



**Michael P. Gallagher**  
Exelon Nuclear  
Vice President  
License Renewal and Decommissioning

200 Exelon Way  
Kennett Square, PA 19348

610 765 5958 Office  
610 765 5658 Fax  
www.exeloncorp.com

michaelp.gallagher@exeloncorp.com

10 CFR 50.90

RA-16-093

November 2, 2016

U.S. Nuclear Regulatory Commission  
ATTN: Document Control Desk  
Washington, DC 20555

Oyster Creek Nuclear Generating Station  
Renewed Facility Operating License No. DPR-16  
NRC Docket Nos. 50-219 and 72-15

**Subject:** Response to Request for Additional Information and Supplemental  
Information Regarding Proposed Changes to Technical Specifications  
Section 6.0 Administrative Controls for Permanently Defueled Condition

- References:**
- 1) Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "License Amendment Request – Proposed Changes to Technical Specifications Section 6.0 Administrative Controls for Permanently Defueled Condition," dated May 17, 2016 (ML16138A129)
  - 2) Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC) – "Oyster Creek Nuclear Generating Station - Request For Additional Information Regarding License Amendment Request For Changes To Technical Specifications Section 6.0, "Administrative Controls" For Permanently Defueled Condition (CAC NO. MF8108)," dated October 7, 2016 (ML16244A311)

By letter dated May 17, 2016 (Reference 1), Exelon Generation Company, LLC (Exelon) submitted a License Amendment Request (LAR) for Oyster Creek Nuclear Generating Station (OCNGS). The LAR proposed changes to the organization, staffing, and training requirements contained in Section 6.0, Administrative Controls, of the OCNGS Technical Specifications (TS) to support the transition of the OCNGS facility to a permanently defueled condition.

In Reference 2, U.S. Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (RAI) indicating that additional clarification is needed to enable the NRC to make an independent assessment regarding its technical review of the proposed changes. The NRC is requesting that Exelon respond to the RAI within 30 days of the date of the Reference 2 letter.

Accordingly, Attachment 1 of this letter contains Exelon's response to the RAI and also includes supplemental information in support of this amendment request. Attachment 2 includes the revised Technical Specifications page mark-ups and Attachment 3 contains clean/camera-ready TS pages. The TS pages being submitted in this letter supersede in entirety those submitted in Reference 1.

Exelon has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the NRC in Reference 1. The additional information provided in this response does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. Furthermore, the additional information provided in this response does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments contained in this submittal.

If you have any questions concerning this submittal, please contact Paul Bonnett at (610) 765-5264.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 2<sup>nd</sup> day of November 2016.

Respectfully,



Michael P. Gallagher  
Vice President, License Renewal & Decommissioning  
Exelon Generation Company, LLC

- Attachments: 1) Response to Request for Additional Information and Supplemental Information Regarding Proposed Changes to Technical Specifications Section 6.0 Administrative Controls for Permanently Defueled Condition  
2) Revised Technical Specifications Page Mark-ups  
3) Clean/camera-ready Technical Specifications Pages

cc: w/ Attachments  
Regional Administrator - NRC Region I  
NRC Senior Resident Inspector - Oyster Creek Nuclear Generating Station  
NRC Project Manager - Oyster Creek Nuclear Generating Station  
Director, Bureau of Nuclear Engineering, New Jersey Department of Environmental Protection  
Mayor of Lacey Township, Forked River, New Jersey

U.S. Nuclear Regulatory Commission  
Response to Request for Additional Information  
and Supplemental Information to OCNGS License Amendment Request  
Docket Nos. 50-219 and 72-15  
November 2, 2016  
Page 3

bcc: w/o Attachments  
Sr. Vice President and COO CENG  
Site Vice President - OCNGS  
Plant Manager - OCNGS  
Director, Operations - OCNGS  
Director, Site Engineering - OCNGS  
Director, Engineering - KSA  
Director, Licensing and Regulatory Affairs - KSA

w/ Attachments  
Manager, Regulatory Assurance - OCNGS  
Manager, Licensing  
Senior Manager, Decommissioning – KSA  
Director, Site Decommissioning – OCNGS  
Commitment Coordinator - KSA  
Records Management – KSA

## **ATTACHMENT 1**

### **Response to Request for Additional Information and Supplemental Information Regarding Proposed Changes to Technical Specifications Section 6.0 Administrative Controls for Permanently Defueled Condition**

Oyster Creek Nuclear Generating Station  
Docket Nos. 50-219 and 72-15

## **ATTACHMENT 1**

### **Response to Request for Additional Information and Supplemental Information Regarding Proposed Changes to Technical Specifications Section 6.0 Administrative Controls for Permanently Defueled Condition**

Oyster Creek Nuclear Generating Station  
Docket Nos. 50-219 and 72-15

#### **SUMMARY**

By letter dated May 17, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16138A129)(Reference 1), Exelon Generation Company, LLC (Exelon) submitted to the U.S. Nuclear Regulatory Commission (NRC) a License Amendment Request (LAR) for changes to Technical Specifications (TS) Section 6.0, Administrative Controls, related to a permanently defueled condition for the Oyster Creek Nuclear Generating Station (OCNGS). Subsequently, in a letter dated October 7, 2016 (Reference 2), the NRC issued a Request for Additional Information (RAI) indicating that additional clarification is needed to enable the NRC to make an independent assessment regarding its technical review of the proposed changes. The NRC is requesting that Exelon respond to the RAI within 30 days of the date of the Reference 2 letter.

Accordingly, this attachment provides Exelon's response to the RAI questions contained in the Reference 2 letter. The specific questions are identified below followed by Exelon's response. Following the responses to the RAI, Exelon is requesting additional proposed administrative changes to supplement the LAR. Attachment 2 includes the revised TS page mark-ups. The revisions are designated by using a box/cloud around the change. Attachment 3 contains clean/camera-ready TS pages. The TS pages being submitted in this letter supersede in entirety those submitted in Reference 1.

#### **RESPONSE TO RAI QUESTIONS**

##### **RAI-1**

*Currently, the Vice President is responsible for overall facility operation in TS 6.1. The Vice President is clearly a "Responsible officer" in accordance with Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Section 2, "Definitions." "Responsible officer means, for the purposes of § 50.55(e) of this chapter, the president, vice-president, or other individual in the organization of a corporation, partnership, or other entity who is vested with executive authority over activities subject to this part." Will the Plant Manager be vested with similar executive authority that the current Vice President has?*

##### **Exelon's Response to RAI-1**

The Site Vice President has overall responsibility and accountability for the safe and efficient operation of a nuclear power plant. The Site Vice President is a corporate officer position. The Site Vice President meets the definition of "Responsible officer" as defined in 10CFR50.2.

To reflect the change in safety concerns from an operating plant to a permanently defueled facility, the responsibility for the control of activities necessary for the safe operation and

maintenance of the plant is changed to the responsibility for safe storage and maintenance of the spent nuclear fuel. The assignment of the position for responsibility at a permanently defueled facility is changed from the OCNGS Site Vice President to the Plant Manager.

The Plant Manager will be the senior position at the facility and will be duly authorized with sufficient authority to carry out the specific responsibilities as described in TS 6.1.1 and TS 6.2.1.b. Although not a corporate officer, the Plant Manager will be vested with similar authority over activities subject to the safe storage and handling of spent nuclear fuel. The Plant Manager will be duly authorized with sufficient authority and organizational freedom to identify, initiate, recommend, and provide solutions at a defueled facility.

As described in the Decommissioning Site Organization (Section 2.5) of the Exelon's Quality Assurance Topical Report (QATR), the management position responsible for facility operations will be "responsible for the management oversight, directing, and implementing appropriate controls to maintain the site within the requirements and constraints applicable to a permanently shutdown unit and to ensure the safe storage of spent nuclear fuel." The QATR further states, "When the Company delegates responsibility for planning, establishing, or implementing any part of the overall QA Program, sufficient authority to accomplish the assigned responsibilities is delegated."

The Plant Manager will report to a "responsible officer" (Vice President or higher) within the corporate organization who will have the necessary authority and full responsibility for the safe management, storage, and handling of spent nuclear fuel.

## **RAI-2**

*In the proposed change to TS 6.1.1, you state the following: "The Plant Manager or delegated designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect safe storage and maintenance."*

*Briefly explain your succession plan for the "delegated designee?"*

## **Exelon's Response to RAI-2**

In a permanently defueled condition, the Plant Manager will be responsible for overall facility operation at the site. In the event the Plant Manager is unavailable, absent, or incapacitated, a senior manager of the facility staff will be designated as responsible for all facility activities.

