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LETTER NUMBER: 2.18.034

September 13, 2018

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

SUBJECT: Technical Specifications Proposed Change - Permanently Defueled Technical Specifications

Pilgrim Nuclear Power Station
Docket No. 50-293
Renewed License No. DPR-035

REFERENCES:

1. Letter, Entergy Nuclear Operations, Inc. to NRC, "Notification of Permanent Cessation of Power Operations," dated November 10, 2015 (Letter Number: 2.15.080) (ML15328A053)

2. Letter, NRC to Entergy Nuclear Operations, Inc., Pilgrim Nuclear Power Station – Issuance of Amendment Regarding Administrative Controls for Permanently Defueled Condition (CAC No. MF9304), dated July 10, 2017 (ML17066A130)

Dear Sir or Madam:

In accordance with Title 10 Code of Federal Regulations (CFR) 50.90, Entergy Nuclear Operations, Inc. (ENO) is proposing an amendment to Renewed Facility Operating License (OL) DPR-35 for Pilgrim Nuclear Power Station (PNPS). This proposed license amendment would revise the OL and the associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

In Reference 1, ENO notified the U.S. Nuclear Regulatory Commission (NRC) that it has decided to permanently cease operations of PNPS no later than June 1, 2019. The proposed changes would revise certain requirements contained within the OL and TS and remove the requirements that would no longer be applicable after it has been certified that all fuel has permanently been removed from the PNPS reactor in accordance with 10 CFR 50.82(a)(1)(ii). After the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed for PNPS, the 10 CFR Part 50 license for PNPS will no longer authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). The proposed changes to the OL and TS are in

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accordance with 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes also include a renumbering of pages, where appropriate, to condense and reduce the number of pages in the TS without affecting the technical content. The TS Table of Contents is also accordingly revised.

In Reference 2, the NRC issued Amendment No. 246 to Renewed Facility Operating License No. DPR-35 for the PNPS. This amendment revises certain staffing and training requirements, reports, programs, and editorial changes contained in the TS Table of Contents; Section 1.0, "Definitions;" Section 4.0, "Design Features;" and Section 5.0, "Administrative Controls," that will no longer be applicable after Pilgrim is permanently defueled. This License Amendment Request reflects the implementation of those changes, because PNPS will be in the permanently shut down and defueled condition when this set of changes is implemented.

ENO has reviewed the proposed amendment in accordance with 10 CFR 50.92 and concludes it does not involve a significant hazards consideration.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, will be provided to the Commonwealth of Massachusetts, Department of Public Health and Agency of Emergency Management.

Attachment 1 to this letter provides a detailed description and evaluation of the proposed change. Attachment 2 contains a markup of the current OL, TS and TS Bases pages. The TS Bases pages are provided for information only. Attachment 3 contains the retyped Renewed Facility License, PDTS, and PDTS Bases pages in their entirety.

ENO requests review and approval of this proposed license amendment by September 13, 2019. The License Amendment will not be implemented until the certifications required by 10 CFR 50.82(a)(1)(i) have been docketed in accordance with 10 CFR 50.82(a)(2) and the decay time requirement established in the analysis of the Fuel Handling Accident in the Spent Fuel Pool (i.e., 24 hours of decay before channeled fuel assemblies can be handled and 46 days of decay (24 hours of decay assumed in the analysis of the FHA + an additional 45 days of decay) before unchanneled fuel assemblies can be handled following shut down) has been met.

There are no new regulatory commitments made in this letter.

If you have any questions on this transmittal, please contact Mr. Peter J. Miner at (508) 830-7127.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 13, 2018.

Sincerely,

BSF/sd

Attachments:

1. Description and Evaluation of the Proposed Changes
2. Markup of the Current Operating License, Technical Specifications and Bases Pages
3. Retyped Renewed Facility License, Permanently Defueled Technical Specifications and Permanently Defueled Technical Specifications Bases Pages

cc: USNRC Regional Administrator, Region I
USNRC Project Manager, Pilgrim
USNRC Resident Inspector, Pilgrim
Planning and Preparedness Section Chief, Massachusetts Emergency Management Agency
Director, Massachusetts Department of Public Health, Radiation Control Program

Attachment 1

Letter Number 2.18.034

Description and Evaluation of Proposed Changes

Description and Evaluation of the Proposed Changes

1. SUMMARY DESCRIPTION

On November 10, 2015, Entergy Nuclear Operations, Inc. (ENO) notified the U.S. Nuclear Regulatory Commission (NRC) that it would permanently cease power operations at Pilgrim Nuclear Power Station (PNPS) no later than June 1, 2019 (Reference 1).

In accordance with Title 10 Code of Federal Regulations (CFR) 50.90, ENO is proposing an amendment to Renewed Facility Operating License (OL) DPR-35 for PNPS. This proposed license amendment would revise the OL and the associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

The proposed changes would revise certain requirements contained within the OL and TS and remove the requirements that would no longer be applicable after it has been certified that all fuel has permanently been removed from the PNPS reactor in accordance with 10 CFR 50.82(a)(1)(ii). After the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed for PNPS, the 10 CFR Part 50 license for PNPS will no longer authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). The proposed changes to the OL and TS are in accordance with 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes also include a renumbering of pages, where appropriate, to condense and reduce the number of pages in the TS without affecting the technical content. The TS Table of Contents is also accordingly revised.

The existing PNPS TS contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the facility being in a defueled condition. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational mode. LCOs and associated Surveillance Requirements (SRs) that will not apply in the permanently defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents associated with a defueled facility.

The changes proposed by this license amendment request would not be effective until the certification of permanent removal of fuel from the reactor vessel has been docketed by the NRC and the specified decay times established in the Fuel Handling Accident (FHA) have occurred.

In Reference 2, the NRC issued Amendment No. 246 to Renewed Facility Operating License No. DPR-35 for the PNPS. This amendment revises certain staffing and training requirements, reports, programs, and editorial changes contained in the TS Table of Contents; Section 1.0, "Definitions;" Section 4.0, "Design Features;" and Section 5.0, "Administrative Controls," that will no longer be applicable after Pilgrim is permanently defueled. This License Amendment Request reflects the implementation of those changes.

Description and Evaluation of the Proposed Changes

Pending Licensing Actions under NRC Review Which Affect This Request

None

2. DETAILED DESCRIPTION

The proposed amendment would modify the PNPS OL and revise PNPS TS into PDTS to comport with a permanently defueled condition.

General Analysis Applicable to Proposed Change

Chapter 14 of the PNPS Updated Final Safety Analysis Report (UFSAR) describes the design basis accident (DBA) and transient scenarios applicable to PNPS during power operations. During normal power operations, the forced inlet flow of water through the reactor coolant system (RCS) removes the heat from the reactor by generating steam. The steam system, operating at high temperatures and pressures, transfers this heat to the turbine generator. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products to the reactor coolant system. Many of the accident scenarios postulated in the UFSAR involve failures or malfunctions of systems which could affect the reactor core.

After the certifications are submitted for permanent cessation of operations and removal of fuel from the reactor vessel for PNPS in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and docketed pursuant to 10 CFR 50.82(a)(2), the majority of DBA scenarios postulated in the UFSAR will no longer be possible. The irradiated fuel will be stored in the Spent Fuel Pool (SFP) and the Independent Spent Fuel Storage Installation (ISFSI) until it is shipped off site in accordance with the schedules to be provided in the Post Shut Down Decommissioning Activities Report (PSDAR) and the Irradiated Fuel Management Plan.

10 CFR 50.36, "Technical Specifications," promulgates the regulatory requirements related to the content of Technical Specifications. As detailed in a subsequent section of this proposed amendment, this regulation lists four criteria to define the scope of equipment and parameters that must be included in TS. In a permanently defueled condition, the scope of equipment and parameters that must be included in the PNPS TS is limited to those needed to address the remaining postulated DBAs that will remain applicable to PNPS in the permanently shut down and defueled condition. These are the FHA and a radioactive waste handling event (i.e., High Integrity Container (HIC) drop event). This is to ensure that the consequences of the accident are maintained within acceptable limits.

High Integrity Container (HIC) Drop Event

HICs are used to contain dewatered solid wastes which include backwash sludge wastes from the Reactor Water Cleanup System; all spent resins and charcoal from the radwaste, SFP, and condensate demineralizers; and Thermex and radwaste filter/demineralizer. Although these types of wastes will no longer be on site after a period of time subsequent to cessation of power operations (they will no longer be generated), the assumed mix of radioisotopes and activity loading in the HIC is expected to bound source terms from all types of dewatered solid waste as well as dry solid wastes (rags, paper, small equipment parts, solid laboratory wastes, etc.) that may be stored onsite. Dewatered solid wastes contained in high integrity containers are placed in cylindrical, concrete storage modules and may be placed within the low-level radwaste storage facility (LLRWSF).

Calculation No. M1421 evaluates the drop of a HIC containing a bounding mix of radioisotopes onto another fully loaded HIC (Reference 3). No station structures, systems, or components were utilized to mitigate the consequences of the event.

Analytical Methodology

The release was assumed to be instantaneous and radiation doses were calculated for: 1) the total body due to cloud submersion; 2) a 2-hour direct shine dose from standing on contaminated ground; and 3) a 50-year Committed Effective Dose Equivalent (CEDE) to the total body based on the inhalation pathway. Radiation dose to the thyroid, based on the inhalation pathway, was also determined for a 50-year period following the intake of the radionuclides. The whole body and thyroid doses were based on the methodology and the applicable dose conversion factors from EPA Federal Guidance Reports No. 11 and No. 12 (References 4 and 5).

Atmospheric dispersion factors for inhalation and submersion doses were calculated for a ground level release based on guidance provided in Regulatory Guide 1.145 (Reference 6). Ground Deposition factors for the 2-hour direct shine dose from standing on contaminated ground was calculated for a ground level release based on guidance provided in Regulatory Guide 1.111 (Reference 7). Both atmospheric dispersion and ground deposition factors were determined using 5 years of PNPS meteorological data.

Assumptions

Sandia National Laboratory has conservatively estimated, for a severity Category 3 transportation accident (which includes 99% of urban and 94% of rural accidents), no more than 1% (0.01) of any package contents would be released (Reference 8). The velocity at impact of a dropped HIC with the ground or another HIC would be less than the velocity of impact for a Category 3 transportation accident. So, the material released due to a HIC drop is bounded by the material released due to a transportation accident.

A HIC is assumed to contain 945 Curies (Ci) of radionuclides with the isotopic mix shown in Table 1 The relative percentage of each isotope results in the bounding radiation dose from the three dose contributors established in the analytical methodology section.

The assumed liner drop is conservatively assumed to occur 100 meters from the exclusion area boundary (EAB) just inside the protected area. This is the limiting distance where HICs could potentially be located. Distances to the Radwaste Building Truck Lock (approximately 549 meters) and to the LLRWSF (approximately 305 meters) where the loading and processing of HICs and the subsequent storage of a loaded HIC typically occur result in lower doses at the site boundary because of the increased distance from the site boundary.

For the HIC drop accident, the dose acceptance criteria were set equal to "a small fraction" of the 10 CFR 100 dose limits of 25 rem whole body and 300 rem thyroid (i.e., to 10% of these values, or 2.5 rem whole body and 30 rem thyroid).

Other assumptions are contained in the footnotes in Table 1.

Inputs

The source term within each container in the HIC drop event is provided in Table 1 from Regulatory Guide 1.3 (Reference 9). The fraction of radioisotopes released in the assumed fire engulfing the released material from the HICs is 0.78% based on data from the U.S. Department of Energy (Reference 10).

Radiological Consequences

	10% of 10 CFR 100 Dose Acceptance Criteria (rem)	Calculated Dose (rem)
EAB (2-hour)	2.5 rem (whole body)	0.337
	30 rem (thyroid)	0.027

Table 1 - HIC Drop Source Term Release Activity

Nuclide[1]	Fraction (%)	Activity per HIC[2] (CI)	Liner Drop Release Activity[3] (CI)
C-14	0.01	9.45E-02	1.47E-05
Cr-51	2.69	2.54E+01	3.97E-03
Mn-54	1.48	1.40E+01	2.18E-03
Fe-55	41.20	3.89E+02	6.07E-02
Fe-59	0.45	4.25E+00	6.63E-04
Co-58	2.48	2.34E+01	3.66E-03
Co-60	39.60	3.74E+02	5.84E-02
Ni-59	0.01	9.45E-02	1.47E-05
Ni-63	3.91	3.69E+01	5.76E-03
Zn-65	0.30	2.84E+00	4.42E-04
Sr-89	0.06	5.67E-01	8.85E-05
Sr-90	0.04	3.78E-01	5.90E-05
Tc-99	0.03	2.84E-01	4.42E-05
Sb-124	0.50	4.73E+00	7.37E-04
Cs-134	0.12	1.13E+00	1.77E-04
Cs-137	6.87	6.49E+01	1.01E-02
Ce-144	0.15	1.42E+00	2.21E-04
Pu-238	0.01	9.45E-02	1.47E-05
Pu-239/240[4]	0.01	9.45E-02	1.47E-05
Pu-241	0.45	4.25E+00	6.63E-04
Am-241	0.02	1.89E-01	2.95E-05

Footnotes:

1. Nuclide listing – Radionuclide mix that bounds dose consequences of mixes determined by laboratory analysis to be present in dewatered solid wastes. Short lived gaseous and volatile radionuclides are not detected in typical radwaste streams.

2. Activity per HIC - The assumed total activity with each HIC in the drop event is 945 Ci. The individual activity for each radionuclide is determined based on the fraction present in the assumed mix.

3. Release activity – The quantity of each radionuclide assumed to be release from the HIC drop event. The release activity is based on: A) HIC is dropped onto another loaded HIC and a release of 1% of the total contents of the two HICs occurs; and B) Of the 1% of the material released, 0.78% is aerosolized to from a "release cloud" source term.

4. In the calculation of EAB doses, 0.01% of both Pu-239 and Pu-240 is included in the release for conservatism.

Fuel Handling Accident Analysis for the Permanently Shut down and Defueled Condition

Summary

On April 28, 2005, the NRC issued License Amendment No. 215 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. The amendment adopted Technical Specifications Task Force Traveler (TSTF-51), "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," and selectively implemented an alternative source term (AST) per 10 CFR 50.67 to perform the radiological consequences analysis of the design-basis FHA to support the changes to the Technical Specifications (Reference 33). The analysis of the FHA that supported these changes did not take credit for secondary containment isolation or filtration by the Standby Gas Treatment System (SGTS) or the Control Room High Efficiency Air Filtration System (CRHEAFS), and assumed the FHA occurred 24 hours after reactor shutdown from full power.

After the reactor has been completely defueled following permanent shut down, an FHA in the reactor cavity is no longer a credible accident. Calculation No. M1422 (Reference 11) concludes that the consequences of a drop of a channeled fuel assembly in the SFP after permanent shut down, i.e., the calculated Total Effective Dose Equivalent (TEDE) values to the CR, EAB, and Low Population Zone (LPZ), are less than the limits set forth in 10 CFR 50.67 and Regulatory Guide 1.183. The analysis assumes: 1) A minimum period of decay of 24 hours before a channeled fuel assembly can be handled; 2) The subsequent drop of a channeled fuel assembly in the SFP; 3) An open Reactor Building with no filtration by the SGTS; and 4) No credit for operation of the Control Room High Efficiency Air Filtration System. This analysis is essentially the same as the analysis that was previously reviewed by the NRC as part of License Amendment No. 215 with the exception of the location of the event.

Before an unchanneled fuel assembly can be handled, the fuel handling procedure requires an additional 45 days of decay beyond the assumed decay period of 24 hours in the design basis FHA (i.e., 46 days of decay (24 hours of decay assumed in the analysis of the FHA + an additional 45 days of decay)). This additional decay time ensures that the consequences of the drop of an unchanneled fuel assembly in the SFP are bounded by the design basis FHA (Reference 11). This is based on a generic analysis of the dose consequences of a drop of an unchanneled fuel assembly in the SFP contained in Reference 13.

Additionally, Reference 11 determined that a 72-hour minimum decay time prior to fuel movement of a channeled fuel assembly would result in the EAB TEDE dose not exceeding the EPA Protective Action Guide (PAG) limit of 1 rem for evacuation (Reference 14).

Fuel Damage

The number of rods assumed failed in an FHA for GE11, GE14, and GNF2 fuel assemblies are obtained from the GESTAR Amendment 22 Reports. For GE11 assemblies, 123 rods are calculated to fail per Reference 15. For GE14 assemblies, 151 rods are calculated to fail per Reference 16. For GNF2 assemblies, 150 rods are calculated to fail per Reference 17. The number of fuel rods calculated to fail for GE14 and GNF2 bound the anticipated fuel rod failures for earlier fuel types (7x7, 8x8, 9x9 lattices). The cladding failure threshold energy is lower for 10x10 designs, compared to earlier designs, due to thinner cladding. Also, older fuel types present in the SFP will be less limiting from a source term perspective given the longer decay time.

Method and Assumptions

The FHA analysis uses the Alternative Source Term (AST) Methodology outlined in NUREG-1465 (Reference 18), Regulatory Guide 1.183 (Reference 19), Regulatory Guide 1.145 (Reference 10), and Regulatory Guide 1.194 (Reference 20).

The following assumptions and initial conditions are used in calculating the fission product release to the environment:

a) The accident is assumed to occur 24 hours after shut down. An evaluation is also performed to show that a decay time of at least 72 hours is sufficient to meet the EPA PAG limit of 1 rem at the EAB for evacuation. After permanent shut down, the decay time for bundles in the SFP will be longer than the assumed 24 or 72 hours.

b) The fuel assembly is dropped from 32.95 feet (the maximum height allowed by the fuel handling equipment over the reactor core). This drop height bounds the significantly lower drop height over the SFP.

c) The FHA results in 151 fuel rods failing, and the release to the environment from the refueling floor occurs within 2 hours.

d) The decontamination factor (DF) of 200 was assumed for the scrubbing effects of water on halogen activity release. The DF was based on a minimum of 23 feet of water over the dropped assembly. No DF was applied to noble gases and the DF for other radionuclides were assumed to be infinite.

e) The core inventory was based on a thermal power level of 2028 megawatt-thermal (MWt), plus a measurement uncertainty of 0.5% (2038 MWt). A radial peaking factor (RPF) of 2.1 was used, which is significantly higher than the generically assumed steady state operation RPF of 1.7 for GE14 and GNF2 assemblies. The bounding core and FHA inventories are given in Table 2.

f) All activity within the gaps of the failed fuel rods is released to the refueling cavity (or SFP) water. The released activity corresponds to 8% of the entire inventory of I-131 in the rods, 10% of the Kr-85, 5% of the remaining halogens and noble gases, and 12% of the alkalis (Cs and Rb).

g) The reactor building is assumed to be open when the FHA occurs, with normal ventilation on, such that all releases to the environment would be via the reactor building vent.

h) 5 years of hourly meteorological data was used for atmospheric dispersion factors shown in Table 3.

i) The control room ventilation system was assumed to remain in the normal operating mode during the entire exposure interval (30 days).

j) Breathing rates, and control room occupancy factors, are as given in Regulatory Guide 1.183 (Reference 19).

k) The dose conversion factors used are from Federal Guidance Reports 11 and 12 (References 4 and 5).

l) The control room air intake rate was assumed to be 1000 cubic feet per minute (cfm) (a low value) and 9000 cfm (a high value).

Drop of an Unchanneled Fuel Assembly

A generic analysis of the dose consequences of a drop of an unchanneled fuel assembly in the SFP was performed (Reference 13). The limiting scenario postulates that the unchanneled assembly is dropped, impacts assemblies in the rack, and subsequently strikes the SFP wall and remains upright. In this scenario, though fewer total rods are calculated to be damaged compared to a drop over the core due to the lower drop height, a number of rods are assumed to fail at the top of the assembly that strikes the SFP wall. This leads to a release of radionuclides at a pool depth of less than 23 feet, which means the assumed decontamination factor for the pool water of 200 would be significantly less. Reference 13 calculates a net increase in the dose consequences relative to the design basis FHA over the core. To counteract the increase in dose consequences an additional decay time of 45 days is recommended on top of what is assumed in the design basis FHA (24 hours). The additional 45 days of decay results in a net reduction in the dose to approximately 40% of the design basis dose.

The additional decay time of 45 days of decay beyond the assumed decay period of 24 hours in the design basis FHA (i.e., 46 days of decay (24 hours of decay assumed in the analysis of the FHA + an additional 45 days of decay)) ensures that the design basis FHA over the core remains bounding. No additional decay beyond 46 days is required to meet the EPA PAG limit of 1 rem at the EAB for the drop of an unchanneled assembly in the SFP due to the magnitude of dose reduction provided by the additional 45 days beyond the assumed decay period of 24 hours in the design basis FHA. Specifically, 40% of the design basis dose at the EAB (1.439 rem) is 0.576 rem.

<u>Radiological Consequences</u>

The radiological consequences of the postulated FHA are as follows:

Location	Exposure Interval	Unfiltered Outside Air Intake Rate (cfm)	TEDE Dose (rem)	Regulatory Limit (rem)	Percent of Regulatory Limit
Control Room	30 days	1000	2.846	5	56.9
		9000	2.863	5	57.3
EAB (24-hour decay)	2 hours	N/A	1.439	6.3	22.8
EAB (72-hour decay)	2 hours	N/A	0.910	1.0[a]	91.0
LPZ	30 days	N/A	0.0920	6.3	1.46
Footnote a – EPA PAG Limit before evacuation					

The calculated TEDE values to the CR, EAB, and LPZ are less than the limits set forth in 10 CFR 50.67 and Regulatory Guide 1.183.

A decay time of at least 72 hours prior to fuel movement ensures that the TEDE dose at the EAB will be less than the EPA PAG recommended threshold for evacuation of 1 rem.

The administrative restriction that prevents movement of an unchanneled fuel assembly prior to 46 days of decay (24 hours of decay assumed in the analysis of the FHA + an additional 45 days of decay) post-shut down ensures that the design basis results presented above remain bounding.

Table 2 – Bounding Core and FHA Inventories

Radionuclide	Undecayed Inventory (Ci)		Fuel Rod Gap Fraction	FHA Undecayed Source Term (Ci)
	Full Core	Peak Assembly		
BR-82	6.872E+05	2.488E+03	0.05	2.042E+02
BR-82M	2.656E+05	9.617E+02	0.05	7.892E+01
BR-83	8.640E+06	3.128E+04	0.05	2.567E+03
BR-84	1.593E+07	5.768E+04	0.05	4.733E+03
BR-84M	4.468E+05	1.618E+03	0.05	1.328E+02
BR-85	1.957E+07	7.086E+04	0.05	5.815E+03
BR-86	1.466E+07	5.308E+04	0.05	4.356E+03
BR-87	3.339E+07	1.209E+05	0.05	9.921E+03
BR-88	3.803E+07	1.377E+05	0.05	1.130E+04
I-128	1.919E+06	6.948E+03	0.05	5.702E+02
I-129	6.033E+00	2.184E-02	0.05	1.793E-03
I-130	4.655E+06	1.685E+04	0.05	1.383E+03
I-130M	1.818E+06	6.582E+03	0.05	5.402E+02
I-131	5.716E+07	2.070E+05	0.08	2.717E+04
I-132	8.113E+07	2.937E+05	0.05	2.411E+04
I-133	1.150E+08	4.164E+05	0.05	3.417E+04
I-134	1.284E+08	4.649E+05	0.05	3.815E+04
I-134M	1.371E+07	4.964E+04	0.05	4.074E+03
I-135	1.071E+08	3.878E+05	0.05	3.182E+04
I-136	5.198E+07	1.882E+05	0.05	1.544E+04
I-136M	3.179E+07	1.151E+05	0.05	9.446E+03
KR-83M	8.638E+06	3.128E+04	0.05	2.567E+03
KR-85	1.439E+06	5.210E+03	0.10	8.551E+02
KR-85M	1.979E+07	7.165E+04	0.05	5.880E+03
KR-87	3.956E+07	1.432E+05	0.05	1.175E+04
KR-88	5.592E+07	2.025E+05	0.05	1.662E+04
KR-89	7.054E+07	2.554E+05	0.05	2.096E+04
KR-90	7.004E+07	2.536E+05	0.05	2.081E+04
XE-131M	6.412E+05	2.322E+03	0.05	1.905E+02
XE-133	1.150E+08	4.164E+05	0.05	3.417E+04
Xe-133M	3.541E+06	1.282E+04	0.05	1.052E+03
XE-135	5.869E+07	2.125E+05	0.05	1.744E+04
XE-135M	2.297E+07	8.317E+04	0.05	6.825E+03
XE-137	1.012E+08	3.664E+05	0.05	3.007E+04
XE-138	1.022E+08	3.700E+05	0.05	3.037E+04
XE-139	8.237E+07	2.982E+05	0.05	2.447E+04

Table 3 – Atmospheric Dispersion Factors (X/Qs) for the Reactor Building Vent Release Point

Receptor Point	Interval	Concentration X/Q (sec/m^3)	Gamma X/Q (sec/m^3)
EAB (actual)*	0-2 hours	7.479E-04	3.199E-04
LPZ (4.25 miles)	0-2 hours	3.692E-05	3.551E-05
	2-8 hours	1.915E-05	1.782E-05
	8-24 hours	1.066E-05	9.627E-05
	24-96 hours	4.339E-06	3.745E-05
	96-720 hours	1.194E-06	9.656E-07
Control Room Fresh Air Intake	0-2 hours	1.76E-03	Not Applicable
	2-8 hours	1.25E-03	
	8-24 hours	4.26E-04	
	24-96 hours	3.67E-04	
	96-720 hours	3.15E-04	

* The EAB distances employed in the atmospheric dispersion analysis are from the closest point of the Reactor Building; as such, they conservatively apply to releases via the Reactor Building vent, which is at the plant Southwest (SW) corner. The critical receptor is in the true Northeast (NE) sector, at a distance of 486 meters, (at the over-water exclusion zone).

