Wisconsin Electric
A WISCONSIN ENERGY COMPANY

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NPL 2000-0142

March 15, 2000

Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
SUPPLEMENT 1 TO APPLICATION FOR AMENDMENT TO
FACILITY OPERATING LICENSE APPENDIX A
TECHNICAL SPECIFICATIONS IMPROVEMENT PROJECT
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On November 15, 1999, Wisconsin Electric Power Company (WE), licensee for the Point Beach Nuclear Plant (PBNP), submitted an application to amend Appendix A, Technical Specifications, for Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Power Plant, Units 1 and 2, respectively (reference letter NPL 99-0669). The application proposed to convert the Point Beach Current Technical Specifications (CTS) to the Point Beach Improved Technical Specifications (ITS). That application contained documentation for ITS Chapters 1.0 and 2.0 and Sections 3.0 through 3.9.

Documentation for ITS Chapters 4.0 and 5.0 is enclosed with this letter. The detailed description and justification for this final section of the proposed license amendment consists of one volume. An explanation of the contents and organization of this volume is included in Attachments 1 and 2 of this letter, and is described below:

Attachment 1, "Summary of the Proposed License Amendment Request," summarizes the organization and content of this submittal.

Attachment 2, "Beyond Scope Changes," provides a listing of those changes that are different than both the CTS and NUREG-1431, Revision 1 (including pending changes).

The conversion to ITS also requires establishment of a Core Operating Limits Report (COLR) and Pressure Temperature Limits Report (PTLR). These applications were provided separately as Technical Specification Change Requests 218 and 219 respectively (reference NPL 2000-0114 dated March 2, 2000 and NPL 2000-0123 dated March 10, 2000).

Any revisions to the Point Beach Final Safety Analysis Report (FSAR) required as a result of this proposed license amendment request will be made in accordance with 10 CFR 50.71(e). As stated in the initial submittal, it is our intent to implement this license amendment request at Point Beach in 2001. This projection is based on the time required for procedure revisions, including the development of new programs, training schedules for both licensed and

A001

non-licensed operators, and the timing of implementation with respect to refueling outages. This projection is also based on the NRC review being completed and a Safety Evaluation issued by approximately March 2001.

The proposed changes in this license amendment request supplement have been reviewed by both the on-site (Manager's Supervisory Staff) and the off-site review committees in accordance with Point Beach Technical Specifications requirements.

Wisconsin Electric has determined that the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, Wisconsin Electric concludes that the proposed amendments meet the categorical exclusion requirements of 10 CFR 51.22(c)(9) and that an environmental impact appraisal need not be prepared.

Wisconsin Electric is notifying the State of Wisconsin of our application for this license amendment request by transmitting a copy of this letter, and its attachments, to the Public Service Commission of Wisconsin.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects, these statements are not based entirely on my personal knowledge, but on information furnished by cognizant Wisconsin Electric employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Sincerely,

Richard L. Ward

FOR Mark Reddemann
Site Vice President
Point Beach Nuclear Plant

Subscribed to and sworn before me
on this 15th day of March, 2000

[signature]
Notary Public, State of Wisconsin

My Commission expires on September 16, 2001.

CAC/tja

Attachments

cc: NRC Regional Administrator
NRC Resident Inspector
NRC Project Manager
PSCW

SUMMARY OF THE IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL
(page 1 of 4)

This submittal supplement for the conversion to Improved Technical Specifications (ITS) was prepared with consideration of the guidance contained in NEI 96-06, "Improved Technical Specifications Conversion Guidance." This portion of the submittal consists of one volume and related attachments to the transmittal letter. The enclosed volume consist of ITS Section packages. Provided below is a brief description of the contents of each of the Section packages, as well as a brief explanation of how the material was prepared and the terminology that is being used.

This supplement to the Technical Specifications Improvement Project (TSIP) for Point Beach is based on Revision 1 of NUREG-1431 and the Point Beach CTS with amendments through February 2000. All approved Technical Specifications Task Force (TSTF) generic change travelers through January 3, 2000 have also been reviewed for incorporation.

This submittal contains TSIP documentation for Chapters 4.0 and 5.0.

The Cross-references, Descriptions of Changes, Justifications for Deviations, and No Significant Hazards Considerations are reports generated from a database that contains the details of the conversion submittal. The CTS and NUREG-1431 mark-ups are Microsoft Word® documents with graphical overlays. The Point Beach ITS sections are also Microsoft Word® documents. The use of graphical overlays rather than hand-marking provides substantial benefits for readability and document searching. Although the use of graphical overlays creates the potential for unintended format changes within the mark-up documents, we have not discovered any significant format errors associated with the use of overlays.

The criteria in 10 CFR 50.36(c)(2)(ii) were applied to the Point Beach Current Technical Specifications (CTS) requirements. For those CTS requirements that do not meet any of the NRC selection criteria and are not retained in the proposed ITS, an evaluation of the CTS requirement against the criteria is provided.

SECTIONS 4.0 THROUGH 5.0

Each Section package corresponds to a Section of NUREG-1431, Revision 1. Each Section package contains the required information to review the conversion to ITS. The information in each package is organized as described below:

Cross-Reference Report

> The cross-reference report contains two tables arranged in alpha-numeric order. One table is the CTS to ITS cross-reference and the other is the ITS to CTS cross-reference.

SUMMARY OF THE IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL
(page 2 of 4)

Descriptions of Changes (DOC)

The DOC report contains the descriptions of changes to the CTS that are proposed for conversion to ITS. The descriptions of changes are listed in alpha-numeric order. The DOCs are categorized as follows:

Designator	Category
A	ADMINISTRATIVE - changes to the CTS that result in no additional or reduced restrictions or flexibility. These changes are supported in aggregate by a single finding of no significant hazards consideration (NSHC).
L	LESS RESTRICTIVE "Specific" - changes to the CTS that result in reduced restrictions or added flexibility. Each less restrictive change is supported by a corresponding evaluation supporting a finding of NSHC.
LA	LESS RESTRICTIVE - changes to the CTS that eliminate detail and relocate the detail to a licensee controlled document. Typically, this involves details of system design and function, or procedural detail on methods of conducting a surveillance. These changes are supported in aggregate by a single NSHC.
LB	LESS RESTRICTIVE "Generic" - changes that remove details that are duplicative of other regulatory requirements. These changes are supported in aggregate by a single NSHC.
M	MORE RESTRICTIVE - changes to the CTS that result in added restrictions or reduced flexibility. These changes are supported in aggregate by a single NSHC.
R	RELOCATIONS - changes to the CTS that encompass the requirements that do not meet the selection criteria of 10 CFR 50.36(c)(2)(ii). These changes are supported in aggregate by a single NSHC.

SUMMARY OF THE IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL
(page 3 of 4)

CTS Mark-up

The CTS mark-up contains annotated copies of the CTS pages which show the disposition of existing requirements into the proposed ITS. The pages are arranged in CTS order. The upper right hand corner of the CTS page is annotated with all the NUREG-1431 Section numbers in which the CTS page occurs. Items on the CTS page that are addressed in other proposed ITS sections are annotated with the appropriate location.

The CTS pages in the Section packages reflect License Amendments issued as of February 2000. Future License Amendment Requests (other than PTLR and COLR) will be incorporated after approval of those License Amendment Requests.

Where a proposed ITS requirement differs from a CTS requirement, individual details of the CTS revision are annotated with alpha-numeric designators which relate to the appropriate Description of Change (DOC). The DOC provides a concise justification for the change. The alpha-numeric designators also correspond to the evaluations supporting a finding of No Significant Hazards Consideration (NSHC) for each Section package.

NUREG-1431 Justifications for Deviations (JFD)

The JFD report describes and justifies the differences between the NUREG-1431, Revision 1, and the proposed ITS specifications and bases.

NUREG-1431 Markup

The NUREG-1431 mark-up contains annotated copies of the applicable NUREG-1431, Revision 1 sections, which show how the proposed ITS differs from the NUREG-1431, Revision 1, requirements. Where a proposed ITS requirement differs from the NUREG-1431, individual details of the change are annotated with alpha-numeric designators which relate to the appropriate Justification for Deviation (JFD). The JFD provides a concise justification for the change. The NUREG-1431 mark-up also shows the incorporation of approved generic changes (Technical Specifications Task Force [TSTF] change travelers) that are applicable.

The JFDs are numbered sequentially for each NUREG-1431 Section.

SUMMARY OF THE IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL

No Significant Hazards Consideration (NSHC)

The NSHC report contains the evaluations required by 10 CFR 50.91(a) supporting a finding of No Significant Hazard Consideration. Based on inherent similarities in the evaluations, generic evaluations for a finding of NSHC have been written for each category of changes, except Category "L," which have specific NSHC evaluations.

Proposed Improved Technical Specifications (ITS)

The proposed ITS contains the Point Beach specific Improved Technical Specifications. The proposed ITS is derived by incorporation of all NUREG-1431 mark-up information to the applicable NUREG-1431 section.

Beyond Scope Changes

NUREG-1431 Section	Explanation			
4.0	NUREG-1431 states that new fuel storage rack design is based on an aqueous foam moderator. The PBNP design basis is optimum moderator density conditions. The PBNP CTS did not state this design basis. Therefore, this design basis has been added into the proposed ITS. The aqueous foam design basis was not adopted.			
	CTS:	**DOC:**	**NUREG:**	**JFD:**
	NEW	M.03	SPEC 4.03.01.02.C	04
5.03	The PBNP CTS does not currently reference Regulatory Guide 1.8. NUREG-1431 identifies RG 1.8, Revision 2, 1987, or more recent revisions, as the valid reference. The appropriate reference for PBNP is RG 1.8, Revision 1, September 1975.			
	CTS:	**DOC:**	**NUREG:**	**JFD:**
	15.06.03.01	M.01	SPEC 5.03.01	01
5.05	NUREG-1431 has been modified by the addition of a Containment Leakage Rate Testing Program. The CTS requirements for the Containment Leakage Rate Testing Program are retained, with the addition of air lock testing requirements.			
	CTS:	**DOC:**	**NUREG:**	**JFD:**
	NEW	M.16	N/A	15
	NEW	M.16	N/A	15
	NEW	M.16	N/A	15

Cross-Reference Report - NUREG-1431 Section 4.0
ITS to CTS

10-Mar-00

ITS	CTS	DOC
SPEC 4.01	15.05.01	A.01
SPEC 4.02	15.05.03.A	A.01
SPEC 4.02.01	15.05.03.A.01	A.01
	15.05.03.A.01	L.01
	15.05.03.A.02	L.01
	15.05.03.A.02	LA.02
SPEC 4.02.02	15.05.03.A.04	A.01
SPEC 4.03	15.05.04	A.01
SPEC 4.03.01.01	15.05.04.02	A.01
SPEC 4.03.01.01.A	15.05.04.02	A.01
SPEC 4.03.01.01.A.01	15.05.04.02	A.01
SPEC 4.03.01.01.A.02	15.05.04.02	A.01
SPEC 4.03.01.01.B	15.05.04.02	A.01
SPEC 4.03.01.01.C	15.05.04.02	M.02
SPEC 4.03.01.02	15.05.04.02	A.01
SPEC 4.03.01.02.A	15.05.04.02	A.01
SPEC 4.03.01.02.C	15.05.04.02	A.01
	NEW	M.03
SPEC 4.03.01.02.D	15.05.04.02	M.02
SPEC 4.03.02	NEW	M.01
SPEC 4.03.03	NEW	A.05

Cross-Reference Report - NUREG-1431 Section 4.0
CTS to ITS

10-Mar-00

CTS	ITS	DOC
15.05.01	SPEC 4.01	A.01
15.05.01 APPL	DELETED	A.02
15.05.01 OBJ	DELETED	A.03
15.05.02	FSAR	LA.01
15.05.02.A	FSAR	LA.01
15.05.02.A.01	FSAR	LA.01
15.05.02.A.02	FSAR	LA.01
15.05.02.B	FSAR	LA.01
15.05.02.B.01	FSAR	LA.01
15.05.02.B.02	FSAR	LA.01
15.05.02.C	FSAR	LA.01
15.05.02.C.01	FSAR	LA.01
15.05.02.C.02	FSAR	LA.01
15.05.03	DELETED	A.04
15.05.03 APPL	DELETED	A.02
15.05.03 OBJ	DELETED	A.03
15.05.03.A	SPEC 4.02	A.01
15.05.03.A.01	FSAR	LA.02
	SPEC 4.02.01	A.01
	SPEC 4.02.01	L.01
15.05.03.A.02	FSAR	LA.02
	SPEC 4.02.01	L.01
	SPEC 4.02.01	LA.02
15.05.03.A.03	FSAR	LA.02
15.05.03.A.04	FSAR	LA.02
	SPEC 4.02.02	A.01
15.05.03.A.05	FSAR	LA.02
15.05.03.A.06	FSAR	LA.02
15.05.03.B	FSAR	LA.02
15.05.03.B.01	FSAR	LA.02
15.05.03.B.02	FSAR	LA.02
15.05.03.B.02.A	FSAR	LA.02
15.05.03.B.02.B	FSAR	LA.02

Cross-Reference Report - NUREG-1431 Section 4.0
CTS to ITS

10-Mar-00

CTS	ITS	DOC
15.05.03.B.03	FSAR	LA.02
15.05.04	SPEC 4.03	A.01
15.05.04 APPL	DELETED	A.02
15.05.04 OBJ	DELETED	A.03
15.05.04.01	FSAR	LA.03
15.05.04.02	FSAR	LA.02
	FSAR	LA.03
	SPEC 4.03.01.01	A.01
	SPEC 4.03.01.01.A	A.01
	SPEC 4.03.01.01.A.01	A.01
	SPEC 4.03.01.01.A.02	A.01
	SPEC 4.03.01.01.B	A.01
	SPEC 4.03.01.01.C	M.02
	SPEC 4.03.01.02	A.01
	SPEC 4.03.01.02.A	A.01
	SPEC 4.03.01.02.C	A.01
	SPEC 4.03.01.02.D	M.02
15.07.02	FSAR	LA.04
15.07.02 F 15.07.02-01	FSAR	LA.04

Description of Changes - NUREG-1431 Section 4.0

10-Mar-00

DOC Number	DOC Text	
A.01	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).	
	CTS:	**ITS:**
	15.05.01	SPEC 4.01
	15.05.03.A	SPEC 4.02
	15.05.03.A.01	SPEC 4.02.01
	15.05.03.A.04	SPEC 4.02.02
	15.05.04	SPEC 4.03
	15.05.04.02	SPEC 4.03.01.01 SPEC 4.03.01.01.A SPEC 4.03.01.01.A.01 SPEC 4.03.01.01.A.02 SPEC 4.03.01.01.B SPEC 4.03.01.02 SPEC 4.03.01.02.A SPEC 4.03.01.02.C
A.02	The CTS provides an introductory statement (Applicability) for the given section. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431.	
	CTS:	**ITS:**
	15.05.01 APPL	DELETED
	15.05.03 APPL	DELETED
	15.05.04 APPL	DELETED

DOC Number	DOC Text
A.03	The CTS provides an introductory statement (Objective) at the beginning of this Section of the Technical Specifications which provides a brief summary of the purpose for this Section. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431.

CTS:	ITS:
15.05.01 OBJ	DELETED
15.05.03 OBJ	DELETED
15.05.04 OBJ	DELETED

DOC Number	DOC Text
A.04	CTS 15.5.3 is provided with references to various FSAR sections. These references are not being retained in ITS, because they do not establish a regulatory requirement. It is unnecessary to provide references in the Technical Specifications. References, when necessary, are provided in the Bases of the ITS. Therefore, deletion of these references is administrative in nature.

CTS:	ITS:
15.05.03	DELETED

DOC Number	DOC Text
A.05	The approved spent fuel storage pool capacity is not included in the CTS and is being added to the ITS. The approved capacity is contained in License Condition 3.E of Operating Licenses DPR-24 and DPR-27 for Point Beach Units 1 and 2, respectively. As this ITS provision is duplicative of an existing License Condition, this change is administrative only.

