Attachment I

Oyster Creek Nuclear Generating Station

Safety Evaluation 000661 - 017

GPU Nuclear

Technical Functions
Safety/Environmental Determination and 50.59 Review
(EP-016)

DESIGNATED ORIGINAL IN RED

Unit Oyster Creek	Page 1 of ~~7~~ 8
Document/Activity Title Removal of Iso Condenser Vent Rad. & Containm. Spray / ESW Water Heat Exchanger Monitoring Systems	SE Rev. No. ~~0~~ 1 2
Document No. *(if applicable)* Doc. Rev. No.	SE No. 000661-017

Type of Activity *(modification, procedure, test, experiment, or document)*: Modification

1. Does this document involve any potential non-nuclear environmental concern? ☐ Yes ☒ No

   To answer this question, review the Environmental Determination (ED) form. Any **YES** answer on the ED form requires an Environmental Impact Assessment by Environmental Controls, per 1000-ADM-4500.03. If in doubt, consult Environmental Controls or Environmental Licensing for assistance. If all answers are NO, further environmental review is not required. In any event, continue with Question 2, below.

2. Is this activity/document listed Section I or II of the matrices in Corporate Procedure 1000-ADM-1291.01? ☒ Yes ☐ No

   If the answer to question 1 is **NO**, stop here. This procedure is not applicable and no documentation is required. (If this activity/document is listed in Section IV of 1000-ADM-1291 review on a case-by-case basis to determine applicability.) If the answer is **YES**, proceed to question 3.

3. Is this a new activity/document or a substantive revision to an activity/document? (See Exhibit 2, paragraph 3, this procedure for examples of non-substantive changes.) ☒ Yes ☐ No

   If the answer to question 3 is **NO**, stop here and complete the approval section below. This procedure is not applicable and no documentation is required. If the answer is **YES**, proceed to answer all remaining questions. These answers become the Safety/Environmental Determination and 50.59 Review.

4. Does this activity/document have the potential to adversely affect nuclear safety or safe plant operations? ☒ Yes ☐ No

5. Does this activity/document require revision of the system/component description in the FSAR or otherwise require revision of the Technical Sepcifications or any other part of the SAR? ☒ Yes ☐ No

6. Does the activity/document require revision of any procedural or operating description in the FSAR or otherwise require revision of the Technical Specifications or any other part of the SAR? ☐ Yes ☒ No

7. Are tests or experiments conducted which are not described in the FSAR, the Technical Specifications or any part of the SAR? ☐ Yes ☒ No

   IF ANY OF THE ANSWERS TO QUESTIONS 4, 5, 6, OR 7 ARE YES, PREPARE A WRITTEN SAFETY EVALUATION FORM.

   If the answers to 4, 5, 6, and 7 are NO, this precludes the occurrence of an Unreviewed Safety Question or Technical Specifications change. Provide a written statement in the space provided below (use back of sheet if necessary) to support the determination, and list the documents you checked.

   NO, because:

   Documents checked: OCFSAR 11.5.2.3, Memos 5350-91-180 & 5350-91-022

8. Are the design criteria as outlined in TMI-1 SDD-T1-000 Div. I or OC-SDD-000 Div. I Plant Level Criteria affected by, or do they affect the activity/document? ☐ Yes ☒ No

   If **YES**, indicate how resolved:

APPROVALS *(print name and sign)*

Engineer/Originator	*[signature]*	Suzanne Buechel	Date 8/9/95
Section Manager	*[signature]*	SM KOWKABANY	Date 8/16/95
Responsible Technical Reviewer	*[signature]*	[illegible] Stevens	Date 8/15/95
Other Reviewer(s)			Date

GPU Nuclear

Technical Functions Safety Evaluation (EP-016)

UNIT Oyster Creek

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ACTIVITY/DOCUMENT TITLE Removal of Iso Condenser Vent Rad & Containment Spray/ES Water Heat Exchanger Monitoring System

SE No. 000661-017

Rev. No. ~~1~~ 2

DOCUMENT NO. (If applicable) – Rev. No.

Type of Activity/Document Modification
(Modification, procedure, test, experiment, or document)

This Safety Evaluation provides the basis for determining whether this activity/document involves an Unreviewed Safety Question or impacts on nuclear safety.

Answer the following questions and provide reason(s) for each answer per Exhibit 7. A simple statement of conclusion in itself is not sufficient. The scope and depth of each reason should be commensurate with the safety significance and complexity of the proposed change.

