value and the maximum core uncovery time limit is in the UNSAFE region."

Justify this deviation from the NUMARC/NESP-007 EAL guidance.

Response to RAI #16:

The Supply System believes that the best indicator of containment barrier potential loss that relates to RPV water level and adequate core cooling is entry to PPM 5.1.7, Primary Containment Flooding. Refer to the response to RAI #9 for discussion of entry to PPM 5.1.7.

The Maximum Core Uncovery Time Limit (MCUTL) is an action level developed for the BWROG Emergency Procedure Guidelines (EPGs). Appendix A of the BWROG EPGs defines the MCUTL as follows:

The Maximum Core Uncovery Time Limit (MCUTL) is defined as the greatest amount of time the reactor core can remain completely uncovered and uncooled without resulting in a peak clad temperature in excess of 1500 °F.

The MCUTL is obtained by linear variation of a generic uncovery time limit for all BWRs provided by the General Electric Company. The variation is based on a ratio of the value of LHGR assumed in the generic uncovery time limit and the plant-specific LHGR for the current fuel load.

The MCUTL is based the following assumptions:

a. The reactor has been shut down from rated power.

b. The core is instantaneously and completely uncovered when recovery from RPV flooding is initiated.

c. Clad temperature is 545 °F when the reactor core is initially uncovered.

The Supply System did not include this limit as part of a loss or potential loss of the fuel clad barrier (and consequently a containment barrier potential loss) because the fuel clad can be seriously damaged long before the Limit is exceeded and, on the other hand, the fuel clad may remain intact and unaffected when the core is uncovered for greater than the Limit under certain conditions. For example:

a. If injection into the RPV prior to start of the MCUTL has failed to restore or maintain clad temperatures less than or equal to 545 °F, significant clad failures could occur because the Limit is based on adiabatic clad heat up from 545 °F to 1500 °F. If the heat up starts at a temperature higher than 545 °F, the clad temperature when the Limit expires must necessarily be greater than 1500 °F. The BWR EPGs ensure that the RPV is flooded for at least the Minimum Core Flooding Interval before the MCUTL is tested, thus ensuring that the clad temperatures are at saturation temperature for the RPV pressure corresponding to the setpoint of the lowest lifting SRV. For this reason, the MCUTL cannot be applied to conditions in which the reactor is not shutdown. The assumptions of the MCUTL do not bound plant parameters that may exist during ATWS events. However, the NUMARC/NESP-007 condition makes no distinction between ATWS events or any other events that can exist which are not governed by the assumptions of the MCUTL.

b. If the RPV water level associated with the NUMARC/NESP-007 containment barrier potential loss is above the Minimum Steam Cooling RPV water level (typically 19/24 of core height), it is possible to maintain that water level indefinitely while providing adequate core cooling. If the MCUTL were implemented with this water level, NUMARC/NESP-007 would require event declaration or escalation for a condition in which neither the fuel clad nor the primary containment barriers are either lost or threatened.

Further, there is no defined UNSAFE region of the MCUTL in either the WNP-2 EOPs or the BWROG EPGs. NUMARC/NESP-007 use of the MCUTL does not specify an UNSAFE region. The reason the WNP-2 EOPs and the BWROG EPGs do not specify an UNSAFE region is based on the manner in which the Limit is applied in the EPGs. The MCUTL is utilized in the EPGs by first determining the time since reactor shutdown, locating that time on the MCUTL curve, and then reading across to the vertical axis to establish the amount of time that RPV injection can be secured while waiting for RPV water level to appear on-scale. Consequently, plant conditions are never above or below the curve - conditions are always at a location on the curve.

RAI #17: WNP-2 IC LA1, EAL #3: Vehicle Crash

WNP-2 EAL #3 under IC LA1 ("Vehicle crash into or projectile impacting safe shutdown plant structures or systems which represents a potential degradation to the safety of the plant") deviates from the corresponding NUMARC/NESP-007 EAL under IC HA1 by including the condition "which represents a potential degradation to the safety of the plant."

Justify this deviation from the NUMARC/NESP-007 guidance and specify how the users of the EAL will determine whether this condition has been met.