CTS:	ITS:
NEW	SPEC 4.03.03

DOC Number	DOC Text
L.01	The CTS does not contain a provision allowing the limited use of lead test assemblies in non-limiting locations in the reactor core. Therefore, addition of this allowance is less restricitive. Use of test assemblies in non-limiting core locations is acceptable, since analyses utilizing NRC approved codes and methods ensure that limiting core locations are identified and are occupied by fuel assemblies of approved design, thereby ensuring all safety analysis and design basis limits are met. The specifics of the fuel types to be used in the PBNP cores is also being relocated and replaced with a general provision that the fuel assembly types be limited to those analyzed with NRC approved codes. This is acceptable in that it assures that analyses are performed using approved codes and demonstrate that existing safety analysis and design limits are met.

CTS:	ITS:
15.05.03.A.01	SPEC 4.02.01
15.05.03.A.02	SPEC 4.02.01

DOC Number	DOC Text
LA.01	CTS 15.5.2 specifies the design features of the Containment System. This information does not establish a regulatory requirement, but rather provides a description of plant equipment/design which is not required to be in the Technical Specifications to provide adequate protection of public health and safety. This information is a reflection of system design and capabilities which are contained in the FSAR. Changes to the FSAR are controlled in accordance with the 10 CFR 50.59 process.

CTS:	ITS:
15.05.02	FSAR
15.05.02.A	FSAR
15.05.02.A.01	FSAR
15.05.02.A.02	FSAR
15.05.02.B	FSAR
15.05.02.B.01	FSAR
15.05.02.B.02	FSAR
15.05.02.C	FSAR
15.05.02.C.01	FSAR
15.05.02.C.02	FSAR

DOC Number	DOC Text	
LA.02	The explicit description of fuel assembly type and design as well as core design and configuration characteristics and fuel in the new fuel storage vault characteristics are being removed from the Technical Specifications. In addition, information related to the Reactor Coolant System is being removed. This information is not contained in NUREG-1431 and will not be retained in the proposed ITS. This information is not required to be in the Specifications to provide adequate protection of public health and safety. This information is a reflection of core and reactor coolant system design information which is contained in the FSAR and controlled in accordance with 10 CFR 50.59.	
	CTS:	**ITS:**
	15.05.03.A.01	FSAR
	15.05.03.A.02	FSAR SPEC 4.02.01
	15.05.03.A.03	FSAR
	15.05.03.A.04	FSAR
	15.05.03.A.05	FSAR
	15.05.03.A.06	FSAR
	15.05.03.B	FSAR
	15.05.03.B.01	FSAR
	15.05.03.B.02	FSAR
	15.05.03.B.02.A	FSAR
	15.05.03.B.02.B	FSAR
	15.05.03.B.03	FSAR
	15.05.04.02	FSAR
LA.03	Certain information presently in the CTS related to the design and characteristics of the new fuel and spent fuel storage racks is being removed from the Technical Specifications. Remaining and added information is in accordance with the guidance in NUREG 1431. The relocated information is contained in the FSAR and is subject to change in accordance with the requirements of 10 CFR 50.59. Requirements and conditions remaining within the TS are sufficient to ensure the health and safety of the public.	
	CTS:	**ITS:**
	15.05.04.01	FSAR
	15.05.04.02	FSAR

Description of Changes - NUREG-1431 Section 4.0

10-Mar-00

DOC Number	DOC Text
LA.04	The site map and information and effluent release points are being relocated to licensee controlled documents and programs including the FSAR and ODCM. These documents are changed via appropriate change mechanisms including 10 CFR 50.59. This information is not necessary within the TS to ensure the health and safety of the public.

CTS:	ITS:
15.07.02	FSAR
15.07.02 F 15.07.02-01	FSAR

DOC Number	DOC Text
M.01	Design provisions for the spent fuel storage pool that limit inadvertent drainage of the pool are not contained in the CTS. Design provisions to prevent inadvertent drainage of the spent fuel pool are described in the Final Safety Analysis Report (FSAR). This is a new condition for the TS thereby making it more restrictive. This addition reflects current design.

CTS:	ITS:
NEW	SPEC 4.03.02

DOC Number	DOC Text
M.02	CTS does not contain parameter specific design criteria related to spacing in the spent fuel and new fuel storage racks, but does contain the criticality (Keff) design limits and fuel enrichment limits. The specific, nominal fuel spacing in the racks is being added consistent with the NUREG recommended presentation. Addition of the design specific spacing information is a new, more restrictive requirement. Overall acceptance criteria for storage of fuel is not changed by this addition.

CTS:	ITS:
15.05.04.02	SPEC 4.03.01.01.C
	SPEC 4.03.01.02.D

DOC Number	DOC Text
M.03	CTS does not contain the additional design restriction on the new fuel storage racks that Keff be maintained </= 0.98 under optimum moderator density conditions. This design restriction is consistent with approved analyses. The addition of this restriction is more restrictive.

CTS:	ITS:
NEW	SPEC 4.03.01.02.C

15.5 DESIGN FEATURES

15.5.1 SITE

Applicability

Applies to the location and extent of the reactor site.

A.2

Objective

To define those aspects of the site which affect the overall safety of the installation.

A.3

Specification

The Point Beach Nuclear Plant is located on property owned by Wisconsin Electric Power Company at a site on the shore of Lake Michigan, approximately 30 miles southeast of the city of Green Bay. The minimum distance from the reactor containment center line to the site exclusion boundary as defined in 10 CFR 100.3 is 1200 meters.

15.5.2 CONTAINMENT

Applicability

Applies to those design features of the Containment System relating to operational and public safety.

Objective

To define the significant design features of the reactor containment structure.

Specifications

A. Reactor Containment

1. The reactor containment completely encloses the entire reactor and Reactor Coolant System and ensures that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even if gross failure of the Reactor Coolant System occurs. The structure provides biological shielding for both normal and accident situations.

2. The containment structure is designed for an internal pressure of 60 psig, plus the loads resulting from an earthquake producing .08g in the vertical and 0.12g in the horizontal planes simultaneously. The containment is also structurally designed to withstand an external pressure 2.0 psi higher than the internal pressure.(1)

B. Penetrations

1. All penetrations through the containment reinforced concrete pressure barrier for pipe, electrical conductors, ducts and access hatches are provided with double barriers against leakage.(2)

LA.1

2. The automatically actuated containment isolation valves are designed to close upon high pressure in the containment (set-point no higher than 6 psig) and on a safety injection signal. The actuation system is designed such that no single component failure will prevent containment isolation if required.

C. Containment Systems

1. The containment vessel has an internal spray system which is capable of providing a distributed borated water spray of at least 1200 gpm. During the initial period of spray operation, sodium hydroxide would be added to the spray water to increase the removal of iodine from the containment atmosphere.[3]

2. The containment vessel has an internal air recirculation system which consists of four ventilation fans and air coolers capable of a total heat removal capability of 41,700 Btu/sec under conditions following a loss-of-coolant accident.[4]

References:

(1) FSAR Section 5.1.2.2
(2) FSAR Section 5.1.2.6
(3) FSAR Section 6.4
(4) FSAR Section 6.3

LA.1

15.5.3 REACTOR

A.2

Applicability

Applies to the ~~reactor core,~~ Reactor Coolant System, and Emergency Core Cooling Systems.

A.3

Objective

To define ~~those design features~~ which are essential in providing for safe system operation.

Specifications

A. Reactor Core

1. General

The uranium fuel is in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 or ZIRLO™ tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. ~~Each fuel assembly nominally contains 179 fuel rods~~[1]. Where safety limits are not violated, limited substitutions of fuel rods by filler rods consisting of Zircaloy 4, ZIRLO™, or stainless steel, or by vacancies, may be made to replace damaged fuel rods if justified by cycle specific reload analysis.

LA.2

LA.2

L.1

Insert 4.0-1

2. Core

A reactor core is a core loading pattern containing any combination of 14x14 OFA and 14x14 upgraded OFA, or any combination of 422V+ and burned 14x14 OFA or burned 14x14 upgraded OFA fuel assemblies. The use of these fuel assemblies will be justified by a cycle specific reload analysis.

LA.2

L.1

LA.2

3. Burnable absorber and/or water displacer rods are incorporated for reactivity and/or power distribution control. The burnable absorber rods consist of borated pyrex glass clad with stainless steel[4]. The water displacer rods are empty burnable absorber rods containing no pyrex glass. Another type of burnable absorber may consist of a thin coating of zirconium diboride on the radial surface of selected fuel rod pellets.

4. There are 33 full-length RCC assemblies in the reactor core. The full-length RCC assemblies contain ~~a 142-inch length of~~ silver-indium-cadmium alloy clad with the stainless steel.

LA.2

LA.2

5. Neutron source assemblies may be used to provide a required minimum count rate during startup operations. A source assembly, if used, would typically consist of four source rodlets comprised of a mixture of antimony and beryllium.

6. Peripheral power suppression assemblies (PPSA) are used to reduce neutron fluence at the welds in the beltline region of the reactor vessel. Peripheral fuel assemblies may contain PPSAs, which utilize part-length hafnium absorber rods in the assembly guide tubes.

B. Reactor Coolant System

1. The design of the Reactor Coolant System complies with the code requirements[6].

2. All high pressure piping, components of the Reactor Coolant System and their supporting structures are designed to Class I requirements, and have been designed to withstand:

   a. The design seismic ground acceleration, 0.06g, acting in the horizontal and 0.04g acting in the vertical planes simultaneously, with stresses maintained within code allowable working stresses.

LA.2

b. The maximum potential seismic ground acceleration, 0.12g, acting in the horizontal and 0.08g acting in the vertical planes simultaneously with no loss of function.

3. The nominal Reactor Coolant System volume (both liquid and steam) at rated operating conditions and zero percent steam generator tube plugging is:

Unit 1 - 6500 ft^3

Unit 2 - 6643 ft^3

LA.2

References

(1) FSAR Section 3.2.3
(2) Deleted
(3) Deleted
(4) FSAR Section 3.2.3
(5) Deleted
(6) FSAR Table 4.1-9

A.4

A.1

15.5.4 FUEL STORAGE

Applicability

Applies ~~to the capacity~~ and storage arrays of new and spent fuel.

A.2

A.3

Objective

To define those aspects of fuel storage ~~relating to~~ prevention of criticality in fuel storage areas.

Specification

~~1. The new fuel storage and spent fuel pool structures are designed to withstand the anticipated earthquake loadings as Class I structures. The spent fuel pool has a stainless steel liner to ensure against loss of water.~~

LA.3

M.2

2. The new and spent fuel storage racks are designed ~~so that it is impossible to store assemblies in other than the prescribed storage locations. The fuel is stored vertically in an array~~ with ~~sufficient~~ center-to-center distance between assemblies to assure K_{eff}<0.95 with the storage pool filled with unborated water and with the fuel loading in the assemblies limited to 5.0 w/o U-235, with or without axial blanket loadings. Each assembly with a fuel loading greater than 4.6 w/o U-235 must contain Integral Fuel Burnable Absorber (IFBA) rods in accordance with Figure 15.5.4-1 for the spent fuel pool. ~~Fresh fuel assemblies with the maximum enrichment of up to 5.0 weight percent U235 and a minimum of 32 1.25X IFBA rods can utilize all available new fuel vault storage cells.~~ ~~An inspection area shall allow rotation of fuel assemblies for visual inspection, but shall not be used for storage.~~

a nominal 9.825 inch

in the spent fuel storage racks and nominal 20 inch center-to-center distance between assemblies in the new fuel storage racks

M.2

Insert 4.0-2

M.3

LA.2

LA.3

3. The spent fuel storage pool shall be filled with borated water at a concentration of at least 2100 ppm boron whenever there are spent fuel assemblies in the storage pool.

4. Spent fuel assembly storage locations immediately adjacent to the spent fuel pool perimeter or divider walls shall not be occupied by fuel assemblies which have been subcritical for less than one year.

Insert 4.0-3

M.1

A.5

< See LCO 3.7.17 >

< See LCO 3.7.16 >

A.1

Spec 4.0 Inserts

Insert 4.0-1:

L.1

Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

Insert 4.0-2:

M.3

, which includes an allowance for uncertainties as described in the FSAR. The new fuel storage racks are designed and shall be maintained with $K_{eff} \leq 0.98$ under optimum moderator density conditions, which includes an allowance for uncertainties as described in Section 9.4 of the FSAR.

Insert 4.0-3:

M.1

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 40 ft, 8 in.

A.5

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1502 fuel assemblies.

15.7.2 SITE DESCRIPTION

Figure 15.7.2-1 is a site map for the Point Beach Nuclear Plant Units 1 and 2. The site map shows the site boundary and points within the site boundary from which gaseous and liquid effluents are released. Fence locations are approximate.

LA.4

LA.4

SITE MAP

Figure 15.7.2-1

Justification For Deviations - NUREG-1431 Section 4.0

10-Mar-00

JFD Number	JFD Text	
01	The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	N/A	SPEC 4.03.01.01.D SPEC 4.03.01.01.E SPEC 4.03.01.01.F
	SPEC 4.01	SPEC 4.01
	SPEC 4.02.01	SPEC 4.02.01
	SPEC 4.02.02	SPEC 4.02.02
	SPEC 4.03.01.01.B	SPEC 4.03.01.01.B
	SPEC 4.03.01.01.C	SPEC 4.03.01.01.C
	SPEC 4.03.01.02.A	SPEC 4.03.01.02.A
	SPEC 4.03.01.02.B	SPEC 4.03.01.02.B
	SPEC 4.03.01.02.C	SPEC 4.03.01.02.C
	SPEC 4.03.01.02.D	SPEC 4.03.01.02.D
	SPEC 4.03.03	SPEC 4.03.03
02	CTS allows limited substitution of vacancies for fuel rods within fuel assemblies as supported by cycle-specific reload analyses to verify safety limits are not violated. This additional allowance must meet the same acceptance criteria as zirconium alloy or stainless steel filler rods as allowed by the CTS and NUREG 1431.	
	ITS:	**NUREG:**
	SPEC 4.02.01	SPEC 4.02.01
03	Fuel is acceptable for storage in the PBNP spent fuel pool up to an enrichment of 4.6% U235 and up to 5% U235 as long as the assemblies contain IFBA as required by CTS figure 15.5.4-1. CTS figure 15.5.4-1 is being replaced by ITS figure 3.7.12-1. Therefore, this change replaces the NUREG recommended requirements with the corresponding plant specific information.	
	ITS:	**NUREG:**
	SPEC 4.03.01.01.A	N/A
	SPEC 4.03.01.01.A.01	SPEC 4.03.01.01.A
	SPEC 4.03.01.01.A.02	N/A

Justification For Deviations - NUREG-1431 Section 4.0

10-Mar-00

JFD Number	JFD Text	
04	The criticality analyses performed for the PBNP new fuel storage racks does not specifically assume the use of aqueous foam in demonstrating Keff remains </= 0.98. The analysis determines the optimum moderator density and demonstrates the acceptance criteria are met.	
	ITS:	**NUREG:**
	SPEC 4.03.01.02.C	SPEC 4.03.01.02.C
05	The spent fuel pool is designed so there is no path for draindown below the bottom edge of the SFP gates (elevation 40' 8"). This corresponds to a decrease in level of 24 feet.	
	ITS:	**NUREG:**
	SPEC 4.03.02	SPEC 4.03.02

4.0 DESIGN FEATURES

4.1 Site Location

[Text description of site location.]