Description and Evaluation of the Proposed Changes

3.0 Technical Evaluation

The following tables identify each section that is proposed to be changed, the proposed changes, and the basis for each change. Changes to the OL are listed first followed by the changes to the TS.

Attachment 2 provides the marked-up version of the PNPS OL, TS, and TS Bases to establish the changes. Additionally, the proposed changes to the TS are considered a major rewrite. Thus, the TS that are deleted in their entirety are identified as such, but the associated deleted pages are not included in Attachment 2. In addition, the following administrative changes are not shown in the marked-up (Attachment 2) OL, TS, and TS Bases pages, because they do not affect the technical content of the OL or TSs:

- Reformatting (margins, font, tabs, line spacing, etc.) content to create a continuous electronic file; and
- Renumbering of pages, where appropriate, to condense and reduce the number of pages.

Attachment 3 provides the re-typed Renewed Facility License, PDTS, and PDTS Bases pages in their entirety. Since the changes to the TS are considered a major rewrite, revision bars are not used.

The mark-ups of the TS Bases and retyped versions of the PDTS Bases are provided for information only. Upon approval of this amendment, changes to the TS Bases will be incorporated in accordance with TS 5.5.6, "Technical Specifications Bases Control Program."

License Title	
<u>Current Title</u> Renewed Facility Operating License	<u>Proposed Title</u> Renewed Facility ~~Operating~~ License
Basis	
The License Title is modified to eliminate the reference to "Operating." After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

License Finding a	
<u>Current License Finding a</u> Except as stated in condition 5, construction of the Pilgrim Nuclear Power Station (the facility) has been substantially completed in conformity with the application, as amended, the Provisional Construction Permit No. CPPR-49, the provisions of the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission as set forth in Title 10, Chapter 1, CFR; and	<u>Proposed License Finding a</u> ***DELETED***

Basis
This license finding is proposed for deletion in its entirety. Decommissioning of PNPS is not dependent on the regulations that govern construction of the facility.

License Finding b	
Current License Finding b The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; and	Proposed License Finding b The facility will ~~operate~~ ***be maintained*** in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; and
Basis	
This license finding is revised to reflect a more accurate description of the future requirements. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Thus, replacing the verb "operate" with the verb "be maintained" will provide accuracy regarding the possession-only 10 CFR Part 50.	

License Finding c	
Current License Finding c There is reasonable assurance (i) that the activities authorized by the renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission; and	Current License Finding c There is reasonable assurance (i) that the activities authorized by the renewed ~~operating~~ license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission; and
Basis	
This license finding is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

<table>
<tr><th colspan="2">License Finding d</th></tr>
<tr><td>Current License Finding d
The Entergy Nuclear Generation Company (Entergy Nuclear) is financially qualified and Entergy Nuclear Operations, Inc. (ENO) is technically and financially qualified to engage in the activities authorized by this renewed operating license, in accordance with the rules and regulations of the Commission; and</td><td>Proposed License Finding d
The Entergy Nuclear Generation Company (Entergy Nuclear) is financially qualified and Entergy Nuclear Operations, Inc. (ENO) is technically and financially qualified to engage in the activities authorized by this renewed ~~operating~~ license, in accordance with the rules and regulations of the Commission; and</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license finding is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).</td></tr>
</table>

<table>
<tr><th colspan="2">License Finding f</th></tr>
<tr><td>Current License Finding f
The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public; and</td><td>Proposed License Finding f
The issuance of this renewed ~~operating~~ license will not be inimical to the common defense and security or to the health and safety of the public; and</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license finding is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).</td></tr>
</table>

<table>
<tr><th colspan="2">License Finding g</th></tr>
<tr><td>Current License Finding g
After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of this renewed operating license (subject to the condition for protection of the environment set forth herein) is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements of said regulations have been satisfied; and</td><td>Proposed License Finding g
After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of this renewed ~~operating~~ license (subject to the condition for protection of the environment set forth herein) is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements of said regulations have been satisfied~~; and~~.</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license finding is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).</td></tr>
</table>

License Finding h	
Current License Finding h Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21 (a)(1); and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21 (c), such that there is reasonable assurance that the activities authorized by the renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.	Proposed License Finding h ***DELETED***
Basis	
This license finding is deleted in its entirety. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). PNPS will not operate during the remaining period of extended operation. Decommissioning of PNPS is not dependent on the requirements of 10 CFR 54 for a renewed license. Therefore, requirements that are unique to a renewed license are not needed.	

License Condition 1	
Current License Condition 1 This renewed operating license applies to the Pilgrim Nuclear Power Station, a single cycle, forced circulation, boiling water nuclear reactor and associated electric generating equipment (the facility), owned by Entergy Nuclear and operated by ENO. The facility is located on the western shore of Cape Cod Bay in the town of Plymouth on the Entergy Nuclear site in Plymouth County, Massachusetts, and is described in the "Final Safety Analysis Report," as supplemented and amended.	Proposed License Condition 1 This renewed ~~operating~~ license applies to the Pilgrim Nuclear Power Station, a single cycle, forced circulation, boiling water nuclear reactor and associated electric generating equipment (the facility), owned by Entergy Nuclear and ~~operated~~ ***maintained*** by ENO. The facility is located on the western shore of Cape Cod Bay in the town of Plymouth on the Entergy Nuclear site in Plymouth County, Massachusetts, and is described in the "Final Safety Analysis Report," as supplemented and amended.
Basis	
This license condition is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

Description and Evaluation of the Proposed Changes

<table>
<tr><th colspan="2">License Condition 2.A</th></tr>
<tr><td><u>Current License Condition 2.A</u>
Pursuant to the Section 104b of the Atomic Energy Act of 1954, as amended (the Act) and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," a) Entergy Nuclear to possess and use and b) ENO to possess, use, and operate the facility as a utilization facility at the designated location on the Pilgrim site;</td><td><u>Proposed License Condition 2.A</u>
Pursuant to the Section 104b of the Atomic Energy Act of 1954, as amended (the Act) and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," a) Entergy Nuclear to possess and use and b) ENO to possess<s>,</s> and use<s>, and operate</s> the facility <s>as a utilization facility</s> at the designated location on the Pilgrim site;</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license condition is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).</td></tr>
</table>

<table>
<tr><th colspan="2">License Condition 2.B</th></tr>
<tr><td><u>Current License Condition 2.B</u>
ENO, pursuant to the Act and 10 CFR 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;</td><td><u>Proposed License Condition 2.B</u>
ENO, pursuant to the Act and 10 CFR 70, to <s>receive,</s> possess<s>, and use</s> at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage <s>and amounts required for reactor operation,</s> as described in the Final Safety Analysis Report, as supplemented and amended;</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license condition is revised to remove the authorization for receipt and use of special nuclear material (SNM) as reactor fuel, eliminate the reference to use of the SNM for reactor operations, and limits the possession of SNM to SNM "that was used" as reactor fuel at PNPS. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As such, PNPS has no need to receive SNM in the form of reactor fuel and cannot use SNM as reactor fuel for reactor operations. The continued authorization to possess SNM "that was used" as reactor fuel is necessary as PNPS currently possesses the reactor fuel that was used for the past operations of the reactor.</td></tr>
</table>

Description and Evaluation of the Proposed Changes

<table>
<tr><th colspan="2">License Condition 2.C</th></tr>
<tr><td><u>Current License Condition 2.C</u>
ENO, pursuant to the Act and 10 CFR Parts 30,40 and 70 to receive, possess and use at any time any byproduct, source or special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;</td><td><u>Proposed License Condition 2.C</u>
ENO, pursuant to the Act and 10 CFR Parts 30,40 and 70 to receive, possess and use at any time any byproduct, source or special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in radiation monitoring equipment <s>calibration</s>, and as fission detectors in amounts as required;</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license condition is revised to remove the authorization for receipt and use of byproduct, source, and SNM as sealed neutron sources for reactor startup. The deletion of the authorization to receive and use sources for reactor startup is consistent with the fact that PNPS will no longer be authorized to operate.
The authorization to possess such sources previously used for reactor startup is retained. The continued authorization to possess neutron sources that were used for reactor startup is consistent with the safe storage of byproduct, source, and SNM. The use of sources for radiation monitoring will continue to be required.
After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). These changes are consistent with the permanently defueled condition.</td></tr>
</table>

<table>
<tr><th colspan="2">License Condition 3</th></tr>
<tr><td><u>Current License Condition 3</u>
This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations; 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50 and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:</td><td><u>Proposed License Condition 3</u>
This renewed <s>operating</s> license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations; 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50 and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license condition is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).</td></tr>
</table>

License Condition 3.A, Maximum Power Level	
Current License Condition 3.A ENO is authorized to operate the facility at steady state power levels not to exceed 2028 megawatts thermal.	Proposed License Condition 3.A ***DELETED***
Basis	
This license condition is deleted in its entirety to reflect the permanently defueled condition of the facility. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

License Condition 3.B, Technical Specifications	
Current License Condition 3.B The Technical Specifications contained in Appendix A. as revised through Amendment No. 247, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.	Proposed License Condition 3.B The Technical Specifications contained in Appendix A. as revised through Amendment No. ~~247~~ ***###***, are hereby ***replaced with the Permanently Defueled Technical Specifications*** ~~incorporated in the renewed operating license~~. The licensee shall ~~operate~~ ***maintain*** the facility in accordance with the ***Permanently Defueled*** Technical Specifications.
Basis	
This license condition is revised to account for the permanently defueled condition of the facility and to incorporate a reference to the PDTS. These nomenclature changes will more accurately describe the remaining TS. Also, the verb "operate" is replaced with the verb "maintained" to better describe the permanently defueled condition. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

Description and Evaluation of the Proposed Changes

License Condition 3.C, Records	
Current License Condition 3.C ENO shall keep facility operating records in accordance with the requirements of the Technical Specifications.	Proposed License Condition 3.C ENO shall keep facility ~~operating~~ records in accordance with the requirements of the Technical Specifications.
Basis	
This license condition is revised to reflect that after the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

License Condition 3.D, Equalizer Valve Restriction	
Current License Condition 3.D Equalizer Valve Restriction - DELETED	Proposed License Condition 3.D ~~Equalizer Valve Restriction -~~ DELETED
Basis	
This license condition is revised to eliminate the title. This is an editorial change, because the content of the license condition was previously deleted.	

License Condition 3.E, Recirculation Loop Inoperable	
Current License Condition 3.E Recirculation Loop Inoperable - DELETED	Proposed License Condition 3.E ~~Recirculation Loop Inoperable -~~ DELETED
Basis	
This license condition is revised to eliminate the title. This is an editorial change, because the content of the license condition was previously deleted.	

License Condition 3.F, Fire Protection	
Current License Condition 3.F ENO shall implement and maintain in effect all provisions of the approved fire protections program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated December 21, 1978 as supplemented subject to the following provision: ENO may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shut down in the event of a fire.	Proposed License Condition 3.F ***DELETED***

Basis
This license condition is deleted to reflect the permanently defueled condition of the facility. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, the fire protection program will be revised to take into account the decommissioning facility conditions and activities. PNPS will continue to utilize the defense-in-depth concept, placing special emphasis on detection and suppression in order to minimize radiological releases to the environment. This condition, which is based on maintaining an operational fire protection program in accordance with 10 CFR 50.48, with the ability to achieve and maintain safe shut down of the reactor in the event of a fire, will no longer be applicable at PNPS. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during facility decommissioning. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. However, the regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shut down and defueled facility is not needed.

License Condition 3.H, Post-Accident Sampling System. NUREG-0737, Item II.B.3. and Containment Atmospheric Monitoring System, NUREG-0737. Item II.F.1(6)	
Current License Condition 3.H The licensee shall complete the installation of a post-accident sampling system and a containment atmospheric monitoring system as soon as practicable, but no later than June 30, 1985.	Proposed License Condition 3.H ***DELETED***
Basis	
This license condition is proposed for deletion to reflect the permanently defueled condition of the facility. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, the post-accident sampling system and containment atmospheric monitoring system will not be required to perform a function in the permanently defueled condition.	

License Condition 3.I, Additional Conditions	
Current License Condition 3.I The Additional Conditions contained in Appendix B, as revised through Amendment No. 177, are hereby incorporated into this renewed operating license. ENO shall operate the facility in accordance with the Additional Conditions.	Proposed License Condition 3.I ***DELETED***

Basis
This license condition is proposed for deletion in its entirety. As discussed below, the conditions contained within Appendix B will no longer be applicable after PNPS is in the permanently defueled condition.

License Condition 3.M	
<u>Current License Condition 3.M</u> Upon Implementation of Amendment No. 231 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage required by SR 4.7.6.2.e in accordance with TS 5.5.8.c.(i), the assessment of CRE habitability as required by Specification 5.5.8.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.8.d shall be considered met as follows: (a) The first performance of SR 4.7.2.6.5.e in accordance with Specification 5.5.8.c.(i) shall be within the specified frequency of 6 years, plus the 18- month allowance as defined by SURVEILLANCE INTERVAL measured from December 5, 2005, the date of the most recent successful tracer gas test, as stated in Entergy's letter "Follow-Up Response to NRC Generic Letter 2003-01" (ENO 2.06.019), dated March 20, 2006, or within 18 months if the time period since the most recent successful tracer gas test is greater than 6 years. (b) The first performance of the periodic assessment of CRE habitability Specification 5.5.8.c.(ii) shall be within 3 years, plus the 9-month allowance of SURVEILLANCE INTERVAL as measured from December 5, 2005, the date of the most recent successful tracer gas test, as stated in Entergy's letter "Follow-Up Response to NRC Generic Letter 2003-01" (ENO 2.06.019), dated March 20, 2006, or within 9 months if the time period since the most recent successful tracer gas test is greater than 3 years. (c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.8.d shall be within 24 months, plus the 180-day allowance of the SURVEILLANCE INTERVAL as measured from the date of the most recent successful pressure measurement test or within 180 days if not performed previously.	<u>Proposed License Condition 3.M</u> ***DELETED***

Basis
This license condition is deleted in its entirety. The license condition defined requirements of TSTF-448 to assess the Control Room Envelope (CRE) Habitability at the specified frequencies for the first performance of the specific test, assessment, and measurement. This is a historical license condition, because the test, assessment, and measurement were completed in accordance with the schedule specified in the license condition.

License Condition 4	
Current License Condition 4 This license is subject to the following condition for the protection of the environment: Boston Edison shall continue, for a period of five years after initial power operation of the facility, an environmental monitoring program similar to that presently existing with the Commonwealth of Massachusetts (and described generally in Section C-111 of Boston Edison's Environmental Report, Operating License Stage dated September, 1970) as a basis for determining the extent of station influence on marine resources and shall mitigate adverse effects, if any, on marine resources.	Proposed License Condition 4 ***DELETED***
Basis	
This license condition is revised to remove historical specified actions that have been completed and are not required to support facility operations in the permanently defueled condition.	

License Condition 5	
Current License Condition 5 Boston Edison has not completed as yet construction of the Rad Waste Solidification System and the Augmented Off-Gas System. Limiting conditions concerning these systems are set forth in the Technical Specifications.	Proposed License Condition 5 ***DELETED***
Basis	
This license condition is revised to remove historical specified actions that have been completed and are not required to support facility operations in the permanently defueled condition.	

Description and Evaluation of the Proposed Changes

License Condition 6	
<u>Current License Condition 6</u> Pursuant to Section 105c(8) of the Act, the Commission has consulted with the Attorney General regarding the issuance of this operating license. After said consultation, the Commission has determined that the issuance of this license, subject to the conditions set forth in this subparagraph 6, in advance of consideration of and findings with respect to matters covered in Section 105c of the Act, is necessary in the public interest to avoid unnecessary delay in the operation of the facility. At the time this operating license is being issued an antitrust proceeding has not been noticed. The Commission, accordingly, has made no determination with respect to matters covered in Section 105c of the Act, including conditions, if any, which may be appropriate as a result of the outcome of any antitrust proceeding. On the basis of its findings made as a result of an antitrust proceeding, the Commission may continue this license as issued, rescind this license or amend this license to include such conditions as the Commission deems appropriate. Boston Edison and others who may be affected hereby are accordingly on notice that the granting of this license is without prejudice to any subsequent licensing action, including the imposition of appropriate conditions, which may be taken by the Commission as a result of the outcome of any antitrust proceeding. In the course of its planning and other activities, Boston Edison will be expected to conduct itself accordingly.	<u>Proposed License Condition 6</u> ***DELETED***
Basis	
This license condition is revised to remove historical specified actions that have been completed and are not required to support facility operations in the permanently defueled condition.	

Description and Evaluation of the Proposed Changes

License Condition 7	
Current License Condition 7 The information in the FSAR supplement, submitted pursuant to 10 CFR 54.21 (d), as supplemented by Commitments Nos. 3, 8, 9, 13, 15, 18, 19, 21, 22, 24, 25, 26, 27, 28, 30, 31, 33, 34, 35, 36, 37, 39, 40, 46, 51, and 52 of Appendix A of NUREG-1891, "Safety Evaluation Report Related to the License Renewal of Pilgrim Nuclear Power Station" dated June 2007, as supplemented, is henceforth part of the FSAR which will be updated in accordance with 10 CFR 50.71(e). In addition, the licensee shall incorporate into its FSAR the "Description of Program" from Table 3.0-1 "FSAR Supplement for Aging Management of Applicable Systems" of License Renewal Interim Staff Guidance LR-ISG-2011-05 "Ongoing Review of Operating Experience." The licensee may make changes to the programs and activities described in the FSAR supplement and Commitments Nos. 3, 8, 9, 13, 15, 18, 19, 21, 22, 24, 25, 26, 27, 28, 30, 31, 33, 34, 35, 36, 37, 39, 40, 46, 51, and 52 of Appendix A of NUREG-1891, as supplemented, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.	Proposed License Condition 7 The information in the FSAR supplement, submitted pursuant to 10 CFR 54.21 (d), as supplemented by Commitments Nos. 3, 8, 9, 13, 15, 18, 19, 21, 22, 24, 25, 26, 27, 28, 30, 31, 33, 34, 35, 36, 37, 39, 40, 46, 51, and 52 of Appendix A of NUREG-1891, "Safety Evaluation Report Related to the License Renewal of Pilgrim Nuclear Power Station" dated June 2007, as supplemented, is henceforth part of the FSAR which will be updated in accordance with 10 CFR 50.71(e). ~~In addition, the licensee shall incorporate into its FSAR the "Description of Program" from Table 3.0-1 "FSAR Supplement for Aging Management of Applicable Systems" of License Renewal Interim Staff Guidance LR-ISG-2011-05 "Ongoing Review of Operating Experience."~~ The licensee may make changes to the programs and activities described in the FSAR supplement and Commitments Nos. 3, 8, 9, 13, 15, 18, 19, 21, 22, 24, 25, 26, 27, 28, 30, 31, 33, 34, 35, 36, 37, 39, 40, 46, 51, and 52 of Appendix A of NUREG-1891, as supplemented, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.
Basis	
This license condition is modified to remove a historical specified action that has been completed.	

License Condition 8	
Current License Condition 8 The licensee's FSAR supplement submitted pursuant to 10 CFR 54.21 (d), as revised during the license renewal application review process, and as supplemented by Commitments Nos. 3, 8, 9, 13, 15, 18, 19, 21, 22, 24, 25, 26, 27, 28, 30, 31, 33, 34, 35, 36, 37, 39, 40, 46, 51, and 52 of Appendix A of NUREG-1891, as supplemented, along with the FSAR description regarding consideration of operating experience for license renewal aging management programs in Condition 7 above, describes certain future programs and activities to be completed before the period of extended operation. The licensee shall complete these activities no later than June 8, 2012, and shall notify the NRC in writing when implementation of these activities is complete.	Proposed License Condition 8 ***DELETED***
Basis	
This license condition is revised to remove historical specified actions that have been completed. On June 8, 2012, ENO notified the NRC of the completion of the implementation of these license renewal activities with a couple of exceptions regarding Condensate Storage Tank "A" testing and neutron absorber testing of Metamic (Reference 25). On October 18, 2012, ENO notified the NRC of the completion of the implementation of the activities associated with Condensate Storage Tank "A" testing and neutron absorber testing of Metamic (Reference 26).	

License Condition 9	
Current License Condition 9 Capsule withdrawal schedule - For the renewed operating license term, all capsules in the reactor vessel that are removed and tested must meet the requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the staff prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the staff, as required by 10 CFR Part 50, Appendix H.	Proposed License Condition 9 ***DELETED***

Basis
10 CFR 50 Appendix H requires that the design of the reactor vessel surveillance capsule program and withdrawal schedule must meet the requirements in the version of ASTM Standard Practice E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Code to which the reactor pressure vessel (RPV) was purchased. The rule also requires the licensee to perform capsule testing and to report the test results in accordance with the requirements in ASTM Standard Practice E 185-82 to the extent practicable for the configuration of the test specimens in the RPV surveillance capsules. The requirements in Appendix H are only applicable to nuclear plants that are performing power operations in the reactor critical operating mode because: (a) this is the plant operating mode that produces high energy neutrons as a result of the reactor's nuclear fission process; and (b) the requirements are set in place to provide assurance that the RPV will maintain adequate levels of fracture toughness throughout the operating life of the reactor. Continued implementation of the applicable surveillance capsule testing and reporting requirements are no longer necessary for PNPS because: (a) ENO has decided to cease power operations of PNPS; and (b) from a fracture toughness perspective, the PNPS RPV will cease to be exposed to further irradiation by high energy neutrons or subjected to any high thermal stress environments, as induced by operating the RCS at an elevated temperature. The physical and radiological control of the remaining surveillance capsules that are located in the RPV will be managed in accordance with the applicable radiological control requirements of 10 CFR Part 20 and with any applicable security or physical protection requirements for components in either 10 CFR Part 37 or 10 CFR Part 73. Therefore, the removal, testing, reporting, and storage requirements for reactor vessel surveillance capsules and their test specimens do not need to be implemented further after PNPS permanently ceases power operations because there will no longer be any need to remove the remaining surveillance capsules from the RPV or perform material testing of the test specimens in those capsules. As such, deletion of this license condition is appropriate. Any corresponding commitments in the PNPS UFSAR will also be deleted under the provisions of 10 CFR 50.59 upon NRC approval of this change.