Response to RAI #17:

The WNP-2 EAL has been changed as follows to delete the phrase "... which represents a potential degradation to the safety of the plant" and add reference to plant vital areas: 45

Vehicle crash or projectile impact which impedes access to or damages equipment in plant vital areas.

RAI #18: WNP-2 IC LA1, EAL #6: Turbine Failure

WNP-2 EAL #6 under IC LA1 ("Turbine failure resulting in casing penetration or damage to turbine or generator seal AND Missiles generated from the turbine failure have affected safety related equipment") deviates from the corresponding NUMARC/NESP-007 EAL under IC HA1 by including the condition "have affected safety related equipment."

Justify this deviation from the NUMARC/NESP-007 guidance.

Response to RAI #18:

The WNP-2 EAL has been changed as follows:

1) The condition "have affected safety related equipment" has been deleted 46

2) visible structural damage to or penetration of a safe shutdown building has been added.

RAI #19: WNP-2 IC NU1, NA1, and NS1: Security Events

It is not clear that the EALs listed under the Security Event ICs are appropriate. For instance, IC NA1 includes as one of its conditions a bomb discovered in the vital area. Under the NUMARC/NESP-007 EAL scheme, a bomb discovered in a vital area is classified as a Site Area Emergency. As another example, IC NU1 includes as an EAL "a Security Officer found incapacitated due to violent circumstances." This EAL should consider the location of the incapacitated security officer in determining the appropriate classification level.

Explain how the EALs under the Security Event ICs are indicative of the ICs and how they relate to the corresponding NUMARC/NESP-007 EALs.

Response to RAI #19:

Based on discussion with plant security and reevaluation of NUMARC/NESP-007, the EALs under Initiating Conditions NU1, NA1, and NS1 have been revised to be more indicative of the NESP-007 Initiating Conditions. EALs under the Unusual Event Initiating Condition are examples of events that are potentially degrading to safety of the plant. The 10 minute compensatory action threshold for Unusual Event is derived from regulatory guidance on implementation of 10 CFR 73.71. The Alert level intrusion by a hostile force into the plant protected area represents an escalated threat which will in turn escalate to a Site Area Emergency in the event of intrusion into a plant vital area. 47

RAI #20: WNP-2 IC NG1: Security Event Resulting in Loss of Ability to Reach and Maintain Cold Shutdown

The WNP-2 EAL under IC NG1 includes the condition "Loss of physical control of both divisions of the Remote Shutdown capability due to a security event."

Justify including "both divisions" in this EAL.

Response to RAI #20:

The WNP-2 EAL has been changed to agree with NUMARC/NESP-007 IC HG1 wording. 48

RAI #21: WNP-2 IC DG1: Loss of Offsite Power

The WNP-2 EAL under IC DG1 includes the condition that "power to either Bus SM-7 or SM-8 is not likely to be restored within 4 hours."

Justify the 4-hour time limit.

Response to RAI #21:

Prolonged loss is quantified by including the 4 hr time limit given in WNP-2 FSAR Section 1.5.2, SBO Coping Study. In accordance with NUMARC/NESP-007, this time limit is based on a site blackout coping analysis performed in conformance with 10CFR50.63 and Regulatory Guide 1.155, "Station Blackout." 49

RAI #22: WNP-2 IC BA1: Inability to Maintain Cold Shutdown

The WNP-2 EAL under IC BA1, "Inability to maintain a reactor temperature of less than 200 °F per PPM 4.4.2.1," deviates from the NUMARC/NESP-007 guidance by not including the condition of "Loss of (site-specific) technical specification required functions to maintain cold shutdown" as specified in the corresponding EAL in NUMARC/NESP-007 IC SA3.

Justify this deviation.

Response to RAI #22:

BWR technical specifications do not state functional requirements per se; they state minimum equipment operability requirements needed to fulfill functions. It is possible for the WNP-2 plant to reach and maintain cold shutdown requirements without the operability of technical specification required equipment for cold shutdown. For example, WNP-2 coolant temperature can be maintained less than 200 °F by rejecting decay heat through the Reactor Water Cleanup System non-regenerative heat exchanger or by utilizing main condenser vacuum to reduce and hold saturation pressure in the RPV below that corresponding to saturation pressure for 200 °F.