Replace with Insert 4.0-1. 1

4.2 Reactor Core

4.2.1 Fuel Assemblies

1 → 121

1 → Zircaloy-4 or ZIRLO™

The reactor shall contain ~~[157]~~ 121 fuel assemblies. Each assembly shall consist of a matrix of ~~[Zircalloy or ZIRLO]~~ Zircaloy-4 or ZIRLO™ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods or vacancies for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

2 → or vacancies

1 → Rod Cluster Control (RCC)

4.2.2 ~~[Control Rod]~~ Rod Cluster Control (RCC) Assemblies

1 → 33, RCC

The reactor core shall contain ~~[48]~~ 33 ~~[control rod]~~ RCC assemblies. The control material shall be ~~[silver indium cadmium, boron carbide, or hafnium metal]~~ silver indium cadmium alloy clad with stainless steel as approved by the NRC.

1 → silver indium cadmium alloy clad with stainless steel

4.3 Fuel Storage

4.3.1 Criticality

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

3 → Replace with Insert 4.0-2.

a. Fuel assemblies having a maximum U-235 enri chment of [4.5] weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[Section 9.1 of the FSAR]~~ Section 9.4 of the FSAR;

1 → Section 9.4 of the FSAR

1

c. A nominal 9.825 inch center to center distance between fuel assemblies placed in the fuel storage racks.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

[c. A nominal [9.15] inch center to center distance between fuel assemblies placed in [the high density fuel storage racks];]

1

~~[d. A nominal [10.95] inch center to center distance between fuel assemblies placed in [low density fuel storage racks];]~~

~~[e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure [3.7.17-1] may be allowed unrestricted storage in [either] fuel storage rack(s); and]~~

~~[f. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure [3.7.17-1] will be stored in compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]~~

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

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ weight percent;

5.0 1

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[Section 9.1 of the FSAR]~~;

1

Section 9.4 of the FSAR

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam which includes an allowance for uncertainties as described in ~~[Section 9.1 of the FSAR]~~; and

1 20

d. A nominal ~~[10.95]~~ inch center to center distance between fuel assemblies placed in the storage racks.

under optimum moderator density conditions 4

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation ~~[23 ft]~~

5 40 ft 8 in

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~[1737]~~ 1502 fuel assemblies.

1

SECTION 4.0 INSERTS

INSERT 4.0-1:

The Point Beach Nuclear Plant is located on property owned by Wisconsin Electric Power Company at a site on the shore of Lake Michigan, approximately 30 miles southeast of the city of Green Bay. The minimum distance from the reactor containment center line to the site exclusion boundary as defined in 10 CFR 100.3 is 1200 meters.

INSERT 4.0-2:

a. Fuel assemblies meeting at least one of the following storage limits may be stored in the spent fuel storage racks:

   1. Fuel assemblies with an enrichment of $\leq$ 4.6 weight percent U-235; or

   2. Fuel assemblies which contain Integral Fuel Burnable Absorber (IFBA) rods in the "acceptable range" of Figure 3.7.12-1.

NSHC Number	NSHC Text
A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
L	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change limits the use of fuel assemblies to designs analyzed by applicable NRC staff approved codes and shown by test or analyses to comply with all fuel safety design bases and allows limited use of lead test assemblies in non-limiting core regions. This requirement is less restrictive than the CTS, which allows use of specifically identified fuel designs. These designs have been shown by analyses, using NRC approved methodologies, to meet all fuel design bases. The requirements are essentially identical since use is dependent on acceptable analyses utilizing approved codes and methodologies. Use of approved codes and methodologies ensurse analyses remain bounding and all design limits are met. Use of lead test assemblies is restricted to non-limiting core locations. This provides assurance that all analyses and core design limits remain bounding. Thus, the probability or consequences of an accident previously evaluated cannot be significantly increased. 2. Does the change create the possibility of new or different kind of accident from any accident previously evaluated? The proposed condition requires that fuel used in the reactor cores be shown to be acceptable based on analyses performed using approved codes and methodology and shown by tests or analyses to meet all fuel design limits. Lead test assemblies are allowed on a limited bases in non-limiting core locations. This ensures that analyses performed, using approved codes and methodologies, remain bounding for core operation. Thus, since analyses remain bounding with acceptable margins of safety, a new or different kind of accident from any previously evaluated cannot be created. 3. Does the change result in a significant reduction in a margin of safety? This ITS condition ensures that fuel utilized in the reactor core is shown to be acceptable by analyses utilizing NRC approved codes and methodologies and is shown to meet all fuel design limits. Lead test assemblies can only be used in non-limiting core locations on a limited basis. Use of approved methodologies and codes, or tests to to ensure all design safety limits are met, and restricting use of lead test assemblies to non-limiting core locations ensures analyses remain bounding and all applicable safety margins are met. Therefore, a significant reduction in a margin of safety cannot result.

NSHC Number	NSHC Text
LA	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

NSHC Number	NSHC Text
M	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not affect any assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

4.0 DESIGN FEATURES

4.1 Site Location

The Point Beach Nuclear Plant is located on property owned by Wisconsin Electric Power Company at a site on the shore of Lake Michigan, approximately 30 miles southeast of the city of Green Bay. The minimum distance from the reactor containment center line to the site exclusion boundary as defined in 10 CFR 100.3 is 1200 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 121 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy-4 or ZIRLO™ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods or vacancies for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Rod Cluster Control (RCC) Assemblies

The reactor core shall contain 33 RCC assemblies. The control material shall be silver indium cadmium alloy clad with stainless steel as approved by the NRC.

4.0 DESIGN FEATURES

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 meeting at least one of the following storage limits may be stored in the spent fuel storage racks:

1. Fuel assemblies with an enrichment of ≤ 4.6 weight percent U-235; or

2. Fuel assemblies which contain Integral Fuel Burnable Absorber (IFBA) rods in the "acceptable range" of Figure 3.7.12-1.

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.4 of the FSAR;

c. A nominal 9.825 inch center to center distance between fuel assemblies placed in the fuel storage racks.

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.4 of the FSAR;

c. $k_{eff} \leq 0.98$ under optimum moderator density conditions, which includes an allowance for uncertainties as described in Section 9.4 of the FSAR; and

d. A nominal 20 inch center to center distance between fuel assemblies placed in the storage racks.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 40 ft 8 in.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1502 fuel assemblies.

Cross-Reference Report - NUREG-1431 Section 5.01

ITS to CTS

15-Mar-00

ITS	CTS	DOC
SPEC 5.01.01	15.06.01.01	M.03
	15.06.01.01	M.01
	15.06.01.01	A.01
SPEC 5.01.02	15.06.01.02	M.02
	15.06.02.02.B	L.01

Cross-Reference Report - NUREG-1431 Section 5.01
CTS to ITS

15-Mar-00

CTS	ITS	DOC
15.06.01.01	SPEC 5.01.01	M.03
	SPEC 5.01.01	M.01
	SPEC 5.01.01	A.01
15.06.01.02	SPEC 5.01.02	M.02
15.06.02.02.B	SPEC 5.01.02	L.01

Description of Changes - NUREG-1431 Section 5.01

15-Mar-00

DOC Number	DOC Text
A.01	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)). **CTS:** 15.06.01.01 — **ITS:** SPEC 5.01.01
L.01	CTS 15.6.2.2.b states the following, "When there is fuel in either unit, an SRO shall be in the control room at all times." This requirement is not a NUREG-1431 requirement and will not be retained in the proposed ITS. Proposed ITS 5.1.2 states the following, "During any absence of the DSS from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the DSS from the control room while both units are in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function." Therefore, NUREG-1431 and the proposed ITS allows the control room command and control function to be assumed by a RO license if the units are in MODE 5 or 6. Having a Reactor Operator licensed individual assume control room command and control provides adequate protection to the public health and safety when the reactor is in MODE 5 or 6. SRO licensed individuals would still be on-site and be able to respond to the control room in a timely manner if necessary. This change is less restrictive. **CTS:** 15.06.02.02.B — **ITS:** SPEC 5.01.02
M.01	CTS 15.6.1.1 specifies that the Plant Manager delegate his responsibilities for overall facility operation in writing when absent from PBNP for greater than 48 hours and where ready contact by telephone or other means is not assured. This attribute will not be maintained in ITS. NUREG-1431 (ISTS) requires that the Plant Manager delegate his responsibilities in writing during his absence, without specifing timeframes or contact availability. Therefore, adopting the ISTS is more restrictive, because it requires delegation in writing regardless of absence timeframes involved or contact availability. **CTS:** 15.06.01.01 — **ITS:** SPEC 5.01.01

Description of Changes - NUREG-1431 Section 5.01

15-Mar-00

DOC Number	DOC Text	
M.02	CTS 15.6.1.2 states that the Duty Shift Superintendent (or during his absence from the control room, the Duty Operating Supervisor) shall be responsible for the Control Room Command function. NUREG-1431 also states this, but specifies in more detail who has the Control Room Command function during the Shift Supervisor's absence, based on what Mode the Unit is in (SRO for Modes 1, 2, 3, or 4 vs. RO for Modes 5 or 6). Therefore, adopting the ISTS is more restrictive.	
	CTS:	**ITS:**
	15.06.01.02	SPEC 5.01.02
M.03	NUREG-1431 item 5.1.1 states the following "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." This requirement will be adopted in proposed ITS 5.1.1. This requirement is not in the CTS; therefore, adopting this requirement is more restrictive.	
	CTS:	**ITS:**
	15.06.01.01	SPEC 5.01.01

15.6 ADMINISTRATIVE CONTROLS

15.6.1 RESPONSIBILITY

See ITS 5.1.1

15.6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during absences from the Point Beach Nuclear Plant area ~~of greater than 48 hours and where ready contact by telephone or other means is not assured.~~

M.01

M.02 See ITS 5.1.2

15.6.1.2 The Duty Shift Superintendent (or during his absence from the control room, the Duty Operating Supervisor) shall be responsible for the Control Room Command function.

15.6.2 ORGANIZATION

15.6.2.1 Onsite and offsite organizations shall be established for plant operation and corporate management, respectively.

See Spec 5.2

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Point Beach Nuclear Plant Final Safety Analysis Report, or plant procedures.

See Spec 5.2

b. The Plant Manager shall be responsible for overall safe plant operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

See Spec 5.2

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.

See ITS 5.1.1

M.03

15.6 ADMINISTRATIVE CONTROLS (Continued)

15.6.2 ORGANIZATION (Continued)

FACILITY STAFF (Continued)

15.6.2.2 (Continued)

L.01

b. When there is fuel in either unit, an SRO* shall be in the control room at all times. In addition to this SRO*, for each unit containing fuel, an RO* or SRO* shall be present at the controls at all times.

c. DELETED

See Spec 5.2

d. An individual qualified in radiation protection procedures shall be on site when fuel is in either reactor.**

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

---

*SRO = NRC Senior Reactor Operator License
RO = NRC Reactor Operator License

**This shift may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of personnel, provided immediate action is taken to restore the shift makeup to within the minimum requirements.

*** A unit is considered to be operating when it is in a mode other than cold shutdown or refueling shutdown.

Justification For Deviations - NUREG-1431 Section 5.01

15-Mar-00

JFD Number	JFD Text	
01	The brackets have been removed and the proper plant specific information has been provided. In addtion, administrative wording changes where necessary to reflect the two (2) Unit Point Beach site design.	
	ITS:	**NUREG:**
	SPEC 5.01.02	SPEC 5.01.02
02	Administrative wording changes where made as necessary to reflect the two (2) Unit, single control room Point Beach design. NUREG-1431 refers to "unit" staff, which could be misinterpreted to mean that the requirements apply to each "unit" or reactor. Therefore, "unit" was replaced with the word "facility" in the proposed ITS to avoid any potential confusion with respect to "unit" specific requirements.	
	ITS:	**NUREG:**
	SPEC 5.01.01	SPEC 5.01.01

Approved
TSTF-65, R1

02

facility

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

Plant Manager

5.1.1 The [Plant ~~Superintendent~~] shall be responsible for overall ~~unit~~ operation and shall delegate in writing the succession to this responsibility during his absence.

The [Plant ~~Superintendent~~] or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

01

Duty Shift Superintendent (DSS)

either

DSS

both units are

5.1.2 The [Shift Supervisor (SS)] shall be responsible for the control room command function. During any absence of the [SS] from the control room while ~~the~~ unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the [SS] from the control room while ~~the unit is~~ in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

No Significant Hazards Considerations - NUREG-1431 Section 5.01

15-Mar-00

NSHC Number	NSHC Text
A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 5.01

15-Mar-00

NSHC Number	NSHC Text
L.01	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change results in a change to an administrative requirement that allows for the control room command and control function to be assumed by a licensed reactor operator when both units are in MODES 5 or 6. The deletion of an administrative requirement does not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revision. The proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to this administrative requirement. All Limiting Conditions of Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, this change does not increase the probability of previously evaluated accidents. This change does not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed change. Existing system and component redundancy and operation is not being changed by this proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, this change does not affect the consequences of previously evaluated accidents. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change to this administrative requirement does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? Changing this administrative requirement has no bearing on any margin of safety. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 5.01

15-Mar-00

NSHC Number	NSHC Text
M	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not affect any assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

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 Duty Shift Superintendent (DSS) shall be responsible for the control room command function. During any absence of the DSS from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the DSS from the control room while both units are in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

Cross-Reference Report - NUREG-1431 Section 5.02

ITS to CTS

15-Mar-00

ITS	CTS	DOC
SPEC 5.01.02	15.06.02.02.A.01	A.02
SPEC 5.02.01	15.06.02.01	A.01
SPEC 5.02.01.A	15.06.02.01.A	A.01
SPEC 5.02.01.B	15.06.02.01.B	A.01
SPEC 5.02.01.C	15.06.02.01.C	A.01
SPEC 5.02.01.D	15.06.02.01.D	M.01
	15.06.02.01.D	A.01
SPEC 5.02.02.A	15.06.02.02.A.05	A.02
SPEC 5.02.02.B	15.06.02 FOOT NOTE (**)	A.02
SPEC 5.02.02.C	15.06.02.02.D	A.02
SPEC 5.02.02.D	15.06.03.05	A.04
SPEC 5.02.02.E	15.06.02.02.A.02	A.02
	15.06.03.04	A.03

Cross-Reference Report - NUREG-1431 Section 5.02
CTS to ITS

15-Mar-00

CTS	ITS	DOC
15.06.02 FOOT NOTE (**)	SPEC 5.02.02.B	A.02
15.06.02.01	SPEC 5.02.01	A.01
15.06.02.01.A	SPEC 5.02.01.A	A.01
15.06.02.01.B	SPEC 5.02.01.B	A.01
15.06.02.01.C	SPEC 5.02.01.C	A.01
15.06.02.01.D	SPEC 5.02.01.D	M.01
	SPEC 5.02.01.D	A.01
15.06.02.02.A.01	SPEC 5.01.02	A.02
15.06.02.02.A.02	SPEC 5.02.02.E	A.02
15.06.02.02.A.03	N/A	LB.01
15.06.02.02.A.04	N/A	LB.01
15.06.02.02.A.05	SPEC 5.02.02.A	A.02
15.06.02.02.D	SPEC 5.02.02.C	A.02
15.06.02.02.E	N/A	LB.01
15.06.03.04	SPEC 5.02.02.E	A.03
15.06.03.05	SPEC 5.02.02.D	A.04

Description of Changes - NUREG-1431 Section 5.02

15-Mar-00

DOC Number	DOC Text
A.01	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.06.02.01	SPEC 5.02.01
15.06.02.01.A	SPEC 5.02.01.A
15.06.02.01.B	SPEC 5.02.01.B
15.06.02.01.C	SPEC 5.02.01.C
15.06.02.01.D	SPEC 5.02.01.D

A.02

Included in CTS 15.6.2.2 are the requirements for the facility staff. The below identified CTS requirements are consistent with the requirements contained in NUREG-1431 and the proposed ITS, and are summarized as follows.