The Plant Manager may delegate the authority to perform their functions provided the delegated senior manager meets the qualification requirements for the Plant Manager functions. This individual will be a senior manager who satisfies the requirements stated in the Plant Manager job description and the training and experience level of the Plant Manager position, as described in ANSI/ANS 3.1-1978, Section 4.2.1. Normally, authority would be delegated no more than one level lower in the organization.

In the event that there are no senior managers available, the Shift Manager is an acceptable position and meets the qualifications as noted above.

Delegation of authority shall be documented in writing. Management may elect to write a delegation of authority memorandum that is effective from the date written until rescinded. The delegation of authority memorandum allows individuals to perform said functions when verbally authorized by management. This delegation of responsibility is communicated in a memorandum that is circulated throughout the leadership team at the facility.

The delegation of Plant Manager responsibility will be included in the Defueled Safety Analysis Report.

TS 6.1.1 (Insert A) is being revised to remove the term "delegated," as follows:

"The Plant Manager or **the** designee shall approve, prior to <...>"

### **RAI-3**

*Currently, in TS 6.2: "The Chief Nuclear Officer shall have corporate responsibility for overall plant nuclear safety..."*

*The proposed change states "A specified corporate officer shall have corporate responsibility for overall facility nuclear safety..."*

*Responsible officer means, for the purposes of § 50.55(e) of this chapter, the president, vice-president, or other individual in the organization of a corporation, partnership, or other entity who is vested with executive authority over activities subject to this part.*

*The term "specified corporate officer" is not defined in the Oyster Creek TSs. Define the definition of "specified corporate officer" in the Oyster Creek TSs or change the "specified corporate officer" to an existing defined term. Will the "specified corporate officer" be located onsite at OCNGS or at another location? Will the "specified corporate officer" be vested with similar executive authority that the current Chief Nuclear Officer has? Will the Plant Manager be the "specified corporate officer" or will it be another person? If the "specified corporate officer" is not the Plant Manager, briefly explain who will fill the "specified corporate position?"*

### **Exelon's Response to RAI-3**

The Chief Nuclear Officer (CNO) is the senior executive officer and the "responsible officer" for Exelon's nuclear fleet. Following plant shutdown and permanent cessation of operations, nuclear safety will focus predominately on ensuring the safe handling and management of spent nuclear fuel. The proposed LAR reassigned this responsibility from the CNO to a "specified corporate officer," which is a generic title used in Improved Technical Specifications (ITS) (i.e., NUREG-1433, Revision 4). The title was applied to specify the corporate officer who would have direct responsibility for the defueled facility. Exelon agrees that this generic title is not defined in the OCNGS TS and has opted to replace the generic title with the term "responsible officer," which is defined in 10CFR 50.2. The "responsible officer" will be an executive level corporate position who is not located at the site. The "responsible officer" will be vested with the corporate responsibility for ensuring the safe storage and handling of spent nuclear fuel. A description of the overall organizational structure and reporting relationships will be maintained in the Defueled Quality Assurance Manual. The Plant Manager at the

defueled facility will report to the "responsible officer," however as discussed in the response to RAI-1, the Plant Manager will not be the "responsible officer".

TS 6.2.1.c is being revised to reflect the term "responsible officer." An additional change to this specifications is being proposed and is discussed below with the additional supplemented changes.

TS 6.2.1.c:

**"A responsible officer shall have corporate responsibility for the safe storage and handling of spent nuclear fuel and shall take measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the facility to ensure safe management of spent nuclear fuel."**

#### **RAI-4**

*Currently, in proposed TS 6.2, the term "non-certified operator" is used. The term "non-certified operator" is not defined in the Oyster Creek TSs. Define the definition of "non-certified operator" in the Oyster Creek TSs or change the "non-certified operator" to an existing defined term.*

#### **Exelon's Response to RAI-4**

The term "non-certified operator" was taken from the draft NUREG-1625, "Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants" (ML082330233). This draft NUREG was developed in Fiscal Year (FY)-1997 by the NRC and is based on the Trojan Nuclear Plant's Permanently Defueled Technical Specification (PDTs). This term has been used in a majority of the PDTs approved by the NRC at other decommissioning sites. Further, the term "non-certified operator" is referenced in the Advanced Notice of Proposed Rulemaking (ANPR) for "Regulatory Improvements for Decommissioning Power Reactors," dated November 19, 2015 (80FR72358) as a common industry term. Exelon considers that "non-certified operator" should be used based on previous precedent and industry usage.

However, Exelon agrees to add the definition for NON-CERTIFIED OPERATOR to describe the new position and ensure consistent understanding and application. The OCNGS TS Section 1.1 "Definitions" is being revised to add the definition for NON-CERTIFIED OPERATOR as follows:

#### **"1.53 NON-CERTIFIED OPERATOR**

**A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 6.3.1, but is not a CERTIFIED FUEL HANDLER."**

All references to the NON-CERTIFIED OPERATOR position in the proposed TS will be capitalized to indicate the position will be a defined term in Sections 6.2.2.a, 6.2.2.c, and 6.2.2.g of the OCNGS TS.

## **RAI-5**

*As documented in Enforcement Guidance Memorandum (EGM) 2012-001, "Dispositioning Noncompliance with Administrative Controls Technical Specifications Programmatic Requirements That Extend Test Frequencies and Allow Performance of Missed Tests," the allowances permitted by TS 4.0.2 and TS 1.24 cannot be applied to TS Administrative Controls Program activities such as Oyster Creek's TS Section 6.8.4.a, "Radioactive Effluent Controls Program" unless such activity is associated with a Limiting Condition for Operation Surveillance Requirement. NRC-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF-545, Revision 3, "TS Inservice Testing [IST] Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing" (ADAMS Accession No. ML15294A555), dated October 21, 2015, addressed the issues identified in EGM 2012-001.*

*Provide a technical justification on the acceptability of allowing the use of TS 4.0.2 and 1.24 for TS 6.8.4.a activities.*

### **Exelon's Response to RAI-5**

The Offsite Dose Calculation Manual (ODCM) is maintained in OCNGS Procedure CY-OC-170-301 and this procedure would be the implementing document referenced in the OCNGS TS. The provision of SR 4.02 and TS 1.24 are currently included within the ODCM procedure. Therefore, there is no need for this proposed revision and it can be withdrawn.

The proposed provisions statement added at the end of TS 6.8.4.a on page 9 of 28 and described on page 21 of 28 is being withdrawn.

## **ADDITIONAL PROPOSED SUPPLEMENTAL CHANGES**

<b>TS Section 1.1 - Definitions</b>	
<b>Proposed TS</b>	
<u>Term</u>	<u>Definition</u>
<b>1.52 CERTIFIED FUEL HANDLER</b>	<b><i>A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Specification 6.3.2.</i></b>
<b>BASIS</b>	
The definition of the term Certified Fuel Handler (CFH) is being added as a definition to describe the new position and ensure consistent understanding and application. All references to the CFH position in the proposed TS will be capitalized to indicate the position is a defined term in Sections 6.2.1.d, 6.2.2.c, 6.2.2.d, 6.2.2.f, 6.2.2.g and 6.3.3 of the OCNGS TS LAR.	