When these alternate decay heat removal mechanisms are operable in the absence of the shutdown cooling mode of RHR, the technical specification reactor temperature limit may not be approached or exceeded. Since cold shutdown conditions can be maintained for such an event, there should be no reason to declare or escalate the event to an Alert classification. 50

Conversely, since the NUMARC/NESP-007 IC SA3 requires both technical specification required functions to maintain cold shutdown AND temperature approaching the technical specification limit, it is possible for the Shutdown Cooling Mode of the RHR system to be operating yet incapable of limiting temperature below the technical specification limit. By the logic structure of NURMARC/NESP-007, the Alert declaration would not be required when clearly the Alert classification is warranted.

To accommodate both of these situations, the WNP-2 EAL specifies the Alert declaration based only on the temperature criterion. Note that the phrase "Inability to maintain ..." includes the condition where the temperature has already been exceeded as well as a recognition that temperature control capability will not stop the temperature increase before the Limit is exceeded.

RAI #23: WNP-2 IC JU1: Loss of Annunciators

The WNP-2 EAL under IC JU1 includes the condition, "Compensatory nonalarming indications ARE available."

Provide information regarding what nonalarming indications are accessible to operators at WNP-2, why the nonalarming indications are not specified in the EAL, and how the user of this procedure will determine the availability of nonalarming indications.

Response to RAI #23:

Nonalarming indications that are accessible to operators at WNP-2 include numerous diverse and, in several instances, redundant indicators and recorders which are mounted on the same panels on which the equipment alarms and annunciators are located. There are indicators and recorders located on control room panels which are not associated with the alarms and annunciators listed in this EAL. There are also alarms and indications available on panels located outside the main control room. The status of these alarms and indications as well as the visible status of equipment in the plant is accessible to the main control room by means of in-plant operator reports by ROLM phone and other communications channels.

Process computer systems and the Graphic Display System have been specifically added to the EAL under compensatory nonalarming indications. Other potential compensatory non alarming indications are not identified in the wording of the EAL because the permanently installed indications are too numerous to list. In addition, a detailed list in the EAL may not include temporary means of monitoring a parameter that would otherwise be considered a nonalarming indication. 51

RAI #24: WNP-2 IC JA1: Loss of Annunciators

The WNP-2 EAL under IC JA1 includes the condition "A significant unplanned plant transient is in progress OR Compensatory nonalarming indication(s) are NOT available."

Explain why the transient must be unplanned for this IC to be applicable.

Justify not defining the conditions which indicate a "significant transient" in the EAL scheme.

Response to RAI #24:

The WNP-2 EAL has been changed to delete the term "unplanned". 52

On page 65 of PPM 13.1.1.A, the basis of WNP-2 EAL JA1 states that a "significant transient" includes response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injection, or thermal power oscillations of 10% or greater. PPM 13.1.1.A is an approved procedure available to the EAL user and may be consulted for supporting information regarding an EAL condition. If during event classification, the user of the classification procedure (PPM 13.1.1) is concerned about the content of an EAL, he may refer to PPM 13.1.1.A.

RAI #25: WNP-2 IC JS1: Inability to Monitor a Significant Transient in Progress

The WNP-2 EAL under IC JS1, "Loss of all Control Room annunciators/indications needed to monitor ANY of the following plant critical safety parameters .... OR Loss of most or all PROCESS Radiation Monitoring System Main Control Room indications for greater than 15 minutes ...." contains indications and a logic sequence which may result in events being classified as a Site Area Emergency when not warranted.

Explain why the loss of indications specified in the WNP-2 EAL is appropriate for this IC and is indicative of a Site Area Emergency classification.

Response to RAI #25:

The Supply System concurs with the reviewer and has restructured the EAL under IC JS1 to agree with the NUMARC/NESP-007 guidance of IC SS6. The functions listed in the EAL include those critical safety parameters that are monitored and controlled in the WNP-2 EOPs for RPV control and primary containment control. 53

RAI #26: WNP-2 ICs CU1, CA1, AND CS1: Anticipated Transient Without Scram

The WNP-2 EALs under ICs CU1, CA1, and CS1 deviate from the corresponding NUMARC/NESP-007 EALs under ICs (SA2 and SS2) as follows:

a. The WNP-2 EAL for failure of the automatic scram is classified at the Unusual Event level, whereas this condition is classified at the Alert level in NESP. As stated in the basis for NUMARC/NESP-007 IC SA2, "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 the 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 the RCS."