CTS 15.6.2.2.a.1 requires one Shift Superintendent (SS) per shift - this requirement is the same as proposed ITS 5.1.2, which states that the SS is responsible for the control room command function. CTS 15.6.2.2.a.2 requires one shift technical advisor per shift - this requirement is the same as proposed ITS 5.2.2.e. CTS 15.6.2.2.a.5 contains the requirements for non-licensed operators - these requirements are the same as those contained in proposed ITS 5.2.2.a. CTS 15.6.2.2.d requires an individual qualified in radiation protection procedures to be on site when there is fuel in either reactor - this requirement is the same as proposed ITS 5.2.2.c. The double asterisk in CTS 15.6 allows for a relaxation of the shift manning requirements for up to 2 hours to accommodate unexpected absences of personnel – this relaxation is included in proposed ITS 5.2.2.b.

These changes are administrative.

CTS:	ITS:
15.06.02 FOOT NOTE (**)	SPEC 5.02.02.B
15.06.02.02.A.01	SPEC 5.01.02
15.06.02.02.A.02	SPEC 5.02.02.E
15.06.02.02.A.05	SPEC 5.02.02.A
15.06.02.02.D	SPEC 5.02.02.C

DOC Number	DOC Text	
A.03	CTS 15.6.3.4 describes the qualification requirements for the Shift Technical Advisor (STA). NUREG- 1431 item 5.2.2.g (proposed ITS 5.2.2.e) states that the STA position meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. Option 2 (the option that PBNP currently employs) of the Policy Statement requires that the STA position shall meet the STA criteria of NUREG-0737, Item I.A.1.1. The requirements specified in this NUREG item (I.A.1.1) are consistent with the requirements specified in CTS 15.6.3.4. The proposed ITS 5.2.2.e adopts NUREG-1431 item 5.2.2.g in whole, which is equivalent to CTS 15.6.3.4; therefore, this change is administrative.	
	CTS:	**ITS:**
	15.06.03.04	SPEC 5.02.02.E
A.04	CTS 15.6.3.5 states that the Operations Manager shall: 1) Hold a Senior Reactor Operator (SRO) license at PBNP; or 2) Have held a SRO license at PBNP or a similar unit; or 3) Have been certified at an appropriate simulator for equivalent senior operator knowledge level, and if the Operations Manager does not hold a SRO license at PBNP, then an operations middle manager to whom the operating crews report shall hold a SRO license at PBNP. NUREG-1431 item 5.2.2.f states that the Operations Manager or Assistant Operations Manager shall hold an SRO license. This requirement will be adopted in whole as proposed ITS 5.2.2.d. This change is administrative because in either case one of the two operations senior managers (in the chain of command to whom the operating crews report) must have an SRO license at PBNP.	
	CTS:	**ITS:**
	15.06.03.05	SPEC 5.02.02.D
LB.01	Included in CTS 15.6.2.2 are the requirements for the facility staff. The below identified CTS requirements are consistent with the requirements already contained in 10 CFR 50.54 and are summarized as follows. CTS 15.6.2.2.a.3 requires one Operating Supervisor (OS) per shift - this requirement is the same as that contained in 10 CFR 50.54(m)(2)(i). CTS 15.6.2.2.a.4 contains the requirements for Reactor Operators - these requirements are the same as those contained in 10 CFR 50.54(m)(2)(i). CTS 15.6.2.2.e contains the supervision requirements for core alterations - these requirements are the same as those contained in 10 CFR 50.54(m)(2)(iv). These CTS requirements will not be retained in the proposed ITS because they are duplicative of the code of federal regulations, which all licensees are required to meet. These changes are less restrictive.	
	CTS:	**ITS:**
	15.06.02.02.A.03	N/A
	15.06.02.02.A.04	N/A
	15.06.02.02.E	N/A

Description of Changes - NUREG-1431 Section 5.02

15-Mar-00

DOC Number	DOC Text
M.01	NUREG-1431 item 5.2.1.d and proposed ITS 5.2.1.d add the "individuals who train the operating staff" to the functions that shall have independence from operating pressures. CTS 15.6.2.1.d contains the requirements for functions that shall have independence from operating pressures, but it does not include these individuals. This item will be adopted in the proposed ITS. This change is more restrictive.

CTS:	ITS:
15.06.02.01.D	SPEC 5.02.01.D

15.6 ADMINISTRATIVE CONTROLS

15.6.1 RESPONSIBILITY

15.6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during absences from the Point Beach Nuclear Plant area of greater than 48 hours and where ready contact by telephone or other means is not assured.

See Spec 5.1

15.6.1.2 The Duty Shift Superintendent (or during his absence from the control room, the Duty Operating Supervisor) shall be responsible for the Control Room Command function.

See Spec 5.1

15.6.2 ORGANIZATION

15.6.2.1 Onsite and offsite organizations shall be established for plant operation and corporate management, respectively.

See ITS 5.2.1

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Point Beach Nuclear Plant Final Safety Analysis Report, or plant procedures.

See ITS 5.2.1.a

b. The Plant Manager shall be responsible for overall safe plant operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

See ITS 5.2.1.b

15.6 ADMINISTRATIVE CONTROLS (Continued)

15.6.2 ORGANIZATION (Continued)

15.6.2.1 (Continued)

See ITS 5.2.1.c

c. The Chief Nuclear Officer shall be an officer of the Company and shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

M.01 → train the operating staff, or

See ITS 5.2.1.d

d. The individuals who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

FACILITY STAFF

A.02

See ITS 5.1.2

15.6.2.2 Facility staffing shall be subject to the following requirements:

a. Each on-duty shift shall normally be composed of at least the minimum shift crew composition as follows:**

1. Shift Superintendent (SRO)* - one per shift

See ITS 5.2.2.e

2. Shift Technical Advisor - one per shift located on-site on ten minute call to the control room

LB.01

3. Operating Supervisor (SRO)* - one per shift

4. Operator (RO)* - three per shift for one or two unit operation***
- two per shift with neither unit in operation***

See ITS 5.2.2.a

5. Non-licensed Operator - one per shift for each unit containing fuel and an additional one per shift when either unit is in operation***

15.6 ADMINISTRATIVE CONTROLS (Continued)

15.6.2 ORGANIZATION (Continued)

FACILITY STAFF (Continued)

15.6.2.2 (Continued)

See Spec 5.1

b. When there is fuel in either unit, an SRO* shall be in the control room at all times. In addition to this SRO*, for each unit containing fuel, an RO* or SRO* shall be present at the controls at all times.

c. DELETED

A.02 See ITS 5.2.2.c

d. An individual qualified in radiation protection procedures shall be on site when fuel is in either reactor.**

LB.01

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

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*SRO = NRC Senior Reactor Operator License
RO = NRC Reactor Operator License

A.02 See ITS 5.2.2.b

**This shift may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of personnel, provided immediate action is taken to restore the shift makeup to within the minimum requirements.

*** A unit is considered to be operating when it is in a mode other than cold shutdown or refueling shutdown.

~~15.6.2.3 DELETED~~

A.01

15.6.3.3 In the event the position of Health Physicist is vacated and the proposed replacement does not meet all the qualifications of 15.6.3.2, but is determined to be otherwise well qualified, the concurrence of NRC shall be sought in approving the qualification of that individual.

See Spec 5.3

15.6.3.4 The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents. The Shift Technical Advisor shall also receive training in plant design and layout including the capabilities of instrumentation and controls in the control room.

A.03

See ITS 5.2.2.e

15.6.3.5 The Operations Manager shall:

1) Hold a Senior Reactor Operator license at PBNP; or

2) Have held a Senior Reactor Operator license at PBNP or a similar unit; or

3) Have been certified at an appropriate simulator for equivalent senior operator knowledge level.

If the Operations Manager does not hold a Senior Reactor Operator license at PBNP, then an operations middle manager to whom the operating crews report shall hold a Senior Reactor Operator license at PBNP.

The Operations Manager or Assistant Operations Manager shall hold an SRO License at Point Beach.

A.04

Justification For Deviations - NUREG-1431 Section 5.02

15-Mar-00

JFD Number	JFD Text
01	The brackets have been removed and the proper plant specific information has been provided. **ITS:** SPEC 5.02.01.A — **NUREG:** SPEC 5.02.01.A
02	NUREG-1431 item 5.2.2.a contains a bracketed requirement that states the following: "two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units". This requirement was not adopted for the proposed PBNP ITS because there is not a CTS requirement for non-licensed operators when both units are defueled. **ITS:** SPEC 5.02.02.A — **NUREG:** SPEC 5.02.02.A
03	NUREG-1431 item 5.2.2.e contains bracketed requirements for facility staff overtime. These requirements were not adopted for the proposed PBNP ITS because there are no CTS requirements for facility staff overtime. Facility staff overtime requirements are appropriately covered in station policies and procedures. Accordingly, the changes to this section made under TSTF-258, Rev. 4 were not incorporated. **ITS:** N/A — **NUREG:** SPEC 5.02.02.E
04	NUREG-1431 item 5.2.2.f states that "the [Operations Manager or Assistant Operations Manager] shall hold an SRO license." TSTF-65, Rev. 1 removed the brackets from this requirement. In addition, this requirement was slightly modified in proposed ITS 5.2.2.d to add "at Point Beach". This was added to avoid the potential for incorrect interpretation with respect to plant applicability for the SRO license. **ITS:** SPEC 5.02.02.D — **NUREG:** SPEC 5.02.02.F
05	Administrative wording changes where made as necessary to reflect the two (2) Unit, single control room Point Beach design. NUREG-1431 refers to "unit" staff, which could be misinterpreted to mean that the requirements apply to each "unit" or reactor. Therefore, "unit" was replaced with the word "facility" in the proposed ITS to avoid any potential confusion with respect to "unit" specific requirements. **ITS:** SPEC 5.02.02.A — **NUREG:** SPEC 5.02.02.A **ITS:** SPEC 5.02.02.C — **NUREG:** SPEC 5.02.02.D **ITS:** SPEC 5.02.02.E — **NUREG:** SPEC 5.02.02.G

Justification For Deviations - NUREG-1431 Section 5.02

15-Mar-00

JFD Number	JFD Text
06	NUREG-1431 item 5.2.1.c states, in part, "a specified corporate officer shall have corporate responsibility". The word "corporate" was deleted in proposed ITS 5.2.1.c based on the following. Wisconsin Electric (WE), license holder for Point Beach, is currently engaged in transferring this license authority to the Nuclear Management Corporation (NMC). The word corporate was deleted to avoid any potential confusion with respect to what is considered corporate (i.e. WE or NMC). When the license transfer is complete, this officer referred to in ITS 5.2.1.c will be employed by the NMC.

ITS:	NUREG:
SPEC 5.02.01.C	SPEC 5.02.01.C

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

Approved
TSTF-65, R1

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 shall be documented in the [FSAR];

, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications,

/QA Plan

FSAR

01

b. The [Plant ~~Superintendent~~] shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

Plant
Manager

c. The [a specified ~~corporate executive~~ position] shall have ~~corporate~~ responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

A specified
~~corporate~~
officer

06

d. The individuals who train the operating staff, 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 independence from operating pressures.

Facility

05

5.2.2 ~~Unit~~ Staff

The ~~unit~~ staff organization shall include the following:

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator

5.2 Organization

5.2.2 Unit Staff (continued)

05 when either

shall be assigned ~~for each control room from which a r~~eactor is operating in MODES 1, 2, 3, or 4.

02

~~[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for t he two units.]~~

Approved TSTF-258, R4

~~b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.~~

Approved TSTF-65, R1

b ~~c~~. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.~~g~~ e 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.

c ~~d~~. A ~~[Health Physics Technician]~~ radiation protection technician shall be on site when fuel is in ~~the~~ either reactor. 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.

05

03

~~[e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators, and key maintenance personnel).~~

~~Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:~~

~~1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;]~~

5.2 Organization

5.2.2 Unit Staff (continued)

03

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed betw een work periods, including shift turnover time;

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be a uthorized in advance by the [Plant Superintendent] or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the [Plant Superintendent] or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

OR

The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

Approved TSTF-258, R4

d

04

at Point Beach

f. The [Operations Manager or Assistant Operations Manager] ~~shall hold an SRO license.~~

05

facility

e

g. ~~The Shift Technical Advisor (STA)~~ shall provide advisory technical support to the ~~Shift Supervisor (SS)~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the ~~unit~~. In ~~addition, the STA~~ shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

An individual

This individual

05

~~unit~~ operations shift crew

No Significant Hazards Considerations - NUREG-1431 Section 5.02

15-Mar-00

NSHC Number	NSHC Text
A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
LB	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change involves deletion of a Specifications/information which is duplicative of information contained in the Code of Federal Regulations (CFRs). This information is more appropriately addressed by the CFRs and serves no purpose in the Technical Specifications. Deletion of this information will not result in an increase in the probability of an accident. Regulatory requirements do not alter plant design or configuration; therefore, this does not alter any event precursor. Accordingly, there will be no effect on the consequences of any accident. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change deletes materials from the Technical Specifications which are adequately addressed in the CFRs. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change deletes materials from the Technical Specifications which are duplicative of requirements contained in the CFRs. These items are not an input to any accident analysis and, therefore, have no impact on margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 5.02

15-Mar-00

NSHC Number	NSHC Text
M	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not affect any assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

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 FSAR;

b. The Plant Manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. A specified officer shall have responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

d. The individuals who train the operating staff, 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 independence from operating pressures.

5.2 Organization

5.2.2 Facility Staff

The facility staff organization shall include the following:

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned when either reactor is operating in MODES 1, 2, 3, or 4.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.e 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.

c. A radiation protection technician shall be on site when fuel is in either reactor. 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.

d. The Operations Manager or Assistant Operations Manager shall hold an SRO License at Point Beach.

e. An individual shall provide advisory technical support to the operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the facility. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

Cross-Reference Report - NUREG-1431 Section 5.03

ITS to CTS

15-Mar-00

ITS	CTS	DOC
SPEC 5.03.01	15.06.03.01	M.01
	15.06.03.02	A.01
SPEC 5.03.02	NEW	A.02
SPEC 5.03.03	15.06.03.03	A.01

Cross-Reference Report - NUREG-1431 Section 5.03
CTS to ITS

15-Mar-00

CTS	ITS	DOC
15.06.03.01	SPEC 5.03.01	M.01
15.06.03.02	SPEC 5.03.01	A.01
15.06.03.03	SPEC 5.03.03	A.01
15.06.04.01	N/A	LB.01
15.06.05.01	N/A	A.03
15.06.05.02	FSAR	LA.01
15.06.05.03	FSAR	LA.01
15.06.06	N/A	A.03

Description of Changes - NUREG-1431 Section 5.03

15-Mar-00

DOC Number	DOC Text
A.01	CTS 15.6.3.1 specifies that the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions or "as clarified in 15.6.3.2 through 15.6.3.5." The attribute "as clarified in 15.6.3.2 through 15.6.3.5" will not be maintained. CTS 15.6.3.2 contains additional requirements for the Health Physicist supervisor. These requirements were added because the requirements for this position in ANSI N18.1-1971 are minimal. The proposed ITS 5.3.1 will add the additional staff qualification requirements contained in Regulatory Guide 1.8, Revision 1, September 1975 (here after referred to as RG 1.8). RG 1.8 endorses ANSI N18.1-1971, but also adds additional requirements for the Radiation Protection supervisor. These requirements are equivalent to the Health Physicist supervisor requirements stated in CTS 15.6.3.2. Therefore, by committing to RG 1.8 in the proposed ITS, retention of 15.6.3.2 is unnecessary. This change is administrative.

CTS:	ITS:
15.06.03.02	SPEC 5.03.01
15.06.03.03	SPEC 5.03.03

DOC Number	DOC Text
A.02	Proposed ITS 5.3.2 adds a clarification definition for licensed senior reactor operator and licensed reactor operator. This was added as a result of Approved TSTF-258, rev. 4. This change is administrative.