TS Section 6.2 - Organization	
Current Proposed TS	Proposed Supplement TS
<p>6.2.1 <del>Corporate</del><b><u>Onsite and Offsite Organizations</u></b></p> <p>c. <del>The Chief Nuclear Officer</del> A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support <del>in the plant so that continued nuclear safety is assured</del><b>to the facility to ensure safe management of spent nuclear fuel.</b></p>	<p>6.2.1 <del>Corporate</del> <b><u>Onsite and Offsite Organizations</u></b></p> <p>c. <del>The Chief Nuclear Officer</del> A <b>responsible</b> officer shall have corporate responsibility for <del>overall plant nuclear safety</del> <b>the safe storage and handling of spent nuclear fuel</b> and shall take measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support <del>in the plant so that continued nuclear safety is assured</del><b>to the facility to ensure safe management of spent nuclear fuel.</b></p>
BASIS	
<p>In response to RAI-3 above, the proposed change was modified to reference responsible officer. This proposed change changes the responsibility of "overall plant safety" to responsibility for "the safe storage and handling of spent nuclear fuel." Following plant shutdown and permanent cessation of operations, nuclear safety will focus predominately on ensuring the safe control and management of spent nuclear fuel.</p>	
Current Proposed TS	Proposed Supplement TS
<p>6.2.2 <del>FACILITY STAFF</del><b><u>Facility Staff</u></b></p> <p>b. <del>Except for the Shift Manager,</del> <b>Shift</b> crew composition may be one less than the minimum requirements <b>of 6.2.2.a</b> for a period of time not to exceed two hours, in order to accommodate unexpected absence of on-duty shift crew members. Immediate action must be taken to restore the shift crew composition to within requirements given above. This provision does not permit any shift crew position to be unmanned upon shift change due to an incoming shift crew member being late or absent.</p>	<p>6.2.2 <del>FACILITY STAFF</del><b><u>Facility Staff</u></b></p> <p>b. <del>Except for the Shift Manager,</del> <b>Shift</b> crew composition may be one less than the minimum requirements <b>of 6.2.2.a</b> for a period of time not to exceed two hours, in order to accommodate unexpected absence of on-duty shift crew members. Immediate action must be taken to restore the shift crew composition to within requirements given above. <b>During such absences, no fuel movement or movement of loads over the spent fuel shall be permitted.</b> This provision does not permit any shift crew position to be unmanned upon shift change due to an incoming shift crew member being late or absent.</p>
BASIS	
<p>The proposed change provides additional provisions to ensure that fuel movement and movement of loads over the spent fuel do not occur during the period when shift crew composition is not below the minimum requirements.</p>	

TS Section 6.3 – Facility Staff Qualifications	
Current Proposed TS	Proposed Supplement TS
6.3 <i>Facility Staff Qualifications</i>	6.3 <del><i>Facility Staff Qualifications</i></del> <b>FACILITY STAFF QUALIFICATIONS</b>
BASIS	
The proposed change corrects the format of the title from "Facility Staff Qualifications" to "FACILITY STAFF QUALIFICATIONS." This change is considered to be editorial in nature.	

TS 6.3 through TS 6.7
TS 6.3 through TS 6.7 are relocated to page 6-2 as part of the reformatting and repagination of TS Section 6.0. The Amendment Numbers listed at the bottom of the deleted pages are being consolidated on page 6-2. Page 6-2a is being deleted and the note on page 6-3 stating "(pages 6-3 through 6-8 deleted)" is being deleted. This proposed change is considered to be editorial.

TS Section 6.8 – Procedures and Programs	
Current Proposed TS	Proposed Supplement TS
6.8.1 <i>Written procedures &lt;...&gt;</i>  a. <i>The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33 as referenced in the QATR.</i>  <...> c. <i>Refueling Operations.</i> <...> j. <i>Plant Staff Overtime pursuant to Technical Specification 6.2.2.2(i), above.</i>	6.8.1 <i>Written procedures &lt;...&gt;</i>  a. The <del>applicable</del> procedures <b><i>applicable to safe storage of nuclear fuel</i></b> recommended in Appendix "A" of Regulatory Guide 1.33 as referenced in the QATR. <...> c. <del><i>Refueling</i></del> <b><i>Fuel Handling</i></b> Operations. <...> j. <del>Plant Staff Overtime pursuant to Technical Specification 6.2.2.2(i), above.</del> [Removed due to last item]
BASIS	
<u>TS 6.8.1.a</u> The applicability of this TS is being revised to address only procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. The proposed change reduces the scope of the TS to requiring only the establishment, implementation, and maintenance of written procedures applicable to the safe storage of nuclear fuel. This change recognizes the reduced requirements associated with the protection of stored nuclear fuel as opposed to the operation of the nuclear power plant.  <u>TS6.8.1.c</u> This proposed change is to specify that procedures are required for fuel	

handling operations, rather than refueling operations, because refueling of the reactor will be prohibited by the 10 CFR Part 50 license once the certifications required by 10 CFR 50.82(a)(1) have been submitted to the NRC. Procedures governing fuel handling operations will provide the guidance necessary to ensure safe handling of spent fuel in the spent fuel pool and transfer from the spent fuel pool to dry fuel storage casks. Procedures governing responses to fuel handling accidents, personnel injuries, spent fuel pool events and external events provide the necessary guidance to mitigate the consequences of such events. No change to OCNGS's response to a fuel handling accident is being proposed in this submittal.

TS 6.8.1.j The proposed change deletes this specification that refers to TS 6.2.2.2(i), which was deleted in license amendment 274. The specification will be removed completely since it is the last item listed. This change is considered to be editorial in nature.

Current Proposed TS	Proposed Supplement TS
<p>6.8.3 <i>Temporary changes to procedures of 6.8.1, above, may be made provided:</i>  &lt;...&gt;  <i>b. The change is approved by two members of the licensee's management staff knowledgeable in the area affected by the procedure. For changes which may affect the operational status of unit systems or equipment, at least one of these individuals shall be a member of unit management or supervision holding a Senior Reactor Operator's License on the unit.</i></p>	<p>6.8.3 Temporary changes to procedures of 6.8.1, above, may be made provided:  &lt;...&gt;  <i>b. The change is approved by two members of the licensee's management staff knowledgeable in the area affected by the procedure. For changes which may affect the operational status of <del>unit</del><b>facility</b> systems or equipment, at least one of these individuals shall be a member of <del>unit</del><b>operations</b> management or supervision holding a Senior Reactor Operator's License on the unit <del>who is a CERTIFIED FUEL HANDLER.</del></i></p>

#### BASIS

The proposed change to this specification changes the approval of temporary procedure changes that may affect the operational status of the unit **[facility] systems or equipment** to be a member of operations management who is a CERTIFIED FUEL HANDLER. The requirement for the manager to hold a Senior Reactor Operator License is being removed and replaced by the requirement to be a CERTIFIED FUEL HANDLER.

Once the certifications required by 10 CFR 50.82(a)(1) have been submitted, the requirements of 10 CFR 50.54(m) will no longer be applicable because the OCNGS Part 50 license no longer will authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel. These certifications also obviate the need for the operators' licenses specified in 10 CFR 55. Therefore, there is no longer a need for operations management staff to hold a SRO license.

The term "unit" is being changed to "facility" to be more appropriate with a defueled condition, which is consistent with other changes in this LAR.

Current Proposed TS	Proposed Supplement TS
6.8.4.a.9 <i>Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from 1-131, 1-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluent released beyond the SITE BOUNDARY conforming to Appendix I of 10 CFR 50,</i>	6.8.4.a.9 Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from <del>1-131, 1-133</del> <b>I-131, I-133</b> , tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluent released beyond the SITE BOUNDARY conforming to Appendix I of 10 CFR 50,
<b>BASIS</b>	
The proposed change corrects a pre-existing typographical error in that "I-131 [and] I-132" were stated as "1-131, 1-132." This change is considered to be editorial in nature.	

TS Section 6.9 – Reporting Requirements	
Current Proposed TS	Proposed Supplement TS
6.9.1 <i>ROUTINE REPORTS</i>	6.9 <del>ROUTINE REPORTS</del> <b>Routine Reports</b>
<b>BASIS</b>	
The proposed change corrects the format of the title from all capital letters, "ROUTINE REPORTS," to capitalization of each word, "Routine Reports." This change is considered to be editorial in nature.	
6.9.3 <i>UNIQUE ROUTINE REQUIREMENTS</i>	6.9.3 <del>UNIQUE REPORTING REQUIREMENTS</del> <b>Unique Reporting Requirements</b>
<b>BASIS</b>	
The proposed change corrects the format of the title from all capital letters, "UNIQUE ROUTINE REQUIREMENTS," to capitalization of each word, "Unique Routine Requirements." This change is considered to be editorial in nature.	

TS Section 6.21 – Reporting Requirements	
Current Proposed TS	Proposed Supplement TS
6.21 <i>Technical Specifications (TS) Basis Control Program</i>	6.21 <del>Technical Specifications (TS) Basis Control Program</del> <b>TECHNICAL SPECIFICATIONS (TS) BASIS CONTROL PROGRAM</b>

BASIS	
The proposed change corrects the format of the title from "Technical Specifications (TS) Basis Control Program," to all capitals letters "TECHNICAL SPECIFICATIONS (TS) BASIS CONTROL PROGRAM." This change is considered to be editorial in nature.	

