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July 6, 1990

Mr. S. D. Ebneter
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta St., NW, Suite 2900
Atlanta, Georgia 30323

Subject: McGuire Nuclear Station
Docket Nos. 50-369, 370
Response to Evaluation of McGuire Nuclear Stations
Emergency Plan Review; Revisions 27, 28 and 29

Gentlemen:

As requested by your letter of May 23, 1990, attached is our response to items relating to the subject revisions, which the NRC staff felt was a decrease in the effectiveness of the McGuire Nuclear Station Emergency Plan.

Upon further evaluation I agree that several items need to be revised. As such, the McGuire Nuclear Station Emergency Plan will be revised. An Emergency Plan Revision, pursuant to 50.54(q), will be provided at a later date. It is my intent to submit these plan changes upon final resolution of all the issues identified by your May 23, 1990 letter. Accordingly, a revised plan will be forwarded to you within 30 days of final resolution of all the issues.

There are several areas which I feel that our proposed revisions don't reduce the effectiveness of the plan, and thus are prudent and appropriate. For these revisions, I have provided additional justification. To further discuss these justifications in more detail, I request a meeting with the appropriate members of the NRC staff prior to a final ruling on these issues. If this is acceptable, please contact R. E. Harris, of my staff, at (704) 373-8669 to make arrangements for the meeting.

I will assure that appropriate resources will be allocated so that a timely resolution of the issues identified can be accomplished. Your timely assistance in this effort is greatly appreciated.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

WTB/202/lcs

Attachment

IE35
1/1

xc: U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Mr. Darl S. Hood
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Mr. P. K. VanDoorn
NRC Resident Inspector
McGuire Nuclear Station

RESPONSE TO 4.1.1-ITEM # 1

[RP/0/A/5700/00-Event Category 4.1.1 (Primary Coolant Leak)-Alert IC # 1 and attendant EAL]

Alert Initiating Condition (IC) # 1, Page D-10: The MNS EAL for primary coolant leak greater than 50 gpm was revised to include the modifier, "Leak cannot be isolated within 15 minutes." Inclusion of such a modifier does not meet the anticipatory intent of NUREG-0654 which provides that classification be rendered upon detection of the condition.

The 15 minute modifier was added to this EAL in recognition that in a plant of this design, leakage can occur (and has occurred) in reactor coolant system auxiliary components and/or systems that is readily identified and easily isolated. One example to consider is leakage in the letdown header of the Chemical and Volume Control System. A leak of this type can be easily isolated from the reactor coolant system by closure of the letdown header isolation valves. Another example is leakage through check valves into a cold leg accumulator. Again, this leakage can be easily isolated by closure of the cold leg accumulator isolation valve. Reactor coolant leakage which is quickly identified and isolated is clearly not an "actual or potential substantial degradation of level of safety of the plant". In these examples, reactor coolant system inventory is not threatened and waste processing systems capacity is not challenged. In fact, the only adverse effects on the plant from the examples given are loss of the normal letdown header functions in the first example and inoperability of one cold leg accumulator in the second example. Loses or adverse effects of this nature do not meet the intent of an Alert level emergency declaration. Duke Power Co and McGuire Nuclear Station share the NRC's interest in making our Emergency Plan as effective as it can be. Over classification and declaration of the resultant emergency class when it's purpose is not served decreases the effectiveness of the Emergency Plan. It is in this interest that use of the modifier is prudent.

The modifier purposely states "cannot be isolated within 15 minutes" vs "is not isolated within 15 minutes". This is to make clear that if prompt isolation is either unsuccessful or not practical, then anticipatory judgement is used and the Alert declaration is made without waiting for the 15 minute clock to "time out". This intent is further explained to the emergency coordinator in the EAL Technical Basis document. When viewed in this manner, this EAL is anticipatory in nature and meets the intent of the NUREG 0654 example Initiating Condition.

It should also be noted that in the Region's previous review of the Oconee Emergency Plan, use of a similar modifier was accepted. In addition to there being sound philosophical and technical justification for using the modifier, we have a strong interest in consistency between Duke Power Co's nuclear stations in the application of EALs.

Basis: A significant reactor coolant system leak exist which cannot be promptly isolated. Isolation attempts have either been unsuccessful or are not practical due to the location or nature of the leak. Continuing leakage will challenge maintenance of reactor coolant system inventory during cooldown to cold shutdown conditions and will also challenge the capacity of waste processing systems. Deterioration at leak location could result in leak rate increase.

RESPONSE TO 4.1.1-ITEM # 2

[RP/0/A/5700/00-Event Category 4.1.1 (Primary Coolant Leak)-Alert IC # 3 and attendant 'a' EAL]

Alert IC # 3, Page D-14: The revised EAL no longer addresses a steam line break inside containment as was described in the previously approved MNS Emergency Plan. The EAL now is applicable only to steam leaks outside containment. This change decreases the effectiveness of the Plan and is contrary to the provisions of NUREG-0654.

The previously approved MNS Emergency Plan did not specify that the steam break location be inside containment. We have always assumed the steam leak location to be outside containment. The previously approved Emergency Plan EAL format included a list "symptoms" which were intended to give guidance in recognizing this particular event. No symptoms were listed which would be indicative of a high energy release inside containment.

It is our view that there is no significant difference in the level of threat to the public between a steam line break inside containment coincident with primary to secondary leakage and a LOCA of the same magnitude. The level of degradation of plant safety is not significantly impacted by the fact that a steam generator is in the flow path for primary coolant to escape the reactor coolant system and enter containment. Although the steam generator blowdown to containment will result in excessive cooling of the reactor coolant system and the reactor core, these are previously analyzed effects and do not significantly add to the level of threat to the public. Therefore, for classification purposes, our Emergency Plan EALs treat this event as a LOCA. For primary to secondary leaks up to 50 gpm, the classification would be a NOUE. For primary to secondary leaks greater than 50 gpm, the classification would be an Alert if subcooling was maintained and a SAE if subcooling was not maintained.

The NUREG 0654 example IC (Alert #4) from which the MNS EAL was derived states:

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

The obvious intent of this example IC as applied to a BWR is to describe a BWR plant event where a steam line break exist coincident with a failure of MSIVs to fully isolate. This event results in a loss of primary coolant outside containment to atmosphere. A similar environmental and public threat at a PWR can only occur for the case of a steam line break outside containment with primary to secondary leakage greater than 10 gpm but less than 50 gpm. The stipulation that the primary to secondary leakage be less than 50 gpm takes into account the fact that for reactor coolant leak rates greater than 50 gpm, the reactor coolant system fission product boundary is considered "lost" and a SAE declaration would be warranted based on the loss of two fission product barriers (the reactor coolant system via leaking steam generator u-tubes and containment via the breached steam line). We will add the less than 50 gpm modifier to the MNS EAL. Modified in this manner, the reactor coolant system boundary is considered intact as relates to fission product barriers and only one barrier (containment via the unisolable steam line break) is considered lost.

