I. OVERVIEW / SIGNATURES

05-035

Facility: Waterford 3

Document Reviewed: *ER-W3-2004-0276-002, AST Implementation* **Change/Rev.:** *0*

System Designator(s)/Description: *None*

Description of Proposed Change

The primary purpose of ER-W3-2004-0276-002 is to provide a basis of increased rigor for the Decontamination Factor assumptions used in the FSAR Chapter 15 dose analysis being implemented under the Alternative Source Term base ER, ER-W3-2004-0276-000. As a result of CR-WF3-2005-0026, it was determined that the Wide Range SG level setpoints used for the EFW system did not result in covering the top of the Steam Generator U-tubes. ER-W3-2005-0019 subsequently revised the EFW level control setpoints to ensure that the top of the SG U-tubes could be covered during postulated design basis events. ER-W3-2004-0276-002 was initiated to ensure that the release assumptions (Decontamination Factor and timing) associated with the AST dose analyses result in releases which are greater than those which would result from a more rigorous calculation of the radioactive releases. Specifically, through Westinghouse calculation CN-TAS-03-30 rev.5, this ER documents that radioactive releases accounting for the calculated flashing fraction and time to cover the SG U-tubes, assuming a worst case single failure, are bounded by the assumed releases in the AST radiological dose calculations with their simplistic and stylized assumptions. Related to this issue, the modeling of releases from the intact Steam Generator for the Steam Generator Tube Rupture event was revised to account for the impact on releases due to flashing, with the assumed primary-to-secondary leakage reduced from 540 gal/day to a value of 150 gal/day (twice the Technical Specification limit of 75 gal/day.)

This ER also addresses several other miscellaneous issues, other than providing increased rigor for radiological release assumptions and updating the SGTR dose calculation, related to radiological analyses, including:

- *Documenting in the FSAR the NRC acceptance comments regarding LOCA ESF filter shine dose*
- *calculation of dose due to Waste Gas Decay Tank Failure*

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**EDITORIAL CHANGE of a Licensing Basis Document**	**Section I**
☐	**SCREENING**	**Sections I and II required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, and III required**
☒	**50.59 EVALUATION (#: 05-021)**	**Sections I, II, and IV required**

Preparer: Paul A. Sicard Paul A Sicard /Entergy/Nuclear Eng./~~6/4/2005~~ 6-5-05
Name (print) / Signature / Company / Department / Date

Reviewer: Christopher G. Pratt CG Pratt /Emerald Coast Services/~~6/4/2005~~ 6-5-05
Name (print) / Signature / Company / Department / Date

OSRC: R A Dodds R. Dodds 6/6/05
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM ENS-LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	DRN 05-845: FSAR Chapter 2 DRN 05-870: FSAR Chapter 10 DRN 05-645: FSAR Chapter 15
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES", perform an Exemption Review per Section III OR perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. AND initiate an LBD change in accordance with NMM ENS-LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM ENS-LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No

If "yes," perform a 50.59 Evaluation per Section IV OR obtain NRC approval prior to implementing the change AND initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. **Basis**

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.

The SGTR reanalysis impacts the FSAR in that Curie release values and assumptions contained in the FSAR are revised as part of the reanalysis. Documenting in the FSAR the basis for NRC acceptance of the LOCA ESF filter shine dose flashing fraction assumptions also obviously impact to the FSAR. The work of calculation ECS04-006 to document that the dose associated with the Loss of Offsite Power event is bounded by the Feedwater Line Break event provides support for the existing FSAR descriptions and therefore does not impact the FSAR. There is no impact on the TRM due to the changes. There is no impact on Technical Specification or the bases of Technical Specifications associated with ER-W3-2004-0276-002. The Waste Gas Decay Tank failure calculation provides support for TS 3.11.2.6. The Westinghouse calculation CN-TAS-03-30 rev.5 supports the analysis assumptions for radiological releases (i.e., the combination of Decontamination Factors and timing) currently documented in the FSAR; FSAR chapter 10 will be revised to document the role of EFW in providing for U-tube submergence. It specifically uses the current value of 575 GPM as a minimum EFW flow assumption, as documented in the FSAR and the TS Bases. The ODCM was not impacted by the base ER implementing AST, ER-W3-2004-0276-000, and is also not impacted by this supporting Linked ER.