License Condition 10	
Current License Condition 10 This license is effective as of the date of issuance and shall expire June 8, 2032.	Proposed License Condition 10 This license is effective as of the date of issuance and ~~shall expire June 8, 2032~~ ***until the Commission notifies the licensee in writing that the license is terminated***.
Basis	
After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Thus, this license condition is revised to conform with 10 CFR 50.51, "Continuation of license," in that the license authorizes ownership and possession by Entergy Nuclear until the Commission notifies the licensee in writing that the license is terminated.	

Attachments	
Current Attachments Attachments: Appendix A – Technical Specifications (Radiological) Appendix B – Additional Conditions Date of Issuance: May 29, 2012	Proposed Attachment Attachment~~s~~: Appendix A – ***Permanently Defueled*** Technical Specifications (Radiological) ~~Appendix B – Additional Conditions~~ Date of Issuance: ~~May 29, 2012~~***TBD***
Basis	
The list of attachments is modified to reflect the renaming of the Technical Specifications as the Permanently Defueled Technical Specifications, elimination of Appendix B, and the modification of the date of issuance to reflect the date that the NRC issues the PDTS that is yet to be determined. These are administrative changes.	

APPENDIX A TO FACILITY OPERATING LICENSE DPR-35	
Current Title Facility Operating License DPR-35 Technical Specification and Bases	Proposed Title Facility ~~Operating~~ License DPR-35 ***Permanently Defueled*** Technical Specification***s*** and Bases
Basis	
The License Title is modified to rename the “Facility Operating License DPR-35 Technical Specification and Bases” as “Facility License DPR-35 Permanently Defueled Technical Specifications and Bases.” These changes reflect the upcoming change in status regarding the PNPS. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).	

<table>
<tr><th colspan="2">APPENDIX B – ADDITIONAL CONDITIONS</th></tr>
<tr><td><u>Current Appendix B</u>
Entergy Nuclear Operations, Inc. shall comply with the following conditions on the schedules noted below:
<u>Amendment Number</u>
177
<u>Additional Conditions</u>
The licensee is authorized to relocate certain Technical Specifications requirements to licensee-controlled documents. Implementation of this amendment shall include relocation of various sections of the technical specifications to the appropriate documents as described in the licensee's application dated September 19, 1997, and in the staff's safety evaluation attached to this amendment.
<u>Implementation Date</u>
The amendment shall be implemented within 30 days from July 31, 1998, except that the licensee shall have until the next scheduled Updated Final Safety Analysis Report (UFSAR) update to incorporate the UFSAR relocations.</td><td><u>Proposed Appendix B</u>
DELETED</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">Appendix B is deleted in its entirety, because it is a historical requirement that was previously met. The Appendix dealt with the relocation of certain requirements from the TS to the UFSAR.</td></tr>
</table>

<table>
<tr><th colspan="2">TS TABLE OF CONTENTS</th></tr>
<tr><td><u>Current PNPS TS</u></td><td><u>Basis for Change</u></td></tr>
<tr><td>Table of Contents</td><td>The Table of Contents is modified to reflect the changes made below.</td></tr>
</table>

TS SECTION 1.0, DEFINITIONS	
TS 1.0, "Definitions," provides defined terms that are applicable throughout the TS and TS Bases. A number of the Definitions are proposed to be deleted, because they have no relevance to and no longer apply to the permanently defueled facility status. Other definitions are modified to reflect the permanently defueled condition.	
Definition	**Basis for Change**
AUTOMATIC PRIMARY CONTAINMENT ISOLATION VALVES	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. The primary containment isolation valves are not credited to mitigate the consequences of any DBAs.
COLD CONDITION	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. This term has no meaning when the Reactor Coolant System (RCS) is no longer in use.
CORE ALTERATION	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. This term is no longer applicable since fuel will be permanently removed from the reactor core.
CORE OPERATING LIMITS REPORT (COLR)	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification and TS 5.6.5 that requires the COLR is also proposed for elimination.
DESIGN POWER	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification and has no meaning when power operations are not permitted.
FIRE SUPPRESSION WATER SYSTEM	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification.
HOT STANDBY CONDITION	This definition is not proposed for inclusion in the PDTS, because operating Modes are not used in any PDTS specification.

IMMEDIATE IMMEDIATE means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.	This definition is modified as follows to reflect the permanently defueled condition: IMMEDIATE means that the required action will be initiated as soon as practicable considering the safe ~~operation~~ ***maintenance*** of the ~~unit~~ ***facility*** and the importance of the required action. The term "operation" is replaced with "maintenance" and the term "unit" is changed to "facility." These are administrative changes that reflect PNPS will be permanently shut down and defueled. The terms "maintenance" and "facility" are more appropriate terms for a site that is undergoing decommissioning. These changes are proposed throughout this license amendment request.
INSTRUMENT CALIBRATION	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There is no instrumentation credited to mitigate the consequences of any DBAs.
INSTRUMENT CHANNEL	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There is no instrumentation credited to mitigate the consequences of any DBAs.
INSTRUMENT CHECK	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There is no instrumentation credited to mitigate the consequences of any DBAs.
INSTRUMENT FUNCTIONAL TEST	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There is no instrumentation credited to mitigate the consequences of any DBAs.
LEAKAGE	This definition is not proposed for inclusion in the PDTS, because none of the structures, systems, or components (SSCs) from or into which leakage is monitored are credited in the analysis of an FHA or the radioactive waste handling event, which are the only remaining credible accidents.

LIMITING CONDITIONS FOR OPERATION (LCO) The LIMITING CONDITIONS FOR OPERATION specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO.	This definition is modified as follows to reflect the permanently defueled condition: The LIMITING CONDITIONS FOR OPERATION specify the minimum acceptable levels of system performance necessary to assure safe ~~startup and operation~~ ***maintenance*** of the facility. When these conditions are met, the ~~plant~~ ***facility*** can be ~~operated~~ ***maintained*** safely and abnormal situations can be safely controlled. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be ***considered a*** failure to meet the LCO. The terms "operation," "operated," and "plant" are replaced with "maintenance," "maintained," and "facility." These are administrative changes that reflect PNPS will be permanently shut down and defueled. The terms "maintenance," "maintained," and "facility" are more appropriate terms for a site that is undergoing decommissioning. These changes are proposed throughout this license amendment request. In addition, an editorial clarification is made to the last paragraph.
LIMITING SAFETY SYSTEM SETTING (LSSS)	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There is no instrumentation credited to mitigate the consequences of any DBAs. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). These specifications do not apply to the safe storage and handling of spent fuel in the SFP.
LOGIC SYSTEM FUNCTIONAL TEST	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There are no logic systems credited in the analysis of the accident that remains credible.

MINIMUM CRITICAL POWER RATIO (MCPR)	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. This definition only applies to an operating reactor core.
MODE	This definition is not proposed for inclusion in the PDTS, because operating Modes are not used in any PDTS specification. Modes are defined for operating or refueling conditions. This term does not apply to a facility in the permanently defueled condition.
OPERABLE – OPERABILITY	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There are no systems or components required to be operable in the PDTS.
OPERATING	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There are no systems or components required to operate in the PDTS.
OPERATING CYCLE	This definition is not proposed for inclusion in the PDTS, because there will no longer be any operating cycles between refueling outages.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. This definition only applies to an operating reactor.
PRIMARY CONTAINMENT INTEGRITY	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. Primary containment integrity is not credited to mitigate the consequences of any DBAs.
PROTECTIVE ACTION	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. The analysis of the accident that remains credible (i.e., the FHA) does not credit the performance of any actions initiated by the protection system.
PROTECTIVE FUNCTION	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. The analysis of the accident that remains credible (i.e., the FHA) does not credit the performance of any actions initiated by the protection system.
REACTOR POWER OPERATION	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. This term is no longer applicable since fuel will be permanently removed from the reactor core.

REACTOR VESSEL PRESSURE	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. This term is no longer applicable since fuel will be permanently removed from the reactor core.
REFUELING INTERVAL	This definition is not proposed for inclusion in the PDTS, because there will no longer be any refueling outages in the permanently defueled condition.
REFUELING OUTAGE	This definition is not proposed for inclusion in the PDTS, because there will no longer be any refueling outages in the permanently defueled condition.
SAFETY LIMIT	Pursuant to 10 CFR 50.36(c)(1), safety limits are limiting parameters necessary to protect the physical barriers that guard against uncontrolled release of radioactivity from a nuclear reactor. The Safety Limits established in TS 2.1 and TS 2.2 protect the integrity of the fuel cladding and reactor coolant system barriers, respectively. This definition is not proposed for inclusion, because the safety limits do not apply to a reactor that is in a permanently defueled condition. The safety limits provided in TS 2.1 and TS 2.2 are also proposed for deletion. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). These specifications do not apply to the safe storage and handling of spent fuel in the SFP.
SECONDARY CONTAINMENT INTEGRITY	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. Secondary containment integrity is not credited to mitigate the consequences of any DBAs.
SIMULATED AUTOMATIC ACTUATION	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There is no instrumentation credited to mitigate the consequences of any DBAs.
SOURCE CHECK	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There is no instrumentation credited to mitigate the consequences of any

	DBAs.
STAGGERED TEST BASIS	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. This definition applies to the performance of surveillance tests on systems with multiple subsystems or channels. There are no surveillance requirements in the PDTS for operating systems.
SURVEILLANCE FREQUENCY ... The SURVEILLANCE FREQUENCY establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance schedule and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages...	This definition is modified as follows to reflect the permanently defueled condition: ... The SURVEILLANCE FREQUENCY establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance schedule and consideration of ~~plant operating~~ ***facility*** conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified ~~for surveillances that are not performed during refueling outages~~... The term "plant operating conditions" is changed to "facility conditions." This is an administrative change that reflects PNPS will be permanently shut down and defueled. The term "facility conditions" are more appropriate terms for a site that is undergoing decommissioning. This change is proposed throughout this license amendment request. Additionally, the language is simplified to eliminate the reference to "surveillances that are not performed during refueling outages." In the PDTS, there are no surveillances that will be performed during refueling outages.

SURVEILLANCE INTERVAL The SURVEILLANCE INTERVAL is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component, or system is not required to be operable, but the instrument, component, or system shall be tested prior to being declared operable. The operating cycle interval is 24 months and the 25% tolerance of the definition of "SURVEILLANCE FREQUENCY" is applicable. The refueling interval is 24 months and the 25% tolerance specified in the definition of "SURVEILLANCE FREQUENCY" is applicable.	This definition is modified as follows to reflect the permanently defueled condition: The SURVEILLANCE INTERVAL is the calendar time between surveillance tests~~, checks, calibrations, and examinations~~ to be performed ***to confirm that a parameter is within limits*** ~~upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component, or system is not required to be operable, but the instrument, component, or system shall be tested prior to being declared operable. The operating cycle interval is 24 months and the 25% tolerance of the definition of "SURVEILLANCE FREQUENCY" is applicable. The refueling interval is 24 months and the 25% tolerance specified in the definition of SURVEILLANCE FREQUENCY" is applicable~~. The only surveillance that will remain in the PDTS ensures that a parameter is within limits. The PDTS will contain no operability requirements, and there will be no instrument or component checks, calibrations, or examinations. In addition, the discussion regarding the operating cycle is no longer applicable during the permanently shut down and defueled condition.
TOTAL PEAKING FACTOR	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. This definition only applies to an operating reactor core.
TRANSITION BOILING	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. The unit will never operate again.
TRIP SYSTEM	This definition is not proposed for inclusion in the PDTS, because the term is not used in any PDTS specification. There is no trip system credited in the analysis of the accident that remains credible.

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TS SECTION 2.0, ~~SAFETY LIMITS~~NOT USED

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to prevent the release of radioactive materials to the environs during operations. TS 2.1 establishes Safety Limits to protect the integrity of these barriers during normal plant operations and anticipated transients.

Pursuant to 10 CFR 50.36(c)(1), safety limits are limiting parameters necessary to protect the physical barriers that guard against uncontrolled release of radioactivity from a nuclear reactor.

TS Section 2.0 is proposed for deletion in its entirety, since the safety limits do not apply to a reactor that is in a permanently defueled condition. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). These specifications do not apply to the safe storage and handling of spent fuel in the SFP.

A mark-up is provided to identify the section as not used, because the TS will not be renumbered.

</td></tr>
<tr><td><u>Current PNPS TS</u></td><td><u>Basis for Change</u></td></tr>
<tr><td><u>TS 2.1, Safety Limits</u></td><td>TS 2.1 will be deleted.

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to prevent the release of radioactive materials to the environs during operations. TS 2.1 establishes Safety Limits to protect the integrity of these barriers during normal plant operations and anticipated transients.

Pursuant to 10 CFR 50.82(a)(2), the facility license for PNPS will no longer authorize operation of the reactor or placement or retention of fuel in the reactor. Since the Safety Limits apply to an operating reactor, they have no function in the permanently defueled condition. Therefore, the safety limits are proposed for deletion.</td></tr>
</table>

TS 2.2 Safety Limit Violation	TS 2.2 will be deleted. The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to prevent the release of radioactive materials to the environs during operations. TS 2.1 establishes Safety Limits to protect the integrity of these barriers during normal plant operations and anticipated transients. TS 2.2 defines the actions to take if there is a non-compliance with a safety limit. Pursuant to 10 CFR 50.82(a)(2), the facility license for PNPS will no longer authorize operation of the reactor or placement or retention of fuel in the reactor. Since the Safety Limits apply to an operating reactor, they have no function in the permanently defueled condition. Therefore, the safety limits are proposed for deletion.

TS SECTION 3.0, ~~LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY~~NOT USED

TS Section 3.0 contains the general requirements applicable to all LCOs and applies at all times unless otherwise stated in TSs. Due to the limited number of LCOs in the proposed PDTS, the PNPS TS provisions in this section are no longer necessary or applicable to the PNPS facility as indicated in the following table.

A mark-up is provided to identify the section as not used, because the TS will not be renumbered.

Current PNPS TS	Basis for Change
Current TS 3.0.1 through TS 3.0.6 Not Used	TS 3.0.1 through TS 3.0.6 will not be included in the PDTS, because they will serve no purpose as there will be no requirements that remain in PDTS Section 3.0.
TS 3.0.7	This TS provides rules for performing special tests and operations in accordance with the LCOs in TS Section 3.14. This TS is proposed to be deleted, because special tests and operations are not applicable in the permanently defueled condition. In addition, all of the requirements in TS Section 3.14 are proposed to be deleted.

TS 3.0.8	LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This TS is proposed to be deleted, because the PDTS do not contain any operability requirements for any systems that rely on snubbers. Thus, this TS is not applicable in the permanently defueled condition.

TS SECTION 4.0, SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

TS Section 4.0 contains the general requirements applicable to all SRs and applies at all times unless otherwise stated in TSs. TS 4.0.3 is maintained in its entirety. However, the Bases for TS 4.0.3 are modified as follows:

TS 4.0.3 establishes the flexibility to defer declaring ~~affected equipment inoperable or~~ an affected variable...

...The basis for this delay period includes consideration of the ~~unit~~ ***facility*** conditions...

~~When a Surveillance with a Surveillance Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, TS 4.0.3 allows for the full delay period of up to the specified Surveillance Frequency to perform the Surveillance. However, since there is no time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.~~

~~TS 4.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of reactor MODE changes imposed by required Actions.~~

...Use of the delay period established by TS 4.0.3 is a flexibility which is not intended to be used as ~~an operational~~ ***a*** convenience to extend Surveillance intervals...

...The determination of the first reasonable opportunity should include consideration of the impact on ~~plant~~***facility*** risk (from delaying the Surveillance as well as any ~~plant~~***facility*** configuration changes required ~~or shutting the plant down~~ to perform the Surveillance) and impact on any (continued) analysis assumptions, in addition to ~~unit~~***facility*** conditions, planning, availability of personnel, and the time required to perform the Surveillance.

~~This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.1 82, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation should be commensurate with the importance of the component. Missed Surveillance for important components should be analyzed~~

~~quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action.~~ All missed Surveillances will be placed in the licensee's Corrective Action Program. If a Surveillance is not completed within the allowed delay period, then ~~the equipment is considered inoperable or~~ the variable is considered outside the specified limits and the completion times or the required actions for the applicable LCO Actions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then ~~the equipment is inoperable, or~~ the variable is outside the specified limits and the completion times of the required actions for the applicable LCO Actions begin immediately upon the failure of the Surveillance.
Basis for the Changes
References to "inoperable equipment" are proposed to be deleted, because the PDTS will contain no operability requirements. Thus, these references will not be applicable in the PDTS. The terms "unit" and "plant" are changed to "facility." These are administrative changes that reflect PNPS will be permanently shut down and defueled. The term "facility" is more appropriate for a site that is undergoing decommissioning. These changes are proposed throughout this license amendment request. The term "operational" is deleted, because the facility will not be allowed to operate. The paragraph that addresses Surveillance Frequencies that are not based on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations is proposed to be deleted, because the only Surveillance Frequency that will exist in the PDTS will be based on a time interval. The paragraph that addresses Surveillances that become applicable as a consequence of reactor MODE changes imposed by required Actions is proposed to be deleted. PNPS will be permanently shut down and defueled; thus, there will be no Mode changes imposed by required Actions. The majority of the paragraph that addresses managing risk due to a missed surveillance is proposed to be deleted. The PNPS will be permanently shut down and defueled. There will be no operability requirements associated with the remaining LCO. The only LCO that remains deals with monitoring a variable (i.e., SPF water level).

TS SECTION 3/4.1, REACTOR PROTECTION SYSTEM	
TS Section 3/4.1 contains requirements to assure the operability of the reactor protection system. It applies to the instrumentation and associated devices which initiate a reactor scram. TS Section 3/4.1 is proposed for deletion in its entirety. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Because the PNPS 10 CFR Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the reactor protection system will not be required and these requirements will not apply in a defueled condition. A mark-up of this TS section is not provided, because it is deleted in its entirety.	
Current PNPS TS	Basis for Change
TS 3/4.1 including Table 3.1.1, Table 4.1.1, and Table 4.1.2	This TS and its Tables are proposed for deletion in PDTS, because the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor after the certifications required by 10 CFR 50.82(a)(1) have been docketed. Thus, there will no longer be a need for protective instrumentation to protect the reactor core.

TS SECTION 3/4.2, PROTECTIVE INSTRUMENTATION

TS Section 3/4.2 contains operability requirements for protective instrumentation that initiate action to mitigate the consequences of accidents which are beyond the operator's ability to control or terminate operator errors before they result in serious consequences.

TS Section 3/4.2 is proposed for deletion in its entirety. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). Because the PNPS 10 CFR Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the protective functions addressed in TS Section 3/4.1 will not be required and these requirements will not apply in a permanently defueled condition.

A mark-up of this TS section is not provided, because it is deleted in its entirety.

Description and Evaluation of the Proposed Changes

Current PNPS TS	Basis for Change
TS 3/4.2.A, Primary Containment Isolation Functions	This TS provides the operability requirements for the instrumentation that initiates primary containment isolation. It is applicable whenever primary containment integrity is required. TS 3.2.A, including Tables 3.2.A and 4.2.A, is not included in the PDTS. After the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Primary containment isolation is no longer required to mitigate the consequences of any DBAs in the permanently defueled condition. Thus, this TS will not apply in a permanently defueled condition.
TS 3/4.2.B, Core and Containment Cooling Systems – Initiation & Control	This TS provides the operability requirements for the instrumentation that initiates or controls the core and containment cooling systems and monitors emergency bus voltage. TS 3/4.2.B, including Tables 3.2.B, 3.2.B.1, and 4.2.B is not included in the PDTS. After the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. The core and containment cooling systems will not be required to mitigate the consequences of any DBAs in the permanently defueled condition. Thus, this TS will not apply in a permanently defueled condition.

TS 3/4.2.C, Control Rod Block Actuation	This TS provides the operability requirements for the Control Rod Block instrumentation. TS 3/4.2.C, including Tables 3.2.C-1, 3.2.C-2, and 4.2.C, is not included in the PDTS. After the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. The control rod system is not required to control core reactivity in the permanently defueled condition. Thus, the control rod block actuation is no longer required.
TS 3/4.2.D, Radiation Monitoring Systems – Isolation & Initiation Functions	This TS provides the operability requirements for the refuel area exhaust monitors that isolate the Reactor Building and initiate the SGTS. It is applicable during movement of recently irradiated fuel assemblies and operations with the potential to drain the reactor vessel. TS 3/4.2.D, including Tables 3.2.D and 4.2.D, is not included in the PDTS. The PNPS will be permanently shut down and defueled. This TS will no longer be required after 24 hours of decay before channeled fuel assemblies can be handled and 46 days of decay (24 hours of decay assumed in the analysis of the FHA + an additional 45 days of decay before an unchanneled fuel assembly can be handled) following shut down, because the nuclear fuel will no longer be considered to be "recently irradiated." In addition, the other condition requiring that secondary containment integrity be met (operations with the potential to drain the reactor vessel) will not be applicable following permanent removal of the fuel from the reactor vessel. Therefore, the conditions requiring the operability of the refuel area exhaust monitors will no longer be applicable and will not be required to mitigate the consequences of any DBAs in the permanently defueled condition.

TS 3/4.2.E, Drywell Leak Detection	This TS provides the operability requirements for the instrumentation that monitors drywell leak detection. TS 3/4.2.E, including the applicable portions of Tables 3.6.C and 4.6.C, is not included in the PDTS. After the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, drywell leak detection instrumentation will no longer be required.
TS 3/4.2.F, Surveillance Information Readouts	This TS provides the operability requirements for the instrumentation that provide the surveillance information readouts. The primary purpose of the instrumentation controlled by TS 3/4.2.F is to display plant variables that provide information required by the control room operators during accident situations. In the Cold Shutdown and Refueling Modes the likelihood of an event that would require use of the instrumentation is extremely low; therefore, the instrumentation does not provide a required protective function in these conditions. As a result, these instruments are not required to be operable in the Cold Shutdown or Refueling Modes. TS 3/4.2.F, including Tables 3.2.F and 4.2.F, is not included in the PDTS. After the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, TS 3/4.2.F will no longer be applicable, and the instrumentation that provides the surveillance information readouts will not be required in the permanently defueled condition.

TS 3/4.2.G, Recirculation Pump Trip/Alternate Rod Insertion	This TS provides the operability requirements for the recirculation pump trip system and alternate rod insertion system instrumentation. These systems are only required when the reactor mode switch is in the RUN mode. TS 3/4.2.G, including Tables 3.2-G and 4.2.G, is not included in the PDTS. After the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, TS 3.2.G will no longer be applicable, and the recirculation pump trip system and alternate rod insertion system instrumentation will not be required in the permanently defueled condition.
TS 3/4.2.H, Drywell Temperature	This TS provides limits regarding drywell temperature to ensure that safety-related equipment will not be subjected to excess temperature. The limits are applicable when the RCS temperature is above 212°F. TS 3/4.2.H, including Tables 3.2.H and 4.2.H, is not included in the PDTS. After the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. Thus, this TS will not be applicable, and the drywell temperature instrumentation will not be required in the permanently defueled condition.

TS SECTION 3/4.3, REACTIVITY CONTROL

TS 3/4.3 contains requirements to assure and verify operability of reactivity control systems. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). As a result, reactivity control systems will not be required and the requirements in TS 3/4.3 will not apply in a defueled condition. Therefore, TS Section 3/4.3 is proposed for deletion in its entirety.

A mark-up of this TS section is not provided, because it is deleted in its entirety.

Current PNPS TS	Basis for Change
TS 3/4.3.A, Reactivity Margin - Core Loading	This TS defines the reactivity margin requirements to ensure: a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events; b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and, c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shut down condition. TS 3/4.3.A is not proposed for inclusion in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, reactivity control systems will not be required and these the requirements in TS 3/4.3.A will not apply in a defueled condition.
TS 3/4.3.B.1, Control Rod Operability	This TS defines the operability requirements for the control rods. TS 3/4.3.B.1 is not proposed for inclusion in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, reactivity control systems will not be required and the requirements in TS 3/4.3B.1 will not apply in a defueled condition.
TS 3/4.3.B.2, Control Rod Drive Housing Support	This TS defines when the control rod drive housing support system is required to be in place. TS 3/4.3.B.2 is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, reactivity control systems will not be required and the requirements in TS 3/4.3B.2 will not apply in a defueled condition.