The Alert IC # 3 wording will be changed to:

"S/G tube leak with an unisolable secondary line break outside containment".

1) The Alert IC # 3 attendant 'a' EAL will be changed as follows:

- S/G tube leak greater than 10 gpm but less than 50 gpm.

  AND

  NC subcooling greater than 0 Deg F.

  AND

  Unisolable secondary (main steam or feedwater) break outside containment on the affected S/G.

Basis: Significant primary to secondary leakage compounded by steam line break to the environment. The primary to secondary leakage is significantly in excess of Tech Spec limits and therefore is in excess of that amount of leakage assumed in accident analysis for a secondary line break coincident with steam generator tube leakage. The reactor coolant system is expected to remain subcooled with core decay heat removal via unaffected steam generators and ECCS flow if safety injected actuated. There is no threat to the fuel clad and the reactor coolant system boundary is considered intact as a fission product barrier for leak rates less than 50 gpm. Without the compounding adverse effects of the secondary line break outside containment, this amount of reactor coolant system leakage would warrant only a NOUE declaration unless effluent release levels required a higher classification.

2) The Alert IC # 3 attendant 'b' EAL will be changed as follows:

- Unisolable secondary (main steam or feedwater) break outside containment.

  AND

  Field monitoring teams detect activity at the protected area fence at greater than or equal to 2 mRem/hr WB.

Basis: Secondary break with no known (or minor) preexisting S/G tube leakage. Field monitoring teams detect activity levels of 2 Mrem/hr conservatively near the plant.

RESPONSE TO 4.1.1-ITEM # 3

[RP/0/A/5700/00-Event Category 4.1.1 (Primary Coolant Leak)-SAE IC # 1 and attendant EAL]

Site Area Emergency IC # 2, Page D-16: This EAL was revised to add the requirement, "Existing NV, NI, and ND flow cannot maintain subcooling greater than 0 degree f." Delaying declaration until loss of subcooling occurs is not consistant with the anticipatory intent of NUREG-0654 which requires declaration immediately upon identification that makeup capacity has been exceeded. It was not clear that this was the intent of the revised EAL.

The NUREG 0654 example IC for this EAL, "Known loss of coolant accident greater than makeup pump capacity", has been one of the most difficult (and important) events for the Emergency Coordinator to recognize and properly classify. The parameters utilized in the previously approved Emergency Plan EAL, such as degrading containment conditions, decreasing pressurizer pressure, and containment EMFs in alarm, did not give any guidance in determining the rate of reactor coolant leakage. The only parameter besides subcooling status that might give this kind of guidance is pressurizer level. However, pressurizer level behavior can give misleading information concerning reactor coolant system inventory status depending on several factors, including the leak location, how long a LOCA event has been in progress, and changes in reactor coolant system temperature. This was very graphically demonstrated at Three Mile Island. For example, a small LOCA coincident with a steam generator steam line break (e.g. S/G safety relief stuck open) can produce an initial pressurizer level and pressure decrease large enough to cause a safety injection. This decrease is magnified by the steam break induced cooldown and may quickly reverse itself if the S/G safety valve reseats. In an event of this type, the ECCS injection flow will act to quickly restore pressurizer level and pressure with likely maintenance of subcooling. Reactor coolant

pumps remain running (or the system remains amenable to natural circulation) and the steam generators remain available to remove core decay heat. The plant status is never degraded to the point that the intent of the Site Area Emergency definition is met. Another example to consider is a LOCA of intermediate size. As reactor coolant pressure decreases, the ECCS injection flow increases and the break flow decreases. This is a very dynamic situation and trying to determine a "Known loss of coolant accident greater than makeup pump capacity" based on ECCS total injection flow and pressurizer level response is extremely difficult. Also, for this scenario, once the intermediate head ECCS pumps begin injecting at 1500 psig, reactor coolant system inventory will likely stabilize such that pressure remains above saturation. Again, reactor coolant pumps will likely remain available and steam generators can be utilized to aid in removing core decay heat.

For larger size LOCAs where loss of subcooling is approached, the Emergency Coordinator can use (and is encouraged to use) anticipatory judgement and make a Site Area Emergency declaration prior to the actual loss occurring. This anticipatory judgement philosophy is referenced in the emergency classification procedure, enclosure 4.4 (Emergency Declaration Guidelines), item A). It should also be noted that even a momentary loss of subcooling, once recognized, will require a SAE declaration in accordance with enclosure 4.4 (Emergency Declaration Guidelines), item C).

The status of subcooling is readily available to the Emergency Coordinator through two separate and independent trains of accident monitoring instrumentation plasma display monitors-very specific and directly observable indicators. The ease of recognition of this parameter during an extremely busy time in the control room, along with the importance of subcooling to core decay heat removal, provides a strong argument for its use as a reliable threshold parameter for determining if makeup capacity has in fact been exceeded. The clear intent of this SAE Example IC being in NUREG-0654 is to recognize the threat to core cooling. There is no significant threat to core cooling as long as subcooling is maintained. With subcooling maintained during a LOCA, an Alert is warranted. Without subcooling being maintained

during a LOCA, a SAE is warranted due to the significant increase in difficulty in removing core decay heat resulting from the degraded secondary heat sink.

Finally, in the Region's review of the Oconee Nuclear Station Emergency Plan EAL procedure, there were no comments regarding the same use of the subcooling parameter during a LOCA. In the on-going effort to achieve consistency within Duke Power Co in the application of EALs, we feel that similar use at McGuire is warranted.

Basis: Reactor coolant system inventory net loss, combined with system depressurization and core decay heat load, has resulted in primary system saturation. Reactor coolant pumps are stopped and natural circulation is degraded, resulting in a severe decrease in heat removal via steam generators. Mitigation of the accident and cool down of the primary system to cold shutdown is impacted by the ineffective secondary heat sink and difficulty in determining reactor coolant system inventory. These adverse effects are triggered by the loss of subcooling. A LOCA without loss of subcooling, while a serious event, is significantly less challenging to manage and recover from.

RESPONSE TO 4.1.1-ITEM # 4

[RP/0/A/5700/00-Event Category 4.1.1 (Primary Coolant Leak)-SAE IC # 2 and attendant EAL]

<u>Site Area Emergency IC # 2, Page D-18</u>: The revised MNS EAL, steam generator tube leak with an unisolable steam line break outside containment and indication of fuel damage, does not appear to be consistant with NUREG-0654. As written, the EAL implies the loss of three fission product barriers which would be classified as a General Emergency, whereas the clear intent of NUREG-0654 is loss of 2 of 3 fission product barriers (steam line break inside containment). The previous revision of the MNS (EAL?) included provisions for steam line breaks inside containment.