TS Section 6.22 – Control Room Envelope Habitability Program	
Current Proposed TS	Proposed Supplement TS
6.22 <i>Control Room Envelope Habitability Program</i>	6.22 <del>Control Room Envelope Habitability Program</del> <b>CONTROL ROOM ENVELOPE HABITABILITY PROGRAM</b>
BASIS	
The proposed change corrects the format of the title from "Control Room Envelope Habitability Program," to all capitals letters " <b>Control Room Envelope Habitability Program</b> ." This change is considered to be editorial in nature.	
<p>c. <i>Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the</i></p> <p>d. <i>CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.</i></p>	<p>c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the <del>the</del> CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.</p>
BASIS	
The proposed change corrects a pre-existing typographical error in that subsection d. was inserted after the first line of subsection c. Subsection d. was removed. This change is considered to be editorial in nature.	

<b>TS Section 6.23 – Reactor Coolant System (RCS) Pressure and Temperature Limits Report</b>	
<b>Current Proposed TS</b>	<b>Proposed Supplement TS</b>
6.23 <i>Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT</i>	6.23 <b>REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT</b>
<b>BASIS</b>	
The proposed change corrects the format of the title to capitalize “Reactor Coolant System” to be consistent with the other headers of this level. This change is considered to be editorial in nature.	

## **REFERENCES**

1. Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "License Amendment Request – Proposed Changes to Technical Specifications Section 6.0 Administrative Controls for Permanently Defueled Condition," dated May 17, 2016 (ML16138A129)
2. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC) – "Oyster Creek Nuclear Generating Station - Request For Additional Information Regarding License Amendment Request For Changes To Technical Specifications Section 6.0, "Administrative Controls" For Permanently Defueled Condition (CAC NO. MF8108)," dated October 7, 2016 (ML16244A311)

## **ATTACHMENT 2**

### **Revised Technical Specifications Page Mark-ups**

#### **Oyster Creek Nuclear Generating Station Renewed Facility Operating License No. DPR-16 NRC Docket No. 50-219**

Proposed supplement changes to the administrative requirements for the OCNGS staff contained in Section 6.0, Administrative Controls, of the OCNGS Technical Specifications (TS) are indicated by a box/cloud designation.

#### **TS Pages**

ii  
iii  
1.0-9  
6-1  
6-2  
6-2a  
6-3  
6-9  
6-10  
6-11  
6-12  
6-13  
6-14  
6-14a  
6-15  
6-16  
6-21  
6-22

### **ATTACHMENT 3**

#### **Clean/Camera-Ready Technical Specifications Pages**

**Oyster Creek Nuclear Generating Station  
Renewed Facility Operating License No. DPR-16  
NRC Docket No. 50-219**

#### **TS Pages**

ii

iii

1.0-9

6-1

6-2

6-3

6-4

6-5

6-6

6-7

6-8

6-9

6-10

6-11

6-12

6-13

6-14

## TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
1.44 Local Linear Heat Generation Rate	1.0-8
1.45 Shutdown Margin (SDM)	1.0-8
1.46 Idle Recirculation Loop	1.0-8
1.47 Isolated Recirculation Loop	1.0-8
1.48 Operational Condition	1.0-8
1.49 Rated Thermal Power	1.0-9
1.50 Thermal Power	1.0-9
1.51 Pressure and Temperature Limits Report (PTLR)	1.0-9
1.52 Certified Fuel Handler	1.0-9
1.53 Non-Certified Operator	1.0-9
 Section 2 Safety Limits and Limiting Safety System Settings	
2.1 Safety Limit - Fuel Cladding Integrity	2.1-1
2.2 Safety Limit - Reactor Coolant System Pressure	2.2-1
2.3 Limiting Safety System Settings	2.2-3
 Section 3 Limiting Conditions for Operation	
3.0 Limiting Conditions for Operation (General)	3.0-1
3.1 Protective Instrumentation	3.1-1
3.2 Reactivity Control	3.2-1
3.3 Reactor Coolant	3.3-1
3.4 Emergency Cooling	3.4-1
3.5 Containment	3.5-1
3.6 Radioactive Effluents	3.6-1
3.7 Auxiliary Electrical Power	3.7-1
3.8 Isolation Condenser	3.8-1
3.9 Refueling	3.9-1
3.10 Core Limits	3.10-1
3.11 (Not Used)	3.11-1
3.12 Alternate Shutdown Monitoring Instrumentation	3.12-1
3.13 Accident Monitoring Instrumentation	3.13-1
3.14 DELETED	3.14-1
3.15 Explosive Gas Monitoring Instrumentation	3.15-1
3.16 (Not Used)	3.16-1
3.17 Control Room Heating, Ventilating and Air Conditioning System	3.17-1
 Section 4 Surveillance Requirements	
4.0 Surveillance Requirement Applicability	4.0-1
4.1 Protective Instrumentation	4.1-1
4.2 Reactivity Control	4.2-1
4.3 Reactor Coolant	4.3-1
4.4 Emergency Cooling	4.4-1
4.5 Containment	4.5-1
4.6 Radioactive Effluents	4.6-1
4.7 Auxiliary Electrical Power	4.7-1
4.8 Isolation Condenser	4.8-1
4.9 Refueling	4.9-1

## TABLE OF CONTENTS (cont'd)

		<u>Page</u>
4.10	ECCs Related Core Limits	4.10-1
4.11	Sealed Source Contamination	4.11-1
4.12	Alternate Shutdown Monitoring Instrumentation	4.12-1
4.13	Accident Monitoring Instrumentation	4.13-1
4.14	DELETED	4.14-1
4.15	Explosive Gas Monitoring Instrumentation	4.15-1
4.16	(Deleted)	4.16-1
4.17	Control Room Heating, Ventilating and Air Conditioning System	4.17-1
 Section 5	 Design Features	
5.1	Site	5.1-1
5.2	Containment	5.2-1
5.3	Auxiliary Equipment	5.3-1
 Section 6	 Administrative Controls	
6.1	Responsibility	6-1
6.2	Organization	6-1
6.3	Facility Staff Qualifications	6-2
6.4	DELETED	6-2
6.5	DELETED	6-2
6-6	DELETED	6-2
6-7	DELETED	6-2
6-8	Procedures and Programs	6-3
6-9	Reporting Requirements	6-6
6-10	Record Retention	6-8
6-11	Radiation Protection Program	6-9
6-12	(Deleted)	6-9
6-13	High Radiation Area	6-9
6-14	Environmental Qualification	6-10*
6-15	Integrity of Systems Outside Containment	6-10
6-16	Iodine Monitoring	6-10
6-17	DELETED	6-11
6-18	Process Control Plan	6-11
6-19	Offsite Dose Calculation Manual	6-11
6-20	DELETED	6-12
6-21	Technical Specification (TS) Bases Control Program	6-12
6-22	Control Room Envelope Habitability Program	6-12
6-23	Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	6-13
6.24	Surveillance Frequency Control Program	6-14
6.25	Snubber Inspection Program	6-14

\*Issued by NRC Order dated 10-24-80

1.49 RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1930 MWt.

1.50 THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

1.51 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.23.

1.52 CERTIFIED FUEL HANDLER

A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Specification 6.3.2.

1.53 NON-CERTIFIED OPERATOR

A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 6.3.1, but is not a CERTIFIED FUEL HANDLER.

## ADMINISTRATIVE CONTROLS

### 6.1 RESPONSIBILITY

- 6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during the Plant Manager's absence.

The Plant Manager or the designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect safe storage and maintenance of spent nuclear fuel.

- 6.1.2 The Shift Manager shall be responsible for the shift command function.

### 6.2 ORGANIZATION

#### 6.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for facility staff and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safe storage and handling of spent nuclear fuel.

- a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including facility organization positions. These relationships shall be documented and updated as appropriate, in the form of organizational descriptions. These organizational descriptions will be documented in the Updated FSAR and updated in accordance with 10 CFR 50.71e.
- b. The Plant Manager shall be responsible for overall facility safe operation and shall have control over those onsite activities necessary for safe storage and maintenance of spent nuclear fuel.
- c. A responsible officer shall have corporate responsibility for the safe storage and handling of spent nuclear fuel and shall take measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the facility to ensure safe management of spent nuclear fuel.
- d. Individuals who train the CERTIFIED FUEL HANDLERS and those who carry out the health physics and quality assurance functions may report to the appropriate manager on site; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

#### 6.2.2 Facility Staff

The facility organization shall meet the following:

- a. Each on duty shift shall include at least the following shift staffing:
  - One (1) Shift Manager (see f. below)
  - One (1) NON-CERTIFIED OPERATOR (see g. below)
- b. Shift crew composition may be one less than the minimum requirements of 6.2.2.a for a period of time not to exceed two hours, in order to accommodate unexpected absence of on-duty shift crew members. Immediate action must be

taken to restore the shift crew composition to within the requirements given above. During such absences, no fuel movement or movement of loads over the spent fuel shall be permitted. This provision does not permit any shift crew position to be unmanned upon shift change due to an incoming shift crew member being late or absent.