4. References

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
FSAR, Technical Specifications, TRM, FSAR Questions, SER (NUREG-0787)	Flashing Fraction SGTR Steam Generator Tube Rupture submergence Emergency Feedwater Waste Gas Decay 0409 (as in NUREG-0409) filter shine shine dose U-tube decontamination factor partition factor partition coefficient cover NEAR3 tube submerge NEAR3 tube

LBDs/Documents reviewed manually:

- FSAR Chapters 2, 9, 10, and 15
- DRN 04-704 (FSAR Chapter 15, ER-W3-2004-0276-000)

5. Is the validity of this Review dependent on any other change?

☐ Yes

☒ No

If "YES", list the required changes/submittals. The changes covered by this 50.59 Review cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

B. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure ENS-EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in a new or additional air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure ENS-EV-117, "Air Emissions Management Program," for guidance in answering this question.

C. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan Review must be perfomed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect (block, move, or alter) security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

(NOTE: This section is not applicable to Waterford 3 and may be removed from 50.59 Reviews performed for Waterford 3 proposed activities.)

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure ENS-LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A.1.

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section III.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # _____ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: _____

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions.

This Section is not required per Section I.

IV. 50.59 EVALUATION

License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?

   ☐ Yes
   ☒ No

   BASIS:

   ER-W3-2004-0276-002 involves documenting the results of the reanalysis of the SGTR event and adding clarifying information related to the function of the EFW system to provide for U-tube submergence following postulated events in order to mitigate radiological releases. There is nothing associated with this change which would impact system performance or reliability, affect any system interface in a way that could lead to an accident, or increase the possibility of operator error to cause any event previously evaluated in the FSAR. Thus, there is no increase in the frequency of occurrence of an event previously evaluated in the FSAR associated with this change.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

   ☐ Yes
   ☒ No

   BASIS:

   ER-W3-2004-0276-002 involves documenting the results of radiological analyses and adding clarifying information related to the function of the EFW system to provide for U-tube submergence following postulated events in order to mitigate radiological releases. There is no EQ impact, which could increase the likelihood of equipment malfunction, associated with this change since the limiting event for EQ is the Large Break LOCA, the results of which are unaffected by this ER. Note U-tube submergence plays no role in the Large Break LOCA event, since releases for this event are based on containment leakage. There is no physical impact on any equipment associated with these changes. LOCA ESF filter shine flashing fraction assumptions only impact the calculated consequences of an event and have no impact on likelihood of accident initiation

   Thus, this change does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

   ☐ Yes
   ☒ No

BASIS:

The FSAR (through DRN 04-0704, Table 15.6-28) reports a Main control room dose of ≤5 Rem TEDE for Steam Generator Tube Rupture. The SGTR has been reanalyzed with different assumptions for the intact SG. Whereas a DF of 100 was previously used for the intact SG for the duration of the event, the revised analysis assumes a DF of 12.5 for the intact SG out to 2.0 hours into the event, and a DF of 100 thereafter. To compensate for this increased release, the calculation assumed a 150 gal/day primary-to-secondary leakage for the intact SG, whereas previously a value of 540 gal/day was assumed The 150 gal/day is an acceptable value to assume for an intact SG, as it exceeds the 75 gal/day operational leakage limit of Technical Specifications. The resulting SGTR doses continue to meet the results reported in the FSAR of ≤25 Rem TEDE offsite doses for the Pre-Existing Iodine Spike (PIS) cases, of ≤2.5 Rem TEDE for the Accident Generated Iodine Spike (GIS) cases, and of ≤5.0 Rem TEDE for Main Control Room dose for both PIS and GIS cases..

Note only about 1% of the releases for an SGTR are associated with the intact SG, with about 99% of the release from the affected SG. Thus, it was expected that the change in calculated dose is small. The SGTR dose calculation also conservatively accounts for a reduced Main Control Room volume, as a contingency in the event ER-W3-2004-0415 is not implemented.