As noted in the MNS response to item #2 (Alert IC # 3, page D-14) under Event Category 4.1.1, the MNS Emergency Plan EALs consider primary to secondary leakage coincident with a steam line break inside containment to be classified based on LOCA criteria. Also, the previously approved MNS Emergency Plan EALs did not make any specific reference to a steam line break inside containment. Upon further review of the McGuire EALs concerning steam line breaks coincident with primary to secondary leakage with or without indications of fuel damage, we have made the following changes:

1) The current EAL referenced in Item # 4 under Event Category 4.1.1 (SAE IC # 2, page D-18) will be deleted. The corresponding EAL under Event Category 4.1.3 will also be deleted.

2) The 4.1.1 SAE IC # 2 wording will be changed to:

   "S/G tube leak with an unisolable steam line break outside containment".

   The attendant SAE IC # 2 EAL will read as follows:

   - Primary to secondary leakage greater than 50 gpm.

     AND

     Unisolable secondary (main steam or feedwater)

break outside containment on the ruptured S/G.

Basis: Failure of two fission product barriers; containment via steam break and reactor coolant system via S/G tube leak (reactor coolant system threshold loss is 50 gpm).

3) A new IC # 3 will be added to 4.1.1 SAE to read:

"S/G tube leak with steam line break inside containment and indication of fuel damage".

The attendant SAE IC # 3 EAL will read as follows:

- Primary to secondary leakage greater than 50 gpm.

AND

Unisolable secondary (main steam or feedwater) break inside containment on the ruptured S/G.

AND

Total fuel clad failure greater than 5% per chemistry analysis (or valid reading on EMF-51a or 51b of 275 R/HR).

Basis: Failure of two fission product barriers; reactor coolant system via a faulted and ruptured S/G and fuel clad via 5% failure (reactor coolant system threshold loss is 50 gpm; clad threshold loss is 5% total fuel failure). EMF-51a or 51b (high range containment area monitors) reading 275 R/HR is indicative of 5% total fuel gap activity released to containment.

4) The fission product barrier IC #1 in Event Category 4.1.2 under General Emergency will be changed as follows to

ensure that steam generator primary to secondary leakage and secondary line breaks outside containment are properly considered when evaluating these plant events relative to loss of fission product barriers.

LOSS OF 2 OF 3 FISSION PRODUCT BARRIERS WITH POTENTIAL FOR LOSS OF 3rd BARRIER.

Note: To classify at this level, at least one condition from two of the three categories (A,B,C) must be satisfied and at least one condition from the third category must be satisfied or have the potential to be satisfied.

A) LOSS OF OR THREAT TO CLAD BARRIER

- Total fuel clad failure greater than 5% per chemistry analysis.

- Containment EMF 51a or 51b valid reading of 275 R/HR (equates to 5% fuel gap activity released to containment).

- Plant conditions require entry into EP/1 or 2/A/5000/12.1 (Response to inadequate core cooling).

B) LOSS OF REACTOR COOLANT SYSTEM BARRIER

- Reactor coolant system leak (including S/G tube leak) greater than 50 gpm.

C) LOSS OF OR THREAT TO CONTAINMENT BARRIER

- Incomplete containment integrity.

- Known containment leakage in excess of tech specs.
- Containment atmosphere hydrogen concentration greater than or equal to 9%.
- Containment pressure greater than or equal to 60 psig.
- Unisolable secondary (main steam or feedwater) break outside containment with S/G tube leak greater than 50 gpm.

Basis: Category B) "Loss of reactor coolant system barrier", will be added to ensure that the amount of leakage constituting a loss of that barrier is recognized. The new EAL to be added to Category C), "Unisolable secondary line break outside containment with S/G tube leakage greater than 50 gpm", ensures that this type event is considered during determination of a loss of the containment barrier.

RESPONSE TO 4.1.3-ITEM # 1

[RP/0/A/5700/00-Event Category 4.1.3 (Steam System Failure)-Alert IC # 1 and attendant EALs]

<u>Alert IC # 1, Page D-34</u>: See comments for Item 2, Event Category, 4.1.1 above.

The Alert IC # 1 wording will be changed to:

"Unisolable secondary line break outside containment with S/G tube leakage".

1) The Alert IC # 1 attendant 'a' EAL will be changed as follows:

   ♦ Unisolable secondary (main steam or feedwater) break outside containment.

   AND

   NC subcooling greater than 0 Deg F.

   AND

   S/G tube leak greater than 10 gpm but less than 50 gpm on the affected S/G.

   Basis: See Event Category 4.1.1 (Primary Coolant Leak)-Alert IC # 3 and attendant 'a' EAL for justification and Basis.

2) The Alert IC # 1 attendant 'b' EAL will be changed as follows:

   ♦ Unisolable secondary (main steam or feedwater) break outside containment.

   AND

   Field monitoring teams detect activity at the protected area fence at greater than or equal to 2 Mrem/hr WB.

   Basis: See Event Category 4.1.1 (Primary Coolant Leak)-Alert IC # 3 and attendant 'b' EAL for justification and Basis.

RESPONSE TO 4.1.3-ITEM # 2

[RP/0/A/5700/00-Event Category 4.1.3 (Steam System Failure)-SAE IC # 1 and attendant EAL]

Site Area Emergency IC # 1, Page D-37: See comments for Item 4, Event Category 4.1.1 above.

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1) The current EAL referenced in Item # 2 under Event Category 4.1.3 (SAE IC # 1, Page D-37) will be deleted.

2) The 4.1.3 SAE IC # 1 wording will be changed to:

"Unisolable steam line break outside containment with a S/G tube leak".

The SAE IC # 1 attendant EAL will be changed as follows:

♦ Unisolable secondary (main steam or feedwater) break outside containment.

AND

Primary to secondary leakage greater than 50 gpm on the faulted S/G.

Basis: See 4.1.1 SAE IC # 2 and attendant EAL for justification and Basis.

3) A new IC # 2 will be added to 4.1.3 SAE to read:

"Steam line break inside containment with S/G tube leak and indication of fuel damage".

The attendant SAE IC # 2 EAL will read as follows:

♦ Unisolable secondary (main steam or feedwater) break inside containment.

AND

Primary to secondary leakage greater than 50 gpm on the faulted S/G.

AND

Total fuel clad failure greater than 5% per chemistry analysis (or valid reading on EMF-51a or 51b of 275 R/HR.

Basis: See 4.1.1 SAE IC # 3 and attendant EAL for justification and Basis.