TS 3/4.3.B.3, Source Range Monitors	This TS defines the operability requirements for the source range monitors. TS 3/4.3.B.3 is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, the source range monitors will not be required and the requirements in TS 3/4.3B.3 will not apply in a defueled condition.
TS 3/4.3.C, Control Rod Scram Times	This TS defines the control rod scram times in Table 3.3.C-1. TS 3/4.3.C is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, reactivity control systems will not be required and the requirements in TS 3/4.3.C will not apply in a defueled condition.
TS 3/4.3.D, Control Rod Scram Accumulators	This TS defines the operability requirements for the control rod scram accumulators. TS 3/4.3.D is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, reactivity control systems will not be required and the requirements in TS 3/4.3D will not apply in a defueled condition.
TS 3/4.3.E, Reactivity Anomalies	This TS defines a reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. TS 3/4.3.E is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, there is no need to continual confirm reactivity during the permanently defueled condition. Thus, the requirements in TS 3/4.3E will not apply in a defueled condition.

TS 3/4.3.F, Rod Worth Minimizer (RWM)	This TS defines the operability requirements for the RWM. TS 3/4.3.F is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, reactivity control systems will not be required and the requirements in TS 3/4.3F will not apply in a defueled condition.
TS 3/4.3.G, Scram Discharge Volume (SDV) Vent and Drain Valves	This TS defines the operability requirements for the SDV vent and drain valves. TS 3/4.3.G is not proposed for inclusion in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, there is no possibility of a reactor scram. Thus, the SDV vent and drain valves will not be required and the requirements in TS 3/4.3B.1 will not apply in a defueled condition.
TS 3/4.3.H, Rod Pattern Control	This TS defines the control rod sequences to assure that the control rod patterns are consistent with the assumptions of the Control Rod Drop Accident analyses. TS 3/4.3.H is not proposed for inclusion in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, reactivity control systems will not be required and the requirements in TS 3/4.3H will not apply in a defueled condition.

TS SECTION 3/4.4, STANDBY LIQUID CONTROL SYSTEM

TS 3/4.4 contains requirements to assure the operability of the Standby Liquid Control System. This system provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods.

After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). As a result, reactivity control systems will not be required. Therefore, TS Section 3/4.4 is proposed for deletion in its entirety.

A mark-up of this TS section is not provided, because it is deleted in its entirety.

Current PNPS TS	Basis for Change
TS 3/4.4, Standby Liquid Control System	This TS defines the operability requirements for the Standby Liquid Control System. TS 3/4.4 is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. As a result, reactivity control systems will not be required.

TS SECTION 3/4.5, CORE AND CONTAINMENT COOLING SYSTEMS

TS Section 3/4.5 contains requirements to assure the operability of core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to station abnormalities.

As discussed in 10 CFR 50.46(a)(1)(i), the requirement to have an Emergency Core Cooling System (ECCS) does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). The core and containment cooling systems do not mitigate the consequences of any DBAs in the permanently defueled condition Therefore, TS Section 3/4.5 is proposed for deletion in its entirety.

A mark-up of this TS section is not provided, because it is deleted in its entirety.

Current PNPS TS	Basis for Change
TS 3/4.5.A, Core Spray and Low Pressure Coolant Injection (LPCI) Systems	This TS defines the operability requirements for the Core Spray and LPCI System. These systems are part of the emergency core cooling systems (ECCS) that provide sufficient cooling to the core to dissipate the energy associated with the entire spectrum of break sizes for a LOCA, to limit calculated fuel clad temperature to less than 2200°F, to limit calculated local metal water reaction to less than or equal to 17%, to limit calculated core wide metal water reaction to less than or equal to 1%, to maintain the core in a coolable geometry and to provide adequate long term cooling. TS 3/4.5.A is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. In this condition, a LOCA is no longer possible and ECCS are no longer needed.
TS 3/4.5.B.1, Residual Heat Removal (RHR) Suppression Pool Cooling	This TS defines the operability requirements for the RHR suppression pool cooling subsystem. The suppression pool is designed to absorb the sudden input of heat from the primary system. In the long term, the pool continues to absorb residual heat generated by fuel in the reactor core. The RHR suppression pool cooling subsystems remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. TS 3/4.5.B.1 is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. In this condition, the RHR suppression pool cooling subsystem is not required to mitigate any DBAs.

TS 3/4.5.B.2, Residual Heat Removal (RHR) Containment Spray	This TS defines the operability requirements for the RHR containment spray subsystem. systems are designed to remove heat energy from primary containment in the event of a LOCA. TS 3/4.5.B.2 is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. A LOCA is no longer possible and the RHR containment spray subsystem is no longer needed.
TS 3/4.5.B.3, Reactor Building Closed Cooling Water (RBCCW) System	This TS defines the operability requirements for the RBCCW system. The RBCCW system is designed to provide a heat sink for the RHR system heat exchangers and the removal of heat from the ECCS equipment, such as RHR pumps' mechanical seal coolers, core spray pump motor thrust bearings, and room coolers, required for a safe reactor shut down following a DBA or transient. TS 3/4.5.B.3 is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. In this condition, the RBCCW system is not required to mitigate any DBAs.

TS 3/4.5.B.4, Salt Service Water (SSW) System and Ultimate Heat Sink (UHS)	This TS defines the operability requirements for the SSW system and UHS. The SSW system provides a supply of cooling water to the secondary side of the RBCCW heat exchangers adequate for the requirements of the RBCCW under transient and accident conditions. The long-term cooling capability of the RHR, Core Spray, and RBCCW pumps is dependent on the cooling provided by the SSW system. TS 3/4.5.B.4 is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. In this condition, the SSW system and UHS is not required to mitigate any DBAs.
TS 3/4.5.C, High Pressure Coolant Injection (HPCI) System	This TS defines the operability requirements for the HPCI system. The HPCI system is provided to assure that the reactor core is adequately cooled to limit fuel dad temperature in the event of a small break in the nuclear system and loss-of-coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. TS 3/4.5.C is not included in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. In this condition, the ECCS are no longer needed to mitigate any DBAs.

TS 3/4.5.D, Reactor Core Isolation Cooling (RCIC) System	This TS defines the operability requirements for the RCIC system. The RCIC system is designed to provide makeup to the nuclear system as part of the planned operation for periods when the normal heat sink is unavailable. The RCIC system also serves as redundant makeup system on total loss of all offsite power in the event that the HPCI system is unavailable. TS 3/4.5.D is not proposed for inclusion in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. In this condition, the ECCS are no longer needed to mitigate any DBAs.
TS 3/4.5.E, Automatic Depressurization System (ADS)	This TS defines the operability requirements for the ADS system. It provides automatic nuclear system depressurization for small breaks in the nuclear system so that the LPCI and the core spray systems can operate to protect the fuel barrier. TS 3/4.5.E is not proposed for inclusion in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. In this condition, the ADS is no longer needed to mitigate any DBAs.
TS 3/4.5.F, Minimum Low Pressure Cooling and Diesel Generator Availability	This TS assures that adequate core cooling equipment is available at all times. TS 3/4.5.F is not proposed for inclusion in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. In this condition, the ECCS are no longer needed to mitigate any DBAs.
TS 3/4.5.G, Deleted	TS 3/4.5.G is not included in the PDTS. This is an administrative change, because the placeholder is no longer required given that TS Section 3/4.5 is proposed to be deleted in its entirety.

TS 3/4.5.H, Maintenance of Filled Discharge Pipe	This TS defines the requirements to ensure that the discharge piping for the core spray systems, LPCI system, HPCI system, or RCIC system is filled from the pump discharge of these systems to the last block valve whenever those systems are required to be operable. TS 3/4.5.H is not proposed for inclusion in the PDTS, because it will not be required after the certifications required under 10 CFR 50.82(a)(1) have been docketed for PNPS. At that time, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel. In this condition, the ECCS are no longer needed to mitigate any DBAs.

TS SECTION 3/4.6, PRIMARY SYSTEM BOUNDARY

TS Section 3/5.6 contains requirements that provide assurance of the integrity and safe operation of the RCS and the operation of the related safety devices. Because the PNPS 10 CFR Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the requirements will not apply (or are no longer needed) in a permanently defueled condition. Therefore, TS Section 3/5.6 is proposed for deletion in its entirety.

A mark-up of this TS section is not provided, because it is deleted in its entirety.

Current PNPS TS	Basis for Change
TS 3/4.6.A, Thermal and Pressurization Limitations	This TS contains thermal and pressurization limitations regarding the RCS as established in the Pressure and Temperature Limits Report (PTLR). The RCS is a primary barrier against the release of fission products to the environs. These limits were established to ensure that this barrier is maintained at a high degree of integrity. TS 3/4.6.A is not proposed for inclusion in the PDTS, because the PNPS license will no longer authorize use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2). Thus, the RCS will remain depressurized.
TS 3/4.6.B, Coolant Chemistry	This TS establishes requirements for RCS water chemistry. TS 3/4.6.B is not included in the PDTS, because PNPS will be permanently shut down and defueled. In this condition, the protection of the reactor coolant pressure boundary is no longer required.

TS 3/4.6.C, Coolant Leakage	This TS provides the allowable leakage rates of reactor coolant from the RCS. The limits provided protection of the reactor coolant pressure boundary from degradation and the core from inadequate cooling. TS 3/4.6.C is not proposed for inclusion in the PDTS, because PNPS will be permanently shut down and defueled. In this condition, the protection of the reactor coolant pressure boundary is no longer required.
TS 3/4.6.D, Safety and Relief Valves	This TS provides the operability requirements for the Safety and Relief Valves (S/RVs). These valves provide overpressure protection to the reactor during operation. TS 3/4.6.D is not proposed for inclusion in the PDTS, because PNPS will be permanently shut down and defueled. In this condition, the S/RVs are not required to operate to mitigate the consequences of a DBA.
TS 3/4.6.E, Jet Pumps	This TS provides the operability requirements for the jet pumps. The jet pumps are part of the reactor vessel internals, and in conjunction with the recirculation loops are designed to provide forced circulation through the core to remove heat from the fuel. TS 3/4.6.E is not proposed for inclusion in the PDTS, because PNPS will be permanently shut down and defueled. In this condition, the jet pumps are not required to operate to mitigate the consequences of a DBA.
TS 3/4.6.F, Recirculation Loops Operating	This TS provides the operability requirements for the Recirculation Loops. The Reactor Water Recirculation System provides forced coolant flow through the core to remove heat from the fuel. TS 3/4.6.F is not included in the PDTS since PNPS will be permanently shut down and defueled. The 10 CFR Part 50 license will prohibit operation of the reactor after the certifications required by 10 CFR 50.82(a)(1) have been docketed. In this condition, the ECCS are no longer needed to mitigate any DBAs.

TS SECTION 3/4.7, CONTAINMENT SYSTEMS

TS Section 3/4.7 contains requirements that assure the integrity of the Primary Containment System and Secondary Containment Systems and the operability of the SGTS and CRHEAFS.

The Primary Containment System provides a barrier against uncontrolled release of fission products to the environs in the event of a LOCA. The SGTS is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. The CRHEAFS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The Secondary Containment System is designed to minimize any ground level release of radioactive materials that might result from an accident.

Calculation No. M1422 (Reference 11) concludes that the consequences of a drop of a channeled fuel assembly in the SFP after permanent shut down (Reference 11), i.e., the calculated TEDE values to the CR, EAB, and LPZ, are less than the limits set forth in 10 CFR 50.67 and Regulatory Guide 1.183. The analysis assumes: 1) A minimum period of decay of 24 hours before a channeled fuel assembly can be handled; 2) The subsequent drop of a channeled fuel assembly in the SFP; 3) An open Reactor Building with no filtration by the SGTS; and 4) No credit for operation of the CRHEAFS.

Before an unchanneled fuel assembly can be handled, the fuel handling procedure requires an additional 45 days of decay beyond the assumed decay period of 24 hours in the design basis FHA (i.e., 46 days of decay (24 hours of decay assumed in the analysis of the FHA + an additional 45 days of decay)). This additional decay time ensures that the consequences of the drop of an unchanneled fuel assembly in the SFP are bounded by the design basis FHA (Reference 11). This is based on a generic analysis of the dose consequences of a drop of an unchanneled fuel assembly in the SFP contained in Reference 13.

TS Section 3/4.7 is proposed for deletion in its entirety. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the TS for the systems addressed in TS Section 3/4.7 will not be required and these requirements will not apply in a permanently defueled condition.

A mark-up of this TS section is not provided, because it is deleted in its entirety.

Current PNPS TS	Basis for Change
TS 3/4.7.A, Primary Containment	This TS provides operability requirements for the primary containment. Its function was to isolate and contain fission products released following a DBA and to confine the postulated release of radioactive material. TS 3/4.7.A is not included in the PDTS, because PNPS will be permanently shut down and defueled. The 10 CFR Part 50 license will prohibit operation of the reactor after the certifications required by 10 CFR 50.82(a)(1) have been docketed in accordance with 10 CFR 50.82(a)(2). Thus, there will no longer be a need for the primary containment, because it will not mitigate the consequences of any DBAs.
TS 3.7.B, Standby Gas Treatment System and Control Room High Efficiency Air Filtration System	This TS provides the operability requirements for the SGTS and CRHEAFS. The SGTS is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. The CRHEAFS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. TS 3/4.7.B is not included in the PDTS, because PNPS will be permanently shut down and defueled. The analysis of the FHA in the SFP in the permanently shut down and defueled condition determines that the radiological consequences in the Control Room are within allowable limits of 10 CFR 50.67 without crediting the operation of the SGTS or CRHEAFS after a 24-day fuel decay period for a channeled fuel assembly or a 46-day fuel decay period (24 hours of decay assumed in the analysis of the FHA + an additional 45 days of decay)) for an unchanneled fuel assembly following permanent reactor shut down.

TS 3.7.C, Secondary Containment	This TS provides the operability requirements for secondary containment. The secondary containment is designed to minimize any ground level release of radioactive materials that might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment during periods when the reactor is shut down, the drywell is open, and activities are ongoing that require secondary containment to be operable. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during movement of "recently irradiated" fuel and during operations with the potential to drain the reactor vessel (OPDRVs). There are two principal accidents for which credit is taken for secondary containment operability. These are a LOCA, although not specifically evaluated for alternate source term methodology, and a FHA involving "recently irradiated fuel." TS 3.7.C1 is not included in the PDTS, because PNPS will be permanently shut down and defueled. This TS will no longer be required after 24 hours of decay before channeled fuel assemblies can be handled and 46 days of decay (24 hours of decay assumed in the analysis of the FHA + an additional 45 days of decay) before unchanneled fuel assemblies can be handled following shut down, because the nuclear fuel will no longer be considered to be "recently irradiated." In addition, the other condition requiring that secondary containment integrity be met (OPDRVs) will not be applicable following permanent removal of the fuel from the reactor vessel. Therefore, the conditions requiring secondary containment integrity will no longer be applicable and secondary containment will not be required to mitigate the consequences of the FHA. Thus, there will no longer be a need for secondary containment.

<table>
<tr><td colspan="2">TS SECTION 3/4.8, PLANT SYSTEMS
TS Section 3/4.8 defines a limit regarding the gross gamma activity rate of noble gases measured at a main condenser pretreatment monitor station and the operability requirements for the Main Steam Line Radiation Monitoring System Radiation - High function for the mechanical vacuum pump.

TS Section 3/4.8 is proposed for deletion in its entirety. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the requirements addressed in TS Section 3/4.8 will not be required and will not apply in a permanently defueled condition.

A mark-up of this TS section is not provided, because it is deleted in its entirety.</td></tr>
<tr><td><u>Current PNPS TS</u></td><td><u>Basis for Change</u></td></tr>
<tr><td>TS 3/4.8.1, Main Condenser Offgas</td><td>This TS defines a limit regarding the gross gamma activity rate of noble gases measured at a main condenser pretreatment monitor station. It is applicable when steam is being exhausted to the main condenser and the resulting non-condensables are being processed via the main condenser offgas system.

TS Section 3/4.8.1 is not included in the PDTS, because PNPS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Thus, this TS is not required and not applicable in the permanently defueled condition.</td></tr>
<tr><td>TS 3/4.8.2, Mechanical Vacuum Pump Isolation Instrumentation</td><td>This TS defines the operability requirements for the Main Steam Line Radiation Monitoring System Radiation - High function for the mechanical vacuum pump.

The mechanical vacuum pump isolation instrumentation initiates a trip of the mechanical vacuum pump and isolation of the associated isolation valve following events in which main steam radiation exceeds predetermined values. Tripping and isolating the mechanical vacuum pump limits the offsite doses in the event of a control rod drop accident (CRDA).

TS Section 3/4.8.2 is not included in the PDTS, because PNPS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel pursuant to 10 CFR</td></tr>
</table>

	50.82(a)(2). Thus, TS 3/4.8.2 is not required and not applicable in the permanently defueled condition.

TS SECTION 3/4.9, AUXILIARY ELECTRICAL SYSTEM

TS 3/4.9 contains operability requirements to assure an adequate source of electrical power to operate the auxiliaries during plant operation, to operate facilities to cool and lubricate the plant during shut down, and to operate the engineered safeguards following an accident. TS 3/4.9.A states the required availability of AC and DC power; i.e., an active off-site AC source, a back-up source of off-site AC power, and the maximum amount of on-site AC and DC sources. TS 3.9.B contains the requirements regarding operation with inoperable equipment.

The design basis accidents and transients analyzed in UFSAR Chapter 14 will no longer be applicable in the permanently defueled condition, with the exception of the FHA in the SFP and a radioactive waste handling accident (HIC Drop Event).

Calculation No. M1421 (Reference 3) establishes that no station structures, systems, or components are required to mitigate the HIC drop event.

Calculation No. M1422 (Reference 11) concludes that the consequences of a drop of a channeled fuel assembly in the SFP after permanent shut down (Reference 11), i.e., the calculated TEDE values to the CR, EAB, and LPZ, are less than the limits set forth in 10 CFR 50.67 and Regulatory Guide 1.183. The analysis assumes: 1) A minimum period of decay of 24 hours before a channeled fuel assembly can be handled; 2) The subsequent drop of a channeled fuel assembly in the SFP; 3) An open Reactor Building with no filtration by the SGTS; and 4) No credit for operation of the CRHEAFS.

Before an unchanneled fuel assembly can be handled, the fuel handling procedure requires an additional 45 days of decay beyond the assumed decay period of 24 hours in the design basis FHA (i.e., 46 days of decay (24 hours of decay assumed in the analysis of the FHA + an additional 45 days of decay)). This additional decay time ensures that the consequences of the drop of an unchanneled fuel assembly in the SFP are bounded by the design basis FHA (Reference 11). This is based on a generic analysis of the dose consequences of a drop of an unchanneled fuel assembly in the SFP contained in Reference 13.

During movement of irradiated fuel assemblies in the SFP, there are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the FHA with the unit permanently defueled. Because the DBA analyses do not rely on any AC or DC power sources for accident mitigation (including any need for providing airborne radiological protection), the AC and DC sources are not required during movement of irradiated fuel assemblies in the SFP for mitigation of a potential FHA. As such, the requirement for AC and DC sources are being deleted because there are no design basis events that rely on these sources for mitigation.

TS Section 3/4.9 is proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the requirements addressed in TS Section 3/4.9 will not be required and will not apply in a permanently defueled condition.

A mark-up of this TS section is not provided, because it is deleted in its entirety.

Current PNPS TS	Basis for Change
TS 3/4.9.A, Auxiliary Electrical Equipment	This TS provides the AC and DC electrical power requirements. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). In this condition, the operational conditions, transients, and postulated DBAs are no longer possible. Therefore, the systems required for reactor safety, which the auxiliary electrical systems were designed to power, are no longer needed. The only DBAs that would apply to the permanently shut down and defueled PNPS reactor would be the FHA and the radioactive waste handling accident. AC and DC sources are not needed during the movement of irradiated fuel assemblies to mitigate the consequences of a potential FHA in the SFP. The FHA analysis does not rely on AC or DC sources for accident mitigation (dose consequences are acceptable without relying on any SSCs to remain functional during and following the event). Thus, these sources are not required. There are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the FHA with the PNPS permanently shut down and defueled. The only electrically powered active system important for the storage of irradiated fuel is the SFP cooling and support systems. The SPF cooling system did not meet the criteria in 10 CFR 50.36 for inclusion in the PNPS TS even when the reactor was authorized to operate. Thus, TS 3/4.9.A is not being proposed for inclusion in the PDTS, because the DBAs that require power for engineered safeguards systems supplied by the AC and DC power systems are no longer applicable in the permanently defueled condition.
TS 3.9.B, Operation with Inoperable Equipment	This TS provides requirements for continued operation of the reactor when the availability of power falls below that required in TS 3/4.9.A. As stated above in the Basis for Change to TS

	3/4.9.A, AC and DC sources are not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA in the SFP. TS 3.9.B is not proposed to be included in the PDTS, because the DBAs that require power for engineered safeguards systems supplied by the AC and DC power sources are no longer applicable in the permanently defueled condition.

TS SECTION 3/4.10, ~~CORE ALTERATIONS~~*SPENT FUEL STORAGE*

TS 3/4.10 contains requirements regarding refueling interlocks, core monitoring, and SFP water level.

TS 3/4.10.A and TS 3/4.10.B address requirements regarding refueling interlocks and core monitoring. These TS are proposed to be deleted in their entirety. TS 3/4.10.A and TS 3/4.10.B do not apply to the safe storage and handling of spent fuel in the SFP. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS 3/4.10.A and 3/4.10.B will not be required and these requirements will not apply in a permanently defueled condition.

TS 3/4.10.C defines requirements for SFP water level. It is retained.

TS Section 3/4.10 will be retitled Spent Fuel Storage to better categorize the remaining requirements.

A markup of this section is provided.

Current PNPS TS	Basis for Change
TS 3/4.10, Core Alterations	The title for this section is proposed to be changed as follows: TS 3/4.10, ~~Core Alterations~~***Spent Fuel Storage*** This change reflects that the requirements that will remain in this TS are those associated with the SFP Water Level.

Description and Evaluation of the Proposed Changes

TS 3.10 Applicability Applies to the fuel handling and core reactivity limitations during refueling and core alterations.	This section is proposed to be modified as follows: TS 3.10 Applicability Applies to the ***safe storage of spent fuel***~~fuel handling and core reactivity limitations during refueling and core alterations~~. This change reflects that the requirements that will remain in this TS are those associated with the SFP Water Level.
TS 3.10 Objective To ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.	This section is proposed to be modified as follows: TS 3.10 Objective To ensure that ***safe storage of spent fuel***~~core reactivity is within the capability of the control rods and to prevent criticality during refueling~~. This change reflects that the requirements that will remain in this TS are those associated with the SFP Water Level.
TS 4.10 Applicability Applies to the period testing of those interlocks and instrumentation used during refueling and core alterations.	This section is proposed to be modified as follows: TS 4.10 Applicability Applies to the ***parameter which monitors the storage of spent fuel***~~period testing of those interlocks and instrumentation used during refueling and core alterations~~. This change reflects that the requirements that will remain in this TS are those associated with the SFP Water Level.
TS 4.10 Objective To verify the operability of instrumentation and interlocks used in refueling and core alterations.	This section is proposed to be modified as follows: TS 4.10 Objective To verify ***that spent fuel is being stored safely***~~the operability of instrumentation and interlocks used in refueling and core alterations~~. This change reflects that the requirements that will remain in this TS are those associated with the SFP Water Level.