There is no increase in consequences associated with the Westinghouse evaluation of radiological releases, since that evaluation was performed to support continued use of the existing assumptions in the radiological calculations for control room and offsite dose.

The NRC approval basis for Large Break LOCA filter shine dose calculations does not impact the consequences of the event, since the NRC explicitly approved the results of the analysis even while not approving the flashing fraction assumptions used in the calculation. Waterford-3 performed this analysis assuming a 10% flashing fraction for 0-24 hours and, based on sump temperature, conservatively assumed a bounding maximum 2% flashing fraction beyond 24 hours. The NRC recommended that the flashing fraction be increased to 10% or that the lower value be justified when the large break LOCA filter shine dose analyses are revised in the future. Waterford-3 internal memo W3C1-2004-0025 dated April 4, 2005, documented justification for the 2% flashing fraction assumption. However, since the NRC approved the results of the analysis (which comprised part of the Control Room dose for Large Break LOCA), the documentation /clarification of the basis of the NRC approval does not result in any increase in the consequence of this event.

The Waste Gas Decay Tank failure calculation continues to support the Technical Specification Basis information that release of the allowed activity results in an offsite dose of less than 0.5 Rem.

This review concludes that the proposed changes do not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The Steam Generator Tube Rupture analysis which is part of this ER involves changes to release assumptions for the intact Steam Generator. The revised analysis allows for an increased release (lower Decontamination Factor) for the intact SG for the first two hours of the event. Westinghouse calculations have demonstrated that these assumptions are conservative. The revised analysis also credits a primary-to-secondary steam generator tube leakage of 150 gal/day for the intact SG. While this value is reduced from the previous assumption of 540 gal/day, this value is twice the value specified in Waterford-3 Technical Specifications of 75 gal/day. (Only the Small Break LOCA dose calculation credits the TS value; all other calculations assume a 150 gal/day value for operational leakage from unfaulted or intact Steam Generators, e.g., steam generators which have not been subjected to significant primary-to-secondary differential pressure transients.)

Westinghouse analyses performed as part of this ER support the conclusion that the radiological releases assumed in the control room and offsite dose calculations are appropriate and conservative. The Westinghouse calculations determined time to U-tube submergence and, using RG 1.183 methodology, radiological releases based upon flashing fraction to show that the calculated releases from this more rigorous approach would be smaller than the simplistic/stylized radiological releases developed in previous Westinghouse calculations and which are used as inputs to the Waterford-3 offsite and control room dose calculations. (Note RG 1.195, for the traditional TID-14844 source terms, provides the same guidance as

RG 1.183 for modeling radiological releases. Thus, this is not an AST issue.)

These Westinghouse analyses credit operator actions, if required, to ensure sufficient EFW flow exists to ensure that the top of the U-tube get covered by SG water level during the event. As discussed in FSAR Section 7.3, the automatic operation of the EFW control system would regulate the flow to the Steam Generators. In the flow control mode, the EFW flow demand would be reduced to 400 GPM as SG level recovers to the "Lo Level" setpoint. Westinghouse analyses were conducted that demonstrated that this flow would be insufficient to raise the SG levels to above the top of the U-tubes without credit operator action to provide additional EFW flow. The Westinghouse analyses show that the EFW control system would otherwise oscillate the flow ; due to the decay heat load, such flows would be insufficient to achieve U-tube submergence within the typical duration of the non-LOCA transient events (generally about 8 hours).

With an SIAS present, the modulating flow control valves are switched immediately to the level control mode of operation and a maximum EFW flow (of greater than 575 GPM) would be provided to the Steam Generators; thus, no operator action needs to be assumed for scenarios where SIAS is present.