RESPONSE TO 4.1.4-ITEM # 1

[RP/0/A/5700/00-Event Category 4.1.4 (High Radiation/Radiological Effluents)-SAE IC # 1 and attendant EAL]

Site Area Emergency # 1.a, Page D-41: This EAL was revised to require dose calculations confirming dose rates at the site boundary of 50 Mrem/hr WB and 250 Mrem/hr thyroid for 30 minutes. This change appears to decrease the effectiveness of the Plan in that specific, directly observable quantities are no longer used as the sole basis for the emergency declaration.

The modifier requiring conformation calculations of dose rates was added in an effort to preclude an unnecessary SAE declaration based on an EMF reading high from an upscale instrument failure. Upon further review of all EALs under event category 4.1.4 (High Radiation/Radiological Effluents), the following changes will be

made in order to address the reviewers specific comment, our concern about instrument failure, and to attain more consistency between Duke Power Co's three nuclear stations.

1) The Alert IC # 1 wording remains unchanged. The attendant Alert IC # 1 EALs will be changed as follows:

- ♦ Any valid area EMF reading greater than or equal to 1000 times background value.

  AND

  Excessive area radiation levels due to either unknown or uncontrolled causes.

Basis: Unexpected increased radiation levels of this magnitude are indicative of significant degradation in the control of radioactive material.

- ♦ Valid indication on EMF-41 reading greater than or equal to 1000 times background value.

  AND

  Excessive airborne levels due to either unknown or uncontrolled causes.

Basis: Unexpected increased airborne levels of this magnitude are indicative of significant degradation in the control of radioactive material.

2) The Alert IC # 2 wording will be changed to:

"Gaseous or liquid radiological effluents exceed 10 times Tech Spec limits".

The attendant Alert IC # 2 EALs will be changed as follows:

- Valid indication on EMF-35 reading greater than or equal to 10 times Trip II setpoint.

Basis: The Trip II setpoint is set below Tech Spec limits. Therefore, 10 times Trip II is a conservative threshold value.

- Valid indication on EMF-36(L) reading greater than or equal to 10 times Trip II setpoint.

Basis: Same as for EMF-35.

- Valid indication on EMF-37 reading greater than or equal to 10 times Trip II setpoint.

Basis: Same as for EMF-35.

- Valid indication on EMF-50 reading greater than or equal to 10 times Trip II setpoint.

Basis: Same as for EMF-35.

- Valid indication on EMF-31 reading greater than or equal to 10 times Trip II setpoint.

Basis: Same as for EMF-35.

- Valid indication on EMF-49 reading greater than or equal to 10 times Trip II setpoint.

Basis: Same as for EMF-35.

- Valid indication on EMF-44 reading greater than or

equal to 10 times Trip II setpoint.

Basis: Same as for EMF-35.

- Gaseous or liquid radiological effluents exceed 10 times Tech Spec limits as determined by RP or chemistry procedures.

Basis: This EAL takes into account field monitoring team results or calculations performed from Tech Spec required sampling while EMFs inoperable during planned releases.

3) The SAE IC # 1 wording will be changed to:

"Accidental release of gaseous effluents".

The attendant SAE IC # 1 EALs will be changed as follows:

- Valid indication on EMF-36(L) reading greater than or equal to 3.25E6 cpm (equates to 50 mrem/hr WB at site boundary).

Basis: Memorandum from L.E. Haynes, Scientist/RP/MNS, dated 1/9/90, which documents calculations of dose rates of 50 mrem/hr to the WB at the site boundary. These dose rates are projected from an instantaneous release based on EMF count rate.

- Valid indication on EMF-36(H) reading greater than or equal to 6.1E2 cpm (equates to 50 mrem/hr WB at site boundary).

Basis: Same as for EMF-36(L).

- Valid indication on EMF-37 reading greater than or equal to 2.8E5 cpm.

Basis: Memorandum from L.E. Haynes, Scientist/RP/MNS, dated 6/28/90, which documents dose rates of 250 Mrem/hr to the thyroid at the site boundary. These dose rates are projected from an instantaneous release based on EMF count rate and adverse meteorology.

- Dose calculations based on containment conditions project dose rates at the site boundary greater than or equal to 50 mrem/hr WB or 250 mrem/hr thyroid.

Basis: Calculations based on design containment assumed bypass leakage and existing containment parameters if no release in progress. If release in progress, calculations based on release rate and existing containment parameters.

- Field monitoring team measurements have determined dose rates at the site boundary greater than or equal to 50 mrem/hr WB or 250 mrem/hr thyroid.

Basis: Criteria met based on actual effluent plume samples.

4) The GE IC # 1 wording will be changed to:

"Accidental release of gaseous effluents".

The attendant GE IC # 1 EALs will be changed as follows:

- Valid indication on EMF-36(H) reading greater than or equal to 6.1E3 cpm (equates to 1 Rem to the WB at site boundary integrated over 2 hours).

Basis: Memorandum from L.E. Haynes, Scientist/RP/MNS, dated 1/9/90, which documents dose calculations of 1 Rem to the WB at the site boundary integrated over 2 hours. This dose is projected based on instantaneous release.

- Valid indication on EMF-37 reading greater than or equal to 2.8E6 cpm (equates to 5 Rem to the thyroid at site boundary integrated over 2 hours).

Basis: Memorandum from L.E. Haynes, Scientist/RP/MNS, dated 6/28/90, which documents dose calculations of 5 Rem to the thyroid at the site boundary integrated over 2 hours. This dose is projected based on instantaneous release.

- Dose calculations based on containment conditions project dose rates at the site boundary greater than or equal to 1 rem/hr WB or 5 rem/hr thyroid integrated over a 2 hour period.

Basis: Calculations based on design containment assumed bypass leakage and existing containment parameters if no release in progress. If release in progress, calculations based on release rate and existing containment parameters.

- Field monitoring team measurements have determined dose rates at the site boundary greater than or equal to 1 rem/hr WB or 5 rem/hr thyroid integrated over a 2 hour period.

Basis: Criteria met based on actual effluent plume samples.

RESPONSE TO 4.1.5-ITEM # 1

[RP/0/A/5700/00-Event Category 4.1.5 (Loss of Shutdown Functions) -Alert IC # 1 and attendant EAL]

<u>Alert IC # 1, Page D-45</u>: This EAL was revised to be mode specific as well as to require that subcooling margin cannot be maintained greater than 0 degrees F. The anticipatory intent of the EAL has been diminished by adding the loss of subcooling requirement.

The Alert IC # 1 wording remains the same. The Alert IC # 1 attendant EAL will be changed as follows:

- Failure of heat sink in modes 5 and 6 results in uncontrolled heat-up.

  AND

  Core exit thermocouples indicate greater than or equal to 200 °F.

Basis: Inability to maintain cold shutdown due to loss of heat sink.