TS 3/4.10.A, Refueling Interlocks	This TS provides the operability requirements for the refueling interlocks. Refueling interlocks restrict the operation of the refueling equipment or the withdrawal of control rods to reinforce unit procedures that prevent the reactor from achieving criticality during refueling. TS 3/4.10.A is not included in the PDTS, because PNPS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2), requirements related to refueling interlocks will not be required. A mark-up is provided to identify the section as not used, because the TS will not be renumbered.
TS 3/4.10.B, Core Monitoring	This TS provides the operability requirements for the source range monitors to monitor the core during periods of station shut down and to guide the operator during refueling operations and station start-up. In addition, it defines requirements for spiral reloading that each control cell to have at least one assembly that meets a minimum exposure requirement. TS 3/4.10.B is not included in the PDTS, because PNPS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2), requirements related to core monitoring will not be required. A mark-up is provided to identify the section as not used, because the TS will not be renumbered.
TS 3/4.10.C, Spent Fuel Pool Water Level	This TS is retained, because it provides the requirements to confirm SPF water level whenever irradiated fuel is stored in the SFP. The Bases for this Technical Specification are modified to define that spent fuel pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

TS SECTION 3/4.11, REACTOR FUEL ASSEMBLY	
TS 3/4.11 contains requirements to ensure that power distribution limits are met. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, in accordance with 10 CFR 50.82(a)(2). Because the PNPS 10 CFR Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, power distribution limits will not be required and these requirements will not apply in a defueled condition. Therefore, TS 3/4.11 is proposed for deletion in its entirety. A mark-up of this TS section is not provided, because it is deleted in its entirety.	
Current PNPS TS	Basis for Change
TS 3/4.11.A, Average Planar Linear Heat Generation Rate (APLGHR)	This TS defines limits for the APLHGR to ensure that the peak cladding temperature during the postulated design basis LOCA does not exceed the limits specified in 10 CFR 50.46. TS 3/4.11.A is not included in the PDTS, because PNPS will be permanently shut down and defueled, and this TS does not provide protection for the cladding of fuel stored in the SFP.
TS 3/4.11.B, Linear Heat Generation Rate (LHGR)	This TS defines limits for the LHGR to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including abnormal operational transients. TS 3/4.11.B is not included in the PDTS, because PNPS will be permanently shut down and defueled, and this TS does not provide protection for the cladding of fuel stored in the SFP.
TS 3/4.11.C, Minimum Critical Power Ratio (MCPR)	This TS defines limits for the MCPR to ensure that no fuel damage results during abnormal operational transients. TS 3/4.11.C is not included in the PDTS, because PNPS will be permanently shut down and defueled, and this TS does not provide protection for the cladding of fuel stored in the SFP.

TS 3/4.11.D, Power/Flow Relationship During Power Operation	This TS defines that the power/flow relationship will not exceed the limiting values specified in the COLR. TS 3/4.11.D is not included in the PDTS, because PNPS will be permanently shut down, defueled, and prohibited from reloading fuel into the reactor vessel. Thus, the power/flow relationship limit in the COLR does not apply in the permanently shut down and defueled condition.

TS SECTION 3/4.12, FIRE PROTECTION

TS 3/4.12 contains requirements regarding the alternate shut down system to effect safe shut down of PNPS in the event of a fire in the Cable Spreading Room.

TS 3/4.12 is proposed for deletion in its entirety. It does not apply to the safe storage and handling of spent fuel in the SFP. After the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the requirements addressed in TS Section 3.12 will not be required and will not apply in a permanently defueled condition.

A mark-up of this TS section is not provided, because it is deleted in its entirety.

Current PNPS TS	Basis for Change
TS 3/412.1, Alternate Shutdown Panels	This TS defines the requirements to ensure the alternate shutdown system can safely shut down of PNPS in the event of a fire in the Cable Spreading Room. TS 3/4.12, including Table 3.12, is not included in the PDTS, because PNPS will permanently cease power operation and will no longer be authorized to operate the reactor or to place or retain fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). The requirements regarding the alternate shutdown system will no longer be applicable.

TS SECTION 3/4.13, INSERVICE CODE TESTING

TS Section 3/4.13 contains requirements to ensure the operational readiness of ASME Code Class 1, 2, and 3 pumps and valves. Because the PNPS 10 CFR Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) will no longer apply in a defueled condition. Therefore, TS 3/4.13 is proposed for deletion in its entirety.

A mark-up of this TS section is not provided, because it is deleted in its entirety.

Current PNPS TS	Basis for Change
TS 3/4.13, Inservice Code Testing	This TS provides the requirements to assure the operational readiness of Code Class 1, 2, and 3 pumps and valves. TS 3/4.13 is not included in the PDTS since PNPS will be permanently shut down and defueled. The 10 CFR Part 50 license will prohibit operation of the reactor after the certifications required by 10 CFR 50.82(a)(1) have been docketed. No Code Class 1, 2, or 3 pumps and valves are utilized to mitigate the consequences of a DBA in the permanently defueled condition. Thus, there will no longer be a need for this TS.

TS SECTION 3/4.14, SPECIAL OPERATIONS

TS Section 3/4.14 contains Special Operations LCOs and SRs that provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. Because the PNPS 10 CFR Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) will no longer apply in a defueled condition. Therefore, TS Section 3/4.14 is proposed for deletion in its entirety.

A mark-up of this TS section is not provided, because it is deleted in its entirety.

Current PNPS TS	Basis for Change
TS 3/4.14.A, Inservice Hydrostatic and Leak Testing Operation	This TS provides the requirements to allow flexibility to perform certain operations by appropriately modifying requirements of other LCOs tor coolant pressure tests to be performed. TS 3/4.14.A is not included in the PDTS since PNPS will be permanently shut down and defueled. The 10 CFR Part 50 license will prohibit operation of the reactor after the certifications required by 10 CFR 50.82(a)(1) have been docketed. Thus, there will no longer be a need for this TS.
TS 3.14.B, (Not Used)	TS 3.14.B will not be included in the PDTS, because it will serve no purpose as TS Section 3/4.14 is proposed to be deleted in its entirety.

TS 3.14.C, Single Control Rod Withdrawal – Hot Shutdown	This TS provides the requirements to permit the withdrawal of a single control rod for testing while in hot shut down, by imposing certain restrictions. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3/4.14.C is not included in the PDTS since PNPS will be permanently shut down and defueled. The 10 CFR Part 50 license will prohibit operation of the reactor after the certifications required by 10 CFR 50.82(a)(1) have been docketed. Thus, there will no longer be a need for this TS.
TS 3/4.14.D, Single Control Rod Withdrawal – Cold Shutdown	This TS provides the requirements to permit the withdrawal of a single control rod for testing while in cold shut down, by imposing certain restrictions. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3/4.14.D is not included in the PDTS since PNPS will be permanently shut down and defueled. The 10 CFR Part 50 license will prohibit operation of the reactor after the certifications required by 10 CFR 50.82(a)(1) have been docketed. Thus, there will no longer be a need for this TS.
TS 3/4.14.E, Multiple Control Rod Removal	This TS provides the requirements to permit multiple control rod withdrawal during refueling by imposing certain administrative controls. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3/4.14.E is not included in the PDTS since PNPS will be permanently shut down and defueled. The 10 CFR Part 50 license will prohibit operation of the reactor after the certifications required by 10 CFR 50.82(a)(1) have been docketed. Thus, there will no longer be a need for this TS.
TS 3/4.14.F, (Not Used)	TS 3/4.14.F will not be included in the PDTS, because they will serve no purpose as TS Section 3/4.14 is proposed to be deleted in its entirety.

TS 3/4.14.G, Control Rod Testing - Operating	This TS provides the requirements to permit control rod testing by imposing certain administrative controls. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. TS 3/4.14.G is not included in the PDTS since PNPS will be permanently shut down and defueled. The 10 CFR Part 50 license will prohibit operation of the reactor after the certifications required by 10 CFR 50.82(a)(1) have been docketed. Thus, there will no longer be a need for this TS.

TS Section 4.0, Design Features	
Current PNPS TS	Basis for Change
Current TS 4.2 Deleted	Proposed TS 4.2 ~~Deleted~~***Not Used*** This change is an administrative change to establish consistency regarding sections that are no longer utilized.
Current TS 4.3 Fuel Storage	Proposed TS 4.3 ***Spent*** Fuel Storage The proposed change to the title of TS 4.3 clarifies that the requirements are applicable to spent fuel storage, because there will be no new fuel storage maintained after PNPS is permanently shut down and defueled. Therefore, the requirements apply only to the spent fuel storage design.

Current 4.3.1.1.b $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.3.5 of the FSAR.	Proposed 4.3.1.1.b $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~Section 10.3.5~~ ***the applicable section*** of the FSAR. This is an administrative change. The PNPS UFSAR will be revised to reflect the permanently shut down and defueled condition. As a result, portions of the FSAR will be re-structured and re-numbered. The FSAR will become the Defueled Safety Analysis Report (DSAR). However, the terms FSAR and DSAR will be interchangeable, as that is the document will be maintained in accordance with 10 CFR 50.59.
Current TS 4.3.1.2	This TS provides requirements regarding the new fuel storage racks. It is proposed to delete these requirements, because there will be no new fuel storage maintained after PNPS is permanently shut down and defueled.

TS Section 5.0, Administrative Controls	
Current PNPS TS	Basis for Change
5.2.2 Facility Staff e. Deleted... g. Deleted... i. Deleted...	5.2.2 Facility Staff e. ~~Deleted~~***Not Used***... g. ~~Deleted~~***Not Used***... i. ~~Deleted~~***Not Used*** These changes are administrative changes to establish consistency regarding sections that are no longer utilized.
5.4.1 Procedures b. Deleted	5.4.1 Procedures b. ~~Deleted~~***Not Used*** This change is an administrative change to establish consistency regarding sections that are no longer utilized.
5.5.1 Offsite Dose Calculation Manual (ODCM)	This specification is modified to correct the numbering of a sub-section. Paragraph c and its subparts were not properly numbered. This is an administrative change.

5.5.5 Component Cyclic or Transient Limit	This specification will not be retained in the PDTS, because it only pertains to reactor support systems that are not required to perform a function in the permanently shut down and defueled condition.
5.5.7 Configuration Risk Management Program (CRMP)	The CRMP is proposed for elimination since the LCO remaining in the PDTS (LCO 3.10.C) does not rely on the operability of any active equipment or systems. LCO 3.10.C establishes a minimum water level in the spent fuel storage pool to ensure that an assumption in the analysis of the FHA is met. Thus, the CRMP is not needed in a permanently shut down and defueled condition.
5.5.8 Control Room Envelope Habitability Program	Following 24 hours of decay before a channeled fuel assembly can be handled or 46 days of decay (24 hours of decay assumed in the analysis of the FHA + an additional 45 days of decay) before an unchanneled fuel assembly can be handled after shut down, the analysis of the FHA demonstrates that the control room envelope is not required for providing airborne radiological protection for the control room operators. As previously discussed, TS 3/4.7.B will not be retained in the PDTS. Thus, TS 5.5.8 will not be retained in the PDTS.
5.5.9 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)	This specification will not be retained in the PDTS, because the PTLR does not apply in the permanently shut down and defueled condition. In previous discussions, the requirements regarding the PTLR were proposed to be deleted from the PDTS. Thus, the need for the PTLR will no longer exist in the permanently shut down and defueled condition.
5.6.4 <u>Not Used</u>	Currently, TS 5.6.4 is not used, and the number is retained as a placeholder for future activities. The reference to TS 5.6.4 will be eliminated to permit reformatting of the PDTS. This is an administrative change.
5.6.5, CORE OPERATING LIMITS REPORT (COLR)	This specification will not be retained in the PDTS, because the plant will be prohibited from reloading fuel into the reactor vessel. Thus, the COLR does not apply in the permanently shut down and defueled condition. In previous discussions, the requirements regarding the COLR were proposed to be deleted from the PDTS.

4. REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENT/CRITERIA

10 CFR 50.82, Termination of License

10 CFR 50.82(a)(1) requires that when a licensee has determined to permanently cease operations the licensee shall, within 30 days, submit a written certification to the NRC, consistent with the requirements of 10 CFR 50.4(b)(8), and after fuel has been permanently removed from the reactor vessel, the licensee shall submit a written certification to the NRC that meets the requirements of 10 CFR 50.4(b)(9). On November 10, 2015, ENO notified the NRC that PNPS would permanently cease operations no later than June 1, 2019 (Reference 1). ENO recognizes that approval of these proposed changes is contingent upon the submittal of the certifications required by 10 CFR 50.82(a)(1) and the docketing of those certifications in accordance with 10 CFR 50.82(a)(2).

10 CFR 50.82(a)(2) states: "Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel."

10 CFR 50.36, Technical Specifications

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TSs. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TSs "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." (Statement of Consideration, "Technical Specification for Facility Licenses; Safety Analysis Reports," 33 FR 18610 (December 17, 1968))

Pursuant to 10 CFR 50.36, TS are required to include items in the following five categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a facility's TS.

These criteria, which were subsequently codified in changes to Section 36 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) (60 FR 36953), also pertain to the Technical Specification requirements for safe storage of spent fuel. A general discussion of these considerations is provided below to address the existing LCOs. As noted in 10 CFR 50.36(c)(2)(iii), a licensee is not required to propose to modify technical specifications that are included in any license issued before August 18, 1995, to satisfy the criteria in paragraph (c)(2)(ii) of 10 CFR 50.36.

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that TS LCOs must be established for "installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary." Since no fuel will be present in the reactor or reactor coolant system at the PNPS facility in the permanently shut down and defueled condition, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. While this criterion was developed for operating reactors, there are some DBAs which continue to apply to a facility authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a facility with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The applicable DBAs for PNPS in the permanently defueled condition, the FHA and the radioactive waste handling accident, are discussed within this license amendment request.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs must be established for SSCs that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The intent of this criterion is to capture into the TSs only those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria. While there are no transients that will continue to apply to PNPS, there are still DBAs that will continue to apply to a facility authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a facility with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The scope of DBAs that will be applicable to PNPS is discussed in more detail within this license amendment request.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that TS LCOs must be established for SSCs that operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs. All of the accident sequences that previously dominated risk at PNPS will no longer be applicable after the reactor is in the permanently shut down and defueled condition.

Addressing administrative controls, 10 CFR 50.36(c)(5) states that they "...are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The particular administrative controls to be included in the TSs, therefore, are the provisions that the Commission deems essential for the safe operation of the facility that are not already covered by other regulations.

10 CFR 50.36(c)(6), "Decommissioning," applies only to nuclear power reactor facilities that have submitted the certifications required by 10 CFR 50.82(a)(1). For such facilities,

TSs involving safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis.

This proposed amendment deletes the portions of the previous PNPS TS that are no longer applicable to a permanently defueled facility while modifying the remaining portions to correspond to the permanently shut down and defueled condition.

10 CFR 50.48(f), Fire Protection during Decommissioning

10 CFR 50.48(f) states in part that licensees that have submitted the certifications required under 10 CFR 50.82(a)(1) shall maintain a fire protection program to address the potential for fires that could cause the release or spread of radioactive materials (i.e., that could result in a radiological hazard).

(1) The objectives of the fire protection program are to—

- (i) Reasonably prevent these fires from occurring;
- (ii) Rapidly detect, control, and extinguish those fires that do occur and that could result in a radiological hazard; and
- (iii) Ensure that the risk of fire-induced radiological hazards to the public environment and plant personnel is minimized.

(2) The licensee shall assess the fire protection program on a regular basis. The licensee shall revise the plan as appropriate throughout the various stages of facility decommissioning.

(3) The licensee may make changes to the fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment that could result in a radiological hazard, taking into account the decommissioning plant conditions and activities.

10 CFR 50.51, Continuation of License

10 CFR 50.51(b) states: "Each license for a facility that has permanently ceased operations, continues in effect beyond the expiration date to authorize ownership and possession of the production or utilization facility, until the Commission notifies the licensee in writing that the license is terminated. During such period of continued effectiveness the licensee shall:

(1) Take actions necessary to decommission and decontaminate the facility and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition, and

(2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the provisions of the specific 10 CFR part 50 license for the facility."

10 CFR 50.82, Termination of License

10 CFR 50.82(a)(2) states: "Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel."

10 CFR 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants

The GDC in place today became effective after the PNPS provisional construction permit was issued. A September 18, 1992 memorandum (Reference 27) to the NRC Executive Director of Operations from the Secretary of the NRC summarized the results of a Commissioners vote in which the Commissioners instructed the NRC staff not to apply the GDC to plants with construction permits issued prior to May 21, 1971. PNPS' provisional construction permit was issued by the Atomic Energy Commission (AEC) on August 26, 1968 (Reference 28).

PNPS' design and licensing basis for fuel storage and handling and radiological controls is detailed in the UFSAR and other plant-specific licensing basis documents.

10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

10 CFR 50.46(a)(1)(i) states: "This section does not apply to a nuclear power reactor facility for which the certifications required under 10 CFR 50.82(a)(1) have been submitted."

10 CFR 50.62, Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants

10 CFR 50.62(a) states: "The requirements of this section apply to all commercial light-water-cooled nuclear power plants, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted."

Design Basis Accidents (DBAs)

Section 14 of the PNPS UFSAR describes the DBA scenarios that are applicable during plant operations. After certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are submitted to the NRC in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and they are docketed for PNPS, the 10 CFR Part 50 license will no longer permit operation of the reactor or placement of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). With the reactor in a permanently shut down and defueled condition, the SFP and its cooling systems are dedicated only to spent fuel storage. In this condition, the spectrum of credible accidents is much smaller than for an operational plant. Therefore, most of the accident scenarios postulated in UFSAR Section 14 will no longer be applicable after PNPS is in the permanently defueled condition. The only remaining DBAs will be the FHA and the radioactive waste handling accident.

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Pursuant to 10 CFR 50.92, Entergy Nuclear Operations, Inc. (ENO) has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10 CFR 50.92(c). These criteria require that operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

This proposed license amendment would revise the Operating License (OL) and revise the associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. The proposed changes would revise certain requirements contained within the Operating License OL and TS and remove the requirements that would no longer be applicable after it has been certified that all fuel has permanently been removed from the PNPS reactor in accordance with 10 CFR 50.82(a)(1)(ii).

On November 10, 2015, ENO notified the U.S. Nuclear Regulatory Commission (NRC) that it would permanently cease power operations at Pilgrim Nuclear Power Station (PNPS) no later than June 1, 2019 (Reference 1). After the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed for PNPS, the 10 CFR Part 50 license for PNPS will no longer authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel, in accordance with 10 CFR 50.82(a)(2).

The existing PNPS TS contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational mode. LCOs and associated Surveillance Requirements (SRs) that will not apply in the permanently defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents associated with a defueled facility.

The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment would not take effect until PNPS has permanently ceased operation, entered a permanently defueled condition, and met the decay requirements established in the analysis of the Fuel Handling Accident (FHA). The proposed amendment would modify the PNPS OL and TS by deleting the portions of the OL and TS that are no longer applicable to a permanently defueled facility, while modifying the other sections to correspond to the permanently defueled condition. This change is consistent with the criteria set forth in 10 CFR 50.36 for the contents of TS.

Section 14 of the PNPS Updated Final Safety Analysis Report (UFSAR) describes the design basis accident (DBA) and transient scenarios applicable to PNPS during power operations. After the reactor is in a permanently defueled condition, the spent fuel pool (SFP) and its cooling systems will be dedicated only to spent fuel storage. In this condition, the spectrum of credible accidents will be much smaller than for an operational plant. After the certifications are docketed for PNPS in accordance with 10 CFR 50.82(a)(1), and the consequent removal of authorization to operate the reactor or to place or retain fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2), the majority of the accident scenarios previously postulated in the UFSAR will no longer be possible and will be removed from the UFSAR under the provisions of 10 CFR 50.59.

The deletion of TS definitions and rules of usage and application requirements that will not be applicable in a defueled condition has no impact on facility structures, systems, and components (SSCs) or the methods of operation of such SSCs. The deletion of design features and safety limits not applicable to the permanently shut down and defueled status of PNPS has no impact on the remaining applicable DBAs, i.e., the FHA and the radioactive waste handling accident (High Integrity Container (HIC) Drop Event).

The removal of LCOs or SRs that are related only to the operation of the nuclear reactor or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents do not affect the applicable DBAs previously evaluated since these DBAs are no longer applicable in the permanently defueled condition. The safety functions involving core reactivity control, reactor heat removal, reactor coolant system inventory control, and containment integrity are no longer applicable at PNPS as a permanently shut down and defueled facility. The analyzed accidents involving damage to the reactor coolant system, main steam lines, reactor core, and the subsequent release of radioactive material will no longer be possible at PNPS.

After PNPS permanently ceases operation, the future generation of fission products will cease and the remaining source term will decay. The radioactive decay of the irradiated fuel following shut down of the reactor will have reduced the consequences of the FHA below those previously analyzed.

The SFP water level and fuel storage TSs are retained to preserve the current requirements for safe storage of irradiated fuel. SFP cooling and makeup related equipment and support equipment (e.g., electrical power systems) are not required to be continuously available since there will be sufficient time to effect repairs, establish alternate sources of makeup flow, or establish alternate sources of cooling in the event of a loss of cooling and makeup flow to the SFP.

The deletion and modification of provisions of the administrative controls do not directly affect the design of SSCs necessary for safe storage of irradiated fuel or the methods used for handling and storage of such fuel in the fuel pool. The changes to the administrative controls do not affect any accidents applicable to the safe management of irradiated fuel or the permanently shut down and defueled condition of the reactor.

The probability of occurrence of previously evaluated accidents is not increased, since extended operation in a defueled condition will be the only operation allowed, and therefore bounded by the existing analyses. Additionally, the occurrence of postulated accidents associated with reactor operation will no longer be credible in a permanently defueled reactor. This significantly reduces the scope of applicable accidents.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes to the PNPS OL and TSs have no impact on facility SSCs affecting the safe storage of irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of irradiated fuel itself. The removal of TS that are related only to the operation of the nuclear reactor or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents, cannot result in different or more adverse failure modes or accidents than previously evaluated because the reactor will be permanently shut down and defueled and PNPS will no longer be authorized to operate the reactor.

   The proposed deletion of requirements of the PNPS OL and TS do not affect systems credited in the accident analyses for the FHA or the HIC Drop Event at PNPS. The proposed OL and TS will continue to require proper control and monitoring of safety significant parameters and activities.

   The TS regarding SFP water level and fuel storage required is retained to preserve the current requirements for safe storage of irradiated fuel. The restriction on the SFP water level is fulfilled by normal operating conditions and preserves initial conditions assumed in the analyses of the postulated DBA.

The proposed amendment does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants (fuel cladding and spent fuel cooling). Since extended operation in a defueled condition will be the only operation allowed, and therefore bounded by the existing analyses, such a condition does not create the possibility of a new or different kind of accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

Because the 10 CFR Part 50 license for PNPS will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel after the certifications required by 10 CFR 50.82(a)(1) are docketed for PNPS as specified in 10 CFR 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation are no longer credible. The only remaining credible accidents are the FHA and a radioactive waste handling accident (HIC Drop Event). The proposed amendment does not adversely affect the inputs or assumptions of any of the design basis analyses that impact the remaining DBAs.

The proposed changes are limited to those portions of the OL and TS that are not related to the safe storage of irradiated fuel. The requirements that are proposed to be revised or deleted from the PNPS OL and TS are not credited in the existing accident analyses for the remaining DBAs; and as such, do not contribute to the margin of safety associated with the accident analyses. Postulated design basis accidents involving the reactor will no longer be possible because the reactor will be permanently shut down and defueled and PNPS will no longer be authorized to operate the reactor.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, ENO concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 PRECEDENT

The proposed changes to the PNPS OL and TSs are consistent with the intent of the license and accompanying PDTS issued to facilities that have been permanently shut down and defueled: (1) Vermont Yankee Nuclear Power Station, for which an amendment was issued on October 7, 2015 (Reference 29); (2) Kewaunee Power Station, for which an amendment was issued on February 13, 2015 (Reference 30); (3) San Onofre Nuclear
Generating Station, Units 2 and 3, for which an amendment was issued on July 17, 2015

(Reference 31); and (4) Crystal River Nuclear Plant, Unit 3, for which an amendment was
issued on September 4, 2015 (Reference 32).

4.4 CONCLUSION

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATIONS

This license amendment request meets the eligibility criteria for categorical exclusion from environmental review set forth in 10 CFR 51.22(c)(9) as follows:

(i) The proposed amendment involves no significant hazards consideration.