Greater reliance is placed on operator action to take manual control of EFW to raise SG water level and submerge the U-tubes. Because both ADV's are assumed available and used for steaming for certain events (e.g., CEA Ejection), the SG U-tubes of both SG's need to be submerged. The Westinghouse analyses show that, for these events, operator actions to ensure the top of the U-tubes are covered by about 7200 seconds would result in an acceptable radiological release, that is would be bounded by the assumptions in the radiological release calculation. The Westinghouse calculations also demonstrate that this timing is achievable; for example, the top of the U-tubes are covered at approximately 6100 seconds if it is assumed that the operators act at 30 minutes to provide flow from 2 of the 3 EFW pumps to the Steam Generators. (assuming 2 of 3 EFW pumps accounts for the worst case single active failure for U-tube submergence and corresponds to an assumed flow of 575 GPM). For Feedwater Line Break type events, the radiological analyses assume that the U-tubes are submerged at 4 hours into the event; this is easily achievable with operator action, as the Westinghouse analyses demonstrate that U-tube submergence can be achieved at slightly less than 2 hours based on operator action to provide >575 GPM to the intact SG at 30 minutes into the event. For an Inside Containment Main Steam Line Break event, Westinghouse analyses demonstrate that the assumed Curie releases are conservative by more than a factor of 10, based on assuming operator action to take manual control of EFW. With operator action at 30 minutes to establish ≥575 GPM EFW flow, the top of the U-tubes would be submerged by 3500 seconds, which is significantly less than the time that would result in an acceptable radiological release. Westinghouse did not explicitly calculate the time by which the tubes must be covered to support the analysis assumptions, but it would be easily greater than the approximately 7200 seconds for CEA Ejection type events. Thus, acceptable radiological consequences are achieved with assumptions of the Westinghouse analyses.

These assumptions are considered consistent with the existing Emergency Operating Procedures; Operations has concluded that no EOP changes are required. EOP's already contain instructions to verify that steam generator level is being maintained or restored to within the operating bands, which provide for covering the top of the U-tubes. Operations has reviewed the ER information and concluded that the operator actions described in the ER are consistent with the expectations of Operator performance during the implementation of EOP's. Operator actions would be required to return EOP process values (e.g., steam generator level) to within the proper band. It is considered reasonable to assume that operator actions would result in restoring the SG level to above the top of the U-tubes within the 2 hour time that supports the radiological release analyses. Thus, Emergency Operating Procedures are considered to provide sufficient guidance for the important operator action to ensure SG U-tube submergence, and thus that there will not be any increase in the consequences associated with the malfunction of a structure, system, or component important to safety.

[As stated in ER-W3-2004-0276-002, the EOP's should not be changed to specify any specific time criteria for operator actions. ER-W3-1997-0335 was written to address the appropriateness of FSAR analysis action times in the Emergency Operating Procedures (EOP's). In general, it is not appropriate to apply the FSAR Chapter 15 assumed action times to the EOP's. The FSAR Chapter 15 events use conservative assumptions and modeling techniques with specific single failures to conservatively increase the consequences of each analyzed design basis accident. As a result, these analyses do not necessarily represent the actual plant response to an accident. The EOP philosophy is a best estimate approach to determine actual operator actions and plant response based upon appropriate safety functions.]

[NRC also recognizes in RG 1.183 the conservative nature of accident analyses and that the scenarios in

the accident analyses are not intended to represent the actual progression of events. Quoting RG 1.183 Section B: "The design basis accidents (DBA's) were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features. These accident analyses are intentionally conservative in order to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion. Although probabilistic risk assessments (PRA's) can provide useful insights into system performance and suggest changes in how the desired depth is achieved, defense in depth continues to be an effective way to account for uncertainties in equipment and human performance."

The greater reliance on the EFW system is recognized by documenting in FSAR Chapter 10 the role of EFW in providing SG U-tube submergence. Note no previous credit for automatic action of the EFW system to cover the top of the U-tubes has been described in the FSAR.