CTS:	ITS:
NEW	SPEC 5.03.02

DOC Number	DOC Text
A.03	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.06.05.01	N/A
15.06.06	N/A

Description of Changes - NUREG-1431 Section 5.03

15-Mar-00

DOC Number	DOC Text
LA.01	Wisconsin Electric Power Company has concluded that CTS 15.6.5.2 "OFF-SITE REVIEW COMMITTEE (OSRC)" and 15.6.5.3 "Fire Protection Audits" can be relocated to licensee control. The basis for this conclusion is as follows: The current PBNP Technical Specifications (CTS) describe the composition and functional requirements of the OSRC and fire protection audit requirements in TS section 15.6.5.2 and 15.6.5.3. These requirements will not be maintained in the proposed ITS and will be relocated to licensee control. NUREG-1431 does not contain these administrative requirements. PBNP proposes to relocate the requirements in these CTS sections to Section 1.4 "Quality Assurance Program" of the PBNP Final Safety Analysis Report (FSAR). This section of the FSAR describes the PBNP Quality Assurance Program (QAP) in detail. Changes to this section of the FSAR are controlled in accordance with the requirements of 10 CFR 50.54(a). Relocating these requirements out of the CTS and into FSAR §1.4 is consistent with the guidance contained in NRC Administrative Letter (AL) 95-06 "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995. NRC AL 95-06 states that the NRC encourages relocation of the review and audit functions out of the licensee's Technical Specifications and into QAP descriptions as long as future revisions to said functions are controlled in accordance with the requirements of 10 CFR 50.54. Based on the above information, the requirements in CTS 15.6.5.2 and CTS 15.6.5.3 can be relocated to licensee control. **CTS:** 15.06.05.02 — **ITS:** FSAR **CTS:** 15.06.05.03 — **ITS:** FSAR
LB.01	Included in CTS 15.6.4.1 are the requirements for the retraining and replacement training program for the facility staff, and states that the program meet or exceed the requirements in section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55. This attribute will not be retained in the proposed ITS. NUREG-1431 does not contain this requirement. The requirements contained in 10 CFR Part 55 go far above and beyond the requirements contained in section 5.5 of ANSI N18.1-1971. Therefore, transferring CTS 15.6.4.1 into the proposed PBNP ITS is not necessary, because the requirement is duplicative of the code of federal regulations (10 CFR Part 55), which all licensees are required to meet. This change removes CTS details that are duplicative of other regulatory requirements. **CTS:** 15.06.04.01 — **ITS:** N/A

Description of Changes - NUREG-1431 Section 5.03

15-Mar-00

DOC Number	DOC Text
M.01	CTS 15.6.3.1 specifies that the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions or as clarified in 15.6.3.2 through 15.6.3.5. The ANSI requirement will be maintained in the proposed ITS, but an additional requirement will be added to state "as supplemented by Regulatory Guide 1.8, Revision 1, September 1975" (here after referred to as RG 1.8), and the attribute "as clarified in 15.6.3.2 through 15.6.3.5" will not be maintained (deletion of this attribute is discussed in DOC A.01 of this section). NUREG-1431 (ISTS) requires that the licensee commit to an ANSI standard or Reg Guide acceptable to the NRC staff for facility qualifications. Because an additional requirement will be imposed in the proposed ITS that does not currently exist in the CTS (RG 1.8), this change is more restrictive. The reason for the addition of RG 1.8 to the proposed ITS 5.3.1 is discussed in DOC A.01 of this section.

CTS:	ITS:
15.06.03.01	SPEC 5.03.01

M.01 → See ITS 5.3.1 → , as supplemented by Regulatory Guide 1.8, Revision 1, September 1975,

15.6.3 FACILITY STAFF QUALIFICATIONS

15.6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions ~~or as clarified in 15.6.3.2 through 15.6.3.5.~~

A.01/M.02

15.6.3.2 ~~Except as provided in 15.6.3.3, the Health Physicist shall be a line supervisor and shall meet the following requirements:~~

~~a. The individual shall have a bachelor's degree or the equivalent in a science or engineering subject, including some formal training in radiation protection. For purposes of this paragraph, "equivalent" is as follows:~~

~~(1) Four years of formal schooling in science or engineering; or~~

~~(2) Four years of applied radiation protection experience at a nuclear facility; or~~

~~(3) Four years of operational or technical experience or training in nuclear power; or~~

~~(4) Any combination of the above totaling four years.~~

~~b. Except as provided in d., below, the individual shall have at least five years of professional experience in applied radiation protection. A master's degree in a related field is equivalent to one year of experience and a doctor's degree in a related field is equivalent to two years of experience.~~

~~c. Except as provided in d., below, at least three of the five years of experience shall be in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power plants.~~

~~d. If the individual has a bachelor's degree specifically in health physics, radiological health, or radiation protection, at least three years of professional experience is required; if the individual has a master's or a doctor's degree specifically in health physics, radiological health, or radiation protection, at least two years of professional experience is required. This experience shall be in applied radiation protection in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power plants.~~

A.01 → See ITS 5.3.1

For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m)

See ITS 5.3.2

A.02

15.6.3.3 In the event the position of Health Physicist is vacated and the proposed replacement does not meet all the qualifications of 15.6.3.2, but is determined to be otherwise well qualified, the concurrence of NRC shall be sought in approving the qualification of that individual.

See ITS 5.3.3

15.6.3.4 The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents. The Shift Technical Advisor shall also receive training in plant design and layout including the capabilities of instrumentation and controls in the control room.

See Spec 5.2

15.6.3.5 The Operations Manager shall:

1) Hold a Senior Reactor Operator license at PBNP; or
2) Have held a Senior Reactor Operator license at PBNP or a similar unit; or
3) Have been certified at an appropriate simulator for equivalent senior operator knowledge level.

If the Operations Manager does not hold a Senior Reactor Operator license at PBNP, then an operations middle manager to whom the operating crews report shall hold a Senior Reactor Operator license at PBNP.

See Spec 5.2

15.6.4 TRAINING

LB.01

15.6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

15.6.5 REVIEW AND AUDIT

15.6.5.1 Manager's Supervisory Staff - DELETED (Relocated to owner controlled documents)

A.03

A.03

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15.6.5.2 OFF-SITE REVIEW COMMITTEE (OSRC)

FUNCTION

15.6.5.2.1 The Off-Site Review Committee shall function to provide independent review and audit of designated activities in the areas of:

a) nuclear power plant operations
b) nuclear engineering
c) chemistry and radiochemistry
d) metallurgy
e) instrumentation and control
f) radiological safety
g) mechanical and electrical engineering
h) quality assurance practices
i) environmental monitoring

COMPOSITION

15.6.5.2.2 The Off-Site Review Committee is made up of a minimum of five regular members appointed by the President and one or more ex-officio members. Of the five or more regular members, at least two will be persons not directly employed by the Licensee. All members will be experienced in one or more aspects of the nuclear industry.

ALTERNATES

15.6.5.2.3 Alternate members may be appointed in writing by the OSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in OSRC activities at any one time.

CONSULTANTS

15.6.5.2.4 Consultants shall be utilized as determined by the OSRC Chairman to provide expert advice to the OSRC.

MEETING FREQUENCY

15.6.5.2.5 The OSRC shall meet at least twice per year at approximately six month intervals.

QUORUM

15.6.5.2.6 A quorum of OSRC shall consist of not less than a majority of the members or designated alternates and shall include the Chairman or his designated alternate. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

15.6.5.2.7 The OSRC shall review:

a) The safety evaluations for 1) changes to procedures, equipment or systems, and 2) tests or experiments completed under the provision of 10 CFR, Section 50.59, to verify that such actions did not constitute an unreviewed safety question.
b) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR, Section 50.59.
c) Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR Section 50.59.
d) Proposed changes in Technical Specifications or Licenses.
e) Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.
f) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.
g) All reportable events.

15.6.5.2.7 (Continued)

h) Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i) Reports and meeting minutes of the Manager's Supervisory Staff.

AUDITS

15.6.5.2.8 Audits of facility activities shall be performed under the cognizance of the OSRC. These audits shall encompass:

a) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions.

b) The performance, training and qualifications of the licensed operating staff.

c) The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety.

d) The results of audits by the quality assurance organization on the performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50.

e) Any other area of facility operation considered appropriate by the President.

AUTHORITY

15.6.5.2.9 The OSRC shall report to and advise the President on those areas of responsibility specified in Section 15.6.5.2.7 and 15.6.5.2.8.

RECORDS

15.6.5.2.10 Records of OSRC activities shall be prepared, approved and distributed as indicated below:

a) Minutes of each OSRC meeting shall be prepared, approved and forwarded to the President within 14 days following each meeting.

b) Reports of reviews encompassed by Section 15.6.5.2.7.e, f and g above shall be prepared, approved and forwarded to the President within 14 days following completion of the review.

c) Audit reports encompassed by Section 15.6.5.2.8 above, shall be forwarded to the President and to the management positions responsible for the areas audited within 30 days after completion of the audit.

15.6.5.3 Fire Protection Audits

a) An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

b) An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

15.6.6 ~~DELETED~~

A.03

Justification For Deviations - NUREG-1431 Section 5.03

15-Mar-00

JFD Number	JFD Text
01	NUREG-1431 item 5.3.1 requires that the licensee commit to an ANSI standard or Reg Guide acceptable to the NRC staff for facility qualifications. CTS 15.6.3.1 specifies that the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions or as clarified in 15.6.3.2 through 15.6.3.5. This requirement will be maintained in the proposed ITS, and an additional requirement will be added to state "as supplemented by Regulatory Guide 1.8, Revision 1, September 1975" (here after referred to as RG 1.8), and the attribute "as clarified in 15.6.3.2 through 15.6.3.5" will not be maintained (deletion of this attribute is discussed in DOC A.01 of section 5.03). The addition of RG 1.8 is being made to the proposed ITS 5.3.1 because it endorses ANSI N18.1-1971, and also adds additional requirements for the Radiation Protection supervisor. These additional requirements for the Radiation Protection supervisor (In RG 1.8) are equivalent to the Health Physicist supervisor requirements stated in CTS 15.6.3.2 (which is not being maintained in the proposed ITS). Approved TSTF-258, rev. 4 expanded the brackets in NUREG-1431 item 5.3.1 to include the entire second sentence which states, "The staff not covered by Regulatory Guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff." This bracketed item was not adopted in the proposed ITS, because there is not a CTS requirement for this, and therefore, it is not part of the current licensing basis. **ITS:** SPEC 5.03.01 — **NUREG:** SPEC 5.03.01
02	Administrative wording changes were made as necessary to reflect the two (2) Unit, single control room Point Beach design. NUREG-1431 refers to "unit" staff, which could be misinterpreted to mean that the requirements apply to each "unit" or reactor. Therefore, "unit" was replaced with the word "facility" in the proposed ITS to avoid any potential confusion with respect to "unit" specific requirements. **ITS:** SPEC 5.03 — **NUREG:** SPEC 5.03
03	CTS 15.6.3.3 states the following "In the event the position of Health Physicist is vacated and the proposed replacement does not meet all the qualifications of 5.3.1, but is determined to be otherwise well qualified, the concurrence of NRC shall be sought in approving the qualification of that individual." PBNP wants to maintain this CTS attribute; therefore, it will be retained in the proposed ITS as 5.3.3. **ITS:** SPEC 5.03.03 — **NUREG:** N/A

5.0 ADMINISTRATIVE CONTROLS

Facility

02

5.3 ~~Unit~~ Staff Qualifications

[Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.]

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].

5.3.3 In the event the position of Health Physicist is vacated and the proposed replacement does not meet all the qualifications of TS 5.3.1, but is determined to be otherwise well qualified, the concurrence of NRC shall be sought in approving the qualification of that individual.

03

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m)

Approved
TSTF-258, R4

Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, as supplemented by Regulatory Guide 1.8, Revision 1, September 1975, for comparable positions.

01

No Significant Hazards Considerations - NUREG-1431 Section 5.03

15-Mar-00

NSHC Number	NSHC Text
A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
LA	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

NSHC Number	NSHC Text
LB	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change involves deletion of a Specifications/information which is duplicative of information contained in the Code of Federal Regulations (CFRs). This information is more appropriately addressed by the CFRs and serves no purpose in the Technical Specifications. Deletion of this information will not result in an increase in the probability of an accident. Regulatory requirements do not alter plant design or configuration; therefore, this does not alter any event precursor. Accordingly, there will be no effect on the consequences of any accident. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change deletes materials from the Technical Specifications which are adequately addressed in the CFRs. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change deletes materials from the Technical Specifications which are duplicative of requirements contained in the CFRs. These items are not an input to any accident analysis and, therefore, have no impact on margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 5.03

15-Mar-00

NSHC Number	NSHC Text
M	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not affect any assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

5.0 ADMINISTRATIVE CONTROLS

5.3 Facility Staff Qualifications

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, as supplemented by Regulatory Guide 1.8, Revision 1, September 1975, for comparable positions.

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.3.3 In the event the position of Health Physicist is vacated and the proposed replacement does not meet all the qualifications of TS 5.3.1, but is determined to be otherwise well qualified, the concurrence of NRC shall be sought in approving the qualification of that individual.

Cross-Reference Report - NUREG-1431 Section 5.04

ITS to CTS

15-Mar-00

ITS	CTS	DOC
SPEC 5.04.01	15.06.08.01	M.01
	15.06.08.01	A.02
	15.06.08.01	A.01

Cross-Reference Report - NUREG-1431 Section 5.04
CTS to ITS

15-Mar-00

CTS	ITS	DOC
15.06.08.01	SPEC 5.04.01	M.01
	SPEC 5.04.01	A.02
	SPEC 5.04.01	A.01
15.06.08.02	FSAR	LA.01

Description of Changes - NUREG-1431 Section 5.04

15-Mar-00

DOC Number	DOC Text	
A.01	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).	
	CTS:	**ITS:**
	15.06.08.01	SPEC 5.04.01
A.02	The CTS requirement for having procedures for quality assurance for effluent and environmental monitoring is simply being relocated from CTS 15.7.8.3 to CTS 15.6.8.1. This requirement is identical to NUREG-1431 item 5.4.1.c and proposed ITS item 5.4.1.i. This change is administrative.	
	CTS:	**ITS:**
	15.06.08.01	SPEC 5.04.01
LA.01	Wisconsin Electric Power Company has concluded that CTS 15.6.8.2 "Approval of Procedures" and 15.6.8.3 "Changes to Procedures" can be relocated to licensee control. The basis for this conclusion is as follows: The current PBNP Technical Specifications (CTS) describe the procedure review and approval process in CTS section 15.6.8.2 and 15.6.8.3. These requirements will not be maintained in the proposed ITS and will be relocated to licensee control. NUREG-1431 does not contain these administrative requirements. PBNP proposes to relocate the requirements in these CTS sections to Section 1.4 "Quality Assurance Program" of the PBNP Final Safety Analysis Report (FSAR). This section of the FSAR describes the PBNP Quality Assurance Program (QAP) in detail. Changes to this section of the FSAR are controlled in accordance with the requirements of 10 CFR 50.54(a). Relocating these requirements out of the CTS and into FSAR §1.4 is consistent with the guidance contained in NRC Administrative Letter (AL) 95-06 "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995. NRC AL 95-06 states that the NRC encourages relocation of the procedure review and approval process out of the licensee's Technical Specifications and into QAP descriptions as long as future revisions to these functions are controlled in accordance with the requirements of 10 CFR 50.54. Based on the above information, the requirements in CTS 15.6.8.2 and CTS 15.6.8.3 can be relocated to licensee control.	
	CTS:	**ITS:**
	15.06.08.02	FSAR

DOC Number	DOC Text
M.01	The NUREG-1431 item 5.4.1.e requirement for having procedures for "all the programs specificed in Specification 5.5" is being adopted in proposed ITS 5.4.1.j. The programs specified in proposed ITS 5.5 include the following: Offsite Dose Calculation Manual, Primary Coolant Sources Outside Containment, Post Accident Sampling, Radioactive Effluent Controls Program, Component Cyclic or Transient Limit, Reactor Coolant Pump Flywheel Inspection Program, Inservice Testing Program, Steam Generator (SG) Tube Surveillance Program, Secondary Water Chemistry Program, Ventilation Filter Testing Program, Explosive Gas Monitoring Program, Diesel Fuel Oil Testing Program, Technical Specifications (TS) Bases Control Program, Safety Function Determination Program (SFDP), Containment Leakage Rate Testing Program, and the Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) Leakage Program. Having implementing procedures for these programs is an inherent attribute of having these individual programs in place at PBNP. However, the CTS does not currently have this requirement; therefore, adopting this requirement is more restrictive.