1. Will implementation of the activity/document adversely affect nuclear safety or safe plant operations? ☐Yes ☒No

   The following questions comprise the 50.59 considerations and evaluation to determine if an Unreviewed Safety Question exists:

2. Is the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report increased? ☐Yes ☒No

3. Is the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report created? ☐Yes ☒No

4. Is the margin of safety as defined in the basis for any Technical Specification reduced? ☐Yes ☒No

   If any answer above is "yes" an impact on nuclear safety or an Unreviewed Safety Question exists. If an adverse impact on nuclear safety exists revise or redesign. If an unreviewed safety question with no adverse impact on nuclear safety exists forward to Licensing with any additional documentation to support a request for NRC approval prior to implementing approval.

5. Specify whether or not any of the following are required, and if "yes" indicate how it was resolved

	Yes	TFAAI/PFU/Other	No
a. Does the activity/document require an update of the FSAR?	X	10-005, 10-124, 10-125	

SECTIONS 11.5.2.3, 6.2.2.2, 6.3.1.1.2, 6.3.1.1.3, 1.9.31

Explain: ~~Chapter 11.5.2.3 needs to be eliminated and Table 11.5-1 needs to be revised, [illegible] delete sentence / paragraph~~ 1
AND TABLES 11.5-1, 6.2-10 REQUIRE REVISION

b. Does the activity/document require a Technical Specification Amendment?

Explain:

1

2

2

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		Yes	TR/TFWR/Other	No
c.	Does the activity/document require a Quality Classification List (QCL) Amendment?			X

Explain: ______________________________

d. Other: (If none, use NA) N/A

This form with the reasons for the answers, together with all applicable continuation sheets constitutes a written Safety Evaluation.

List of Effective Pages

Page No.	Rev. No.	Page No.	Rev. No.	Page No.	Rev. No.
1	~~0~~ 2	5350-91-180	~~0~~ 2		
2	~~1~~ 2	5350-91-022	~~0~~ 2		
3	~~1~~ 2	5310-96-011	~~1~~ 2		
4	~~1~~ 2	6632-96-001	~~1~~ 2		
5	~~0~~ 2	5310-96-036	~~1~~ 2		
6	~~1~~ 2	LETTER TO NRC	DATED	APRIL 30, 1981	2
7	~~1~~ 2				
~~6~~	~~1~~	PRG 97-06			2
~~8~~	~~0~~				
8	2				

2

Approvals (Print Name and Sign)		Date
Engineer/Originator	John Stevens [signature]	2-23-96
Section Manager	R. PANICKER / Rom Panicker	2-23-96
Responsible Technical Reviewer	[signature] (John Y ...)	2/23/96
Independent Safety Reviewer	John Rogers [signature]	3-12-96 *
Other Reviewer(s)		

* resigned on 3/12/96 as designated original. True original lost. JJR

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1.0 PURPOSE

The purpose of this safety evaluation is to evaluate the removal of the following two systems, Isolation Condenser Vent Radiation Monitoring System and Containment Spray/Emergency Service Water Heat Exchanger Radiation Monitoring System, from the plant configuration as these monitoring systems presents the potential for operator confusion. 2

The scope of the modification for the Isolation Condenser Vent Radiation Monitoring System involves disconnection and physical removal of log ratemeters RN0007A3, 4, 5 and 6 from Control Room Panel 2R and replacement with blank panels, disconnection of four (4) input signals to Recorder RN006B on Control Room Panel 10F, disconnection of signal circuitry to Alarm Window 10F-1-e (A Vent Hi) and to Alarm Window 10F-2-e (B vent Hi) on Panel 10F and replacement with two (2) blank windows, and physical removal of all associated abandoned wiring within the Control Room. Detectors RE-RN0004A3, 4, 5 and 6 and their associated shielding and cabling are to be physically removed from Reactor Building Elevation 95'3". 2

The scope of the modification for the Containment Spray/Emergency Service Water Heat Exchanger Monitoring System involves disconnection and physical removal of log ratemeters RN0040A1, 2, 3 and 4 from Control Room Panel 2R and replacement with blanks, disconnection of four (4) input signals to Recorder RN006B on Control Panel 10F, disconnection of input signals to Alarm Window 10F-4-g (Area/Vent/Effl. Dnscl.) on Panel 10F, disconnection of signal circuitry to Alarm Window 10F-1-g (ESW A/B Hi) and to Alarm Window 10F-2-g (ESW C/D Hi) on Panel 10F and replacement with two (2) blank windows, and physical removal of all associated abandoned wiring within the Control Room. Detectors RN0038A1, 2, 3 and 4 and their associated shielding are to be abandoned in place at Reactor Building Elevation 23'6".