As described in Section 4.2 of this evaluation, the proposed changes do not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed amendment does not involve any physical alterations to the facility configuration that could lead to a change in the type or amount of effluent release offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above, ENO concludes that the proposed change meets the eligibility criteria for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6. REFERENCES

1. Letter, Entergy Nuclear Operations, Inc. to NRC, "Notification of Permanent Cessation of Power Operations," dated November 10, 2015 (Letter Number: 2.15.080) (ML15328A053)

2. Letter, NRC to Entergy Nuclear Operations, Inc., Pilgrim Nuclear Power Station – Issuance of Amendment Regarding Administrative Controls for Permanently Defueled Condition (CAC No. MF9304), dated July 10, 2017 (ML17066A130)

3. Calculation No. M1421, "Offsite Doses Following the Drop of a High Integrity Container," Revision 0

4. EPA 520/1-88-020, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration, and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (ORNL, September 1988)

5. EPA 402-R-93-081, Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil" (ORNL, September 1993)

6. Regulatory Guide 1.145, "Atmospheric Dispersion Models or Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982

7. Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors," Revision 1, July 1977

8 SAND87-2808, "The Potential Consequences and Risks of Highway Accidents Involving Gamma-Emitting Low Specific Activity (LSA) Waste" (Sandia National Laboratories, August 1988)

9. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2 (U.S. NRC, June 1974)

10. DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities, Volume 1 – Analysis of Experimental Data," (United States Department of Energy, December 1994)

11. Calculation No. M1422, "Radiological Consequences of a Design Basis Fuel Handling Accident Based on the Alternate Source Term Methodology – Update for Permanent Shutdown," Revision 0

12. Calculation 32-5052589-03, "Radiological Consequences of a Design Basis Fuel Handling Accident based on the Alternative Source Term Methodology (2038 MWt)"

13. BWROG TP-10-006 "Fuel Handling Accident in the Spent Fuel Pool Generic Dose Assessment" (EC54296)

14. EPA 400/R-17/001, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," January 2017

15. GE Report NEDE-31917P, "GE11 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)"

16. GE Report NEDC-32868P, "GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)"

17. GE Report NEDC-33270P, "GNF2 Advantage Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)"

18. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995

19. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

20. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003

21. NUREG-1891, "Safety Evaluation Report Related to the License Renewal of Pilgrim Nuclear Power Station," published November 2007 (ML073241016)

22. NRC Generic Letter 2016-01, "Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools," dated April 7, 2016 (ML16097A169)

23. Letter from John A. Dent, Jr (Entergy Nuclear Operations, Inc.) to NRC, "Response to NRC Generic Letter 2016-01, "Monitoring of Neutron-Absorbing Material in Spent Fuel Pools," dated November 3, 2016 (ML16319A131)

24. Letter from Mandy Halter (Entergy Nuclear Operations, Inc.) to NRC, "Response to Request for Supplemental Information Regarding Generic Letter 2016-01, "Monitoring of Neutron Absorbing Materials in the Spent Fuel Pools" for Grand Gulf Nuclear Station Unit 1 and Pilgrim Nuclear Power Station," dated February 8, 2018 (ML18039A843)

25. Letter from Robert G. Smith (Entergy Nuclear Operations, Inc.) to NRC, "Pilgrim Nuclear Power Station (PNPS) Completion of Activities to Support Entry into the Period of Extended Operation," dated June 8, 2012 (ML12164A334)

26. Letter from Robert G. Smith (Entergy Nuclear Operations, Inc.) to NRC, "Pilgrim Nuclear Power Station (PNPS) Followup to Completion of Activities to Support Entry into the Period of Extended Operation," dated October 18, 2012 (ML12307A432)

27. Memorandum from Samuel J. Chilk (Secretary, NRC) to James M. Taylor (Executive Director for Operations, NRC), SECY-92-223 – Resolution of Deviations Identified during the Systematic Evaluation Program, dated September 18, 1992 (ML003763736)

28. Letter from Peter A. Morris, (AEC, Division of Reactor Licensing) to James M. Carroll (Boston Edison Company), Provisional Construction Permit No. CPPR-49 for PNPS, dated August 26, 1968 (ML011900193)

29. Letter, NRC to Entergy Nuclear Operations, Inc., "Vermont Yankee Nuclear Power Station – Issuance of Amendment for Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition (CAC No. MF 3714)," dated October 7, 2015 (ADAMS Accession No. ML15117A551)

30. Letter, NRC to Dominion Energy Kewaunee, Inc., "Kewaunee Power Station – Issuance of Amendment for Permanently Shutdown and Defueled Technical Specifications and Certain License Conditions (TAC No. MF 1952)," dated February 13, 2015 (ADAMS Accession No. ML14237A045)

31. Letter, NRC to Southern California Edison Company, "San Onofre Nuclear Generating Station, Units 2 and 3 – Issuance of Amendment for Permanently Shutdown and Defueled Operating License and Technical Specifications (TAC Nos. MF3774 and MF3775)," dated July 17, 2015 (ADAMS Accession No. ML15139A390)

32. Letter, NRC to Crystal River Nuclear Plant, "Crystal River Unit 3 Nuclear Generating Plant – Issuance of Amendment for Permanently Shutdown and Defueled Operating License and Technical Specifications (TAC No MF 3089)," dated September 4, 2015 (ADAMS Accession No. ML152248B286)

33. Letter, NRC to Entergy Nuclear Operations, Inc., "Pilgrim Nuclear Power Station - Issuance of Amendment Re: Alternative Source Term for the Fuel Handling Accident Dose Consequences (TAC No. MC2705)," dated April 28, 2005 (ADAMS Accession No. ML051040065)

Attachment 2

Letter Number 2.18.034

Markup of the Current Operating License, Technical Specifications and Bases Pages

ENTERGY NUCLEAR GENERATION COMPANY *

And ENTERGY NUCLEAR OPERATIONS, INC.

(PILGRIM NUCLEAR POWER STATION)

DOCKET NO. 50-293

RENEWED FACILITY ~~OPERATING~~ LICENSE

Renewed License No. DPR-35

The Nuclear Regulatory Commission (the Commission) has found that:

a. ~~Except as stated in condition 5, construction of the Pilgrim Nuclear Power Station (the facility) has been substantially completed in conformity with the application, as amended, the Provisional Construction Permit No. CPPR-49, the provisions of the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission as set forth in Title 10, Chapter 1, CFR; and~~

DELETED

b. The facility will ~~operate~~ be maintained in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; and

c. There is reasonable assurance (i) that the activities authorized by the renewed ~~operating~~ license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission; and

d. The Entergy Nuclear Generation Company (Entergy Nuclear) is financially qualified and Entergy Nuclear Operations, Inc. (ENO) is technically and financially qualified to engage in the activities authorized by this renewed ~~operating~~ license, in accordance with the rules and regulations of the Commission; and

e. Entergy Nuclear and ENO have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations; and

f. The issuance of this renewed ~~operating~~ license will not be inimical to the common defense and security or to the health and safety of the public; and

g. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of this renewed ~~operating~~ license (subject to the condition for protection of the environment set forth herein) is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements of said regulations have been satisfied~~; and~~.

h. ~~Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under~~

DELETED

* The Nuclear Regulatory Commission approved the transfer of the license from Boston Edison Company to Entergy Nuclear Generation Company on April 29, 1999.

~~10 CFR 54.21(a)(1); and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by the renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.~~

Facility Operating License No. DPR-35, dated June 8, 1972, issued to the Boston Edison Company (Boston Edison) is hereby amended in its entirety, pursuant to an Initial Decision dated September 13, 1972, by the Atomic Safety and Licensing Board, to read as follows:

1. This renewed ~~operating~~ license applies to the Pilgrim Nuclear Power Station, a single cycle, forced circulation, boiling water nuclear reactor and associated electric generating equipment (the facility), owned by Entergy Nuclear and ~~operated~~ maintained by ENO. The facility is located on the western shore of Cape Cod Bay in the town of Plymouth on the Entergy Nuclear site in Plymouth County, Massachusetts, and is described in the "Final Safety Analysis Report," as supplemented and amended.

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Entergy Nuclear:

   A. Pursuant to the Section 104b of the Atomic Energy Act of 1954, as amended (the Act) and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," a) Entergy Nuclear to possess and use and b) ENO to possess~~,~~ and use~~, and operate~~ the facility ~~as a utilization facility~~ at the designated location on the Pilgrim site;

   B. ENO, pursuant to the Act and 10 CFR 70, to ~~receive,~~ possess~~, and use~~ at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage ~~and amounts required for reactor operation,~~ as described in the Final Safety Analysis Report, as supplemented and amended;

   C. ENO, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess and use at any time any byproduct, source or special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in radiation monitoring equipment ~~calibration~~, and as fission detectors in amounts as required;

   D. ENO, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

   E. ENO, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed ~~operating~~ license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations; 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50 and Section 70.32 of 10 CFR Part 70; and is subject to all applicable

provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. ~~Maximum Power Level~~ DELETED

~~ENO is authorized to operate the facility at steady state power levels not to exceed 2028 megawatts thermal.~~

B. Technical Specifications

###

replaced with the Permanently Defueled Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~247~~, are hereby ~~incorporated in the renewed operating license~~. The licensee shall ~~operate~~ the facility in accordance with the Technical Specifications.

maintain

Permanently Defueled

C. Records

ENO shall keep facility ~~operating~~ records in accordance with the requirements of the Technical Specifications.

D. ~~Equalizer Valve Restriction -~~ DELETED

E. ~~Recirculation Loop Inoperable -~~ DELETED

F. ~~Fire Protection~~ DELETED

~~ENO shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated December 21, 1978 as supplemented subject to the following provision:~~

~~ENO may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.~~

G. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Pilgrim Nuclear Power Station Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0" submitted by letter dated October 13, 2004, as supplemented by letter dated May 15, 2006.

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee's CSP was approved by License Amendment No. 236, as supplemented by changes approved by Amendment Nos. 238, 241, 244, and 247.

DELETED

H. ~~Post-Accident Sampling System. NUREG 0737, Item II.B.3. and Containment Atmospheric Monitoring System, NUREG 0737. Item II.F.1(6)~~

~~The licensee shall complete the installation of a post-accident sampling system and a containment atmospheric monitoring system as soon as practicable, but no later than June 30, 1985.~~

DELETED

I. ~~Additional Conditions~~

~~The Additional Conditions contained in Appendix B, as revised through Amendment No. 177, are hereby incorporated into this renewed operating license. ENO shall operate the facility in accordance with the Additional Conditions.~~

J. Conditions Related to the Sale and Transfer

(1) For purposes of ensuring public health and safety, Entergy Nuclear shall provide decommissioning funding assurance of no less than $396 million, after payment of any taxes, in the decommissioning trust fund for Pilgrim upon the transfer of the Pilgrim licenses to Entergy Nuclear.

(2) Entergy Nuclear shall maintain the decommissioning trust funds in accordance with the Order, the related Safety Evaluation dated April 29, 1999, and the related application for approval of the transfer.

(3) Entergy Nuclear shall provide a Provisional Trust fund in the amount of $70 million, after payment of any taxes, in the Provisional Trust for Pilgrim upon the transfer of the Pilgrim licenses to Entergy Nuclear. The Provisional Trust shall be established and maintained in conformance with the representations made in the application for approval of the transfer.

(4) Entergy Nuclear shall have access to a contingency fund of not less than fifty million dollars ($50m) for payment, if needed, of Pilgrim operating and maintenance expenses, the cost to transition to decommissioning status in the event of a decision to permanently shut down the unit, and decommissioning costs. Entergy Nuclear will take all necessary steps to ensure that access to these funds will remain available until the full amount has been exhausted for the purposes described above. Entergy Nuclear shall inform the Director, Office of Nuclear Regulation, in writing, at such time that it utilizes any of these contingency funds. This provision does not affect the NRC's authority to assure that adequate funds will remain available in the plant's separate decommissioning fund(s), which Entergy Nuclear shall maintain in accordance with NRC regulations. Once the plant has been placed in a safe-shutdown condition following a decision to decommission, Entergy Nuclear will use any remainder of the $50m contingency fund that has not been used to safely operate and maintain the plant to support the safe and prompt decommissioning of the plant, to the extent such funds are needed for safe and prompt decommissioning.

(5) The Decommissioning Trust agreement(s) shall be in a form which is acceptable to the NRC and shall provide, in addition to any other clauses, that:

a) Investments in the securities or other obligations of Entergy Nuclear, Entergy Corporation, their affiliates, subsidiaries or associates, or their successors or assigns shall be prohibited. In addition, except for investments tied to market indexes or other non-nuclear sector mutual funds, investments in any entity owning one or more nuclear power plants is prohibited.

b) The Director, Office of Nuclear Reactor Regulation, shall be given 30 days prior written notice of any material amendment to the trust agreement(s).

K. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

L. The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

DELETED

M. ~~Upon Implementation of Amendment No. 231 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage required by SR 4.7.6.2.c in accordance with TS 5.5.8.c.(i), the assessment of CRE habitability as required by Specification 5.5.8.c.(ii), and the measurement~~

~~of CRE pressure as required by Specification 5.5.8.d shall be considered met as follows:~~

~~(a) The first performance of SR 4.7.2.6.5.c in accordance with Specification 5.5.8.c.(i) shall be within the specified frequency of 6 years, plus the 18 month allowance as defined by SURVEILLANCE INTERVAL measured from December 5, 2005, the date of the most recent successful tracer gas test, as stated in Entergy's letter "Follow Up Response to NRC Generic Letter 2003-01" (EN0 2.06.019), dated March 20, 2006, or within 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.~~

~~(b) The first performance of the periodic assessment of CRE habitability Specification 5.5.8.c.(ii) shall be within 3 years, plus the 9 month allowance of SURVEILLANCE INTERVAL as measured from December 5, 2005, the date of the most recent successful tracer gas test, as stated in Entergy's letter "Follow Up Response to NRC Generic Letter 2003-01" (EN0 2.06.019), dated March 20, 2006, or within 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.~~

~~(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.8.d shall be within 24 months, plus the 180 day allowance of the SURVEILLANCE INTERVAL as measured from the date of the most recent successful pressure measurement test or within 180 days if not performed previously.~~

DELETED

4. ~~This license is subject to the following condition for the protection of the environment: Boston Edison shall continue, for a period of five years after initial power operation of the facility, an environmental monitoring program similar to that presently existing with the Commonwealth of Massachusetts (and described generally in Section C-III of Boston Edison's Environmental Report, Operating License Stage dated September, 1970) as a basis for determining the extent of station influence on marine resources and shall mitigate adverse effects, if any, on marine resources.~~

DELETED

5. ~~Boston Edison has not completed as yet construction of the Rad Waste Solidification System and the Augmented Off-Gas System. Limiting conditions concerning these systems are set forth in the Technical Specifications.~~

DELETED

6. ~~Pursuant to Section 105c(8) of the Act, the Commission has consulted with the Attorney General regarding the issuance of this operating license. After said consultation, the Commission has determined that the issuance of this license, subject to the conditions set forth in this subparagraph 6, in advance of consideration of and findings with respect to matters covered in Section 105c of the Act, is necessary in the public interest to avoid unnecessary delay in the operation of the facility. At the time this operating license is being issued an antitrust proceeding has not been noticed. The Commission, accordingly, has made no determination with respect to matters covered in Section 105c of the Act, including conditions, if any, which may be appropriate as a result of the outcome of any antitrust proceeding. On the basis of its findings made as a result of an antitrust proceeding, the Commission may continue this license as issued, rescind this license or amend this license to include such conditions as the Commission~~

~~deems appropriate. Boston Edison and others who may be affected hereby are accordingly on notice that the granting of this license is without prejudice to any subsequent licensing action, including the imposition of appropriate conditions, which may be taken by the Commission as a result of the outcome of any antitrust proceeding. In the course of its planning and other activities, Boston Edison will be expected to conduct itself accordingly.~~

7. The information in the FSAR supplement, submitted pursuant to 10 CFR 54.21(d), as supplemented by Commitments Nos. 3, 8, 9, 13, 15, 18, 19, 21, 22, 24, 25, 26, 27, 28, 30, 31, 33, 34, 35, 36, 37, 39, 40, 46, 51, and 52 of Appendix A of NUREG-1891, "Safety Evaluation Report Related to the License Renewal of Pilgrim Nuclear Power Station" dated June 2007, as supplemented, is henceforth part of the FSAR which will be updated in accordance with 10 CFR 50.71(e). ~~In addition, the licensee shall incorporate into its FSAR the "Description of Program" from Table 3.0-1 "FSAR Supplement for Aging Management of Applicable Systems" of License Renewal Interim Staff Guidance LR-ISG-2011-05 "Ongoing Review of Operating Experience."~~

   The licensee may make changes to the programs and activities described in the FSAR supplement and Commitments Nos. 3, 8, 9, 13, 15, 18, 19, 21, 22, 24, 25, 26, 27, 28, 30, 31, 33, 34, 35, 36, 37, 39, 40, 46, 51, and 52 of Appendix A of NUREG-1891, as supplemented, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

DELETED

8. ~~The licensee's FSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as supplemented by Commitments Nos. 3, 8, 9, 13, 15, 18, 19, 21, 22, 24, 25, 26, 27, 28, 30, 31, 33, 34, 35, 36, 37, 39, 40, 46, 51, and 52 of Appendix A of NUREG-1891, as supplemented, along with the FSAR description regarding consideration of operating experience for license renewal aging management programs in Condition 7 above, describes certain future programs and activities to be completed before the period of extended operation. The licensee shall complete these activities no later than June 8, 2012, and shall notify the NRC in writing when implementation of these activities is complete.~~

DELETED

9. ~~Capsule withdrawal schedule – For the renewed operating license term, all capsules in the reactor vessel that are removed and tested must meet the requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the staff prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the staff, as required by 10 CFR Part 50, Appendix H.~~

until the Commission notifies the licensee in writing that the license is terminated

10. This license is effective as of the date of issuance and ~~shall expire June 8, 2032~~.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signature on File

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Permanently Defueled

Attachment~~s:~~
Appendix A - Technical Specifications (Radiological)
~~Appendix B - Additional Conditions~~
Date of Issuance: ~~May 29, 2012~~

TBD

NOTE THAT THE FOLLOWING INCORPORATES AMENDMENT 246 - ADMINISTRATIVE CHANGES DUE TO PERMANENT SHUTDOWN BECAUSE IT WILL BE IMPLEMENTED PRIOR TO THIS THE IMPLEMENTATION OF THIS AMENDMENT - THIS NOTE WILL NOT BE INCLUDED IN THE RETYPED TECHNICAL SPECIFICATIONS

APPENDIX A

TO

FACILITY ~~OPERATING~~ LICENSE DPR-35

PERMANENTLY DEFUELED

TECHNICAL SPECIFICATION AND BASES

S

FOR

PILGRIM NUCLEAR POWER STATION

PLYMOUTH, MASSACHUSETTS

ENTERGY NUCLEAR and ENTERGY NUCLEAR OPERATIONS, INC.

~~APPENDIX B~~

~~ADDITIONAL CONDITIONS~~

~~OPERATING LICENSE NO. DPR-35~~

~~Entergy Nuclear Operations, Inc. shall comply with the following conditions on the schedules noted below:~~

~~Amendment Number~~	~~Additional Conditions~~	~~Implementation Date~~
~~177~~	~~The licensee is authorized to relocate certain Technical Specifications requirements to licensee-controlled documents. Implementation of this amendment shall include relocation of various sections of the technical specifications to the appropriate documents as described in the licensee's application dated September 19, 1997, and in the staff's safety evaluation attached to this amendment.~~	~~The amendment shall be implemented within 30 days from July 31, 1998, except that the licensee shall have until the next scheduled Updated Final Safety Analysis Report (UFSAR) update to incorporate the UFSAR relocations.~~

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1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

ACTION	ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.
~~AUTOMATIC PRIMARY CONTAINMENT ISOLATION VALVES~~	~~Are primary containment isolation valves which receive an automatic primary containment group isolation signal.~~
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with the provisions of the CERTIFIED FUEL HANDLER Training and Retraining Program.
~~COLD CONDITION~~	~~Reactor coolant temperature equal to or less than 212°F.~~
~~CORE ALTERATION~~	~~CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:~~ ~~a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and~~ ~~b. Control rod movement, provided there are no fuel assemblies in the associated core cell.~~ ~~Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~
~~CORE OPERATING LIMITS REPORT (COLR)~~	~~The COLR is a reload-cycle specific document that provides core operating limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these operating limits is addressed in individual specifications.~~
~~DESIGN POWER~~	~~DESIGN POWER means a steady state power level of 2028 thermal megawatts.~~
~~FIRE SUPPRESSION WATER SYSTEM~~	~~A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source(s); gravity tank(s) or pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include hydrant post indicator valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.~~
~~HOT STANDBY CONDITION~~	~~HOT STANDBY CONDITION means operation with coolant temperature greater than 212°F, system pressure less than 600 psig, the main steam isolation valves closed and the mode switch in startup.~~
IMMEDIATE	IMMEDIATE means that the required action will be initiated as soon as practicable considering the safe ~~operation~~ maintenance of the ~~unit~~ facility and the importance of the required action.

~~INSTRUMENT CALIBRATION~~

~~An INSTRUMENT CALIBRATION means the adjustment of an instrument signal output so that it corresponds, within acceptable range and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm or trip.~~

~~INSTRUMENT CHANNEL~~

~~An INSTRUMENT CHANNEL means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.~~

~~INSTRUMENT CHECK~~

~~An INSTRUMENT CHECK is a determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.~~

~~INSTRUMENT FUNCTIONAL TEST~~

~~An INSTRUMENT FUNCTIONAL TEST means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.~~

~~LEAKAGE~~

~~a. Identified LEAKAGE:~~

~~1. Reactor coolant LEAKAGE into drywell collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or~~

~~2. Reactor coolant LEAKAGE into the drywell atmosphere from sources which are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be Pressure Boundary Leakage.~~

~~b. Unidentified LEAKAGE:~~

~~Unidentified LEAKAGE shall be all reactor coolant leakage which is not Identified Leakage.~~

~~c. Pressure Boundary LEAKAGE~~

~~Pressure Boundary LEAKAGE shall be leakage through a non- isolable fault in a reactor coolant system component body, pipewall or vessel wall.~~

LIMITING CONDITIONS FOR OPERATION (LCO)

The LIMITING CONDITIONS FOR OPERATION specify the minimum acceptable levels of system performance necessary to assure safe ~~startup and operation~~ maintenance of the facility. When these conditions are met, the ~~plant~~ facility can be ~~operated~~ maintained safely and abnormal situations can be safely controlled.

Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be considered a failure to meet the LCO.