CTS:	**ITS:**
15.06.08.01	SPEC 5.04.01

15.6.8 PLANT OPERATING PROCEDURES

15.6.8.1 The plant shall be operated and maintained in accordance with approved procedures. Procedures shall be provided for the following operations where these operations involve nuclear safety of the plant:

1. Normal sequences of startup, operation and shutdown of components, systems and overall plant.
2. Refueling.
3. Specific and foreseen potential malfunctions of systems or components including abnormal reactivity changes.
4. Security Plan Implementation.
5. Emergencies which could involve release of radioactivity.
6. Nuclear core testing.
7. Surveillance and Testing of safety related equipment.
8. Fire Protection Implementation.

A.02

Quality Assurance for effluent and environmental monitoring

M.01

All programs specified in Specification 5.5

15.6.8.2 Approval of Procedures.

LA.01

A. ~~Each procedure, or change thereto, of the categories listed in 15.6.8.1 (except 15.6.8.1.4) and 15.6.11 shall be reviewed by an individual or group other than the individual who prepared the procedure, or change thereto. All procedures of the categories listed in 15.6.8.1 (except 15.6.8.1.4) and 15.6.11, and modifications to the intent thereof, shall be approved by the Plant Manager or a department manager assigned responsibility for those procedures (hereafter referred to as the Approval Authority) prior to implementation. Non-intent changes shall be reviewed and approved in accordance with 15.6.8.2 or 15.6.8.3.~~

B. ~~Individuals responsible for reviews in accordance with 15.6.8.2 and 15.6.8.3 shall be members of the plant staff previously designated by the Plant Manager and meet or exceed the qualifications of Technical Specification 15.6.3.~~

15.6.8 PLANT OPERATING PROCEDURES (Continued)

C. Each review shall include a determination of whether or not additional, cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by qualified personnel of the appropriate discipline.

D. Each review shall include an assessment for applicability of 10 CFR 50.59 and when necessary appropriate evaluations shall be performed.

15.6.8.3 Changes to Procedures

LA.01

A. Changes to procedures, of the categories in 15.6.8.2A, that may involve a change to the intent of the original procedures shall be approved in accordance with 15.6.8.2.

B. Temporary changes to procedures, of the categories listed in 15.6.8.2A, which do not change the intent of the approved procedure, may be made provided such changes are approved by two members of the plant staff, at least one of whom holds a Senior Reactor Operator's License.

C. All temporary changes to procedures of the categories listed in 15.6.8.2A shall subsequently by reviewed and approved in accordance with 15.6.8.2 within 2 weeks. Temporary changes only become permanent changes after approval by the Approval Authority.

Justification For Deviations - NUREG-1431 Section 5.04

15-Mar-00

JFD Number	JFD Text
01	The CTS 15.6.8.1 requirements for plant procedures is being retained in the proposed ITS 5.4.1. The NUREG-1431 item 5.4.1.a requirements for plant procedures (procedures recommended in Reg Guide 1.33) is not being adopted in the proposed ITS. Point Beach is not committed to Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; therefore, it is not part of the current licensing basis. The NUREG-1431 item 5.4.1.b requirements for plant procedures (emergency operating procedures (EOPs) required to implement the requirements of NUREG-0737 and Supplement 1, as stated in Generic Letter 82-33) is not being adopted in the proposed ITS. Point Beach responded to GL 82-33 in an April 15, 1983 letter from C. W. Fay (WE) to H. R. Denton (NRC). The response stated that the PBNP EOPs were completely rewritten based on Westinghouse Owners Group (WOG) guidelines. The PBNP response to GL 82-33 was accepted by the NRC and closed out. Therefore, PBNP has already met the requirements of GL 82-33 and stating this in the ITS is unnecessary. The NUREG-1431 item 5.4.1.c requirements for plant procedures (quality assurance for effluent and environmental monitoring) will be adopted in the proposed ITS. The CTS already contains this requirement in CTS 15.7.8.3. The NUREG-1431 item 5.4.1.d requirements for plant procedures (fire protection program implementation) will be adopted in the proposed ITS and is consistent with CTS 15.6.8.1.8. The NUREG-1431 item 5.4.1.e requirements for plant procedures (all programs specified in specification 5.5) will be adopted in the proposed ITS.

ITS:	NUREG:
SPEC 5.04.01	SPEC 5.04.01

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

a. ~~The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;~~

b. ~~The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in [Generic Letter 82-33];~~

c. ~~Quality assurance for effluent and environmental monitoring;~~

d. ~~Fire Protection Program implementation; and~~

e. ~~All programs specified in Specification 5.5.~~

a. Normal sequences of startup, operation and shutdown of components, systems and overall plant;
b. Refueling;
c. Specific and foreseen potential malfunctions of systems or components including abnormal reactivity changes;
d. Security Plan Implementation;
e. Emergencies which could involve release of radioactivity;
f. Nuclear core testing;
g. Surveillance and Testing of safety related equipment;
h. Fire Protection Implementation;
i. Quality Assurance for effluent and environmental monitoring;
j. All programs specified in Specification 5.5

01

No Significant Hazards Considerations - NUREG-1431 Section 5.04

15-Mar-00

NSHC Number	NSHC Text
A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 5.04

15-Mar-00

NSHC Number	NSHC Text
LA	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

No Significant Hazards Considerations - NUREG-1431 Section 5.04

15-Mar-00

NSHC Number	NSHC Text
M	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not affect any assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

a. Normal sequences of startup, operation and shutdown of components, systems and overall plant;
b. Refueling;
c. Specific and foreseen potential malfunctions of systems or components including abnormal reactivity changes;
d. Security Plan Implementation;
e. Emergencies which could involve release of radioactivity;
f. Nuclear core testing;
g. Surveillance and Testing of safety related equipment;
h. Fire Protection Implementation;
i. Quality Assurance for effluent and environmental monitoring;
j. All programs specified in Specification 5.5.

Cross-Reference Report - NUREG-1431 Section 5.05

ITS to CTS

15-Mar-00

ITS	CTS	DOC
ODCM	15.07.01.A	LA.01
	15.07.01.B	LA.01
	15.07.01.C	LA.01
	15.07.01.D	LA.01
SPEC 5.05.01.A	15.07.08.03.A	A.03
	15.07.08.03.C	A.04
	15.07.08.03.C	A.05
SPEC 5.05.01.B	15.07.08.03.A	A.04
	15.07.08.03.A	A.03
	15.07.08.03.B.08	A.04
SPEC 5.05.01.C	15.07.08.07.B	A.05
SPEC 5.05.01.C.01	15.07.08.07.B.01	A.05
SPEC 5.05.01.C.01.i	15.07.08.07.B.01.a	A.04
SPEC 5.05.01.C.01.ii	15.07.08.07.B.01.b	A.04
SPEC 5.05.01.C.02	15.07.08.07.B.02	A.04
SPEC 5.05.01.C.03	15.07.08.07.B.03	A.05
SPEC 5.05.02	NEW	M.02
SPEC 5.05.02.a	NEW	M.02
SPEC 5.05.02.b	NEW	M.02
SPEC 5.05.03	15.06.08.04.A	A.07
	15.06.08.04.A	M.01
SPEC 5.05.03.A	15.06.08.04.A.I	A.04
SPEC 5.05.03.B	15.06.08.04.A.II	A.04
SPEC 5.05.03.C	15.06.08.04.A.III	A.04
SPEC 5.05.04	15.07.08.03	A.05
	15.07.08.03.B	A.05
SPEC 5.05.04.B	15.07.08.03.B.03	A.04
SPEC 5.05.04.C	15.07.08.03.B	A.05
	15.07.08.03.B.02	A.05
	15.07.08.03.B.02	A.04
SPEC 5.05.04.D	15.07.08.03.C	A.04
	15.07.08.03.C	A.05
	15.07.08.03.C	M.05
SPEC 5.05.04.E	15.07.08.03.B.04	A.04

ITS	CTS	DOC
SPEC 5.05.04.F	15.07.08.03.B.05	M.04
SPEC 5.05.04.G	15.07.08.03.B.06	A.04
	15.07.08.03.B.06.a	A.04
	15.07.08.03.B.06.b	A.04
	15.07.08.03.B.06.c	A.04
SPEC 5.05.04.H	NEW	M.06
SPEC 5.05.04.I	15.07.08.03.B.07	A.04
SPEC 5.05.04.J	15.07.08.03.C	M.05
SPEC 5.05.05	NEW	M.07
SPEC 5.05.06	NEW	M.08
SPEC 5.05.07	15.04.02	A.04
	15.04.02.B	A.04
	15.04.02.B.03	A.04
SPEC 5.05.07.a	NEW	M.09
SPEC 5.05.07.b	NEW	M.09
SPEC 5.05.07.c	NEW	M.09
SPEC 5.05.07.d	15.04.02.B.03.a	A.04
SPEC 5.05.08	15.04.02.A	A.04
	NEW	M.10
SPEC 5.05.08 T 5.05.08-01	15.04.02 T 15.04.02-01	A.04
SPEC 5.05.08.a	15.04.02.A.05.A	A.04
SPEC 5.05.08.a.01	15.04.02.A.01	A.04
SPEC 5.05.08.a.02	15.04.02.A.05.A	A.04
SPEC 5.05.08.a.03	15.04.02.A.05.A	A.04
SPEC 5.05.08.a.04	15.04.02.A.05.A	A.04
SPEC 5.05.08.a.05	15.04.02.A.05.A	A.04
SPEC 5.05.08.a.06	15.04.02.A.05.A	A.04
SPEC 5.05.08.b	15.04.02.A.02	A.04
SPEC 5.05.08.b.01	15.04.02.A.02.A	A.04
SPEC 5.05.08.b.01.i	15.04.02.A.02.A.01	A.04
SPEC 5.05.08.b.01.ii	15.04.02.A.02.A.02	A.04
SPEC 5.05.08.b.02	15.04.02.A.02.B	A.04
SPEC 5.05.08.b.02.i	15.04.02.A.02.B	A.04

ITS	CTS	DOC
SPEC 5.05.08.b.02.ii	15.04.02.A.02.B	A.04
SPEC 5.05.08.b.02.iii	15.04.02.A.02.B	A.04
SPEC 5.05.08.b.03	15.04.02.A.02.C	A.04
SPEC 5.05.08.b.04	15.04.02.A.02.D	A.04
SPEC 5.05.08.b.05	15.04.02.A.02.F	A.04
SPEC 5.05.08.c	15.04.02.A.03	A.10
SPEC 5.05.08.d	15.04.02.A.04	A.04
SPEC 5.05.08.d.01	15.04.02.A.04.A	A.04
SPEC 5.05.08.d.02	15.04.02.A.04.B	A.04
SPEC 5.05.08.d.03	15.04.02.A.04.C	A.04
SPEC 5.05.08.d.04	15.04.02.A.04.D	A.04
SPEC 5.05.08.d.05	15.04.02.A.04.E	A.04
SPEC 5.05.08.e	15.04.02.A.06	A.09
SPEC 5.05.09	DPR-24 OL 3.I	A.04
	DPR-27 OL 3.I	A.04
SPEC 5.05.09.A	DPR-24 OL 3.I.01	A.04
	DPR-27 OL 3.I.01	A.04
SPEC 5.05.09.B	DPR-24 OL 3.I.02	A.04
	DPR-27 OL 3.I.02	A.04
SPEC 5.05.09.C	DPR-24 OL 3.I.03	A.04
	DPR-27 OL 3.I.03	A.04
SPEC 5.05.09.D	DPR-24 OL 3.I.04	A.04
	DPR-27 OL 3.I.04	A.04
SPEC 5.05.09.E	DPR-24 OL 3.I.05	A.04
	DPR-27 OL 3.I.05	A.04
SPEC 5.05.09.F	DPR-24 OL 3.I.06	A.04
	DPR-27 OL 3.I.06	A.04
SPEC 5.05.10	NEW	A.04
	NEW	M.12
SPEC 5.05.10.a	15.03.12.02.a	A.11
	15.04.11.04.a	A.04
	15.04.11.04.b	A.04

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ITS	CTS	DOC
SPEC 5.05.10.b	15.03.12.02.a	A.11
	15.04.11.04.b	M.11
	15.04.11.04.c	M.11
SPEC 5.05.10.c	15.03.12.02.b	A.11
	15.04.11.04.d	A.04
	BASES	A.04
SPEC 5.05.10.d	15.04.11.01	LA.06
	15.04.11.01	M.03
SPEC 5.05.11	NEW	M.13
SPEC 5.05.11.A	NEW	M.13
SPEC 5.05.12	15.04.06.A.06	A.13
SPEC 5.05.12.A	NEW	M.14
SPEC 5.05.12.A.1	NEW	M.14
SPEC 5.05.12.A.2	NEW	M.14
SPEC 5.05.12.A.3	NEW	M.14
SPEC 5.05.12.B	NEW	M.14
SPEC 5.05.12.C	NEW	M.14
SPEC 5.05.13	NEW	M.15
SPEC 5.05.13.A	NEW	M.15
SPEC 5.05.13.B.1	NEW	M.15
SPEC 5.05.13.B.2	NEW	M.15
SPEC 5.05.13.C	NEW	M.15
SPEC 5.05.13.D	NEW	M.15
SPEC 5.05.14	NEW	M.15
SPEC 5.05.14.01.A	NEW	M.15
SPEC 5.05.14.01.B	NEW	M.15
SPEC 5.05.14.01.C	NEW	M.15
SPEC 5.05.14.01.D	NEW	M.15
SPEC 5.05.14.02.A	NEW	M.15
SPEC 5.05.14.02.B	NEW	M.15
SPEC 5.05.14.02.C	NEW	M.15
SPEC 5.05.15	15.06.12	A.04
SPEC 5.05.15.A	15.06.12.A	A.04

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ITS	CTS	DOC
SPEC 5.05.15.B	15.06.12.B	A.04
SPEC 5.05.15.C	15.06.12.C	A.04
SPEC 5.05.15.D	15.06.12.D	A.04
SPEC 5.05.15.D.01	15.06.12.D.01	A.04
SPEC 5.05.15.D.02	15.06.12.D.02	A.04
SPEC 5.05.15.D.03	NEW	M.16
SPEC 5.05.15.D.03.a	NEW	M.16
SPEC 5.05.15.D.03.b	NEW	M.16
SPEC 5.05.15.E	15.06.12.E	A.04
SPEC 5.05.15.F	15.06.12.F	A.04
SPEC 5.05.16	NEW	A.08
SPEC 5.05.16.01	15.04.16 T 15.04.16-01 FOOTNOTE (a).01	A.04
SPEC 5.05.16.02	15.04.16 T 15.04.16-01 FOOTNOTE (a).02	A.04
SPEC 5.05.16.03	15.04.16 T 15.04.16-01 FOOTNOTE (a).03	A.04
SPEC 5.05.16.04	15.04.16 T 15.04.16-01 FOOTNOTE (a).04	A.04