- c. At all times when nuclear fuel is stored in the spent fuel pool, at least one person qualified in the control room (NON-CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER) shall be present in the control room.
- d. Oversight of fuel handling operations shall be provided by a CERTIFIED FUEL HANDLER.
- e. An individual qualified in radiation protection measures shall be on site during movement of fuel and during the movement of loads over the fuel.
- f. The Shift Manager shall be a CERTIFIED FUEL HANDLER.
- g. The position of NON-CERTIFIED OPERATOR may be filled by a CERTIFIED FUEL HANDLER.

### 6.3 FACILITY STAFF QUALIFICATIONS

- 6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1 of 1978 for comparable positions unless otherwise noted in the Technical Specifications. Technicians and maintenance personnel who do not meet ANSI/ANS 3.1 of 1978, Section 4.5, are permitted to perform work for which qualification has been demonstrated.
- 6.3.2 The management position responsible for radiological controls shall meet or exceed the qualifications of Regulatory Guide 1.8 (Rev. 1-R, 9/75). Each other member of the radiation protection organization for which there is a comparable position described in ANSI N18.1-1971 shall meet or exceed the minimum qualifications specified therein, or in the case of radiation protection technicians, they shall have at least one year's continuous experience in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations and shall have been certified by the management position responsible for radiological controls as qualified to perform assigned functions. This certification must be based on an NRC approved, documented program consisting of classroom training with appropriate examinations and documented positive findings by responsible supervision that the individual has demonstrated his ability to perform each specified procedure and assigned function with an understanding of its basis and purpose.
- 6.3.3 The NRC approved training and retraining program for CERTIFIED FUEL HANDLERs shall be maintained.

6.4 DELETED

6.5 DELETED

6.6 DELETED

6.7 DELETED

OYSTER CREEK

6-2

Amendment No.: ~~69, 78, 84, 89, 92, 102, 108, 117, 125, 134, 161, 180, 181, 194, 203, 210, 213, 220, 224, 226, 232, 251, 267, 273, 274, 281,~~

## 6.8 PROCEDURES AND PROGRAMS

- 6.8.1 Written procedures shall be established, implemented, and maintained covering the items referenced below:
- a. The procedures applicable to safe storage of nuclear fuel recommended in Appendix "A" of Regulatory Guide 1.33 as referenced in the QATR.
  - b. Surveillance and test activities of equipment that affects nuclear safety and radioactive waste management equipment.
  - c. Fuel Handling Operations.
  - d. Security Plan Implementation.
  - e. Fire Protection Program Implementation.
  - f. Emergency Plan Implementation.
  - g. Process Control Plan Implementation.
  - h. Offsite Dose Calculation Manual Implementation.
  - i. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1.
- 6.8.2 Each procedure required by 6.8.1 above, and substantive changes thereto, shall be reviewed and approved prior to implementation and shall be reviewed periodically as set forth in administrative procedures.
- 6.8.3 Temporary changes to procedures of 6.8.1, above, may be made provided:
- a. The intent of the original procedure is not altered;
  - b. The change is approved by two members of the licensee's management staff knowledgeable in the area affected by the procedure. For changes which may affect the operational status of facility systems or equipment, at least one of these individuals shall be a member of operations management or supervision who is a CERTIFIED FUEL HANDLER.
  - c. The change is documented, reviewed and approved within 14 days of implementation.

6.8.4 The following programs shall be established, implemented and maintained:

a. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluent and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluent as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including the surveillance tests and setpoint determination in accordance with the methodology in the ODCM,
2. Limitations on the concentrations of radioactive material released in liquid effluent to the UNRESTRICTED AREA conforming to less than the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402.
3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluent in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM.
4. Limitations on the annual and quarterly doses and dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluent released to the UNRESTRICTED AREA conforming to Appendix I of 10 CFR 50,
5. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.
6. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in the 31 day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR 50,
7. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to the UNRESTRICTED AREA shall be limited to the following:
  - a. For noble gases: Less than or equal to a dose rate of 500 mRems/yr to the total body and less than or equal to a dose rate of 3000 mRems/yr to the skin, and
  - b. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mRems/yr to any organ.
8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to the UNRESTRICTED AREA conforming to Appendix I of 10 CFR 50,

9. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from I-131, I-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluent released beyond the SITE BOUNDARY conforming to Appendix I of 10 CFR 50,
10. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from Uranium fuel cycle sources conforming to 40 CFR Part 190.

b. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,
2. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and
3. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.8.5 Station Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications," of the following:

- a. Actions to restore station battery cells with float voltage < 2.13 volts, and
- b. Actions to equalize and test station battery cells that have been discovered with electrolyte level below the top of the plates.

## 6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of 10 CFR, the following identified reports shall be submitted to the Administrator of the NRC Region I office unless otherwise noted.

### 6.9.1 Routine Reports

- a. DELETED
- b. DELETED
- c. DELETED
- d. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluent and solid waste released from the facility. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

- e. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the facility during the previous calendar year shall be submitted prior to May 1 of each year.

The Report shall include summaries, interpretations, and an analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in: (1) the ODCM; and, (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

Basis: 6.9.1.e - RELOCATED TO THE ODCM.

- f. DELETED

### 6.9.2 DELETED

### 6.9.3 Unique Reporting Requirements

Special reports shall be submitted to the Director of Regulatory Operations Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

- a. Materials Radiation Surveillance Specimen Reports (4.3A)
- b. (Deleted)
- c. Results of required leak tests performed on sealed sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination.
- d. Deleted
- e-j. Pursuant to the ODCM.
- k. Records of results of analyses required by the Radiological Environmental Monitoring Program.
- l. Failures and challenges to Relief and Safety Valves which do not constitute an LER will be the subject of a special report submitted to the Commission within 60 days of the occurrence. A challenge is defined as any automatic actuation (other than during surveillance or testing) of Safety or Relief Valves.
- m. Plans for compliance with standby liquid control Specifications 3.2.C.3(b) and 3.2.C.3(e)(1) or plans to obtain enrichment test results per Specification 4.2.E.5.
- n. Inoperable high range radioactive noble gas effluent monitor (3.13.H)

## 6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

- a. Records and logs of facility operation covering time interval at each power level.
- b. Records and logs of principle maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
- c. All Licensee Event Reports.
- d. Records of surveillance activities, inspections and calibrations required by these technical specifications.
- e. Records of reactor tests and experiments.
- f. Records of changes made to operating procedures.
- g. Deleted.
- h. Records of sealed source leak tests and results.
- i. Records of annual physical inventory of all source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

- a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.
- b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
- c. Records of facility radiation and contamination surveys.
- d. Records of doses received by all individuals for whom monitoring was required
- e. Records of gaseous and liquid radioactive material released to the environs.
- f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.
- g. Records of training and qualification for current members of the plant staff.
- h. Records of inservice inspections performed pursuant to these technical specifications.
- i. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.
- j. Deleted.

- k. Records of Environmental Qualification which are covered under the provisions for paragraph 6.14.
- l. Deleted.
- m. Records of results of analyses required by the Radiological Environmental Monitoring Program.
- n. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PLAN.
- o. Records of radioactive shipments

6.10.3 Quality Assurance Records shall be retained as specified by the QATR.

#### 6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

#### 6.12 DELETED

#### 6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by Section 20.1601 of 10 CFR 20, each high radiation area in which the intensity of radiation at 30 cm (11.8 in.) is greater than deep dose equivalent of 100 mRem/hr but less than 1,000 mRem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).

NOTE: Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are following plant radiation protection procedures for entry into high radiation areas.

An individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

- a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.
- b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a pre-set integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.
- c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive exposure control over the activities within the area and who will perform periodic radiation surveillance at the frequency in the RWP. The surveillance frequency will be established by the management position responsible for radiological controls.

6.13.2 Specification 6.13.1 shall also apply to each high radiation area in which the intensity of radiation is greater than deep dose equivalent of 1,000 mRem/hr at 30 cm (11.8 in.) but less than 500 rads in 1 hour at 1 meter (3.28 ft.) from sources of radioactivity. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of operations and/or radiation protection supervision on duty.