2.0 SYSTEMS AFFECTED

2.1 The plant systems affected by this modification consist of the following:

2.1.1 Radiation Monitoring System (661)

2.1.2 Main Control Room Panels (611)

2.1.3 Plant Annunciator System (616)

2.1.4 Emergency Condenser System (Isolation Condenser) (211)

2.1.5 Containment Spray System /Emergency Service Water System (241/532) 2

2.2 The systems affected by this modification are shown on the following baseline drawings:

2.2.1 Process Radiation Monitor System, GE Dwg No. 846D686, Sh. 1, Rev. 18.

2.3 The affected systems are described in Section 6.2, 6.3 and 11.5.2.3 of the Oyster Creek FSAR. 2

3.0 EFFECTS ON SAFETY

3.1 Will implementation of the activity/document adversely affect nuclear safety or safe plant operations?

No. The existence of the Isolation Condenser Vent Radiation Monitoring System and the Containment Spray/Emergency Service Water Heat Exchanger Radiation Monitoring System 2 represents a potential source of operator confusion since background radiation in the vicinity of the monitors masks their capability for leak detection. Existing surveillance procedures minimize the potential for loss of integrity during normal operating and emergency conditions. Although original plant design took credit for vent radiation monitors for leak detection capability, alternate means are available to detect leakage in a timely manner.

The isolation condenser vent radiation monitors were intended to provide a leak detection function and not an effluent quantification function (Ref. memo 5350-91-180, dated 8/21/91). Tube wall integrity is verified on an ongoing basis by surveillance requirements, including tracking of temperature and level every four hours. With respect to leak detection capability, the Technical Specifications require isolation condenser integrity, but do not specify the use of radiation monitors for verification.

In order to increase the availability of the Isolation Condensers (ICs) as heat sinks, NUREG 0737 Item II.K.3.14 required that their isolation on high radiation be switched from the Main Steam Line Radiation Monitors to the IC Vent Radiation Monitors. In response, GPUN, by a letter from Ivan R. Finfrock to the NRC, dated April 30, 1981 (copy attached), clarified the 2 fact that Oyster Creek did not isolate the ICs on high radiation. Isolation is provided by detection of excessive flow in the steam line to and condensate line from the IC. In its response letter, dated December 18, 1981, the NRC stated that the GPUN response to Item II.K.3.14 was acceptable and the item was considered resolved.

The Containment Spray/Emergency Service Water Heat Exchanger Radiation Monitoring 2 System was part of the original plant operation. Similarly, the detectors do not adequately perform their design function because background radiation in the vicinity of the monitors masks their effectiveness for leak detection. The Containment Spray/ESW heat exchanger does not function during normal plant operation, when the tube side is fed with service water. The system is fed with Emergency Service Water during monthly surveillance testing and following an accident.

The tubes are not subject to significant temperature or pressure excursions during post-LOCA conditions. The tube side is normally maintained at a positive differential pressure with respect to the shell side (exception identified in LER 84-26). Should the pressure differential drop below 5 psid (indication of a possible leak) the Control Room is alarmed. On a monthly basis, surveillance testing is performed which includes verification of tube and shell side pressures and differential pressure indication.

Leakage from the torus into the ESW through the heat exchangers during non accident conditions has negligible off site dose consequences.

Per Rad engineering the contamination concentration of the torus water will not be permitted to increase significantly from current average concentrations during operation because higher levels of contamination present operational problems in terms of contamination of plant areas and personnel dose. Given this upperbound to the contamination of the torus water Rad Engineering calculated, Memo 6632-96-001, (See references) that a leak rate of 10,000 gpm through the ESW heat exchangers would be necessary to exceed 10CFR50 Appendix I limits. The maximum annual dose to an individual in an unrestricted area per 10CFR50 Appendix I is specified as 5 mR. 10CFR20.1301 has a less restrictive annual dose limit to an individual in an unrestricted area of 100 mR. During normal operation the torus water may be at a one psi greater pressure than the ESW water. Memo 5310-96-011 (See references) concluded that a very conservative torus to ESW leak rate is 1 gpm. Given the Rad Engineering calculation that a 1 gpm leak rate has the potential to produce an off site dose of 1/10000 of Appendix I limits a 1 gpm leak rate has negligible off site consequences. | 2

