UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ON REVISED EMERGENCY ACTION LEVELS FOR

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-317 AND 50-318

1.0 INTRODUCTION

By letter dated July 20, 1993, as supplemented by letters dated March 11, 1994, and March 31, 1994, Baltimore Gas and Electric Company (the licensee) proposed changes to the Calvert Cliffs Emergency Action Levels (EALs). Specifically, the licensee provided a technical basis document that described how the proposed EALs incorporated the guidance in NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels," Revision 2, January 1992. The NRC has endorsed NUMARC/NESP-007 as an acceptable method by which licensees may develop site-specific emergency classification schemes. The licensee will incorporate the revised EALs into site procedure ERPIP 3.0, "Immediate Actions," Revision 18.

2.0 BACKGROUND

The EAL changes associated with Revision 18 to the Calvert Cliffs emergency classification procedure were reviewed against the requirements in 10 CFR 50.47(b)(4) and Appendix E to 10 CFR Part 50.

The requirements in 10 CFR 50.47(b)(4) specifies that onsite emergency plans must meet the requirements in the following standard: "A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee ...".

Appendix E, Subsection IV.C, specifies that "Emergency action levels (based not only on onsite and offsite radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, such as pressure in containment and response of the Emergency Core Cooling System) for notification of offsite agencies shall be described... The emergency classes defined shall include: (1) notification of unusual events, (2) alert, (3) site area emergency, and (4) general emergency."

In Revision 3 to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," the NRC endorsed NUMARC/NESP-007, Revision 2, (NESP-007), "Methodology for Development of Emergency Action Levels," as an acceptable method for licensees to meet the requirements of 10 CFR 50.47(b)(4) and Appendix E to 10 CFR Part 50. The NRC staff relied upon the guidance in NUMARC/NESP-007 as the basis for its review of the Calvert Cliffs EAL changes.

3.0 EVALUATION

The licensee has divided the fifty site-specific initiating conditions (ICs) into eight subcategories: (1) Radioactivity Release, (2) Fission Product Barrier Degradation, (3) Security, (4) Equipment Failure, (5) Fire, (6) Natural Phenomenon, (7) Electrical, and (8) Other Hazards. Each IC is identified by a unique letter-number designation. The first letter identifies the applicable subcategory the IC falls under while the second letter identifies the emergency classification level for the IC. The number designator identifies separate ICs with identical letter designations. A majority of the proposed EALs under these ICs conform closely to the guidance; however, several of the licensee's proposed changes depart from the example EALs in NUMARC/NESP-007. Review of the licensee's justification for these variations, as discussed below, found the variations to be acceptable.

1. EAL #2 under NESP-007 IC AU1, "Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Technical Specifications for 60 Minutes or Longer," states:

   Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates with a release duration of 60 minutes or longer in excess of two times (site-specific technical specifications).

   The licensee's technical basis for IC RU1 explains that liquid effluents are monitored by the liquid waste discharge radiation monitor. A high radiation alarm from this monitor will result in a signal to close the liquid waste discharge valves. If these valves fail to shut on the automatic signal, the operators are directed to secure the pump being used for discharge and shut the liquid waste radiation monitoring system outlet valve. It is extremely improbable that a liquid effluent discharge could exist for greater than 60 minutes following a valid monitor alarm. Thus, the licensee has not included a specific EAL for liquid effluents.

2. Under NESP-007 IC AA2, "Major Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in Uncovering of Irradiated Fuel Outside the Reactor Vessel," the following example EALs are provided:

   1. *A (site-specific set point) alarm on one or more of the following radiation monitors: (site-specific monitors)*

      Refuel Floor Area Radiation Monitor
      Fuel Handling Building Ventilation Monitor
      Fuel Bridge Area Radiation Monitor

   2. *Report of visual observation of irradiated fuel uncovered.*

   3. *Water level less than (site-specific) feet for the Reactor Refueling Cavity that will result in irradiated fuel uncovering.*

4. *Water level less than (site-specific) feet for the Spent Fuel Pool and Fuel Transfer Canal that will result in irradiated fuel uncovering.*

In addressing this NESP-007 IC, the licensee has described their site-specific thresholds based upon entry into their abnormal operating procedures (AOPs) for loss of refueling pool level and fuel handling incidents with associated valid area radiation alarms. Thus, the licensee has effectively combined the example EALs 2, 3, and 4 with EAL 1. Because the radiation monitors provide an accurate, objective indication of the severity of any uncovery of or damage to irradiated fuel, the staff finds this approach acceptable.

3. NESP-007 provides an IC for declaration of an Unusual Event when reactor coolant system (RCS) leakage exceeds a predetermined value. The example EAL in NESP-007 for this IC reads:

1. *The following conditions exist:*

   a. *Unidentified or pressure boundary leakage greater than 10 gpm.*

   OR

   b. *Identified leakage greater than 25 gpm.*

The licensee has applied this example EAL into their classification scheme as:

AOP-2A, Excessive Reactor Coolant Leakage, is implemented for RCS leakage exceeding the capacity of one charging pump AND reactor shutdown is required.