#### 6.14 ENVIRONMENTAL QUALIFICATION

- A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position of Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License DPR-16 dated October 24, 1980.
- B. By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

#### 6.15 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

- 1. Provisions establishing preventative maintenance and periodic visual inspection requirements, and
- 2. System leak test requirements, to the extent permitted by system design and radiological conditions, for each system at a frequency of once every 24 months. The systems subject to this testing are (1) Core Spray, (2) Containment Spray, (3) Reactor Water Cleanup, (4) Isolation Condenser, and (5) Shutdown Cooling.

#### 6.16 IODINE MONITORING

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas\* under accident conditions. This program shall include the following:

- a. Training of personnel,
- b. Procedures for monitoring, and
- c. Provisions for maintenance of sampling and analysis equipment.

---

\*Areas requiring personnel access for establishing hot shutdown condition.

6.17 Deleted

6.18 PROCESS CONTROL PLAN

- a. Licensee initiated changes to the PCP:
  - 1. Shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:
    - a. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;
    - b. a determination that the changes did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and
    - c. documentation that the changes have been reviewed and approved pursuant to Section 6.8.2.
  - 2. Shall become effective upon review and approval by licensee management.

6.19 OFF SITE DOSE CALCULATION MANUAL

- a. The ODCM shall be approved by the Commission prior to implementation.
- b. Licensee initiated changes to the ODCM shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:
  - 1. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;
  - 2. a determination that the changes did not reduce the accuracy or reliability of dose calculations or setpoint determination; and,
  - 3. documentation that the changes have been reviewed and approved pursuant to Section 6.8.2.
- c. Change(s) shall become effective upon review and approval by licensee management.

6.20 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

DELETED.

6.21 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
  - 1. A change in the TS incorporated in the license or
  - 2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.
- d. Proposed changes that meet the criteria of Specification 6.21.b.1 or 6.21.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.22 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room HVAC System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of a 30-day integrated dose of 5 rem TEDE. The program shall include the following elements:

- a. The definition of the CRE and the CRE boundary.
- b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.
- c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

1. The Oyster Creek CRE boundary operability is not dependent on a measured unfiltered air leakage value (Reference Oyster Creek letter to NRC dated November 17, 2005, Letter No. 2130-05-20218). No leakage testing for determining the unfiltered air leakage past the CRE boundary into the CRE is required at the Oyster Creek site.
- d. Measurement, at designated locations, of the CRE pressure relative to areas adjacent to the CRE boundary during the pressurization mode of operation by one subsystem (train) of the Control Room Ventilation System operating at the design flow rate, at a Frequency of 24 months. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.
- e. The unfiltered air leakage limit for radiological challenges is the leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air leakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.
- f. The provisions of Section 1.24 are applicable to the frequencies for assessing CRE habitability measuring CRE pressure and assessing the CRE boundary as required by paragraphs d and c, respectively.

6.23 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
  - i) Limiting Conditions for Operation Section 3.3, "Reactor Coolant"
  - ii) Surveillance Requirements Section 4.3, "Reactor Coolant"
- b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:
  - i) SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors"
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

#### 6.24 SURVEILLANCE FREQUENCY CONTROL PROGRAM

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
- b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
- c. The provisions of Definition 1.24 and Surveillance Requirement 4.0.2 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

#### 6.25 SNUBBER INSPECTION PROGRAM

This program conforms to the examination, testing, and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

- a. This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.
- b. The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(a), 50.55a(a)(1), 50.55a(a)(1)(i), and 50.55a(a)(1)(iv), subject to its limitations and modifications, and subject to Commission approval.
- c. The program shall, as allowed by 10 CFR 50.55a(b)(3)(v)(B), meet Subsection ISTA, "General Requirements," and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," in lieu of Section XI of the ASME B&PV Code ISI requirements for snubbers, or meet authorized alternatives pursuant to 10 CFR 50.55a(z).
- d. The 120-month program updates shall be made in accordance with 10 CFR 50.55a (including 10 CFR 50.55a(g)(4)(ii)) subject to the conditions listed therein.
- e. Records of the service life of all snubbers, including the date which the service life commences, and associated installation and maintenance records shall be maintained for the duration of the Facility Operating License.

## TABLE OF CONTENTS (Cont'd)

		<u>Page</u>
	1.52 Certified Fuel Handler	
	1.53 Non-Certified Operator	
1.44	Local Linear Heat Generation Rate	1.0-8
1.45	Shutdown Margin (SDM)	1.0-8
1.46	Idle Recirculation Loop	1.0-8
1.47	Isolated Recirculation Loop	1.0-8
1.48	Operational Condition	1.0-8
1.49	Rated Thermal Power	1.0-9
1.50	Thermal Power	1.0-9
1.51	Pressure and Temperature Limits Report (PTLR)	1.0-9
Section 2	Safety Limits and Limiting Safety System Settings	
2.1	Safety Limit - Fuel Cladding Integrity	2.1-1
2.2	Safety Limit - Reactor Coolant System Pressure	2.2-1
2.3	Limiting Safety System Settings	2.2-3
Section 3	Limiting Conditions for Operation	
3.0	Limiting Conditions for Operation (General)	3.0-1
3.1	Protective Instrumentation	3.1-1
3.2	Reactivity Control	3.2-1
3.3	Reactor Coolant	3.3-1
3.4	Emergency Cooling	3.4-1
3.5	Containment	3.5-1
3.6	Radioactive Effluents	3.6-1
3.7	Auxiliary Electrical Power	3.7-1
3.8	Isolation Condenser	3.8-1
3.9	Refueling	3.9-1
3.10	Core Limits	3.10-1
3.11	(Not Used)	3.11-1
3.12	Alternate Shutdown Monitoring Instrumentation	3.12-1
3.13	Accident Monitoring Instrumentation	3.13-1
3.14	DELETED	3.14-1
3.15	Explosive Gas Monitoring Instrumentation	3.15-1
3.16	(Not Used)	3.16-1
3.17	Control Room Heating, Ventilating and Air Conditioning System	3.17-1
Section 4	Surveillance Requirements	
4.0	Surveillance Requirement Applicability	4.0-1
4.1	Protective Instrumentation	4.1-1
4.2	Reactivity Control	4.2-1
4.3	Reactor Coolant	4.3-1
4.4	Emergency Cooling	4.4-1
4.5	Containment	4.5-1
4.6	Radioactive Effluents	4.6-1
4.7	Auxiliary Electrical Power	4.7-1
4.8	Isolation Condenser	4.8-1
4.9	Refueling	4.9-1

## TABLE OF CONTENTS (cont'd)

		<u>Page</u>
4.10	ECCs Related Core Limits	4.10-1
4.11	Sealed Source Contamination	4.11-1
4.12	Alternate Shutdown Monitoring Instrumentation	4.12-1
4.13	Accident Monitoring Instrumentation	4.13-1
4.14	DELETED	4.14-1
4.15	Explosive Gas Monitoring Instrumentation	4.15-1
4.16	(Deleted)	4.16-1
4.17	Control Room Heating, Ventilating and Air Conditioning System	4.17-1
 Section 5	 Design Features	
5.1	Site	5.1-1
5.2	Containment	5.2-1
5.3	Auxiliary Equipment	5.3-1
 Section 6	 Administrative Controls	
6.1	Responsibility	6-1
6.2	Organization	6-1
6.3	Facility Staff Qualifications	6-2a
6.4	DELETED	6-3
6.5	DELETED	6-3
6.6	<del>Reportable Event Action</del>	6-9
6.7	<del>Safety Limit Violation</del>	6-9
6.8	Procedures and Programs	6-10
6.9	Reporting Requirements	6-13
6.10	Record Retention	6-17
6.11	Radiation Protection Program	6-18
6.12	(Deleted)	6-18
6.13	High Radiation Area	6-18
6.14	Environmental Qualification	6-19*
6.15	Integrity of Systems Outside Containment	6-19
6.16	Iodine Monitoring	6-19
6.17	<del>Post Accident Sampling</del>	6-20
6.18	Process Control Plan	6-20
6.19	Offsite Dose Calculation Manual	6-20
6.20	DELETED	6-20
6.21	Technical Specification (TS) Bases Control Program	6-21
6.22	Control Room Envelope Habitability Program	6-21
6.23	Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	6-22
6.24	Surveillance Frequency Control Program	6-23
6.25	Snubber Inspection Program	6-23

\*Issued by NRC Order dated 10-24-80

1.49 RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1930 MWt.

1.50 THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

1.51 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.23.

1.52 CERTIFIED FUEL HANDLER

A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Specification 6.3.2.

1.53 NON-CERTIFIED OPERATOR

A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 6.3.1, but is not a CERTIFIED FUEL HANDLER.