RESPONSE TO 4.1.5-ITEM # 2

[RP/0/A/5700/00-Event Category 4.1.5 (Loss of Shutdown Functions) -SAE IC # 1 and attendant EALs]

<u>Site Area Emergency IC # 1, Page D-47</u>: See comments for Item 1, Event Category 4.1.5 above.

The SAE IC # 1 wording remains the same. The SAE IC # 1 attendant EALs will be changed as follows:

- Failure of heat sink in mode 4 results in uncontrolled heat-up.

  AND

  Core exit thermocouples indicate greater than or equal to 350 °F.

Basis: Inability to maintain hot shutdown due to loss of heat sink.

- Inability to feed steam generators from any source in modes 1-3.

  AND

  Feed and bleed cooling of the reactor core is necessary to remove core decay heat.

Basis: Steam generators inventory is depleting and heat-up to saturation in the reactor coolant system is anticipated. When feed and bleed is established, subcooling is likely lost and core decay heat is removed via ECCS flow.

RESPONSE TO 4.1.5-ITEM # 3

[RP/0/A/5700/00-Event Category 4.1.5 (Loss of Shutdown Functions) -SAE IC # 2 and attendant EAL]

Site Area Emergency IC # 2, Page D-49: This EAL was revised to require confirmation of core damage. NUREG-0654 nor the previous MNS Plan require core damage for declaration of the ATWS Site Area Emergency. This change, therefore, decreases the effectiveness of the Plan.

The SAE IC # 2 wording remains the same. The SAE IC # 2 attendant EAL will be changed as follows:

- Transient with failure of the reactor protection system to automatically initiate and complete a Rx trip which brings the reactor subcritical (ATWS Event).

  AND

  Control rods cannot be manually tripped or inserted from the Control Room.

Basis: Reactor remains critical during a transient requiring automatic Rx trip. Shutdown of the reactor will likely be accomplished from outside the Control Room. Response time assumed in accident analysis for reactor shutdown is exceeded.

RESPONSE TO 4.1.5-ITEM # 4

[RP/0/A/5700/00-Event Category 4.1.5 (Loss of Shutdown Functions) -GE IC # 2 and attendant EAL]

General Emergency IC # 2, Page D-54: The revised EAL for ATWS required confirmed fuel damage and entry into the inadequate core cooling emergency procedure. This change decreases the effectiveness of the Plan in that the accepted interpretation of the NUREG-0654 is ATWS with core damage or failure of core cooling.

The GE IC # 2 wording will be changed as follows:

"Transient requiring operation of shutdown systems with failure to trip which results in core damage or failure of core cooling capability".

The GE IC # 2 attendant EALs will be changed as follows:

- Transient with failure of reactor protection systems to automatically initiate and complete a Rx trip which brings the reactor subcritical.

  AND

  Actions taken per EP/1 or 2/A/5000/11.1 (Response to Nuclear Power Generation/ATWS) fail to bring the reactor subcritical.

  AND

  Chemistry analysis indicates greater than or equal to 5% total fuel clad failure.

Basis: Same as for SAE IC # 2 attendant EAL. Also, chemistry analysis has confirmed substantial fuel damage resulting from the ATWS event.

- Transient with failure of reactor protection systems to automatically initiate and complete a Rx trip which brings the reactor subcritical.

  AND

  Actions taken per EP/1 or 2/A/5000/11.1 (Response to Nuclear Power Generation/ATWS) fail to bring the reactor subcritical.

  AND

Plant conditions require entry into EP/1 or 2/A/5000/12.1 (Response to Inadequate Core Cooling).

Basis: Same as for SAE IC # 2 attendant EAL. Also, any core damage resulting from the ATWS event will likely be accentuated by the failure of core cooling.

RESPONSE TO 4.1.5-ITEM # 5

[RP/0/A/5700/00-Event Category 4.1.5 (Loss of Shutdown Functions) -GE IC # 3 and attendant 'c' EAL]

<u>General Emergency IC # 3, Page D-65 (55?)</u>: This EAL was an addition to the Plan which provided greater clarification; however, the requirement for Emergency Coordinator Judgement is inappropriate. The conditions for this EAL specifically meet the definition of General Emergency without an additional judgement call.

The intent of adding this EAL was to provide guidance to the Emergency Coordinator for determination of a General Emergency in the event of a loss of decay heat removal compounded by a lack of core exit thermocouple and Lower Range RVLIS instrumentation. With the reactor vessel head in place, this degraded core monitoring capability is allowed for only a total time period of approximately 4 hours per outage. Since this instrumentation is not required with the reactor vessel head removed, there are long periods of time during a refueling outage (other than the 4 hours) when core coolant direct status is unavailable. A loss of decay heat removal leading to boiling in the core cannot produce even a small amount of system pressurization with the vessel head removed. Therefore, available loop level instrumentation will initially provide reliable level indication. This is especially true with the recent addition of ultrasonic loop level indicators

during refueling outages. Boil-off combined with inadequate make-up resulting in decreasing net inventory can be tracked by observation of the loop level instrumentation. When level drops to the bottom of the loops, the core is still covered by 3 feet of coolant. Therefore, the specifics of the IC (i.e., core uncovered) are not yet met. The Emergency Coordinator judgement is necessary for anticipatory declaration of a GE. Factors to consider include rate of decrease of observed loop level prior to loss of indication and status of efforts to provide adequate make-up to the system. In consideration of the above justifications, the EAL remains unchanged.

RESPONSE TO 4.1.6-ITEM # 1

[RP/0/A/5700/00-Event Category 4.1.6 (Loss of Power)-Alert IC # 1, Alert IC # 2, SAE IC # 1, SAE IC # 2- and attendant EALs]

Alert and Site Area Emergency ICs, Pages D-60 through D-63: The MNS EAL scheme was revised to be mode specific. The new EAL did not require declaration of an Alert for momentary loss (up to 15 minutes) for loss of all AC power or loss of DC power during modes 5 and 6. In addition, the MNS EAL did not require declaration of the Site Area Emergency for loss (greater than 15 minutes) of all AC power or loss of DC power during modes 5 and 6. These changes are considered a degradation from the previous Plan and are inconsistent with the provisions of NUREG-0654.

Upon further review of the Loss of Power event category, it was determined that no EALs existed to account for a momentary loss of all AC or DC in modes 1-4. To address this inadquacey, the following changes will be made:

1) The Alert IC # 1 wording will be changed to:

"Loss of offsite power and loss of all onsite AC power for greater than 1 minute but less than or equal to 15 minutes in modes 1-4".

The Alert IC # 1 attendant EAL will be changed to:

- Both 4160 V ESS busses are deenergized for greater than 1 minute but less than or equal to 15 minutes in modes 1-4.

Basis: Momentary loss of offsite power and loss of all onsite AC power in modes 1-4. The stipulation of "greater than 1 minute" takes into account the time necessary for the emergency diesel generators to energize the 4160 V ESS busses.