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CTS to ITS

15-Mar-00

CTS	ITS	DOC
15.03.09	N/A	A.01
15.03.12.02.a	N/A	LA.06
	SPEC 5.05.10.a	A.11
	SPEC 5.05.10.b	A.11
15.03.12.02.b	N/A	LA.06
	SPEC 5.05.10.c	A.11
15.04.02	SPEC 5.05.07	A.04
15.04.02 APPL	N/A	A.12
15.04.02 OBJ	N/A	A.12
15.04.02 T 15.04.02-01	SPEC 5.05.08 T 5.05.08-01	A.04
15.04.02.A	SPEC 5.05.08	A.04
15.04.02.A.01	SPEC 5.05.08.a.01	A.04
15.04.02.A.02	SPEC 5.05.08.b	A.04
15.04.02.A.02.A	SPEC 5.05.08.b.01	A.04
15.04.02.A.02.A.01	SPEC 5.05.08.b.01.i	A.04
15.04.02.A.02.A.02	SPEC 5.05.08.b.01.ii	A.04
15.04.02.A.02.B	SPEC 5.05.08.b.02	A.04
	SPEC 5.05.08.b.02.i	A.04
	SPEC 5.05.08.b.02.ii	A.04
	SPEC 5.05.08.b.02.iii	A.04
15.04.02.A.02.C	SPEC 5.05.08.b.03	A.04
15.04.02.A.02.D	SPEC 5.05.08.b.04	A.04
15.04.02.A.02.E	N/A	A.09
15.04.02.A.02.F	SPEC 5.05.08.b.05	A.04
15.04.02.A.03	SPEC 5.05.08.c	A.10
15.04.02.A.04	SPEC 5.05.08.d	A.04
15.04.02.A.04.A	SPEC 5.05.08.d.01	A.04
15.04.02.A.04.B	SPEC 5.05.08.d.02	A.04
15.04.02.A.04.C	SPEC 5.05.08.d.03	A.04
15.04.02.A.04.D	SPEC 5.05.08.d.04	A.04
15.04.02.A.04.E	SPEC 5.05.08.d.05	A.04

CTS	ITS	DOC
15.04.02.A.05.A	N/A	A.09
	SPEC 5.05.08.a	A.04
	SPEC 5.05.08.a.02	A.04
	SPEC 5.05.08.a.03	A.04
	SPEC 5.05.08.a.04	A.04
	SPEC 5.05.08.a.05	A.04
	SPEC 5.05.08.a.06	A.04
15.04.02.A.06	SPEC 5.05.08.e	A.09
15.04.02.B	N/A	LB.05
	SPEC 5.05.07	A.04
15.04.02.B.01	N/A	LB.05
15.04.02.B.01.a	N/A	LB.05
15.04.02.B.03	N/A	LB.05
	SPEC 5.05.07	A.04
15.04.02.B.03.a	SPEC 5.05.07.d	A.04
15.04.04.II	N/A	LB.02
15.04.04.II.A	N/A	LB.02
15.04.04.II.B	N/A	LB.02
15.04.04.II.C	N/A	LB.02
15.04.04.II.C.01	N/A	LB.02
15.04.04.II.C.01.A	N/A	LB.02
15.04.04.II.C.02	N/A	LB.02
15.04.04.II.C.02.A	N/A	LB.02
15.04.04.II.C.02.B	N/A	LB.02
15.04.04.II.C.02.B.I	N/A	LB.02
15.04.04.II.C.02.B.II	N/A	LB.02
15.04.04.II.C.02.B.III	N/A	LB.02
15.04.04.II.C.02.B.IV	N/A	LB.02
15.04.04.II.C.02.C	N/A	LB.02
15.04.04.II.C.02.D	N/A	LB.02
15.04.04.II.C.02.E	N/A	LB.02
15.04.04.II.C.02.E.01	N/A	LB.02
15.04.04.II.C.02.E.02	N/A	LB.02

CTS	ITS	DOC
15.04.04.II.C.02.E.03	N/A	LB.02
15.04.04.II.D	N/A	LB.02
15.04.04.III	N/A	LB.03
15.04.04.III.A	N/A	LB.03
15.04.04.III.B	N/A	LB.03
15.04.04.III.C	N/A	LB.03
15.04.04.III.C.01	N/A	LB.03
15.04.04.III.C.02	N/A	LB.03
15.04.04.III.C.03	N/A	LB.03
15.04.04.III.C.04	N/A	LB.03
15.04.04.III.C.05	N/A	LB.03
15.04.04.III.C.06	N/A	LB.03
15.04.04.III.D	N/A	LB.03
15.04.04.III.E	N/A	LB.03
15.04.04.IV	N/A	LB.04
15.04.04.IV.A	N/A	LB.04
15.04.04.IV.A.01	N/A	LB.04
15.04.04.IV.A.02	N/A	LB.04
15.04.04.IV.B	N/A	LB.04
15.04.04.IV.C	N/A	LB.04
15.04.04.IV.D	N/A	LB.04
15.04.04.IV.E	N/A	LB.04
15.04.06.A.06	SPEC 5.05.12	A.13
15.04.10	N/A	A.01
15.04.11.01	SPEC 5.05.10.d	LA.06
	SPEC 5.05.10.d	M.03
15.04.11.04.a	N/A	LA.06
	SPEC 5.05.10.a	A.04
15.04.11.04.b	N/A	LA.06
	SPEC 5.05.10.a	A.04
	SPEC 5.05.10.b	M.11
15.04.11.04.c	N/A	LA.06
	SPEC 5.05.10.b	M.11

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CTS	ITS	DOC
15.04.11.04.d	N/A	LA.06
	SPEC 5.05.10.c	A.04
15.04.16 T 15.04.16-01 FOOTNOTE (a).01	SPEC 5.05.16.01	A.04
15.04.16 T 15.04.16-01 FOOTNOTE (a).02	SPEC 5.05.16.02	A.04
15.04.16 T 15.04.16-01 FOOTNOTE (a).03	SPEC 5.05.16.03	A.04
15.04.16 T 15.04.16-01 FOOTNOTE (a).04	SPEC 5.05.16.04	A.04
15.06.08.04.A	SPEC 5.05.03	A.07
	SPEC 5.05.03	M.01
15.06.08.04.A FOOT NOTE *	N/A	A.07
15.06.08.04.A FOOT NOTE **	N/A	A.07
15.06.08.04.A.I	SPEC 5.05.03.A	A.04
15.06.08.04.A.II	SPEC 5.05.03.B	A.04
15.06.08.04.A.III	SPEC 5.05.03.C	A.04
15.06.12	SPEC 5.05.15	A.04
15.06.12.A	SPEC 5.05.15.A	A.04
15.06.12.B	SPEC 5.05.15.B	A.04
15.06.12.C	SPEC 5.05.15.C	A.04
15.06.12.D	SPEC 5.05.15.D	A.04
15.06.12.D.01	SPEC 5.05.15.D.01	A.04
15.06.12.D.02	SPEC 5.05.15.D.02	A.04
15.06.12.E	SPEC 5.05.15.E	A.04
15.06.12.F	SPEC 5.05.15.F	A.04
15.07	N/A	A.02
15.07.01.A	ODCM	LA.01
15.07.01.B	ODCM	LA.01
15.07.01.C	ODCM	LA.01
15.07.01.D	ODCM	LA.01
15.07.03	N/A	A.01
15.07.04	N/A	A.01
15.07.05	N/A	A.01
15.07.05 APPL	N/A	A.12
15.07.05 OBJ	N/A	A.12
15.07.05.A	N/A	LA.07

CTS	ITS	DOC
15.07.05.A.01	N/A	LA.07
15.07.05.A.02	N/A	LA.07
15.07.06	N/A	A.01
15.07.07	N/A	A.01
15.07.08.02	N/A	LA.04
15.07.08.02.A	N/A	LA.04
15.07.08.02.B	N/A	LA.04
15.07.08.03	SPEC 5.05.04	A.05
15.07.08.03.A	N/A	LA.02
	SPEC 5.05.01.A	A.03
	SPEC 5.05.01.B	A.03
	SPEC 5.05.01.B	A.04
15.07.08.03.B	SPEC 5.05.04	A.05
	SPEC 5.05.04.C	A.05
15.07.08.03.B.02	SPEC 5.05.04.C	A.04
	SPEC 5.05.04.C	A.05
15.07.08.03.B.03	SPEC 5.05.04.B	A.04
15.07.08.03.B.04	SPEC 5.05.04.E	A.04
15.07.08.03.B.05	SPEC 5.05.04.F	M.04
15.07.08.03.B.06	SPEC 5.05.04.G	A.04
15.07.08.03.B.06.a	SPEC 5.05.04.G	A.04
15.07.08.03.B.06.b	SPEC 5.05.04.G	A.04
15.07.08.03.B.06.c	SPEC 5.05.04.G	A.04
15.07.08.03.B.07	SPEC 5.05.04.I	A.04
15.07.08.03.B.08	SPEC 5.05.01.B	A.04
15.07.08.03.C	SPEC 5.05.01.A	A.04
	SPEC 5.05.01.A	A.05
	SPEC 5.05.04.D	A.05
	SPEC 5.05.04.D	M.05
	SPEC 5.05.04.D	A.04
	SPEC 5.05.04.J	M.05
15.07.08.03.D	N/A	LB.01
15.07.08.05	N/A	LA.03
15.07.08.05.A	N/A	LA.03

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CTS	ITS	DOC
15.07.08.05.B	N/A	LA.03
15.07.08.05.C	N/A	LA.03
15.07.08.05.D	N/A	LA.03
15.07.08.05.E	N/A	LA.03
15.07.08.05.F	N/A	LA.03
15.07.08.07.A	N/A	LB.01
15.07.08.07.A.01	N/A	LB.01
15.07.08.07.A.02	N/A	LB.01
15.07.08.07.A.03	N/A	LB.01
15.07.08.07.B	SPEC 5.05.01.C	A.05
15.07.08.07.B.01	SPEC 5.05.01.C.01	A.05
15.07.08.07.B.01.a	SPEC 5.05.01.C.01.i	A.04
15.07.08.07.B.01.b	SPEC 5.05.01.C.01.ii	A.04
15.07.08.07.B.02	SPEC 5.05.01.C.02	A.04
15.07.08.07.B.03	SPEC 5.05.01.C.03	A.05
15.07.08.07.B.04	N/A	A.06
BASES	N/A	LA.05
	SPEC 5.05.10.c	A.04
DPR-24 OL 3.I	SPEC 5.05.09	A.04
DPR-24 OL 3.I.01	SPEC 5.05.09.A	A.04
DPR-24 OL 3.I.02	SPEC 5.05.09.B	A.04
DPR-24 OL 3.I.03	SPEC 5.05.09.C	A.04
DPR-24 OL 3.I.04	SPEC 5.05.09.D	A.04
DPR-24 OL 3.I.05	SPEC 5.05.09.E	A.04
DPR-24 OL 3.I.06	SPEC 5.05.09.F	A.04
DPR-27 OL 3.I	SPEC 5.05.09	A.04
DPR-27 OL 3.I.01	SPEC 5.05.09.A	A.04
DPR-27 OL 3.I.02	SPEC 5.05.09.B	A.04
DPR-27 OL 3.I.03	SPEC 5.05.09.C	A.04
DPR-27 OL 3.I.04	SPEC 5.05.09.D	A.04
DPR-27 OL 3.I.05	SPEC 5.05.09.E	A.04
DPR-27 OL 3.I.06	SPEC 5.05.09.F	A.04

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DOC Number	DOC Text
A.01	The information contained in CTS sections 15.3.9, 15.4.10, 15.7.3, 15.7.4, 15.7.5, 15.7.6 and 15.7.7 is not being retained in ITS. This information does not provide any regulatory requirements necessary to protect the public health and safety, but rather states that the requirements previously contained in the above CTS sections were relocated to the Radiological Effluents and Materials Control and Accountability Program Manual (REMCAP). Therefore, deletion of this information is administrative.

CTS:	ITS:
15.03.09	N/A
15.04.10	N/A
15.07.03	N/A
15.07.04	N/A
15.07.05	N/A
15.07.06	N/A
15.07.07	N/A

DOC Number	DOC Text
A.02	The information contained in CTS 15.7 is not being retained in ITS. This information does not provide any regulatory requirements necessary to protect the public health and safety, but rather states that the RETS do not expand the responsibilities of the licensed operators, and the material contained therein will not be the subject of SRO/RO licensing examinations. Therefore, deletion of this information is administrative.

CTS:	ITS:
15.07	N/A

DOC Number	DOC Text
A.03	CTS 15.7.8.3.a is revised to reflect the format of the ISTS. The Environmental Manual (EM) will become the ODCM, which will contain the methodology and parameters used in the conduct of the radiological environmental monitoring program. The ODCM will also contain the radiological effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Monitoring Report.

CTS:	ITS:
15.07.08.03.A	SPEC 5.05.01.A
	SPEC 5.05.01.B

DOC Number	DOC Text
A.04	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.04.02	SPEC 5.05.07
15.04.02 T 15.04.02-01	SPEC 5.05.08 T 5.05.08-01
15.04.02.A	SPEC 5.05.08
15.04.02.A.01	SPEC 5.05.08.a.01
15.04.02.A.02	SPEC 5.05.08.b
15.04.02.A.02.A	SPEC 5.05.08.b.01
15.04.02.A.02.A.01	SPEC 5.05.08.b.01.i
15.04.02.A.02.A.02	SPEC 5.05.08.b.01.ii
15.04.02.A.02.B	SPEC 5.05.08.b.02 SPEC 5.05.08.b.02.i SPEC 5.05.08.b.02.ii SPEC 5.05.08.b.02.iii
15.04.02.A.02.C	SPEC 5.05.08.b.03
15.04.02.A.02.D	SPEC 5.05.08.b.04
15.04.02.A.02.F	SPEC 5.05.08.b.05
15.04.02.A.04	SPEC 5.05.08.d
15.04.02.A.04.A	SPEC 5.05.08.d.01
15.04.02.A.04.B	SPEC 5.05.08.d.02
15.04.02.A.04.C	SPEC 5.05.08.d.03
15.04.02.A.04.D	SPEC 5.05.08.d.04
15.04.02.A.04.E	SPEC 5.05.08.d.05
15.04.02.A.05.A	SPEC 5.05.08.a SPEC 5.05.08.a.02 SPEC 5.05.08.a.03 SPEC 5.05.08.a.04 SPEC 5.05.08.a.05 SPEC 5.05.08.a.06
15.04.02.B	SPEC 5.05.07
15.04.02.B.03	SPEC 5.05.07

DOC Number	DOC Text	
	15.04.02.B.03.a	SPEC 5.05.07.d
	15.04.11.04.a	SPEC 5.05.10.a
	15.04.11.04.b	SPEC 5.05.10.a
	15.04.11.04.d	SPEC 5.05.10.c
	15.04.16 T 15.04.16-01 FOOTNOTE (a).01	SPEC 5.05.16.01
	15.04.16 T 15.04.16-01 FOOTNOTE (a).02	SPEC 5.05.16.02
	15.04.16 T 15.04.16-01 FOOTNOTE (a).03	SPEC 5.05.16.03
	15.04.16 T 15.04.16-01 FOOTNOTE (a).04	SPEC 5.05.16.04
	15.06.08.04.A.I	SPEC 5.05.03.A
	15.06.08.04.A.II	SPEC 5.05.03.B
	15.06.08.04.A.III	SPEC 5.05.03.C
	15.06.12	SPEC 5.05.15
	15.06.12.A	SPEC 5.05.15.A
	15.06.12.B	SPEC 5.05.15.B
	15.06.12.C	SPEC 5.05.15.C
	15.06.12.D	SPEC 5.05.15.D
	15.06.12.D.01	SPEC 5.05.15.D.01
	15.06.12.D.02	SPEC 5.05.15.D.02
	15.06.12.E	SPEC 5.05.15.E
	15.06.12.F	SPEC 5.05.15.F
	15.07.08.03.A	SPEC 5.05.01.B
	15.07.08.03.B.02	SPEC 5.05.04.C
	15.07.08.03.B.03	SPEC 5.05.04.B
	15.07.08.03.B.04	SPEC 5.05.04.E
	15.07.08.03.B.06	SPEC 5.05.04.G
	15.07.08.03.B.06.a	SPEC 5.05.04.G
	15.07.08.03.B.06.b	SPEC 5.05.04.G
	15.07.08.03.B.06.c	SPEC 5.05.04.G
	15.07.08.03.B.07	SPEC 5.05.04.I
	15.07.08.03.B.08	SPEC 5.05.01.B
	15.07.08.03.C	SPEC 5.05.01.A SPEC 5.05.04.D
	15.07.08.07.B.01.a	SPEC 5.05.01.C.01.i
	15.07.08.07.B.01.b	SPEC 5.05.01.C.01.ii
	15.07.08.07.B.02	SPEC 5.05.01.C.02

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DOC Number	DOC Text	
	BASES	SPEC 5.05.10.c
	DPR-24 OL 3.I	SPEC 5.05.09
	DPR-24 OL 3.I.01	SPEC 5.05.09.A
	DPR-24 OL 3.I.02	SPEC 5.05.09.B
	DPR-24 OL 3.I.03	SPEC 5.05.09.C
	DPR-24 OL 3.I.04	SPEC 5.05.09.D
	DPR-24 OL 3.I.05	SPEC 5.05.09.E
	DPR-24 OL 3.I.06	SPEC 5.05.09.F
	DPR-27 OL 3.I	SPEC 5.05.09
	DPR-27 OL 3.I.01	SPEC 5.05.09.A
	DPR-27 OL 3.I.02	SPEC 5.05.09.B
	DPR-27 OL 3.I.03	SPEC 5.05.09.C
	DPR-27 OL 3.I.04	SPEC 5.05.09.D
	DPR-27 OL 3.I.05	SPEC 5.05.09.E
	DPR-27 OL 3.I.06	SPEC 5.05.09.F
	NEW	SPEC 5.05.10

A.05 15.7.8.3, 15.7.8.3.b, 15.7.8.3.c and 15.7.8.7.B have been revised to reflect the concurrent reorganization of the Radiological Effluents and Materials Control and Accountability Program Manual (REMCAP), Environmental Manual (EM), Radiological Environmental Monitoring Program (REMP) and Radiological Effluent Control Program (RECP) into the Offsite Dose Calculation Manual (ODCM), consistent with the recommendation of GL 89-01. The revisions to the CTS are necessary to adopt certain wording preferences or conventions which do not result in technical changes.