Catastrophic failure of a heat exchanger tube is not a credible event because, per drawing PX-D9361-S-100, the heat exchanger is tubed with titanium, rated for operation at 250 psi and subjected to only a 1 psi differential pressure (Ref. Memo 5310-96-036, dated 2/16/96). Nevertheless an order of magnitude estimate was made of the consequences of a catastrophic tube failure. The 18 foot 3/4 inch tube is assumed to experience a guillotine break 3 feet from one end. The ends of the tube deflect so that both ends discharge freely. Flow is estimated using Crane Technical Paper 410 Appendix B page 14, Flow of Water Through Schedule 40 Steel Pipe. The use of a pipe table to estimate flow from tubing is conservative because 3/4 inch Sch 40 pipe has a larger inner diameter than 3/4 inch tubing. Given a 1 psi differential pressure the total leak rate can be estimated to be less than 30 gpm. A 30 gpm leak rate has the potential to produces an off site dose of 3/1000 of Appendix I limits. This has negligible off site consequences. From these cases it may be concluded that the leakage of torus water into the ESW through the heat exchanger tubes including the case of catastrophic tube failure has negligible off site dose consequences.

The system is subject to leak testing during refueling outages, with an acceptance criterion of zero leakage. The heat exchanger tubes are of titanium which is resistant to corrosion.

Shell side (torus water) chemistry analysis is performed on a monthly basis (Ref. Memo 5350-91-022, dated 4/9/91).

3.2 Is the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report increased?

No. Removal of these radiation monitors has no impact on the operation of any plant system and as such has no impact on the probability or consequences of any previously evaluated event.

The radiation monitors are not relied upon to perform a post-accident function.

3.3 Is the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report created?

No. These radiation monitors do not affect operation of any plant systems, and as such their removal will not create the possibility for an accident or malfunction of a different type.

3.4 Is the margin of safety as defined in the basis for any Tech Spec reduced?

No. The containment spray/emergency service water radiation monitors are not discussed in the Technical Specifications. The isolation condenser vent radiation monitors are discussed in the basis for Technical Specifications Section 3.8. as ". . . provided to alert the operator of a tube leak in the isolation condenser . . .". As described in Section 3.1 above, the background radiation in the vicinity of these monitors, while the isolation condensers are inservice, masks their capability for leak detection. However, Oyster Creek procedures would lead to manual isolation in the event of a tube leak, using a combination of on-site monitoring for radioactivity and control room indications and alarms, such as steam line temperatures and shell levels and temperature. 2

Based on the above, it is concluded that the removal of these radiation monitors has no impact on any margin of safety as defined in Technical Specifications.

The basis for Technical Specification Section 3.8 will be revised to delete the description of isolation condenser vent radiation monitors.

3.5 This activity will require revision of Design Basis Document for 661 System and the following FSAR Sections and Tables:

+ Sections 11.5.2.3, 6.2.2.2, 6.3.1.1.2, 6.3.1.1.3, 1.9.31
+ Tables 11.5-1, 6.2-10 2

4.0 CONCLUSION

From the justifications provided above, it is concluded that the proposed changes will not have any adverse effect on plant safety, and do not represent an unreviewed Safety Question. The changes will however require a change to the Oyster Creek FSAR as described in Section 3.5. 2

GPU Nuclear

GPU Nuclear
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Parsippany, New Jersey 07054
201 263-6500
TELEX 136-482
Writer's Direct Dial Number:

April 30, 1981

Director,
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
NUREG 0737, Item II.K.3.14

NUREG 0737, Item II.K.3.14 specifies a design change to the isolation condenser system isolation logic. The description as provided in the NUREG is incorrect for the Oyster Creek station. The isolation condenser system utilized at the Oyster Creek Nuclear facility does not isolate upon a high-radiation signal in the main steam line leading to the isolation condensers. The Oyster system utilizes excessive flow in the steam lines to and condensate lines from the isolation condensers as the only isolation signal for the system isolation valves.

Because the Oyster Creek system is different from the one described in the NUREG, we feel no additional modifications are required to meet the objective of this NUREG item.

If you should have any questions concerning this matter, please call Mr. James Knubel (201-455-8753).

Very truly yours,

Ivan R. Finfrock

Ivan R. Finfrock, Jr.
Vice President

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