Based on this review it is concluded that this changes does not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

Because there are no new system interactions or relationships created by ER-W3-2004-0276-002, there is no possibility of creating an accident of a different type than previously evaluated. There are no physical changes to the plant or changes to plant procedures associated with this change. No new accident initiators are introduced by this ER. Thus, since this ER does not impact any equipment, it does not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

There are no physical changes to the plant or changes to plant procedures associated with this change. This ER is concerned with the calculation inputs and methodology used to assess accident consequences and with analyses performed to demonstrate that the simplistic/stylized release assumptions (combination of Decontamination Factor and timing) used to evaluate radiological consequences are appropriate and conservative. Thus, this change which basically implements and documents analysis results does not have the possibility of creating any malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

This ER is concerned with radiological analyses, which determine the radiological consequences (addressed in Questions 3 and 4) associated with FSAR Chapter 15 events, including those that involve postulated fuel failure. Thus, there is no impact on any fission product barriers which are associated with calculations for offsite and control room dose, but rather these calculations determine the consequences which can result as a result of fuel failure or other failures of fission product barriers. For example, for Steam Generator tube rupture, the rupture itself involves the failure of a Steam Generator U-tube, which is part of the Reactor Coolant System boundary and is thus a part of a fission product barrier. However, there is no impact on the design basis limits for the RCS (e.g., primary pressure or stresses) associated with how the release is determined and there is no impact on the design basis limits associated with the efficiency of the removal mechanisms (e.g., scrubbing efficiency or Decontamination Factors) for radiological releases. Thus, ER-W3-2004-0276-002 does not result in exceeding or altering any design basis limit for fission product barriers.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

The radiological analyses associated with ER-W3-2004-0276-002 continue to follow Regulatory Guide

1.183 for the methodology for Alternative Source Term radiological analyses. Regulatory Guide 1.183 provides guidance for implementing the AST dose methodology into the plant design and licensing basis. The Steam Generator Tube Rupture analysis revised the inputs for the Decontamination Factor as a function of time for the intact (unaffected) steam generator based upon more detailed Westinghouse analyses. The Westinghouse analyses of CN-TAS-03-30 rev.5 provide support of increased rigor that the release assumptions (i.e., the combination of Decontamination Factor and timing) for the various FSAR Chapter 15 events remain appropriate and bounding. Specifically, the Westinghouse analyses determined the Curie releases that correspond to the time, given a worst case Single Active Failure, to cover the top of the Steam Generator U-tubes; it was assumed that the fraction of iodine (and other particulates) that becomes airborne is assumed equal to the flashing fraction of the primary coolant leaking to the SG secondary side. This is consistent with the guidance of RG 1.183. RG 1.183 allows credit for fission product scrubbing in the SG (i.e., a Decontamination Factor of 100) if the top of the U-tubes are submerged, per the models of NUREG-0409. The more rigorous release calculations performed by Westinghouse are consistent with this methodology and show that these releases would be conservatively less than the simplified/stylized releases assumed for the original Westinghouse Curie release calculations and used in the calculations for offsite and control room dose. The Westinghouse calculations also demonstrate the ability to submerge the top of the U-tubes by 4 hours into a Feedwater Line Break event; this is the only event for which the AST License Amendment Request and the subsequent FSAR changes explicitly states that the U-tubes will be covered by a specified time. For all other events, the initial lower Decontamination Factor is applied on the basis of accounting for potential elevated releases due to the initial SG level transient.

The SGTR reanalysis changes assumptions for the modeling of the intact SG. Specifically, the change in assumed primary-to-secondary leakage is a change in an input parameter and is not a method change. The change in assumed DF is a conservative change that supports a more rigorous (and conservative) accounting for releases from the intact SG (which is only a minor contributor to the SGTR event). Since this remains consistent with RG 1.183, there is no departure from the method of evaluation described in the FSAR.

Documenting the NRC acceptance basis for the flashing fraction assumptions used in Large Break LOCA control room filter shine dose calculations does not cause a departure from a method of evaluation described in the FSAR. Documenting this information in the FSAR provides required clarifying information concerning the complex approval basis for the control room filter shine dose calculations; addition of this information to the FSAR can be considered as previously approved by the NRC since this information is extracted from the NRC Safety Evaluation Report for Amendment 198 (Alternative Source Term).

The Waste Gas Decay Tank release calculation is a simple calculation that also is consistent with RG 1.183 and FSAR Appendix 15.B thus involves no departure from approved methods of evaluation. FSAR Chapter 6 is revised to fix a typo to provide the correct reference for the evaluation of velocities in EQ evaluations.

ER-W3-2004-0276-02 does not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "YES", obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure ENS-LI-113.