2) The Alert IC # 2 wording will be changed to:

"Loss of all vital DC power for up to 15 minutes in modes 1-4".

The Alert IC # 2 attendant EAL will be changed to:

- Both unit related EVDA and EVDD busses deenergized for up to 15 minutes in modes 1-4.

Basis: Momentary loss of vital DC power in modes 1-4.

3) The original Alert IC # 1 will be renumbered Alert IC # 3. The wording of this IC and the attendant EAL remains unchanged.

4) The original Alert IC # 2 will be renumbered Alert IC # 4. The wording of this IC and the attendant EAL remains unchanged.

We do not agree that a momentary loss of all AC power or DC power in modes 5 and 6 warrant an immediate Alert emergency declaration. The worse case plant reaction to a power loss of this type would be no more than a momentary loss of decay heat removal. During the loss, manual action would be taken if necessary to provide gravity make-up to the reactor coolant system for cooling and Emergency Class declaration would be based on the Loss of Shutdown Functions event category.

We also do not agree that a separate EAL is warranted under SAE in the Loss of Power event category for an extended loss of all AC power or DC power in modes 5 and 6. Again, the worse case plant reaction to a power loss of this type would be an extended loss of decay heat removal. We feel that adequate guidance is provided in the Loss of Shutdown Functions event category to determine the appropriate emergency class.

RESPONSE TO 4.1.7-ITEM # 1

[RP/0/A/5700/00-Event Category 4.1.7 (Fires and Security Actions) -NOUE IC/EAL # 1]

NOUE IC # 1, Page D-65: This EAL was changed from "Fire within the plant lasting 10 minutes" to "Fire within the plant that takes more than 10 minutes to extinguish." The new EAL is inconsistent with NUREG-0654 in that it is dependent on the response of the fire brigade rather than the duration of the fire.

This IC/EAL will be changed to:

"Fire situation (as determined by the Shift Supervisor or designee) within the plant (see Note) lasting longer than 10 minutes".

Basis: A responsible individual has confirmed the existence of a fire and the fire has continued for at least 10 minutes.

RESPONSE TO 4.1.7-ITEM # 2

[RP/0/A/5700/00-Event Category 4.1.7 (Fires and Security Actions) -Alert IC # 1 and attendant EAL]

<u>Alert IC # 1, Page D-68</u>: The revised MNS EAL, Fire Resulting in Loss of any Required Function (Both Trains), does not meet the intent of NUREG-0654, Fire Potentially (Affecting?) Safety Systems. The previous MNS EAL was consistant with NUREG-0654; therefore, the Plan was revised in a non-conservative manner.

The Alert IC # 1 wording will be changed to:

"Fire potentially affecting safety systems".

The Alert IC # 1 attendant EAL will be changed to :

- Fire resulting in potential deterioration (visible or assumed) to any ESF component (see ENC 4.2) or ESF component subsystem (attendant instrumentation, controls, power supply, cooling or seal water, or lubrication) required by Tech Specs for the current operating mode.

Basis: Observed or assumed deterioration can potentially affect the capability of an ESF component to perform its design function.

RESPONSE TO 4.1.7-ITEM # 3

[RP/0/A/5700/00-Event Category 4.1.7 (Fires and Security Actions) -SAE IC # 1 and attendant 'a' EAL]

<u>Site Area Emergency IC # 1.a, Page D-72</u>: The EAL was changed from "Observation of a Major Fire that Defeats Redundant Safety Systems or Function" to "Fire That Results in the Inability to Maintain Hot Shutdown and NC Subcooling Cannot be Maintained Greater than 0 degrees F." The new EAL is inconsistent with the corresponding NUREG-0654 EAL, Fire Compromising the Functions of Safety Systems. The previous MNS EAL was adequate; therefore, the Plan was revised in a non-conservative manner.

The SAE IC # 1 wording will be changed to:

"Fire compromising the function of safety systems".

The SAE IC # 1 attendant 'a' EAL will be changed as follows:

- Fire resulting in redundant trains of ESF components (see ENC 4.2) or ESF components subsystems (attendant instrumentation, controls, power supply, cooling or seal water, or lubrication) required by Tech Specs for the current operating mode becoming incapable of performing their design function (inoperable).

Basis: It has been determined from either judgement or testing that a fire has rendered inoperable redundant trains of ESF components.

RESPONSE TO 4.1.7-ITEM # 4

[RP/0/A/5700/00-Event Category 4.1.7 (Fires and Security Actions) -SAE IC # 1 and attendant 'b' EAL]

Site Area Emergency, IC # 1.b, Page D-72: The revised MNS EAL, Fire Requiring Control Evacuation and Control Cannot be Established from Stand-by Shutdown Facility and NC Subcooling Cannot be Maintained Greater than 0 degrees F, is inconsistent with the corresponding NUREG-0654 EAL. Specifically, the licensee deleted the 15 minutes requirement for establishing plant control and replaced it with the loss of subcooling requirement. This change is not anticipatory in nature and is a degradation from the previous MNS Plan.

The SAE IC # 1 wording will be changed to:

"Fire compromising the function of safety systems".

The SAE IC # 1 'b' attendant EAL will be changed as follows:

- Fire requiring Control Room evacuation.

  AND

  Control established or in process of being established from the SSF.

Basis: Loss of Control Room safety systems function due to fire. This event at MNS requires control to be established at the SSF vs the Auxiliary Shutdown Panel since the Control Room and ASD Panel share common cables. Due to the limited capabilities of the SSF, a SAE is warranted anytime it is the sole means of control of either unit regardless of the length of time taken to establish control.

RESPONSE TO 4.1.7-ITEM # 5

[RP/0/A/5700/00-Event Category 4.1.7 (Fires and Security Actions) -GE IC # 2 and attendant EAL]

General Emergency IC # 2, Page D-75: This EAL is acceptable as revised; however, it appeared the Stand-by Shutdown Facility should be included here rather than the Auxiliary Shutdown Panel.

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1) The GE IC # 1 wording will be changed to:

"Any major fire which could result in massive common damage or loss of control of the plant".

The GE IC # 1 attendant EAL will be changed as follows:

♦ Fire requiring evacuation of the Control Room.

AND

Control of shutdown systems cannot be established from any plant location.

Basis: Fire has resulted loss of plant status input and control of vital plant systems.

2) The GE IC # 2 wording will be changed to:

"Loss of physical control of the plant due to physical attack".

The GE IC # 2 attendant EAL will be changed as follows:

- Adversaries commandeer the Control Room or other vital area.

Basis: Security breach has resulted in loss of plant status input and control of vital plant systems.