1.0 DEFINITIONS

~~LIMITING SAFETY SYSTEM SETTING (LSSS)~~	~~The LIMITING SAFETY SYSTEM SETTINGS are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.~~
~~LOGIC SYSTEM FUNCTIONAL TEST~~	~~A LOGIC SYSTEM FUNCTIONAL TEST means a test of all relays and contacts of a logic circuit from sensor to activated device to insure components are operable per design intent. Where practicable, action will go to completion (i.e., pumps will be started and valves opened)~~
~~MINIMUM CRITICAL POWER RATIO (MCPR)~~	~~The value of critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.~~
~~MODE~~	~~The reactor MODE is that which is established by the mode-selector-switch. The MODES include:~~

~~Startup MODE~~

~~In this MODE the reactor protection scram trip, initiated by main steam line isolation valve closure, is bypassed when reactor pressure is less than 600 psig, the low pressure main steam line isolation valve closure trip is bypassed, the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service.~~

~~Run MODE~~

~~In this MODE the reactor system pressure is at or above 785 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.~~

~~Shutdown MODE~~

~~The reactor is in the shutdown MODE when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.~~

a. ~~Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.~~

b. ~~Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.~~

~~Refuel MODE~~

~~The reactor is in the refuel MODE when the mode switch is in the refuel mode position. When the mode switch is in the refuel position, the refueling interlocks are in service.~~

NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 5.3.1, but is not a CERTIFIED FUEL HANDLER.
~~OPERABLE - OPERABILITY~~	~~A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).~~
~~OPERATING~~	~~OPERATING means that a system or component is performing its intended functions in its required manner.~~
~~OPERATING CYCLE~~	~~Interval between the end of one refueling outage and the end of the next subsequent refueling outage.~~
~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~	~~The PTLR is the Pilgrim-Specific document that provides the reactor vessel Pressure-Temperature (P-T) Curves, including heat up and cool down rates and fluence and Adjusted Reference Temperature limits for Specification 3.6.A. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.5.9.~~
~~PRIMARY CONTAINMENT INTEGRITY~~	~~PRIMARY CONTAINMENT INTEGRITY means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:~~

~~1.~~ ~~All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.~~

~~2.~~ ~~At least one door in each airlock is closed and sealed~~

~~3.~~ ~~All blind flanges and manways are closed.~~

~~4.~~ ~~All automatic primary containment isolation valves and all instrument line check valves are operable or at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition.~~

~~5.~~ ~~All containment isolation check valves are operable or at least one containment valve in each line having an inoperable valve is secured in the isolated position.~~

~~PROTECTIVE ACTION~~	~~An action initiated by the protection system when a limit is reached. A PROTECTIVE ACTION can be at a channel or system level.~~

~~PROTECTIVE FUNCTION~~	~~A system PROTECTIVE ACTION which results from the PROTECTIVE ACTION of the channels monitoring a particular plant condition.~~
~~REACTOR POWER OPERATION~~	~~REACTOR POWER OPERATION is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% design power.~~
~~REACTOR VESSEL PRESSURE~~	~~Unless otherwise indicated, REACTOR VESSEL PRESSURES listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.~~
~~REFUELING INTERVAL~~	~~REFUELING INTERVAL applies only to In-service Code Testing Program surveillance tests. For the purpose of designating frequency of these code tests, a REFUELING INTERVAL shall mean at least once every 24 months.~~
~~REFUELING OUTAGE~~	~~REFUELING OUTAGE is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling. For the purpose of designating frequency of testing and surveillance, a REFUELING OUTAGE shall mean a regularly scheduled outage; however, where such outages occur within 11 months of completion of the previous REFUELING OUTAGE, the required surveillance testing need not be performed until the next regularly scheduled outage.~~
~~SAFETY LIMIT~~	~~The SAFETY LIMITS are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences, but it indicates an operational deficiency subject to regulatory review.~~
~~SECONDARY CONTAINMENT INTEGRITY~~	~~SECONDARY CONTAINMENT INTEGRITY means that the reactor building is intact and the following conditions are met:~~ ~~1. At least one door in each access opening is closed.~~ ~~2. The standby gas treatment system is operable.~~ ~~3. All automatic ventilation system isolation valves are operable or secured in the isolated position.~~
~~SIMULATED AUTOMATIC ACTUATION~~	~~SIMULATED AUTOMATIC ACTUATION means applying a simulated signal to the sensor to actuate the circuit in question.~~
~~SOURCE CHECK~~	~~A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.~~
~~STAGGERED TEST BASIS~~	~~A STAGGERED TEST BASIS shall consist of: (a) a test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals; (b) the testing of one system, subsystem, train or other designated components at the beginning of each subinterval.~~

SURVEILLANCE FREQUENCY

Each Surveillance Requirement shall be performed within the specified SURVEILLANCE INTERVAL with a maximum allowable extension not to exceed 25 percent of the specified SURVEILLANCE INTERVAL.

facility

The SURVEILLANCE FREQUENCY establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance schedule and consideration of ~~plant operating~~ conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified ~~for surveillances that are not performed during refueling outages~~.

This limitation of this definition is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

to confirm that a parameter is within limits.

SURVEILLANCE INTERVAL

The SURVEILLANCE INTERVAL is the calendar time between surveillance~~ tests~~~~, checks, calibrations, and examinations~~ to be performed ~~upon an instrument or component when it is required to be operable~~. ~~These tests may be waived when the instrument, component, or system is not required to be operable, but the instrument, component, or system shall be tested prior to being declared operable. The operating cycle interval is 24 months and the 25% tolerance of the definition of "SURVEILLANCE FREQUENCY" is applicable. The refueling interval is 24 months and the 25% tolerance specified in the definition of "SURVEILLANCE FREQUENCY" is applicable.~~

~~**TOTAL PEAKING FACTOR**~~

~~The ratio of the fuel rod surface heat flux to the heat flux of an average rod in an identical geometry fuel assembly operating at the core average bundle power.~~

~~**TRANSITION BOILING**~~

~~TRANSITION BOILING means the boiling regime between nucleate and film boiling. TRANSITION BOILING is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.~~

~~**TRIP SYSTEM**~~

~~A TRIP SYSTEM means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A TRIP SYSTEM may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.~~

NOT USED

2.0 ~~SAFETY LIMITS~~

Not Used

~~2.1 Safety Limits~~

~~2.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% of rated core flow:~~

~~THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.~~

~~2.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% of rated core flow:~~

~~MINIMUM CRITICAL POWER RATIO shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.~~

~~2.1.3 Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.~~

~~2.1.4 Reactor steam dome pressure shall be ≤ 1340 psig at any time when irradiated fuel is present in the reactor vessel.~~

~~2.2 Safety Limit Violation~~

~~With any Safety Limit not met within two hours the following actions shall be met:~~

~~2.2.1 Restore compliance with all Safety Limits, and~~

~~2.2.2 Insert all insertable control rods.~~

~~BASES:~~

~~2.0 SAFETY LIMITS~~

~~INTRODUCTION~~

~~The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish a Safety Limit such that the Minimum Critical Power Ratio (MCPR) is not less than the limit specified in Specification 2.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.~~

~~The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions.~~

~~While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling (i.e., MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling.~~

~~Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.~~

~~FUEL CLADDING INTEGRITY (2.1.1)~~

~~GE critical power correlations are applicable for all critical power calculations at pressures at or above 685 psig or core flows at or above 10% of rated flow. For operation at low pressures and low flows another basis is used as follows:~~

(Cont)

~~FUEL CLADDING INTEGRITY (2.1.1) (Cont)~~

~~Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of~~ ~~28×10^3~~ ~~lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than~~ ~~28×10^3~~ ~~lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 685 psig is conservative.~~

~~MINIMUM CRITICAL POWER RATIO (2.1.2)~~

~~The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (2), which is a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. Instead of the standard GETAB model uncertainties, revised uncertainties in accordance with references 3 and 4 were used to calculate the SLMCPR. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.~~

~~The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not result in damage to BWR fuel rods, the critical power at~~

(Cont)

~~**BASES:**~~

~~2.0 SAFETY LIMITS (Cont)~~

~~MINIMUM CRITICAL POWER RATIO (2.1.2) (Cont)~~

~~which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.~~

~~The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity Safety Limit calculation are given in Reference 1. References 3 and 4 include a tabulation of the uncertainties used in the determination of the Safety Limit MCPR and of the nominal values of the parameters used in the Safety Limit MCPR statistical analysis.~~

~~REACTOR WATER LEVEL (Shutdown Condition) (2.1.3)~~

~~With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored.~~

(Cont)

~~REACTOR STEAM DOME PRESSURE (2.1.4)~~

~~The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the reactor coolant system is not endangered. The reactor pressure limit of 1340 psig as measured in the vessel steam dome was derived from the design pressure of the reactor vessel. The peak pressures for the piping systems connected to the reactor vessel have been recalculated based on a reactor steam dome peak pressure of 1340 psig. These peak pressures are below the lowest of the transient pressures permitted by the applicable design code: ASME Boiler and Pressure Vessel (B&PV) Code (1965 Edition, including the January 1966 Addendum), for the pressure vessel, USAS Piping Code B31.1 for the steam space piping and ASME Section III for the reactor coolant system recirculation piping. The ASME B&PV Code permits pressure transients up to 10% over the design pressure (110%x1250 = 1375 PSIG). The USAS Piping Code and ASME Section III permit pressure transients and other occasional loads whose combined effect do not exceed stress levels based on the duration of the loads and the applicable service limit.~~

~~REFERENCES~~

~~1) "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (through the latest approved amendment at the time the reload analyses are performed as specified in the CORE OPERATING LIMITS REPORT).~~

~~2) General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, January 1977, NEDE-10958-PA and NEDO-10958-A.~~

~~3) "Methodology & Uncertainties for SLMCPR Evaluations," NEDC-32601-P-A (August 1999).~~

~~4) "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC-32694-P-A (August 1999).~~

~~5) "Pilgrim Nuclear Power Station Safety Valve Setpoint Increase," GE Hitachi Nuclear Energy Report, NEDC-33532P, Rev. 2 (January 2011).~~

~~6) NEDC-33292P, Rev. 3, GEXL17 Correlation for GNF2 Fuel, dated June 2009~~

~~7) NEDE-33270P, Rev. 7, GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II), Oct. 2016~~

NOT USED

3.0 ~~LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY~~

Not Used

~~3.0.1 Not Used~~

~~3.0.2 Not Used~~

~~3.0.3 Not Used~~

~~3.0.4 Not Used~~

~~3.0.5 Not Used~~

~~3.0.6 Not Used~~

~~3.0.7 Special Operations LCOs in Section 3.14 allow specified Technical Specifications requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other Technical Specification requirements remain unchanged. Compliance with Special Operations LCOs is optional. When a Special Operations LCO is desired to be met but is not met, the ACTIONS of the Special Operations LCO shall be met. When a Special Operations LCO is not desired to be met, entry into a Mode or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.~~

~~3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:~~

~~a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or~~

~~b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.~~

~~At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.~~

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

4.0.1 Not Used

4.0.2 Not Used

4.0.3 If it is discovered that a Surveillance was not performed within its specified Surveillance Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Surveillance Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

NOT USED

BASES:

Not Used

3.0 ~~LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY~~

~~3.0.1 Not Used~~

~~3.0.2 Not Used~~

~~3.0.3 Not Used~~

~~3.0.4 Not Used~~

~~3.0.5 Not Used~~

~~3.0.6 Not Used~~

~~3.0.7 There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.14 allow specified Technical Specification requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with those Technical Specification requirements. Unless otherwise specified, all the other Technical Specification requirements remain unchanged. This ensures all appropriate requirements of the Mode or other specified condition, not directly associated with or required to be changed to perform the special test or operation, will remain in effect.~~

~~The Applicability of a Special Operations LCO represents a condition not necessarily in compliance with the normal requirements of the Technical Specifications. Compliance with Special Operations LCOs is optional. A special operation may be performed either under the provisions of the appropriate Special Operations LCO or under the other applicable Technical Specification requirements. If it is desired to perform the special operation under the provisions of the Special Operations LCO, the requirements of the Special Operations LCO shall be followed. When a Special Operations LCO requires another LCO to be met, only the requirements of the LCO statement are required to be met regardless of that LCO's Applicability (i.e., should the requirements of this other LCO not be met, the ACTIONS of the Special Operations LCO apply, not the ACTIONS of the other LCO). However, there are instances where the Special Operations LCO ACTIONS may direct the other LCOs' ACTIONS be met.~~

~~It is not required to meet the Surveillances of the other LCO, unless specified in the Special Operations LCO. If conditions exist such that the Applicability of any other LCO is met, all the other LCO's requirements (ACTIONS and Surveillance Requirements) are required to be met concurrent with the requirements of the Special Operations LCO.~~

~~3.0.8 LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO~~

~~states that the supported system is not considered to be inoperable solely due to one or more snubbers not being capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.~~

~~If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the conditions and required actions entered.~~

~~LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.~~

~~LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.~~

~~LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.~~

BASES:

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

4.0.1 Not Used

4.0.2 Not Used

4.0.3 TS 4.0.3 establishes the flexibility to defer declaring ~~affected equipment inoperable or~~ an affected variable outside the specified limits when a Surveillance has not been completed within the specified Surveillance Frequency. A delay period of up to 24 hours or up to the limit of the specified Surveillance Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with the definition of "Surveillance Frequency" and not at the time that the specified Surveillance Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of the facility ~~unit~~ conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

~~When a Surveillance with a Surveillance Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, TS 4.0.3 allows for the full delay period of up to the specified Surveillance Frequency to perform the Surveillance. However, since there is no time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.~~

~~TS 4.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of reactor MODE changes imposed by required Actions.~~

Failure to comply with specified Frequencies for surveillance intervals is expected to be an infrequent occurrence. Use of the delay period established by TS 4.0.3 is a flexibility which is not intended to be used as a ~~an operational~~ convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Surveillance Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on facility ~~plant~~ risk (from delaying the Surveillance as well as any facility ~~plant~~ configuration changes required ~~or shutting the plant down~~ to perform the Surveillance) and impact on any (continued) analysis assumptions, in addition to facility ~~unit~~ conditions, planning, availability of personnel, and the time required to perform the Surveillance.

BASES:

4.03 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY (Cont'd)

~~This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation should be commensurate with the importance of the component. Missed Surveillance for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action.~~ All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then ~~the equipment is considered inoperable or~~ the variable is considered outside the specified limits and the completion times or the required actions for the applicable LCO Actions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then ~~the equipment is inoperable, or~~ the variable is outside the specified limits and the completion times of the required actions for the applicable LCO Actions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the completion time of the Actions, restores compliance with "Surveillance Frequency."

ADD HEADER - 3/4.10 SPENT FUEL STORAGE

SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.10 ~~CORE ALTERATIONS~~

storage of spent fuel

Applicability:

Applies to the ~~fuel handling and core reactivity limitations during refueling and core alterations.~~

safe storage of spent fuel

Objective:

To ensure ~~that core reactivity is within the capability of the control rods and to prevent criticality during refueling.~~

Not Used

Specification:

A. ~~Refueling Interlocks~~

~~1. During in-vessel fuel movement with equipment associated with the interlocks the refueling equipment interlocks shall be operable with the reactor mode switch locked in the "Refuel" position. If one or more required refueling equipment interlocks are inoperable:~~

~~a. Suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s) immediately.~~

~~OR~~

~~b. Insert a control rod withdrawal block AND verify all control rods are fully inserted.~~

~~2. When the reactor vessel head is removed and any control rod is withdrawn the one-rod-out interlock shall be operable with the reactor mode switch locked in the "Refuel" position. If the one-rod-out interlock is inoperable:~~

~~a. Suspend control rod withdrawal immediately.~~

~~AND~~

~~b. Initiate action to fully insert all control rods in core cells containing one or more fuel assemblies immediately.~~

SPENT FUEL STORAGE

SURVEILLANCE REQUIREMENTS

4.10 ~~CORE ALTERATIONS~~

parameter which monitors the storage of spent fuel

Applicability:

Applies to the ~~period testing of those interlocks and instrumentation used during refueling and core alterations.~~

that spent fuel is being stored safely

Objective:

To verify ~~the operability of instrumentation and interlocks used in refueling and core alterations.~~

Not Used

Specification:

A. ~~Refueling Interlocks~~

~~1. Prior to in-vessel fuel movement with equipment associated with the refueling equipment interlocks, the interlocks shall be functionally tested. They shall be tested at weekly intervals thereafter until no longer required.~~

~~2. When the reactor vessel head is removed and any control rod is withdrawn the one-rod-out interlock shall be functionally tested at weekly intervals. The functional test is not required to be performed until 1 hour following withdrawing a control rod.~~

LIMITING CONDITION FOR OPERATION

Not Used

~~3.10 CORE ALTERATIONS (Cont)~~

B. ~~Core Monitoring~~

~~During core alterations when fuel is in the vessel two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:~~

~~1. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)~~

SURVEILLANCE REQUIREMENTS

~~4.10 CORE ALTERATIONS (Cont)~~

Not Used

B. ~~Core Monitoring~~

~~Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response.~~

LIMITING CONDITION FOR OPERATION

~~3.10 CORE ALTERATIONS (Cont)~~

~~B. Core Monitoring (Cont)~~

~~2. The SRM shall have a minimum of 3 cps except as specified in 3 and 4 below.~~

~~3. Prior to spiral unloading, the SRM's shall have an initial count rate of ≥ 3 cps. During spiral unloading, the count rate on the SRM's may drop below 3 cps.~~

~~4. During spiral reload, each control cell shall have at least one assembly with a minimum exposure of 1000 MWD/ST.~~

C. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 33 feet.

SURVEILLANCE REQUIREMENTS

~~4.10 CORE ALTERATIONS (Cont)~~

~~B. Core Monitoring (Cont)~~

~~Spiral Reload~~

~~During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, up to two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these assemblies have loaded, the cps requirement is not necessary.~~

C. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

3/4.10

SPENT FUEL STORAGE

BASES:

~~3.10~~ ~~CORE ALTERATIONS~~

~~A.~~ ~~Refueling Interlocks~~

~~1. Refueling Equipment Interlocks~~

~~BACKGROUND~~

~~Refueling equipment interlocks restrict the operation of the refueling equipment or the withdrawal of control rods to reinforce unit procedures that prevent the reactor from achieving criticality during refueling. The refueling interlock circuitry senses the conditions of the refueling equipment and the control rods. Depending on the sensed conditions, interlocks are actuated to prevent the operation of the refueling equipment or the withdrawal of control rods.~~

~~One channel of instrumentation is provided to sense the position of the refueling platform, the loading of the refueling platform fuel grapple, and the full insertion of all control rods, except control rods withdrawn in accordance with LCO 3/4.14.E or fully inserted and disarmed. Additionally, inputs are provided for the loading of the refueling platform frame mounted hoist, the loading of the refueling platform monorail mounted hoist, the full retraction of the fuel grapple, and the loading of the service platform hoist. With the reactor mode switch in the shutdown or refueling position, the indicated conditions are combined in logic circuits to determine if all restrictions on refueling equipment operations and control rod insertion are satisfied.~~

~~A control rod not at its full-in position interrupts power to the refueling equipment and prevents operating the equipment over the reactor core when loaded with a fuel assembly. Conversely, the refueling equipment located over the core and loaded with fuel inserts a control rod withdrawal block in the Control Rod Drive System to prevent withdrawing a control rod.~~

~~The refueling platform has two mechanical switches that open before the platform or any of its hoists are physically located over the reactor vessel. All refueling hoists have switches that open when the hoists are loaded with fuel.~~

~~The refueling interlocks use these indications to prevent operation of the refueling equipment with fuel loaded over the core whenever any control rod is withdrawn, or to prevent control rod withdrawal whenever fuel loaded refueling equipment is over the core.~~

~~To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.~~

~~APPLICABLE SAFETY ANALYSES~~

~~A prompt reactivity excursion during refueling could potentially result in fuel failure with subsequent release of radioactive material to the environment. Criticality and, therefore, subsequent prompt reactivity excursions are prevented during the insertion of fuel, provided all control rods are fully inserted during the fuel insertion. The refueling interlocks accomplish this by preventing loading of fuel into the core with any control rod withdrawn or by preventing withdrawal of a rod from the core during fuel loading.~~

~~Refueling equipment interlocks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).~~

BASES:

~~3.10~~ ~~CORE ALTERATIONS (Cont)~~

~~A.~~ ~~Refueling Interlocks (Cont)~~

~~1. Refueling Equipment Interlocks (Cont)~~

~~SPECIFICATION 3.10.A.1 REQUIREMENTS~~

~~To prevent criticality during refueling, the refueling interlocks ensure that fuel assemblies are not loaded with any control rod withdrawn. To prevent these conditions from developing, the all-rods-in, the refueling platform position, the refueling platform fuel grapple fuel loaded, the refueling platform frame mounted hoist fuel loaded, the refueling platform monorail mounted hoist fuel loaded, the refueling platform fuel grapple fully retracted position, and the service platform hoist fuel loaded inputs are required to be operable. These inputs are combined in logic circuits, which provide refueling equipment or control rod blocks to prevent operations that could result in criticality during refueling operations.~~

~~The interlocks are required to be operable with the reactor mode switch locked in the "Refuel" position during in-vessel fuel movement with refueling equipment associated with the interlocks.~~

~~With one or more of the required refueling equipment interlocks inoperable (does not include the one-rod-out interlock addressed in Specification 3.10.A.2), the unit must be placed in a condition in which the Specification does not apply or the interlocks are not needed. This can be performed by ensuring fuel assemblies are not moved in the reactor vessel or by ensuring that the control rods are inserted and cannot be withdrawn.~~

~~Therefore, 3.10.A.1.a requires that in-vessel fuel movement with the affected refueling equipment must be immediately (i.e., in a time frame consistent with safety) suspended. This action ensures that operations are not performed with equipment that would potentially not be blocked from unacceptable operations (e.g., loading fuel into a cell with a control rod withdrawn). Suspension of in-vessel fuel movement shall not preclude completion of movement of a component to a safe position.~~

~~Alternately, 3.10.A.1.b requires that a control rod withdrawal block be inserted and that all control rods subsequently verified to be fully inserted. This action ensures that control rods cannot be inappropriately withdrawn because an electrical or hydraulic block to control rod withdrawal is in place. To the extent practicable, in the event of a failure(s) of an individual interlock, the effects of a failed interlock will be isolated to allow refueling activities to continue while the other interlocks are maintained available. As a result, the unaffected interlocks will continue to provide partial protection. Like 3.10.A.1.a these actions ensure that unacceptable operations are blocked (e.g., loading fuel into a cell with the control rod withdrawn).~~

BASES:

~~3.10~~ ~~CORE ALTERATIONS (Cont)~~

~~A.~~ ~~Refueling Interlocks (Cont)~~

~~2. Refuel Position One-Rod-Out Interlock~~

~~BACKGROUND~~

~~The refuel position one-rod-out interlock restricts the movement of control rods to reinforce unit procedures that prevent the reactor from becoming critical during refueling operations. During refueling operations, no more than one control rod is permitted to be withdrawn except as allowed by Specification 3/4.14.E.~~

~~The refuel position one-rod-out interlock prevents the selection of a second control rod for movement when any other control rod is not fully inserted. It is a logic circuit that has redundant channels. It uses the all-rods-in signal (from the control rod full-in position indicators) and a rod selection signal (from the Reactor Manual Control System).~~

~~APPLICABLE SAFETY ANALYSES~~

~~A prompt reactivity excursion during refueling could potentially result in fuel failure with subsequent release of radioactive material to the environment.~~
~~The refuel position one-rod-out interlock and adequate shutdown margin prevent criticality by preventing withdrawal of more than one control rod. With one control rod withdrawn, the core will remain subcritical, thereby preventing any prompt critical excursion.~~

~~The refuel position one-rod-out interlock satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).~~

~~SPECIFICATION 3.10.A.2 REQUIREMENTS~~

~~To prevent criticality, the refuel position one-rod-out interlock ensures no more than one control rod may be withdrawn. Therefore, the one-rod-out interlock must be operable when any control rod is withdrawn (except as allowed by Specification 3/4.14.E). The reactor mode switch must be locked in the refuel position to support the operability of the interlock.~~

~~With the refueling position one-rod-out interlock inoperable, the refueling interlocks may not be capable of preventing more than one control rod from being withdrawn. This condition may lead to criticality. Therefore, control rod withdrawal must be immediately suspended, and action must be immediately initiated to fully insert all control rods in core cells containing one or more fuel assemblies. Action must continue until all such control rods are fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and, therefore, do not have to be inserted.~~

BASES:

~~3.10~~ ~~CORE ALTERATIONS (Cont)~~

~~B.~~ ~~Core Monitoring~~

~~The source range monitors (SRMs) are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRMs in or adjacent to any core quadrant where fuel or control rods are being moved ensures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second (cps) provides assurance that neutron flux is being monitored and ensures startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident.~~

~~The limiting conditions for operation of the SRM subsystem of the neutron monitoring system are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a functional analysis of the neutron monitoring system. The specification is based on the Nuclear Safety Requirements for Plant Operation in Subsection 7.5.10 of the FSAR.~~

~~A spiral unloading program is one by which the fuel in the outermost cells (four fuel bundles surrounding a control blade) is removed first. Unloading continues by removing the remaining outermost fuel cell by cell. The center cell would be the last removed.[(1)] A spiral loading program is one by which fuel is loaded on the periphery of the previously loaded fueled region beginning around a single SRM. Spiral unloading and reloading will preclude the creation of flux traps (moderator filled cavities surrounded on all sides by fuel).~~

~~During spiral unloading, the SRMs shall have an initial count rate of ≥ 3 cps with all rods fully inserted. The count rate will diminish during fuel removal. Under the special condition of complete spiral core unloading, it is expected that the count rate of the SRMs will drop below 3 cps before all of the fuel is unloaded.~~

~~Since there will be no reactivity additions, a lower number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, the SRMs will no longer be required. Requiring the SRMs to be operational prior to fuel removal assures that the SRMs are operable and can be relied on even when the count rate may go below 3 cps.~~

~~During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, up to two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these assemblies have been loaded, the 3 cps requirement is not necessary.~~

~~[(1)] Prior to initiating spiral unloading, up to five cells may be unloaded, provided the remaining fueled portion of the core is contiguous and connected to all four SRMs. Fuel bundles are considered contiguous when loaded face adjacent.~~

BASES:

~~3.10 CORE ALTERATIONS (Cont)~~

C. Spent Fuel Pool Water Level

To ensure there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (-1 foot) and is well above the level to assure adequate cooling.