## ADMINISTRATIVE CONTROLS

### 6.1 RESPONSIBILITY

Plant Manager

and

6.1.1 The Vice President ~~Oyster Creek~~ shall be responsible for overall facility operation. These responsibilities delegated to the Vice President as stated in the ~~Oyster Creek Technical Specifications~~ may also be fulfilled by the Plant Manager. The Vice President shall delegate in writing the succession to this responsibility during his and/or the Plant Manager's absence.

INSERT A

INSERT B

### 6.2 ORGANIZATION

Onsite and Offsite Organizations

#### 6.2.1 Corporate

Onsite

organizations

facility staff

safe storage and handling of spent nuclear fuel.

6.2.1.1 An onsite and offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

6.2.1.2 Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including operating organization positions. These relationships shall be documented and updated as appropriate, in the form of organizational charts. These organizational charts will be documented in the Updated FSAR and updated in accordance with 10 CFR 50.71e.

a.

INSERT C

c.

6.2.1.3 The Chief Nuclear Officer shall have corporate responsibility for overall plant nuclear safety and shall take measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

INSERT D

#### 6.2.2 FACILITY STAFF

Facility Staff

6.2.2.1 The Vice President ~~Oyster Creek~~ shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

6.2.2.2 The facility organization shall meet the following:

a. Each on duty shift shall include at least the following shift staffing:

One (1) NON-CERTIFIED OPERATOR (see g. below)

- One (1) Shift Manager (see h. below)
- Two (2) licensed Nuclear Plant Operators
- Three (3) licensed or non-licensed Nuclear Plant Operators
- One (1) Shift Technical Adviser (see h. below)

f.

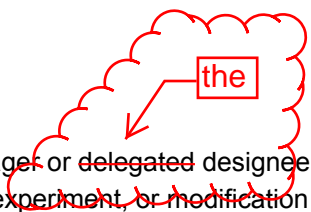
b. Shift

Except for the Shift Manager, shift crew composition may be one less than the minimum requirements, for a period of time not to exceed two hours, in order to accommodate unexpected absence of on-duty shift crew members. Immediate action must be taken to restore the shift crew

of 6.2.2.a

the safe storage and handling of spent nuclear fuel

Insert A

The Plant Manager or ~~delegated~~ designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect safe storage and maintenance of spent nuclear fuel.

Insert B

6.1.2 The Shift Manager shall be responsible for the shift command function.

Insert C

- b. The Plant Manager shall be responsible for overall facility safe operation and shall have control over those onsite activities necessary for safe storage and maintenance of spent nuclear fuel.

Insert D

- d. Individuals who train the CERTIFIED FUEL HANDLERS and those who carry out the health physics and quality assurance functions may report to the appropriate manager on site; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

nuclear fuel is stored in the spent fuel pool

During such absences, no fuel movement or movement of loads over the spent fuel shall be permitted.

composition to within requirements given above. This provision does not permit any shift crew position to be unmanned upon shift change due to an incoming shift crew member being late or absent.

c.

person qualified to stand watch in the control room (NON-CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER)

b.

At all times when there is fuel in the vessel, at least one licensed senior reactor operator shall be on site and one licensed reactor operator should be at the controls.

present in the control room.

c.

At all times when there is fuel in the vessel, except when the reactor is in COLD SHUTDOWN or REFUEL modes, two licensed senior reactor operators and two licensed reactor operators shall be on site, with at least one licensed senior reactor operator in the control room and one licensed reactor operator at the controls.

Oversight of fuel handling operations shall be provided by a CERTIFIED FUEL HANDLER.

d.

At least two licensed reactor operators shall be in the control room during all reactor startups, shutdowns, and other periods involving planned control rod manipulations.

e.

All CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e.

f.

f. The Shift Manager shall be a CERTIFIED FUEL HANDLER.

An individual qualified in radiation protection measures shall be on site when fuel is in the reactor.

during movement of fuel and during the movement of loads over the fuel.

g.

(deleted)

h.

The position of NON-CERTIFIED OPERATOR may be filled by a CERTIFIED FUEL HANDLER.

Each on duty shift shall include a Shift Technical Advisor except that the Shift Technical Advisors position need not be filled if the reactor is in the refuel or shutdown mode and the reactor is less than 212 F. The Shift Technical Advisor position may be filled by an on shift Senior Reactor Operator (dual role SRO/STA) provided the individual meets the requirements of 6.3.3.

i.

(deleted)

- j- ~~The Senior Manager Operations or an Operations Manager, and the Shift Manager require Senior Reactor Operators licenses. The licensed Nuclear Plant Operators require a Reactor Operators license.~~

## FACILITY STAFF QUALIFICATIONS

### 6.3 Facility Staff Qualifications

~~6.2.2.3 Individuals who train the operating staff and those who carry out the health physics and quality assurance function shall have sufficient organizational freedom to be independent of operational pressures, however, they may report to the appropriate manager on site.~~

- 6.3.1 Each member of the ~~unit~~ **facility** staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1 of 1978 for comparable positions unless otherwise noted in the Technical Specifications, ~~with the following exceptions: 1) the licensed operators who shall comply only with the requirements of 10 CFR 55, and 2) technicians and maintenance personnel~~ who do not meet ANSI/ANS 3.1 of 1978, Section 4.5, are permitted to perform work for which qualification has been demonstrated.

**Technicians**

- 6.3.2 The management position responsible for radiological controls shall meet or exceed the qualifications of Regulatory Guide 1.8 (Rev. 1-R, 9/75). Each other member of the radiation protection organization for which there is a comparable position described in ANSI N18.1-1971 shall meet or exceed the minimum qualifications specified therein, or in the case of radiation protection technicians, they shall have at least one year's continuous experience in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations and shall have been certified by the management position responsible for radiological controls as qualified to perform assigned functions. This certification must be based on an NRC approved, documented program consisting of classroom training with appropriate examinations and documented positive findings by responsible supervision that the individual has demonstrated his ability to perform each specified procedure and assigned function with an understanding of its basis and purpose.

- 6.3.3 ~~The Shift Technical Advisors shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, response and analysis of the plant for transients and accidents.~~

**The NRC approved training and retraining program for CERTIFIED FUEL HANDLERs shall be maintained.**

~~6.4 DELETED~~

~~6.5 DELETED~~

This Page Deleted

~~6.3~~

~~(Pages 6.4 through 6.8 deleted)~~

~~OYSTER CREEK~~

~~Amendment No.: 69,78,89,108,117,125,134,161,  
180,181,194,203,210,213,224,232,251, 273~~



← 6.4 DELETED  
6.5 DELETED

6.6 REPORTABLE EVENT ACTION ← DELETED

6.6.1 ~~The following actions shall be taken for REPORTABLE EVENTS:~~

- a. ~~The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50; and~~
- b. ~~Each REPORTABLE EVENT shall be reported to the cognizant manager and the cognizant department director and the Vice President Oyster Creek. The functionally cognizant department staff shall prepare a Licensee Event Report (LER) in accordance with the guidance outlined in 10 CFR 50.73(b). Copies of all such reports shall be submitted to the functionally cognizant department director and the Vice President Oyster Creek.~~

6.7 SAFETY LIMIT VIOLATION ← DELETED

6.7.1 ~~The following actions shall be taken in the event a Safety Limit is violated:~~

- a. ~~If any Safety Limit is exceeded, the reactor shall be shut down immediately until the Commission authorizes the resumption of operation.~~
- b. ~~The Safety Limit violation shall be reported to the Commission and the Vice President Oyster Creek.~~
- c. ~~A Safety Limit Violation Report shall be prepared. The report shall be submitted to the Vice President Oyster Creek. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components systems or structures, (3) corrective action taken to prevent recurrence.~~
- d. ~~The Safety Limit Violation Report shall be submitted to the Commission within ten days of the violation.~~

## 6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the items referenced below:

- a. The ~~applicable~~ procedures recommended in Appendix "A" of Regulatory Guide 1.33 as referenced in the QATR. applicable to safe storage of nuclear fuel
- b. Surveillance and test activities of equipment that affects nuclear safety and radioactive waste management equipment.
- c. ~~Refueling~~ Operations. Fuel Handling
- d. Security Plan Implementation.
- e. Fire Protection Program Implementation.
- f. Emergency Plan Implementation.
- g. Process Control Plan Implementation.
- h. Offsite Dose Calculation Manual Implementation.
- i. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1.
- j. ~~Plant Staff Overtime pursuant to Technical Specification 6.2.2.2(i), above.~~

6.8.2 Each procedure required by 6.8.1 above, and substantive changes thereto, shall be reviewed and approved prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1, above, may be made provided:

- a. The intent of the original procedure is not altered; facility
- b. The change is approved by two members of the licensee's management staff knowledgeable in the area affected by the procedure. For changes which may affect the operational status of ~~unit~~ systems or equipment, at least one of these individuals shall be a member of ~~unit~~ management or supervision holding a ~~Senior Reactor Operator's License on the unit.~~ operations who is a CERTIFIED FUEL HANDLER.
- c. The change is documented, reviewed and approved within 14 days of implementation.