The licensee's chemical volume control system (CVCS) utilizes three positive displacement charging pumps with a capacity of 44 gpm each. Letdown flow in the CVCS can be varied from a minimum of 28 gpm to 128 gpm. Normal configuration of the CVCS has a single charging pump operating with approximately 40 gpm letdown. During excessive RCS leakage, letdown flow will initially be controlled to maintain pressurizer programmed level. When letdown flow is reduced to its minimum of 28 gpm without the ability to maintain pressurizer level within the programmed band using the single charging pump, this indicates a leak in excess of 11 gpm - taking into consideration 5 gpm for reactor coolant pump seal leakage. In accordance with procedure AOP-2A, the reactor must be shut down under these conditions. AOP-2A provides for a clear, objective threshold that the operators can readily apply in evaluating the RCS leakage IC. The difference between the licensee's 11 gpm effective threshold and NESP-007's 10 gpm threshold for unidentified leakage is insignificant. The licensee does not distinguish between unidentified and identified leakage and declaration would be made with indifference to the source(s) of the leakage.

4. In accordance with NUMARC/NESP-007, a potential loss of the fuel clad barrier should be declared when critical safety function status trees (CSFSTs) indicate a core cooling ORANGE path or heat sink RED path.

An ORANGE path is established for core cooling and the fuel clad is considered potentially lost when:

RCS subcooling is less than 50 °F

AND

No RCP [Reactor Coolant Pump] is running

AND

Core exit thermocouples are greater than 700 °F with RVLIS [Reactor Vessel Level Instrumentation System] greater than 39%
OR
Core exit thermocouples are less than 700 °F with RVLIS less than 39%

An ORANGE path also exists when:

RCS subcooling is less than 50 °F

AND

RVLIS dynamic head range less than:

80% - 4 RCPs running
70% - 3 RCPs running
60% - 2 RCPs running
50% - 1 RCP running

A heat sink RED path is established and the fuel clad is considered potentially lost when:

Narrow range level in at least one SG is NOT greater than 35%

AND

Total feedwater flow to SGs is less than 260 klbm/hr

Therefore, the CSFSTs provide indications to the operators on whether or not the fuel is being cooled and will continue to be cooled. These indications encompass steam generator inventory and feed, status of forced circulation, reactor vessel water level, reactor coolant system subcooling and core exit thermocouples.

The licensee operates two nuclear units designed by Combustion Engineering (CE). CE's emergency operating procedure (EOP) guidance does not utilize CSFSTs to monitor critical safety functions. Instead CE guidance establishes safety function status checks (SFSCs) to evaluate performance of critical safety functions. SFSCs utilize acceptance criteria that must be verified by operators to conclude a function is being satisfactorily performed. These SFSCs are similar to the CSFST decision branches. In determining whether a potential loss of the fuel clad exists, the licensee has utilized core and RCS heat removal SFSCs in EOP-8, "Functional Recovery Procedure." The acceptance criteria are categorized by success path:

CORE AND RCS HEAT REMOVAL SAFETY FUNCTION ACCEPTANCE CRITERIA

Forced Circulation/No SIS Operation:	Natural Circulation/No SIS Operation:	SG Heat Sink with SIS Operation:	Once-Through-Cooling:
At least one S/G has level between -170 and +50 inches with feedwater available to maintain level. OR S/G level is being restored by feedwater flow greater than 150 gpm.	At least one S/G has level between -170 and +50 inches with feedwater available to maintain level. OR S/G level is being restored by feedwater flow greater than 150 gpm.	At least one S/G has level between -170 and +50 inches with feedwater available to maintain level. OR S/G level is being restored by feedwater flow greater than 150 gpm.	CET temperatures are less than superheated, and are constant or lowering.
T_{HOT} minus T_{COLD} is less than 10 °F and NOT rising.	T_{HOT} minus T_{COLD} is between 10 °F and 50 °F and not rising.	All available charging pumps are operating.	All available charging pumps are operating.
T_{COLD} is less than 535 °F and not rising.	T_{COLD} is less than 535 °F and not rising.	HPSI and LPSI Pumps are injecting water into the RCS.	HPSI Pumps are injecting water into the RCS.
RCS subcooling is greater than 25 °F based on CET temperatures.	RCS subcooling is greater than 25 °F based on CET temperatures.	CET temperatures are less than superheated.	Pressurizer pressure is less than 1270 psia or is lowering.
RVLMS indicates that the core is covered.	RVLMS indicates that the core is covered.	RVLMS indicates that the core is covered.	RVLMS indicates that the core is covered.

These criteria closely relate to the RED and ORANGE paths of the CSFSTs used by Westinghouse plants such that if any one of these criteria cannot be met, the function of removing heat from the core is severely challenged or lost. Therefore, the SFSC acceptance criteria in EOP-8 for core and RCS heat removal are appropriate thresholds for defining a potential loss of the fuel clad.

5. NESP-007 establishes a coolant activity threshold for loss of the fuel clad barrier at 300 μCi/cc dose-equivalent I-131. The licensee's site-specific analyses demonstrate that this activity corresponds to approximately 1% gap activity release and is extremely close to the licensee's technical specification limit for iodine spiking. To maintain consistency with other EALs for loss of the fuel clad barrier and to ensure that the threshold is well above the technical specification limit, the licensee has chosen 600 μCi/cc dose-equivalent I-131 as their coolant activity threshold. The licensee has provided adequate justification for their site-specific value.