RESPONSE TO 4.1.8-ITEM # 1

[RP/0/A/5700/00-Event Category 4.1.8 (Spent Fuel Damage)-Alert IC # 1 and attendant EAL]

Alert IC # 1, Page D-76: The revised MNS EAL requires verification of a release to the environment. This change decreases the effectiveness of the Plan in that the previous EAL used directly observable plant parameters for indication. In addition, NUREG-0654 only provides for activity release to the fuel handling building or containment not to the environment.

The Alert IC # 1 wording remains unchanged. The Alert IC # 1 attendant EALs will be changed as follows:

CONTAINMENT

- Valid Trip II alarm on 1 EMF-39

AND

Report of fuel damage during core alterations or movement of spent fuel in Containment.

- Valid Trip II alarm on 2 EMF-39

AND

Report of fuel damage during core alterations or movement of spent fuel in Containment.

Basis: Observed or assumed damage to spent fuel with release of radioactivity to Containment confirmed by Containment gaseous monitor EMF-39 Trip II alarm.

FUEL HANDLING BUILDING

- Valid Trip II alarm on 1 EMF-17 (or equivalent).

AND

Valid Trip II alarm on 1 EMF-42.

AND

Report of fuel damage during movement of spent fuel or loads over the spent fuel pool in the fuel building.

- Valid Trip II alarm on 2 EMF-4 (or equivalent).

AND

Valid Trip II alarm on 2 EMF-42.

AND

Report of fuel damage during movement of spent fuel or

loads over the spent fuel pool in the fuel building.

Basis: Observed or assumed damage to spent fuel with release of radioactivity to the spent fuel building confirmed by the fuel building gaseous and area radiation monitors Trip II alarm.

RESPONSE TO 4.1.8-ITEM # 2

[RP/0/A/5700/00-Event Category 4.1.8 (Spent Fuel Damage)-SAE IC # 1 and attendant EAL]

Site Area Emergency IC # 1, Page D-78: Again, this EAL was revised to require confirming dose calculations for declaration, and this is inconsistent with NUREG-0654 which does not provide for activity release to the environment. The previous EAL used observation of damage, fuel pool water level, or radiation alarm to determine classification. With these two considerations, this change decreases the effectiveness of the Plan.

The SAE IC # 1 wording remains unchanged. The SAE IC # 1 attendant EAL will be changed as follows:

CONTAINMENT

- Valid 1 EMF-16 (for Unit-1) or 2 EMF-3 (for Unit-2) Trip II alarm.

AND

Valid 1 (2) EMF-39 off scale high.

AND

Report of spent fuel damage during core alterations or movement of spent fuel in Containment.

AND

Dose rate inside Containment coupled with known Containment leak rate results in calculated dose rate at the Site Boundary of greater than or equal to 50 mrem/hr WB or 250 mrem/hr thyroid.

Basis: Observed or assumed damage to spent fuel with gross release of radioactivity to Containment confirmed by Low Range area and Containment gaseous monitor indications. Dose rate at Site Boundary is calculated using Containment pressure and a known Containment leak path size to determine leak rate. Also, see Basis for the Fuel Handling Building EALs below.

FUEL HANDLING BUILDING

♦ 1 EMF-17 Trip II alarm.

AND

1 EMF-42 Trip II alarm.

AND

Valid indication on 1 EMF-36 reading greater than or equal to 3.26E6 cpm.

AND

Report of fuel damage during movement of spent fuel or loads over spent fuel in Unit 1 fuel building.

◊ 2 EMF-4 Trip II alarm.

AND

2 EMF-42 Trip II alarm.

AND

Valid indication on 2 EMF-36 reading greater than or equal to 3.25E6 cpm.

AND

Report of fuel damage during movement of spent fuel or loads over spent fuel in Unit 2 fuel building.

Basis: Observed or assumed damage to spent fuel with gross release of radioactivity to the spent fuel building confirmed by low range area and fuel building gaseous monitor indications. The unit vent gaseous monitor reading is calculated assuming a dose rate at the site boundary of greater than or equal to 50 Mrem/hr WB.

In order to classify at the Site Area Emergency level, criteria must be established to quantify the level of threat to the public from major damage to spent fuel. Assuming a public threat of the magnitude defined by Site Area Emergency based only on initiating events such as low spent fuel pool water level or large dropped objects damage fuel is unnecessary over classification that could decrease the effectiveness of the Emergency Plan. Note that low water level in the reactor vessel would be classified based on the Loss of Shutdown Functions event category and is more conservative than the # 10 NUREG-0654 example IC under

SAE (water level below the top of fuel assemblies would be classified as a GE vs SAE).

Also, the region has previously approved the Oconee method of classifying major damage to spent fuel with release of radioactivity. We will patterned the proposed revised MNS equivalent EALs to reflect their methods. This will been done for the justifications previously stated in addition to Duke Power Co's desire for EAL consistency between its three nuclear stations.

RESPONSE TO 4.1.9-ITEM # 1

[RP/0/A/5700/00-Event Category 4.1.9 (Natural Disasters and Other Hazards]-NOUE IC # 3 and attendant EALs]

NOUE IC # 3, Page D-82: These MNS EALs were revised to require "Physical Damage Observed to Equipment/Structures Within the Site Boundary." The changes decrease the effectiveness of the Plan in that they require damage to occur before declaration. The clear intent of NUREG-0654 is declaration based on occurrence of the event (i.e., sustained winds greater than 60 mph).

The NOUE IC # 3 wording remains unchanged. The NOUE IC # 3 attendant EALs will be changed as follows:

- Tornado observed touching down within the protected area fence.

Basis: Damage to plant structures, systems, or components is considered likely from this event.

- Su.:ained (greater than 15 minutes) winds greater than or equal to 60 mph.

Basis: Damage to plant structures, systems, or components is considered likely from this event. Also, the on-site meteorological instrumentation maximum indication capability is 60 mph wind speed.

RESPONSE TO 4.1.9-ITEM # 2

[RP/0/A/5700/00-Event Category 4.1.9 (Natural Disasters and Other Hazards)-NOUE Ics 5, 6, & 7, and attendant EALs]

<u>NOUE Ics # 5,6,7 Pages D-85 through D-87</u>: These MNS EALs were revised to require physical damage, evacuation of protected area personnel, and or injury. Again, the clear intent of NUREG-0654 is declaration based on occurrence of the event.

The intent of NUREG 0654 in including these example IC's is to recognize that most incidents of this type will cause damage to plant structures, systems, or components; injury to personnel, or result in plant areas becoming inaccessible. <u>Minor</u> incidents of this type which do not result in the above adverse effects do not cause a "potential degradation in the level of safety of the plant". Therefore, the MNS EALs are considered to meet the intent of NUREG-0654 and remain unchanged.