Justify this deviation.

Response to RAI #26a:

Emergency classifications associated with scram failure conditions have been revised to agree with the NUMARC/NESP-007 example EALs as modified by the clarifications provided in "NUMARC/NRC Questions and Answers, June 1994". 54

Note that Systems Malfunction Question #7 in "NUMARC and NRC Questions and Answers" states that "... a scram is considered unsuccessful if it does not result in achieving a state in which the reactor will remain shut down under all conditions without boron." This BWROG EPG definition of a scram failure condition is implemented in the WNP-2 EOPs by the phrase "... control rod pattern alone [can/cannot] always assure reactor shutdown ...". The WNP-2 EOP wording has been carried forth in the WNP-2 EAL wording for consistency of understanding and interpretation.

RAI #26:

b. The WNP-2 EAL scheme includes an Alert level EAL for the failure of the automatic and manual scram with the condition that "operator actions WERE successful in reducing reactor power to less than 5% AND Existing control rod pattern alone cannot always assure reactor shutdown" and also includes a Site Area Emergency level EAL for the failure of the automatic and manual scram with the condition that "operator actions WERE NOT successful in reducing reactor power to less than 5%." The corresponding NUMARC/NESP-007 EAL under IC SS2 does not include this power level condition and instead specifies that the scram was not successful. The NRC has determined that it is acceptable to have an EAL classified at the Site Area Emergency level with the condition that reactor power is less than 5% as long as the EAL also includes a condition indicating that the heat removal capability of safety systems is not exceeded: for example, that the suppression pool temperature is not above a site-specific value. It is not acceptable, however, to have an EAL for the failure of the automatic and manual scram classified at the Alert level.

   The Licensee should modify the Alert and Site Area Emergency EALs for the failure of the automatic and manual scram to conform with the NUMARC/NESP-007 guidance. The WNP-2 EAL may include a Site Area Emergency EAL with a reactor power level indication and a heat removal indication as described above.

Response to RAI #26b:

Emergency classifications associated with scram failure conditions have been revised to agree with the NUMARC/NESP-007 example EALs as modified by the clarifications provided in "NUMARC/NRC Questions and Answers, June 1994". Refer to the response to RAI #26a. 55

RAI #27: WNP-2 IC CG1: ATWS with Challenge to Core Cooling

The WNP-2 EAL under IC CG1 includes the conditions, "Suppression Pool temperature cannot be maintained less than the HCTL curve OR Entry into Primary Containment Flooding, PPM 5.1.7, is required", corresponding to the NUMARC/NESP-007 (IC SG2) condition of "(site-specific) indications exist that the core cooling is extremely challenged OR (site-specific) indications exist that heat removal is extremely challenged."

a. Explain how the conditions specified in the WNP-2 EAL correspond to the NUMARC/NESP-007 conditions.

Response to RAI #27a:

The conditions specified in the WNP-2 EAL correspond to the NUMARC/NESP-007 conditions as follows:

a. (Site-specific) indications exist that the core cooling is extremely challenged:

Entry to PPM 5.1.7 and its relationship to core cooling is discussed under the response to RAI #9.

b. (Site-specific) indication exists that heat removal is extremely challenged:

In accordance with the WNP-2 EOPs, heat removal is extremely challenged when suppression pool temperature cannot be maintained below the Heat Capacity Temperature Limit (HCTL). The HCTL is an indication of a challenge to heat removal because it is the highest wetwell temperature at which initiation of RPV depressurization will not result in exceeding the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer to the containment is within the capacity of the containment vent. For all suppression pool temperatures below the HCTL, all energy produced by the reactor can be discharged either through normal heat removal systems (e. g., main condenser, RHR shutdown cooling mode, etc.) or, when normal heat removal systems are inoperable or incapable of heat removal, through primary containment vent paths. 56

The HCTL is used in the WNP-2 EOPs to preclude failure of the containment or equipment necessary for safe shutdown by assuring that RPV blowdown does not cause containment pressure to exceed the pressure at which containment venting (irrespective of the offsite release) must proceed. The HCTL is determined assuming all heat removal capability from the containment is lost, and the airspace and water in the wetwell are in thermal equilibrium. Any challenge to heat removal that raises suppression pool temperature above the HCTL may also raise primary containment pressures to levels that require manual operation of the containment vent which is a containment barrier loss.