6. In accordance with NUMARC/NESP-007, the RCS barrier should be declared potentially lost when CSFSTs indicate a heat sink RED path. As stated in #4 above, a heat sink RED path is established when:

   Narrow range level in at least one SG is NOT greater than 35%

   AND

   Total feedwater flow to SGs is less than 260 klbm/hr

   Because the licensee does not utilize CSFSTs in evaluating safety functions, they developed a site-specific EAL for potential loss of the RCS based upon their SFSCs. As with the potential loss of the fuel clad barrier, the licensee used the acceptance criteria for core and RCS heat removal SFSCs to establish the appropriate threshold. These criteria clearly address the ability to remove heat from the RCS via the steam generators based on inventory and feed to the generators. The licensee has provided sufficient justification for their site-specific EAL.

7. In accordance with NESP-007, the RCS barrier should be declared potentially lost when there is an unisolable RCS leak exceeding the capacity of one charging pump in the normal charging mode. However, as discussed in #3 above, the normal charging mode for the licensee is a single pump injecting 44 gpm and letdown flow of approximately 40 gpm with the difference arising from leakage at the reactor coolant pump seals. Based on the operation of the licensee's CVCS, the RCS leakage needed to exceed the normal charging and letdown mode could be as little as 11 gpm. Therefore, to provide a clear escalation from the Unusual Event IC for RCS leakage, and to provide a clear and objective threshold for the potential loss of the RCS barrier, the licensee wrote their site-specific EAL as:

   RCS Leakage Exceeds Available CVCS Capacity.

   This equates to an RCS leak rate of approximately 125 gpm assuming letdown is isolated, and is readily observable based on the safety injection actuation signal (SIAS) that will be received when available CVCS capacity cannot maintain pressurizer level. The 125 gpm is also commensurate with the capacity of a single Westinghouse centrifugal charging pump operating at normal system pressure. The licensee has provided sufficient justification for the site-specific EAL.

8. In accordance with NESP-007, the containment barrier is considered potentially lost when:

Core exit thermocouples [are] in excess of 1200 °F and restoration procedures not effective within 15 minutes; or, core exit thermocouples [are] in excess of 700 °F with reactor vessel water level below the top of active fuel and restoration procedures not effective within 15 minutes.

The licensee's reactor vessel level monitoring system (RVLMS) does not measure water levels below the top of active fuel; therefore, the licensee has not applied the second argument of the NESP-007 example EAL to their site-specific classification scheme. Based upon the licensee's justification, omission of the second argument is acceptable.

In lieu of applying the first argument of the NESP-007 example EAL directly, the licensee chose to establish their site specific threshold as:

Valid Core Exit Thermocouple Readings GREATER THAN 1300 °F AND Increasing.

According to the licensee, core exit thermocouple readings above 1300 °F would clearly indicate that functional recovery of RCS heat removal was ineffective and that core conditions were continuing to degrade. According to the licensee's emergency response plan implementing procedure (ERPIP) 802, 1300 °F also corresponds to approximately 20% clad damage. This would be consistent with other EALs based on source term in the containment; therefore, the licensee's site-specific threshold is acceptable.

9. In accordance with NESP-007 IC HS2, a site area emergency should be declared when the following conditions exist:

a. Control room evacuation has been initiated.

AND

b. Control of the plant cannot be established per (site-specific) procedure (site-specific) minutes.

The guidance provides that the site-specific time to transfer control should be based on analysis or assessment as to how quickly control must be reestablished without core uncovering and/or core damage; however, the time should not exceed 15 minutes.

The licensee's site-specific EAL is written as:

Control room evacuation initiated AND either of the following:
- *Inability to establish auxiliary feedwater to AT LEAST one steam generator within 30 minutes.*
- *Inability to establish reactor coolant make-up (charging pump flow) within 60 minutes.*

The licensee has provided a site specific analysis (LER 50-317/89-009, Rev. 2) to demonstrate the ability to safely shut down the Calvert Cliffs units in accordance with AOP-9A, "Control Room Evacuation and Safe Shutdown Due to a Severe Control Room Fire." A RETRAN analysis by the licensee indicates that steam generators will go dry in approximately 47 minutes following a reactor trip (AOP-9A implemented) and no feedwater flow to the generators. Therefore, the licensee has set their threshold for reestablishing feedwater at 30 minutes. Appendix R analyses allow 90 minutes for restoring RCS inventory; therefore, the licensee has set their site-specific threshold for reestablishing RCS make-up at 60 minutes. The licensee has provided sufficient justification for the site-specific thresholds.

4.0 CONCLUSION

The proposed EAL changes for Revision 18 of ERPIP 3.0, "Immediate Actions," are consistent with the guidance in NUMARC/NESP-007, with variations as identified and accepted in this review, and therefore meet the requirements of 10 CFR 50.47(b)(4) and Appendix E to 10 CFR Part 50.

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