~~4.10 CORE ALTERATIONS~~

~~A. Refueling Interlocks~~

~~SPECIFICATION 4.10.A.1 REQUIREMENTS~~

~~Performance of a functional test demonstrates that each required refueling equipment interlock will function properly when a simulated or actual signal indicative of a required condition is injected into the logic. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable functional test of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.~~

~~The function test may be performed by any series of sequential, overlapping, or total channel steps so that the entire channel is tested.~~

~~The weekly frequency is based on engineering judgment and is considered adequate in view of other indications of refueling interlocks and their associated input status that are available to unit operations personnel.~~

The fuel handling accident evaluates the dropping of an irradiated fuel assembly into the spent fuel pool. The water level in the spent fuel pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

~~BASES:~~

~~4.10 CORE ALTERATIONS (Cont)~~

~~A. Refueling Interlocks (Cont)~~

~~SPECIFICATION 4.10.A.2 REQUIREMENTS~~

~~Performance of a functional test demonstrates the associated refuel position one-rod-out interlock will function properly when a simulated or actual signal indicative of a required condition is injected into the logic. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable functional test of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The functional test may be performed by any series of sequential, overlapping, or total channel steps so that the entire channel is tested. The weekly frequency of testing is considered adequate because of demonstrated circuit reliability, procedural controls on control rod withdrawals, and visual and audible indications available in the control room to alert the operator to control rods not fully inserted. To perform the required testing, if the surveillance is not current, the applicable condition may be required to be entered (i.e., a control rod must be withdrawn from its full-in position). Therefore, 4.10.A.2 is not required to be performed until 1 hour after any control rod is withdrawn.~~

~~B. Core Monitoring~~

~~Requiring the SRM's to be functionally tested prior to any core alteration ensures the SRM's will be operable at the start of that alteration. The daily response check of the SRM's ensures their continued operability.~~

4.0 DESIGN FEATURES

4.1 Site Location

Pilgrim Nuclear Power Station is located on the western shore of Cape Cod Bay in the Town of Plymouth, Plymouth County, Massachusetts and contains approximately 517 acres owned by Entergy Nuclear as shown on FSAR Figures 2.2-1 and 2.2-2. The site boundary is posted and a perimeter security fence provides a distinct security boundary for the protected area of the station.

The reactor (center line) is located approximately 1800 feet from the nearest property boundary.

Not Used

4.2 ~~Deleted~~

SPENT

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum k-infinity of 1.32 for standard core geometry, calculated at the burnup of maximum bundle reactivity, and an average U-235 enrichment of 4.6 % averaged over the axial planar zone of highest average enrichment; and

b. $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~Section 10.3.5~~ of the FSAR.

the applicable section

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

~~4.3.1.2 The new fuel storage racks are designed and shall be maintained with:~~

~~a.~~ ~~K_{eff} ≤0.95 if fully flooded with water, which includes an allowance for uncertainties as described in Section 10.2.5 of the FSAR;~~

~~b.~~ ~~K_{eff} ≤0.90 when dry, which includes an allowance for uncertainties as described in Section 10.2.5 of the FSAR; and~~

~~c.~~ ~~A nominal 6.60 inch center to center distance between fuel assemblies placed in storage racks.~~

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 115 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3859 fuel assemblies .

4.3.4 Heavy Loads

a. Loads in excess of 2000 lb. shall be prohibited from travel over fuel assemblies in the spent fuel storage pool with the exception that heavy load handling over irradiated fuel in the Multi-Purpose Canister is permitted using a single-failure-proof handling system.

b. No fuel which has decayed for less than 200 days shall be stored in racks within an arc described by the height of the cask around the periphery of the leveling platform during cask handling operations in the spent fuel pool or when a cask is in the spent fuel pool.

5.2 Organization

5.2.2 Facility Staff (continued)

b. At least one person qualified to stand watch in the control room (NON-CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER) shall be present in the Control Room when nuclear fuel is stored in the spent fuel pool.

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

d. Shift crew composition may be less than the minimum requirement of 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements and all of the following conditions are met:

1) No fuel movements are in progress;

2) No movement of loads over fuel are in progress; and

3) No unmanned shift positions during shift turnover shall be permitted while the shift crew is less than the minimum.

Not Used

e. ~~Deleted~~

f. An individual qualified in radiation protection procedures shall be on site during fuel handling operations and during movement of heavy loads over the fuel storage racks. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

Not Used

g. ~~Deleted~~

Not Used

h. The control room supervisor shall be a CERTIFIED FUEL HANDLER.

i. ~~Deleted~~

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

Not Used

b. ~~Deleted~~

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release, reports required by Specification 5.6.2 and Specification 5.6.3.

Correct the alignment. Subsection c should have the same alignment as Subsections a and b

c. Licensee initiated changes to the ODCM:

1. ~~a.~~ Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. ~~1.~~ sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   b. ~~2.~~ a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. ~~b.~~ Shall become effective after the approval of the plant manager; and

3. ~~c.~~ Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, Tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 ~~Component Cyclic or Transient Limit~~

Not Used

~~This program provides controls to track the FSAR Section C.3.4.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.~~

5.5.6 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 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 FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.6b 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).

~~5.5 Programs and Manuals~~

~~5.5.7~~ ~~Configuration Risk Management Program (CRMP)~~

~~CRMP provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed allowed outage time has been granted.~~

~~The CRMP includes the following elements:~~

~~a.~~ ~~Provisions for the control and implementation of a Level 1 at power internal event PRA-informed methodology. The assessment is capable of evaluating the applicable plant configuration.~~

~~b.~~ ~~Provisions for performing an assessment prior to entering the LCO Action Statement for preplanned activities.~~

~~c.~~ ~~Provisions for performing an assessment after entering the LCO Action Statement for unplanned entry into the LCO Action Statement activities.~~

~~d.~~ ~~Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Action Statement.~~

~~e.~~ ~~Provisions for considering other applicable risk significant contributors such as Level 2 issues and external events, quantitatively or qualitatively.~~

~~5.5.8~~ ~~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 Main Control Room Heating, Ventilation and Air Conditioning 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 5 rem whole body or its equivalent to any part of the body 5 rem total effective dose equivalent (TEDE) for the duration of the accident. 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,~~

~~5.5 Programs and Manuals~~

~~"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.~~

~~d.~~ ~~Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one subsystem of the Main Control Room Heating, Ventilation and Air Conditioning System, operating at the flow rate required by the Ventilation Filter Testing Program (VFTP), at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.~~

~~e.~~ ~~The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.~~

~~f.~~ ~~Each Surveillance Requirement shall be performed within the specified SURVEILLANCE INTERVAL with a maximum allowable extension not to exceed 25 percent of the specified SURVEILLANCE INTERVAL. The SURVEILLANCE INTERVAL requirement is applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.~~

~~5.5.9~~ ~~Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)~~

~~a.~~ ~~RCS pressure and temperature limits for heatup, cool-down, low temperature operation criticality and hydrostatic testing as well as heatup and cool-down rates shall be established and documented in the PTLR for the following:~~

~~i)~~ ~~Limiting conditions for Operation Section 3.6.A.2~~

~~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", April 2007~~

~~c.~~ ~~The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any reason or supplement thereto.~~

(continued)

5.6 Reporting Requirements

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility shall be submitted in accordance with 10 CFR 50.36a by May 15th of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be consistent with the objectives outlined in the ODCM and process control procedures and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

~~5.6.4~~ ~~Not Used~~

~~5.6.5~~ ~~Core Operating Limits Report (COLR)~~

~~a.~~ ~~Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:~~

~~1.~~ ~~Table 3.1.1 – APRM High Flux trip level setting~~

~~2.~~ ~~Table 3.2.C – APRM Upscale trip level setting~~

~~3.~~ ~~3.11.A – Average Planar Linear Heat Generation Rate (APLHGR)~~

~~4.~~ ~~3.11.B – Linear Heat Generation Rate (LHGR)~~

~~5.~~ ~~3.11.C – Minimum Critical Power Ratio (MCPR)~~

~~6.~~ ~~3.11.D – Power/Flow Relationship During Power Operation~~

~~b.~~ ~~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:~~

~~1.~~ ~~NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (through the latest NRC approved amendment at the time the reload analyses are performed as specified in the COLR).~~

(Continued)

5.6 Reporting Requirements

~~5.6.5~~ ~~(continued)~~

~~c.~~ ~~The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.~~

~~d.~~ ~~The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.~~

Attachment 3

Letter Number 2.18.034

Retyped Renewed Facility License, Permanently Defueled Technical Specifications, and Permanently Defueled Technical Specifications Bases Pages

ENTERGY NUCLEAR GENERATION COMPANY *

And ENTERGY NUCLEAR OPERATIONS, INC.

(PILGRIM NUCLEAR POWER STATION)

DOCKET NO. 50-293

RENEWED FACILITY LICENSE

Renewed License No. DPR-35

The Nuclear Regulatory Commission (the Commission) has found that:

a. DELETED

b. The facility will be maintained in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; and

c. There is reasonable assurance (i) that the activities authorized by the renewed license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission; and

d. The Entergy Nuclear Generation Company (Entergy Nuclear) is financially qualified and Entergy Nuclear Operations, Inc. (ENO) is technically and financially qualified to engage in the activities authorized by this renewed license, in accordance with the rules and regulations of the Commission; and

e. Entergy Nuclear and ENO have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations; and

f. The issuance of this renewed license will not be inimical to the common defense and security or to the health and safety of the public; and

g. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of this renewed license (subject to the condition for protection of the environment set forth herein) is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements of said regulations have been satisfied.

h. DELETED

* The Nuclear Regulatory Commission approved the transfer of the license from Boston Edison Company to Entergy Nuclear Generation Company on April 29, 1999.

Facility Operating License No. DPR-35, dated June 8, 1972, issued to the Boston Edison Company (Boston Edison) is hereby amended in its entirety, pursuant to an Initial Decision dated September 13, 1972, by the Atomic Safety and Licensing Board, to read as follows:

1. This renewed license applies to the Pilgrim Nuclear Power Station, a single cycle, forced circulation, boiling water nuclear reactor and associated electric generating equipment (the facility), owned by Entergy Nuclear and maintained by ENO. The facility is located on the western shore of Cape Cod Bay in the town of Plymouth on the Entergy Nuclear site in Plymouth County, Massachusetts, and is described in the "Final Safety Analysis Report," as supplemented and amended.

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Entergy Nuclear:

   A. Pursuant to the Section 104b of the Atomic Energy Act of 1954, as amended (the Act) and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," a) Entergy Nuclear to possess and use and b) ENO to possess and use the facility at the designated location on the Pilgrim site;

   B. ENO, pursuant to the Act and 10 CFR 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage as described in the Final Safety Analysis Report, as supplemented and amended;

   C. ENO, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess and use at any time any byproduct, source or special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in radiation monitoring equipment, and as fission detectors in amounts as required;

   D. ENO, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

   E. ENO, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations; 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50 and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. DELETED

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ###, are hereby replaced with the Permanently Defueled Technical Specifications. The licensee shall maintain the facility in accordance with the Permanently Defueled Technical Specifications.

C. Records

ENO shall keep facility records in accordance with the requirements of the Technical Specifications.

D. DELETED

E. DELETED

F. DELETED

G. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Pilgrim Nuclear Power Station Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0" submitted by letter dated October 13, 2004, as supplemented by letter dated May 15, 2006.

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee's CSP was approved by License Amendment No. 236, as supplemented by a change approved by Amendment No. 238.

H. DELETED

I. DELETED

J. Conditions Related to the Sale and Transfer

(1) For purposes of ensuring public health and safety, Entergy Nuclearshall provide decommissioning funding assurance of no less than $396 million, after payment of any taxes, in the decommissioning trust fund for Pilgrim upon the transfer of the Pilgrim licenses to Entergy Nuclear.

(2) Entergy Nuclear shall maintain the decommissioning trust funds in accordance with the Order, the related Safety Evaluation dated April 29, 1999, and the related application for approval of the transfer.

(3) Entergy Nuclear shall provide a Provisional Trust fund in the amount of $70 million, after payment of any taxes, in the Provisional Trust for Pilgrim upon the transfer of the Pilgrim licenses to Entergy Nuclear. The Provisional Trust shall be established and maintained in conformance with the representations made in the application for approval of the transfer.

(4) Entergy Nuclear shall have access to a contingency fund of not less than fifty million dollars ($50m) for payment, if needed, of Pilgrim operating and maintenance expenses, the cost to transition to decommissioning status in the event of a decision to permanently shut down the unit, and decommissioning costs. Entergy Nuclear will take all necessary steps to ensure that access to these funds will remain available until the full amount has been exhausted for the purposes described above. Entergy Nuclear shall inform the Director, Office of Nuclear Regulation, in writing, at such time that it utilizes any of these contingency funds. This provision does not affect the NRC's authority to assure that adequate funds will remain available in the plant's separate decommissioning fund(s), which Entergy Nuclear shall maintain in accordance with NRC regulations. Once the plant has been placed in a safe-shutdown condition following a decision to decommission, Entergy Nuclear will use any remainder of the $50m contingency fund that has not been used to safely operate and maintain the plant to support the safe and prompt decommissioning of the plant, to the extent such funds are needed for safe and prompt decommissioning.

(5) The Decommissioning Trust agreement(s) shall be in a form which is acceptable to the NRC and shall provide, in addition to any other clauses, that:

a) Investments in the securities or other obligations of Entergy Nuclear, Entergy Corporation, their affiliates, subsidiaries or associates, or their successors or assigns shall be prohibited. In addition, except for investments tied to market indexes or other non-nuclear sector mutual funds, investments in any entity owning one or more nuclear power plants is prohibited.

b) The Director, Office of Nuclear Reactor Regulation, shall be given 30 days prior written notice of any material amendment to the trust agreement(s).

K. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

L. The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

M. DELETED

4. DELETED

5. DELETED

6. DELETED

7. The information in the FSAR supplement, submitted pursuant to 10 CFR 54.21(d), as supplemented by Commitments Nos. 3, 8, 9, 13, 15, 18, 19, 21, 22, 24, 25, 26, 27, 28, 30, 31, 33, 34, 35, 36, 37, 39, 40, 46, 51, and 52 of Appendix A of NUREG-1891, "Safety Evaluation Report Related to the License Renewal of Pilgrim Nuclear Power Station" dated June 2007, as supplemented, is henceforth part of the FSAR which will be updated in accordance with 10 CFR 50.71(e).

The licensee may make changes to the programs and activities described in the FSAR supplement and Commitments Nos. 3, 8, 9, 13, 15, 18, 19, 21, 22, 24, 25, 26, 27, 28, 30, 31, 33, 34, 35, 36, 37, 39, 40, 46, 51, and 52 of Appendix A of NUREG-1891, as supplemented, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

8. DELETED

9. DELETED

10. This license is effective as of the date of issuance and until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signature on File

William Dean, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendix A – Permanently Defueled Technical Specifications
(Radiological)
Date of Issuance: TBD

APPENDIX A TO FACILITY LICENSE DPR-35

PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS AND BASES

FOR PILGRIM NUCLEAR POWER STATION

PLYMOUTH, MASSACHUSETTS

ENTERGY NUCLEAR and ENTERGY NUCLEAR OPERATIONS, INC.

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

ACTION	ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with the provisions of the CERTIFIED FUEL HANDLER Training and Retraining Program.
IMMEDIATE	IMMEDIATE means that the required action will be initiated as soon as practicable considering the safe maintenance of the facility and the importance of the required action.
LIMITING CONDITIONS FOR OPERATION (LCO)	The LIMITING CONDITIONS FOR OPERATION specify the minimum acceptable levels of system performance necessary to assure safe maintenance of the facility. When these conditions are met, the facility can be maintained safely and abnormal situations can be safely controlled. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be considered a failure to meet the LCO.
NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 5.3.1, but is not a CERTIFIED FUEL HANDLER.
SURVEILLANCE FREQUENCY	Each Surveillance Requirement shall be performed within the specified SURVEILLANCE INTERVAL with a maximum allowable extension not to exceed 25 percent of the specified SURVEILLANCE INTERVAL. The SURVEILLANCE FREQUENCY establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance schedule and consideration of facility conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified. This limitation of this definition is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.
SURVEILLANCE INTERVAL	The SURVEILLANCE INTERVAL is the calendar time between surveillance tests to be performed to confirm that a parameter is within limits.

2.0 NOT USED

Not Used

3.0 NOT USED

Not Used

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

4.0.1 Not Used

4.0.2 Not Used

4.0.3 If it is discovered that a Surveillance was not performed within its specified Surveillance Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Surveillance Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

3.0 NOT USED

Not Used

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

4.0.1 Not Used

4.0.2 Not Used

4.0.3 TS 4.0.3 establishes the flexibility to defer declaring an affected variable outside the specified limits when a Surveillance has not been completed within the specified Surveillance Frequency. A delay period of up to 24 hours or up to the limit of the specified Surveillance Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with the definition of "Surveillance Frequency" and not at the time that the specified Surveillance Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of the facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

Failure to comply with specified Frequencies for surveillance intervals is expected to be an infrequent occurrence. Use of the delay period established by TS 4.0.3 is a flexibility which is not intended to be used as a convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Surveillance Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on facility risk (from delaying the Surveillance as well as any facility configuration changes required to perform the Surveillance) and impact on any (continued) analysis assumptions, in addition to facility conditions, planning, availability of personnel, and the time required to perform the Surveillance.

All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the variable is considered outside the specified limits and the completion times or the required actions for the applicable LCO Actions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the variable is outside the specified limits and the completion times of the required actions for the applicable LCO Actions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the completion time of the Actions, restores compliance with "Surveillance Frequency."

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.10 SPENT FUEL STORAGE	4.10 SPENT FUEL STORAGE
Applicability: Applies to the storage of spent fuel.	Applicability: Applies to the parameter which monitors the storage of spent fuel.
Objective: To ensure safe storage of spent fuel.	Objective: To verify that spent fuel is being stored safely.
Specification:	Specification:
A. Not Used	A. Not Used
B. Not Used	B. Not Used
C. Spent Fuel Pool Water Level	C. Spent Fuel Pool Water Level
Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 33 feet.	Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

3/4.10 SPENT FUEL STORAGE

C. Spent Fuel Pool Water Level

To ensure there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (-1 foot) and is well above the level to assure adequate cooling.

The fuel handling accident evaluates the dropping of an irradiated fuel assembly into the spent fuel pool. The water level in the spent fuel pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

4.0 DESIGN FEATURES

4.1 Site Location

Pilgrim Nuclear Power Station is located on the western shore of Cape Cod Bay in the Town of Plymouth, Plymouth County, Massachusetts and contains approximately 517 acres owned by Entergy Nuclear as shown on FSAR Figures 2.2-1 and 2.2-2. The site boundary is posted and a perimeter security fence provides a distinct security boundary for the protected area of the station.

The reactor (center line) is located approximately 1800 feet from the nearest property boundary.

4.2 Not Used

4.3 Spent Fuel Storage

4.3.1 <u>Criticality</u>

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum k-infinity of 1.32 for standard core geometry, calculated at the burnup of maximum bundle reactivity, and an average U-235 enrichment of 4.6 % averaged over the axial planar zone of highest average enrichment; and

b. $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in the applicable section of the FSAR.

4.3.2 <u>Drainage</u>

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 115 ft.

4.3.3 <u>Capacity</u>

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3859 fuel assemblies.

4.3.4 <u>Heavy Loads</u>

a. Loads in excess of 2000 lb. shall be prohibited from travel over fuel assemblies in the spent fuel storage pool with the exception that heavy load handling over irradiated fuel in the Multi-Purpose Canister is permitted using a single-failure-proof handling system.

b. No fuel which has decayed for less than 200 days shall be stored in racks within an arc described by the height of the cask around the periphery of the leveling platform during cask handling operations in the spent fuel pool or when a cask is in the spent fuel pool.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

5.1.2 The control room supervisor (CRS) shall be responsible for the shift command function.

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for facility staff and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear fuel.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Pilgrim Station Final Safety Analysis Report (FSAR);

b. The plant manager shall be responsible for overall safe operation of the facility and shall have control over those onsite activities necessary for safe storage and maintenance of the nuclear fuel;

c. The specified corporate officer for Pilgrim shall have corporate responsibility for the safe storage and handling of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the facility to ensure safe management of nuclear fuel; and

d. The individuals who train the CERTIFIED FUEL HANDLERS, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

5.2.2 Facility Staff

The facility staff organization shall include the following:

a. Each duty shift shall be composed of at least one control room supervisor and one NON-CERTIFIED OPERATOR. The NON-CERTIFIED OPERATOR position may be filled by a CERTIFIED FUEL HANDLER.

(continued)

5.2.2 Facility Staff (continued)

b. At least one person qualified to stand watch in the control room (NON-CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER) shall be present in the Control Room when nuclear fuel is stored in the spent fuel pool.

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

d. Shift crew composition may be less than the minimum requirement of 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements and all of the following conditions are met:

   1) No fuel movements are in progress;

   2) No movement of loads over fuel are in progress; and

   3) No unmanned shift positions during shift turnover shall be permitted while the shift crew is less than the minimum.

e. Not Used

f. An individual qualified in radiation protection procedures shall be on site during fuel handling operations and during movement of heavy loads over the fuel storage racks. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

g. Not Used

h. The control room supervisor shall be a CERTIFIED FUEL HANDLER.

i. Not Used

5.3 Facility Staff Qualifications

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions with exceptions specified in the Quality Assurance Program Manual (QAPM).

5.3.2 An NRC approved training and retraining program for CERTIFIED FUEL HANDLERS shall be maintained.

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. Not Used

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

5.5 Programs and Manuals

The following programs shall be established, implemented and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release, reports required by Specification 5.6.2 and Specification 5.6.3.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

    b. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after the approval of the plant manager; and

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

(continued)

5.5 Programs and Manuals (continued)

5.5.2 Not Used

5.5.3 Not Used

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative 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;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

(continued)

5.5.4 Radioactive Effluent Controls Program (continued)

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site boundary to areas at or beyond the site boundary conforming to the following:

1. For noble gases: Less than or equal to 500 mrem/yr to the whole body and less than or equal to 3000 mrem/yr to the skin, and

2. For Iodine-131, Iodine-133, Tritium, and all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, Tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Not Used

5.5.6 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 or Bases that requires NRC approval pursuant to 10 CFR 50.59.

(continued)

5.5.6 Technical Specifications (TS) Bases Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.6b 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).

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not Used

5.6.2 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 by May 15 of each year. The report shall include summaries, interpretations, and analyses 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 the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include a summary of the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility shall be submitted in accordance with 10 CFR 50.36a by May 15th of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be consistent with the objectives outlined in the ODCM and process control procedures and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 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). Individuals qualified in radiation protection procedures (e.g., radiation protection personnel) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following facility radiation protection procedures for entry into such high radiation areas.

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

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset 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 are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the radiation protection manager in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the control room supervisor on duty or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

(continued)

5.7 High Radiation Area (continued)

5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.