RAI #27: WNP-2 IC CG1: ATWS with Challenge to Core Cooling

b. The WNP-2 EAL under IC CG1 includes the condition "operator actions WERE NOT successful in reducing reactor power to less than 5%," which is not specified in the corresponding NUMARC/NESP-007 EAL.

Justify this deviation from the NUMARC/NESP-007 guidance.

Response to RAI #27b:

The 5% is the APRM downscale trip setpoint and is the power level that closely approximates the thermal energy produced by the reactor at decay heat levels shortly after shutdown. This power level, therefore, should be within the capacity of normal decay heat removal systems. The basis discussion of NUMARC/NESP-007 IC SG2 states that this classification is necessary if extreme challenges to core cooling or heat removal exist and "... the reactor has not been brought below ... typically 3 to 5% power." 57

However, the SG2 example EAL specifically indicates that the criterion for this IC is a scram that is not successful. As mentioned in the response to RAI #26a, Systems Malfunction Question #7 in "NUMARC and NRC Questions and Answers" states that "... a scram is considered unsuccessful if it does not result in achieving a state in which the reactor will remain shut down under all conditions without boron." Since the Questions and Answers response is a later clarification of the NUMARC/NESP-007 example IC wording, the Supply System assumes that it is NUMARC/NESP-007 intent to define an unsuccessful scram as given in the Questions and Answers. Accordingly, the WNP-2 EAL has been changed as discussed in the response to RAI #26a. 58

WNP-2 NRC SUBMITTAL CERTIFICATION FORM

SUBMITTAL:

TITLE: RESPONSE TO REQUESTS FOR ADDITIONAL INFORMATION ON EMERGENCY CLASSIFICATION

STATEMENT NUMBER	OBJECTIVE EVIDENCE LIST EXHIBITS/REFERENCES (ATTACHED)	COMMENTS
1	Submittal Tab C (PPM 13.1.1) and Tab D (PPM 13.1.1A)	Defueled mode now indicated where applicable.
2	Submittal Tab C (PPM 13.1.1) and Tab D (PPM 13.1.1A)	Safe shutdown buildings now indicated where applicable.
3	WNP-2 ODCM, Section 6.2.2.1 (not attached)	
4	Submittal Tab C, Initiating Condition (IC) GU1 and GA1	
5	TECHNICAL BASES FOR GU1, GA1, GS1 & GG1 EALS, INCREASED RADIATION RELEASE TO THE ENVIRONMENT (attached).	
6	Submittal Tab B.1 (PPM 4.12.2.1 ABNORMAL RELEASE OF RADIOACTIVITY)	
7	Submittal Tab A (PPM 13.1.1, IC GU1)	
8	WNP-2 ODCM, Section 6.2.2 (not attached)	
9	RASCAL output using containment bypass and fuel gap release (attached)	
10	Submittal Tab C (PPM 13.1.1, IC GA1)	
11	Submittal Tab C (PPM 13.1.1, IC GA1)	
12	Submittal Tab C (PPM 13.1.1, IC GU1, GA1, GS1, & GG1)	