6-3

6.8.4 The following programs shall be established, implemented and maintained:

a. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluent and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluent as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including the surveillance tests and setpoint determination in accordance with the methodology in the ODCM,
2. Limitations on the concentrations of radioactive material released in liquid effluent to the UNRESTRICTED AREA conforming to less than the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402.
3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluent in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM.
4. Limitations on the annual and quarterly doses and dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluent released to the UNRESTRICTED AREA conforming to Appendix I of 10 CFR 50,
5. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.
6. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in the 31 day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR 50,
7. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to the UNRESTRICTED AREA shall be limited to the following:
  - a. For noble gases: Less than or equal to a dose rate of 500 mRems/yr to the total body and less than or equal to a dose rate of 3000 mRems/yr to the skin, and
  - b. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mRems/yr to any organ.
8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to the UNRESTRICTED AREA conforming to Appendix I of 10 CFR 50,

I-131, I-133

9. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from ~~I-131, I-133~~, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluent released beyond the SITE BOUNDARY conforming to Appendix I of 10 CFR 50,
10. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from ~~Uranium fuel cycle sources conforming to 40 CFR Part 190.~~

b. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,
2. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and
3. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.8.5 Station Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications," of the following:

- a. Actions to restore station battery cells with float voltage < 2.13 volts, and
- b. Actions to equalize and test station battery cells that have been discovered with electrolyte level below the top of the plates.

6-5

OYSTER CREEK

6-12

Amendment No.: ~~69,78,84,108,125,134,166,205,245~~  
~~Corrected by letter of 10/15/2004~~

## 6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of 10 CFR, the following identified reports shall be submitted to the Administrator of the NRC Region I office unless otherwise noted.

### 6.9.1 ROUTINE REPORTS

**Routine Reports**

**DELETED**

- a. ~~Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specified details required in license conditions based on other commitments shall be included in this report.~~

~~Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

b. DELETED

c. DELETED

OYSTER CREEK

6-13

6-6

Amendment No.: 69,78,84,108,125,  
134,166,183,191,  
254

d. Radioactive Effluent Release Report

facility

The Radioactive Effluent Release Report covering the operation of the ~~unit~~ during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluent and solid waste released from the ~~unit~~. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

e. Annual Radiological Environmental Operating Report

facility

The Annual Radiological Environmental Operating Report covering the operation of the ~~unit~~ during the previous calendar year shall be submitted prior to May 1 of each year.

The Report shall include summaries, interpretations, and an analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in: (1) the ODCM; and, (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

Basis: 6.9.1.e -  
RELOCATED TO  
THE ODCM.

f.

~~CORE OPERATING LIMITS REPORT (COLR)~~

~~DELETED~~

1. ~~Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:~~
  - a. ~~The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.10.A.~~
  - b. ~~The K<sub>i</sub> core flow adjustment factor for Specification 3.10.C.~~
  - c. ~~The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.10.C.~~
  - d. ~~The LOCAL LINEAR HEAT GENERATION RATE (LLHGR) for Specification 3.10.B.~~
  - e. ~~The Average Power Range Monitor (APRM) stability protection settings for Specifications 2.3.A.1 and 2.3.B.~~

~~and shall be documented in the COLR.~~

- ~~2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.~~
- ~~a. GPU Nuclear (GPUN) Topical Report (TR) 020, Methods for the Analysis of Boiling Water Reactors Lattice Physics, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)~~
  - ~~b. GPUN TR 021, Methods for the Analysis of Boiling Water Reactors Steady State Physics, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)~~

- ~~c. GPUN TR 033, Methods for the Generation of Core Kinetics Data for RETRAN-02, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)~~
  - ~~d. GPUN TR 040, Steady State and Quasi Steady State Methods Used in the Analysis of Accidents and Transients, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)~~
  - ~~e. GPUN TR 045, BWR-2 Transient Analysis Model Using the Retran Code, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)~~
  - ~~f. NEDE 31462P and NEDE 31462, Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss of Coolant Accident Analysis, (The approved revision at the time reload analyses are performed shall be identified in the COLR.)~~
  - ~~g. NEDE 24011 P-A, General Electric Standard Application for Reactor Fuel, (GESTAR II) (The approved revision at the time reload analyses are performed shall be identified in the COLR.)~~
  - ~~h. DELETED~~
  - ~~i. XN 75-55 (A); XN 75-55, Supplement 1 (A); XN 75-55, Supplement 2 (A), Revision 2, "Exxon Nuclear Company WREM Based NJP BWR ECCS Evaluation Model and Application to the Oyster Creek Plant," April 1977~~
  - ~~j. XN 75-36(NP) (A); XN 75-36(NP), Supplement 1 (A), "Spray Cooling Heat Transfer Phase Test Results, ENC-8x8 BWR Fuel 60 and 63 Active Rods, Interim Report," October 1975~~
  - ~~k. NEDC 33065P, Rev. 0, "Application of Stability Long Term Solution Option II for Oyster Creek," April 2002.~~
- ~~3. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.~~
- ~~4. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~

~~Basis: 6.9.1.e RELOCATED TO THE ODCM.~~

#### 6.9.2 REPORTABLE EVENTS

← **DELETED**

~~The submittal of Licensee Event Reports shall be accomplished in accordance with the requirements set forth in 10 CFR 50.73.~~

6.9.3 UNIQUE REPORTING REQUIREMENTS

Unique Reporting  
Requirements

Special reports shall be submitted to the Director of Regulatory Operations Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

- a. Materials Radiation Surveillance Specimen Reports (4.3A)
- b. (Deleted)
- c. Results of required leak tests performed on sealed sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination.
- d. Deleted
- e-j. Pursuant to the ODCM.
- k. Records of results of analyses required by the Radiological Environmental Monitoring Program.
- l. Failures and challenges to Relief and Safety Valves which do not constitute an LER will be the subject of a special report submitted to the Commission within 60 days of the occurrence. A challenge is defined as any automatic actuation (other than during surveillance or testing) of Safety or Relief Valves.
- m. Plans for compliance with standby liquid control Specifications 3.2.C.3(b) and 3.2.C.3(e)(1) or plans to obtain enrichment test results per Specification 4.2.E.5.
- n. Inoperable high range radioactive noble gas effluent monitor (3.13.H)

6.20 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

DELETED.

**TECHNICAL SPECIFICATION (TS) BASIS CONTROL PROGRAM**

6.21 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
  1. A change in the TS incorporated in the license or
  2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.
- d. Proposed changes that meet the criteria of Specification 6.21.b.1 or 6.21.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

**CONTROL ROOM ENVELOPE HABITABILITY PROGRAM**

6.22 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room HVAC System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of a 30-day integrated dose of 5 rem TEDE. The program shall include the following elements:

- a. The definition of the CRE and the CRE boundary.
- b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.
- c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

1. The Oyster Creek CRE boundary operability is not dependent on a measured unfiltered air leakage value (Reference Oyster Creek letter to NRC dated November 17, 2005, Letter No. 2130-05-20218). No leakage testing for determining the unfiltered air leakage past the CRE boundary into the CRE is required at the Oyster Creek site.
- d. Measurement, at designated locations, of the CRE pressure relative to areas adjacent to the CRE boundary during the pressurization mode of operation by one subsystem (train) of the Control Room Ventilation System operating at the design flow rate, at a Frequency of 24 months. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.
- e. The unfiltered air leakage limit for radiological challenges is the leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air leakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.
- f. The provisions of Section 1.24 are applicable to the frequencies for assessing CRE habitability measuring CRE pressure and assessing the CRE boundary as required by paragraphs d and e, respectively.

REACTOR COOLANT SYSTEM

6-23 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
  - i) Limiting Conditions for Operation Section 3.3, "Reactor Coolant"
  - ii) Surveillance Requirements Section 4.3, "Reactor Coolant"
- b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:
  - i) SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors"
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.