RESPONSE TO 4.1.9-ITEM # 3

[RP/0/A/5700/00-Event Category 4.1.9 (Natural Disasters and Other Hazards)-Alert IC # 2 and attendant EAL]

<u>Alert IC # 2, Page D-89</u>: The previous EALs for Tornado, Sustained winds, aircraft crash, train derailment, missile impact and explosion were combined to read "Damage to plant equipment causing the inability to maintain cold shutdown in modes 5 and 6." The previous EALs also included directly observable quantities for each EAL (i.e., 95 mph winds, tornado striking facility). The changes are considered a Plan degradation and are inconsistent with the corresponding NUREG-0654 EALs.

The Alert IC # 2 wording will be changed to:

"Damage from tornado, sustained winds, aircraft crash, missile, or explosion".

The Alert IC # 2 attendant EALs will be changed as follows:

- Any tornado striking plant structures within the protected area fence.

- Approaching hurricane with sustained (greater than 15 minutes) winds greater than 75 but less than 95 mph as reported by the National Weather Service.

- Aircraft crash within the protected area fence affecting safe operation of the unit.

- Missile impact within the protected area fence affecting safe operation of the unit.

- Explosion damage within the protected area fence affecting safe operation of the unit.

Basis: These EALs will be changed back to the previously

approved wording with minor clarification improvements.

RESPONSE TO 4.1.9-ITEM # 4

[RP/0/A/5700/00-Event Category 4.1.9 (Natural Disasters and Other Hazards)-Alert IC # 3 and attendant EAL]

Alert IC # 3, Page D-90: See comments for Item 2, Event Category 4.1.9 above.

The Alert IC # 3 wording remains unchanged. The Alert IC # 3 attendant EAL will be changed as follows:

- Uncontrolled entry of toxic or flammable gas within protected area fence affecting safe operation of the unit.

Basis: This EAL will be changed back to the previously approved wording.

RESPONSE TO 4.1.9-ITEM # 5

[RP/0/A/5700/00-Event Category 4.1.9 (Natural Disasters and Other Hazards)-SAE IC # 1 and attendant EAL]

Site Area Emergency IC # 1, Page D-91: This EAL, Earthquake of greater than SSE, (Modes1-4), was revised to require loss of subcooling margin for emergency declaration. This addition reduces the anticipatory intent of the EAL and is a degradation from the previous Plan. The mode specifically, in this area, is acceptable.

The SAE # 1 wording will be changed to:

"Earthquake greater than SSE in modes 1-4".

The SAE # 1 attendant EALs will be changed to:

- Greater than 0.15 g horizontal.

- Greater than 0.10 g vertical.

Basis: These EALs will be changed back to the previously approved wording.

RESPONSE TO 4.1.9-ITEM # 6

[RP/0/A/5700/00-Event Category 4.1.9 (Natural Disasters and Other Hazards)-SAE Ics # 2 & 3 and attendant EALs]

Site Area Emergency IC # 2 and # 3, Page D-92 and D-93: These EALs were changed from very specific, observable quantities (i.e., greater than 95 mph winds, aircraft crash affecting vital structure) to "Failure of heat sink results in the inability to maintain hot shutdown condition and NC subcooling cannot be maintained greater than 0 degrees F." These revised EALs are inconsistent with NUREG-0654 and were degraded from those previously in place.

1) The SAE IC # 2 wording will be changed to:

"Damage from tornado, sustained winds, aircraft crash, missile, or explosion".

The SAE IC # 2 attendant EALs will be changed as follows:

- Any tornado striking any plant vital area structure resulting in loss of any ESF function required for the current operating mode.

Basis: Damage from tornado is in excess of that assumed in design structure analysis.

- Approaching hurricane with sustained (greater than 15 minutes) winds greater than 95 mph as reported by the National Weather Service.

Basis: This EAL will be changed back to the previously approved wording.

- Aircraft crash causing damage or fire to the Containment Building, Control Room, Auxiliary Building, Fuel Building, or RN Intake Structure in modes 1-4.

Basis: This EAL will be changed back to the previously approved wording.

- Damage from missile or explosion in modes 1-4 causes inability to maintain or establish functions required for hot shutdown.

Basis: This EAL will be changed back to the previously approved wording with minor clarification improvements.

2) The SAE IC # 3 wording remains unchanged. The SAE IC # 3 attendant EAL will be changed to:

- Entry of uncontrolled toxic or flammable gases into the Control Room, Cable Spreading Rooms, Containment Building, Switchgear Rooms, Auxiliary Shutdown Panels area (CA pumps room), or Emergency Diesel Generator

Rooms affecting safe operation of the unit in modes 1-4.

Basis: This EAL will be changed back to the previously approved wording.

RESPONSE TO 4.1.10-ITEM # 1

[RP/0/A/5700/00-Event Category 4.1.10 (Other Abnormal Plant Conditions)-NOUE IC # 6 and attendant 'c' EAL]

NOUE IC # 6, Page D-101: The loss of meteorological capability EAL was revised to require not only a loss of onsite instrumentation, but also the inability to contact the National Weather Service (NWS). This is contrary to NUREG-0654 and is a degradation from the previous MNS Plan which did not require loss of NWS capability.

This EAL was not changed from the previously approved Emergency Plan. Therefore, its wording remains unchanged in this response.

RESPONSE TO 4.1.10-ITEM # 2

[RP/0/A/5700/00-Event Category 4.1.10 (Other Abnormal Plant Conditions)-Alert IC # 1 and attendant EALs]

Alert IC # 2 (1?), Page D-104: The revised EAL is acceptable, however, it appeared that the Stand-by Shutdown Facility should be addressed rather than the Auxiliary Shutdown Panel.

The Alert IC # 1 and attendant 'a' EAL wording remains unchanged.

The Alert IC # 1 attendant 'b' EAL will be deleted.

Basis: Control room evacuation requiring unit control from the SSF is considered to be a SAE emergency level due to the limited controls and indication available at the SSF.

RESPONSE TO 4.1.10-ITEM # 3

[RP/0/A/5700/00-Event Category 4.1.10 (Other Abnormal Plant Conditions)-SAE IC # 1

<u>Site Area Emergency IC # 1, Page D-108</u>: See comments for Item 4, Event Category 4.1.7 above.

The SAE IC # 1 wording remains unchanged. The SAE IC # 1 EALs will be changed as follows:

♦ Evacuation of Control Room.

AND

Control not established from the Auxiliary Shutdown Panel within 15 minutes.

Basis: Unit status is unknown and could rapidly degrade such that core decay heat removal is challenged.

♦ Evacuation of Control Room

AND

Control established or in the process of being established from the SSF.

Basis: Establishment of control from the SSF justifies SAE declaration due to SSF limited capabilities.

The SAE IC # 1 'b' EAL will be deleted.

Basis: This EAL is adequately covered under Event Category 4.1.5 (Loss of Shutdown Functions).