STATEMENT NUMBER	OBJECTIVE EVIDENCE LIST EXHIBITS/REFERENCES (ATTACHED)	COMMENTS
13	TECHNICAL BASES FOR GU1, GA1, GS1 & GG1 EALS, INCREASED RADIATION RELEASE TO THE ENVIRONMENT (attached).	
14	TECHNICAL BASES FOR GU1, GA1, GS1 & GG1 EALS, INCREASED RADIATION RELEASE TO THE ENVIRONMENT (attached).	
15	Submittal Tab C (PPM 13.1.1, IC GS1)	
16	TECHNICAL BASES FOR GU1, GA1, GS1 & GG1 EALS, INCREASED RADIATION RELEASE TO THE ENVIRONMENT (attached).	
17	TECHNICAL BASES FOR GU1, GA1, GS1 & GG1 EALS, INCREASED RADIATION RELEASE TO THE ENVIRONMENT (attached).	
18	Submittal Tab C (PPM 13.1.1, IC GS1)	
19	TECHNICAL BASES FOR GU1, GA1, GS1 & GG1 EALS, INCREASED RADIATION RELEASE TO THE ENVIRONMENT (attached).	
20	Appendix B of EPA-400-R-92-001 (attached)	
21	Submittal Tab C, (PPM 13.1.1, IC, GU1, GA1, GS1, & GG1)	
22	Submittal Tab C (PPM 13.1.1, IC FA1)	
23	FSAR Table 12.3-1 (attached)	
24	Instrument Master Data Sheet ARM-RIS-2, Setpoint References Section (attached).	

STATEMENT NUMBER	OBJECTIVE EVIDENCE LIST EXHIBITS/REFERENCES (ATTACHED)	COMMENTS
25	Instrument Master Data Sheet ARM-RIS-1 (attached).	
26	Submittal Tab C (PPM 13.1.1, IC FA1)	
27	Table of representative ARM readings (attached)	
28	Instrument Master Data Sheets for ARM-RIS-4,5,6,7,8,9,10,11,12,13,23,24,32,& 33 (attached)	
29	Instrument Master Data Sheet ARM-RIS-2 (attached).	
30	Instrument Master Data Sheet ARM-RIS-1, Setpoint References Section (attached).	
31	Submittal Tab B.2 (PPM 5.1.7, Primary Containment Flooding)	
32	Submittal Tab B.2 (PPM 5.1.7, Primary Containment Flooding)	
33	Submittal Tab B.2 (PPM 5.1.7, PRIMARY CONTAINMENT FLOODING)	
34	PPM 5.1.6, RPV FLOODING - ATWS (attached)	
35	PPM 5.1.4, RPV FLOODING (attached)	
36	Submittal Tab B.3 (Engineering Calculation NE-02-94-57)	
37	Submittal Tab C (PPM 13.1.1, Fission Product Barrier Degradation table)	
38	NUMARC/NESP-007, page 5-21 (attached)	

STATEMENT NUMBER	OBJECTIVE EVIDENCE LIST EXHIBITS/REFERENCES (ATTACHED)	COMMENTS
39	NUMARC/NRC QUESTIONS & ANSWERS, page 17 (attached)	
40	Submittal Tab C (PPM 13.1.1, Fission Product Barrier Degradation table)	
41	Submittal Tab C (PPM 13.1.1, Fission Product Barrier Degradation table).	
42	Submittal Tab C (PPM 13.1.1, Fission Product Barrier Degradation table)	
43	FSAR page 6.2-5 (attached)	
44	PPM 5.2.1, Primary Containment Control (attached)	
45	Submittal Tab C (PPM 13.1.1, IC LA1)	
46	Submittal Tab C (PPM 13.1.1, IC LA1)	
47	Submittal Tab C (PPM 13.1.1, IC NU1, NA1, & NS1)	
48	Submittal Tab C (PPM 13.1.1, IC NG1)	
49	FSAR page 1.5-13 (attached)	
50	NUMARC/NRC Questions and Answers, page 27 (attached)	
51	Submittal Tab C, IC JU1	
52	Submittal Tab C, IC JA1	
53	Submittal Tab C, IC JS1	
54	Submittal Tab C, IC CA1, CS1, and CG1	
55	Submittal Tab C, IC CA1, CS1, and CG1	

STATEMENT NUMBER	OBJECTIVE EVIDENCE LIST EXHIBITS/REFERENCES (ATTACHED)	COMMENTS
56	PPM 5.2.1, PRIMARY CONTAINMENT CONTROL, Wetwell Temp (attached)	
57	NUMARC/NESP-007, page 5-80 (attached)	
58	NUMARC/NRC Questions and Answers, page 28 (attached)	

COMMENTS: _____

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Based on the objective evidence provided above, the Material Statements, identified above in the subject document are complete and accurate in all material aspects.

PREPARED: D.E. Larson October 11, 1994

REVIEWER: DB Holmes / DA Barnes 10-12-94