CTS:	ITS:
15.07.08.03	SPEC 5.05.04
15.07.08.03.B	SPEC 5.05.04 SPEC 5.05.04.C
15.07.08.03.B.02	SPEC 5.05.04.C
15.07.08.03.C	SPEC 5.05.01.A SPEC 5.05.04.D
15.07.08.07.B	SPEC 5.05.01.C
15.07.08.07.B.01	SPEC 5.05.01.C.01
15.07.08.07.B.03	SPEC 5.05.01.C.03

DOC Number	DOC Text
A.06	CTS 15.7.8.7.B.4 requires all changes regarding explosive gas to be made via the 50.59 process. The Explosive Gas Monitoring Program is required by procedure to be controlled by the 10 CFR 50.59 process. It is unnecessary to state this requirement in Technical Specifications. Therefore, deletion of this statement is administrative in nature.

CTS:	ITS:
15.07.08.07.B.04	N/A

DOC Number	DOC Text
A.07	CTS 15.6.8.4.A is modified by foot note *, "Post-Accident Coolant Sampling and Post-Accident Containment Atmospheric Sampling Systems" and foot note **, "It is acceptable if the licensee maintains details of the program in plant operation manuals." These footnotes do not establish or relax any requirement and these details are not required in ITS to provide adequate protection of the public health and safety.

CTS:	ITS:
15.06.08.04.A	SPEC 5.05.03
15.06.08.04.A FOOT NOTE *	N/A
15.06.08.04.A FOOT NOTE **	N/A

DOC Number	DOC Text
A.08	CTS 15.4.16, Table 15.4.16-1, footnotes (a) and (b) are retained in ITS as the requirements of the RCS PIV Leakage Program. These footnotes are being preceded by a statement that the program shall be established to verify the leakage from each RCS PIV is within the limits specified, in accordance with the Event V Order, issued April 20, 1981. This statement does not impose any additional requirements, but rather provides information necessary to apply the specified limits to the RCS PIVs.

CTS:	ITS:
NEW	SPEC 5.05.16

DOC Number	DOC Text
A.09	CTS 15.4.2.A.2(e) and associated footnote 1, and 15.4.2.A.5(a) Definitions for F* Distance and F* Tube and associated footnote 2, have not been retained in ITS. These items were applicable only to Westinghouse Model 44 steam generators in Unit 2. According to the footnotes, these requirements, definitions, and repair options are null and void following Unit 2 steam generator replacement. Due to the replacement of the Unit 2 steam generators, these requirements, definitions, and repair options are no longer required to be in the Technical Specifications, and are therefore deleted.

CTS:	ITS:
15.04.02.A.02.E	N/A
15.04.02.A.05.A	N/A
15.04.02.A.06	SPEC 5.05.08.e

DOC Number	DOC Text
A.10	CTS 15.4.2.A.3 has been modified by replacing reference to CTS 15.4.2.B.1 with a reference to 10 CFR 50.55a(g). CTS 15.4.2.B.1 provided Inservice Inspection requirements, which have been removed from the Technical Specifications, because they are duplicative of the 10 CFR 50.55a(g) requirements.

CTS:	ITS:
15.04.02.A.03	SPEC 5.05.08.c

DOC Number	DOC Text
A.11	CTS 15.3.12.2.a states the results of the in-place cold DOP and halogenated hydrocarbon tests on the HEPA and charcoal adsorber banks shall show a "minimum of 99% DOP removal and 99% halogenated hydrocarbon removal." CTS 15.3.12.2.b states the laboratory charcoal adsorbent tests shall show a "minimum of 99% removal of methyl iodide." The requirements of CTS 15.3.12.2.a have been changed to "penetration and system bypass </= 1.0%." The requirement of CTS 15.3.12.2.b has been changed to "methyl iodide penetration </= 1.0%." These revisions do not change the requirements, but rather restate the same requirement in different terms. Therefore, this change is administrative.

CTS:	ITS:
15.03.12.02.a	SPEC 5.05.10.a
	SPEC 5.05.10.b
15.03.12.02.b	SPEC 5.05.10.c

DOC Number	DOC Text
A.12	CTS 15.4.2 and 15.7.5 provide introductory statements (Applicability / Objectives) which simply state which systems/components are addressed within each section and provide a brief summary of the purpose for each Section. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431.

CTS:	ITS:
15.04.02 APPL	N/A
15.04.02 OBJ	N/A
15.07.05 APPL	N/A
15.07.05 OBJ	N/A

DOC Number	DOC Text
A.13	Editorial changes to CTS 15.4.6.A.6 have been made to clarify the diesel fuel oil testing program. The program will include sampling and testing requirements and acceptance criteria in accordance with applicable ASTM standards.

CTS:	ITS:
15.04.06.A.06	SPEC 5.05.12

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DOC Number	DOC Text	
LA.01	The information contained in CTS sections 15.7.1 is not being retained in ITS. This information does not provide any regulatory requirements necessary to protect the public health and safety, but provides definitions for frequently used terms in the RETS. The requirements of the RETS were removed from the CTS in Amendments 184/188 and placed in the Radiological Effluents and Materials Control and Accountability Program (REMCAP). In conjunction with the ITS project, the REMCAP is being reorganized to reflect the recommendations of GL 89-01, and will become the Offsite Dose Calculation Manual (ODCM). The information contained in CTS 15.7.1 will be moved to the ODCM. This information is not necessary to adequately describe the actual regulatory requirement and can be moved to other documents without impact on safety. Changes to the ODCM will be controlled by the ODCM process in Section 5 of the proposed ITS.	
	CTS:	**ITS:**
	15.07.01.A	ODCM
	15.07.01.B	ODCM
	15.07.01.C	ODCM
	15.07.01.D	ODCM
LA.02	The information contained in CTS sections 15.7.8.3.a regarding an annual milk survey is not being retained in ITS. This information will be located in the ODCM. This information is not necessary to adequately protect the public and can be moved to other documents without impact on safety. Changes to the ODCM will be controlled by the ODCM process in Section 5 of the proposed ITS.	
	CTS:	**ITS:**
	15.07.08.03.A	N/A
LA.03	The information contained in CTS 15.7.8.5 regarding major changes to radioactive liquid, gaseous and solid waste treatment systems is not being retained in ITS. This information will be located in the ODCM. This information is not necessary to adequately protect the public and can be moved to other documents without impact on safety. Changes to the ODCM will be controlled by the ODCM process in Section 5 of the proposed ITS.	
	CTS:	**ITS:**
	15.07.08.05	N/A
	15.07.08.05.A	N/A
	15.07.08.05.B	N/A
	15.07.08.05.C	N/A
	15.07.08.05.D	N/A
	15.07.08.05.E	N/A
	15.07.08.05.F	N/A

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DOC Number	DOC Text	
LA.04	The information contained in CTS 15.7.8.2 regarding audits of the activities encompassed by the Radioactive Effluent and Materials and Accountability Program (REMCAP) is not being retained in ITS. In conjunction with the ITS project, the REMCAP is being reorganized to reflect the recommendations of GL 89-01, and will become the Offsite Dose Calculation Manual (ODCM). The information contained in CTS 15.7.8.2 will be moved to the ODCM. This information is not necessary to adequately protect the public and can be moved to other documents without impact on safety. Changes to the ODCM will be controlled by the ODCM process in Section 5 of the proposed ITS.	
	CTS:	**ITS:**
	15.07.08.02	N/A
	15.07.08.02.A	N/A
	15.07.08.02.B	N/A
LA.05	The Bases associated with CTS 15.4.2 is not being retained in ITS, but is moved to the FSAR. This information provides details which are not directly pertinent to the actual requirements. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to other documents without impact on safety. Changes to the FSAR are controlled in accordance with the 10 CFR 50.59 process.	
	CTS:	**ITS:**
	BASES	N/A
LA.06	CTS 15.3.12.A, Control Room Emergency Filtration, has been modified by removing the testing requirements of the Control Room Emergency Filtration (CREF) system. The CREF testing requirements will instead be in accordance with the applicable portions of Regulatory Guide (RG) 1.52, Revision 2, ASTM D3803-1989 and ASME N510-1989, as applicable. Although this change will result in less restrictive testing requirements for the HEPA filters and charcoal adsorbers, Regulatory Guide 1.52 contains methods acceptable to the NRC for implementing the regulations in 10 CFR 50, Appendix A, with regard to the testing criteria for air filtration and adsorption units of ESF atmospheric cleanup systems designed to mitigate the consequences of a postulated accident.	
	CTS:	**ITS:**
	15.03.12.02.a	N/A
	15.03.12.02.b	N/A
	15.04.11.01	SPEC 5.05.10.d
	15.04.11.04.a	N/A
	15.04.11.04.b	N/A
	15.04.11.04.c	N/A
	15.04.11.04.d	N/A

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DOC Number	DOC Text	
LA.07	The Gas Decay Tank oxygen concentration limit and the required actions if the limit is exceeded are not being retained in ITS. This information will be contained in the Explosive Gas Monitoring Program. This information is not necessary to adequately protect the public and can be moved to other documents without impact on safety. Changes to the Explosive Gas Monitoring Program will be controlled via the 10 CFR 50.59 process.	
	CTS:	**ITS:**
	15.07.05.A	N/A
	15.07.05.A.01	N/A
	15.07.05.A.02	N/A
LB.01	CTS 15.7.8.3.d and 15.7.8.7 contain requirements to establish and maintain a Process Control Program (PCP) to assure compliance with 10 CFR Parts 20, 61 and 71. These requirements duplicate current regulations which provide sufficient and appropriate control of these requirements. Therefore, these details are not required to be in the ITS to provide adequate protection of public health and safety. Since this information is contained in 10 CFR Parts 20, 61 and 71, the requirements will continue to be applicable to Point Beach. Therefore, this change is an administrative relocation of information.	
	CTS:	**ITS:**
	15.07.08.03.D	N/A
	15.07.08.07.A	N/A
	15.07.08.07.A.01	N/A
	15.07.08.07.A.02	N/A
	15.07.08.07.A.03	N/A

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15-Mar-00

DOC Number	DOC Text
LB.02	The Tendon Surveillance Program of CTS 15.4.4.II is not being retained in the ITS. 10 CFR 50.55.a requires facilities to adopt the ASME Section XI, Subsection IWE and IWL programs by September 2001. Point Beach will adopt these Section XI programs prior to ITS implementation. Therefore, the Tendon Surveillance Program will be duplicative of the requirements specified by ASME Section XI, as endorsed and required under 10 CFR 50.55a. Inclusion of these requirements via reference into 10 CFR 50.55a makes these requirement applicable to Point Beach without the need to duplicate these requirements in the Technical Specifications.

CTS:	ITS:
15.04.04.II	N/A
15.04.04.II.A	N/A
15.04.04.II.B	N/A
15.04.04.II.C	N/A
15.04.04.II.C.01	N/A
15.04.04.II.C.01.A	N/A
15.04.04.II.C.02	N/A
15.04.04.II.C.02.A	N/A
15.04.04.II.C.02.B	N/A
15.04.04.II.C.02.B.I	N/A
15.04.04.II.C.02.B.II	N/A
15.04.04.II.C.02.B.III	N/A
15.04.04.II.C.02.B.IV	N/A
15.04.04.II.C.02.C	N/A
15.04.04.II.C.02.D	N/A
15.04.04.II.C.02.E	N/A
15.04.04.II.C.02.E.01	N/A
15.04.04.II.C.02.E.02	N/A
15.04.04.II.C.02.E.03	N/A
15.04.04.II.D	N/A

Description of Changes - NUREG-1431 Section 5.05

15-Mar-00

DOC Number	DOC Text
LB.03	The End Anchorage Concrete Surveillance requirements of CTS 15.4.4.III are not being retained in the ITS. The Inservice Inspection of ASME Code Class 1, Class 2, and Class 3 components are required to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55.a(g) modified by Section 50.55.a(b), except where specific relief is granted by the NRC, pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Therefore, the Inservice Inspection requirements in the CTS are duplicative of the above ASME Section XI requirements and removing these requirements from CTS is an administrative relocation of the information.

CTS:	ITS:
15.04.04.III	N/A
15.04.04.III.A	N/A
15.04.04.III.B	N/A
15.04.04.III.C	N/A
15.04.04.III.C.01	N/A
15.04.04.III.C.02	N/A
15.04.04.III.C.03	N/A
15.04.04.III.C.04	N/A
15.04.04.III.C.05	N/A
15.04.04.III.C.06	N/A
15.04.04.III.D	N/A
15.04.04.III.E	N/A

DOC Number	DOC Text	
LB.04	The Liner Plate examination requirements of CTS 15.4.4.IV are not being retained in the ITS. The Inservice Inspection of ASME Code Class 1, Class 2, and Class 3 components are required to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55.a(g) modified by Section 50.55.a(b), except where specific relief is granted by the NRC, pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Therefore, the Inservice Inspection requirements in the CTS are duplicative of the above ASME Section XI requirements and removing these requirements from CTS is an administrative relocation of the information.	
	CTS:	**ITS:**
	15.04.04.IV	N/A
	15.04.04.IV.A	N/A
	15.04.04.IV.A.01	N/A
	15.04.04.IV.A.02	N/A
	15.04.04.IV.B	N/A
	15.04.04.IV.C	N/A
	15.04.04.IV.D	N/A
	15.04.04.IV.E	N/A
LB.05	The Inservice Inspection requirements of CTS 15.4.2.B, 15.4.2.B.1 and 15.4.2.B.3 are not being retained in the ITS. The Inservice Inspection of ASME Code Class 1, Class 2, and Class 3 components are required to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55.a(g) modified by Section 50.55.a(b), except where specific relief is granted by the NRC, pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Therefore, the Inservice Inspection requirements in the CTS are duplicative of the above ASME Section XI requirements and removing these requirements from CTS is an administrative relocation of the information.	
	CTS:	**ITS:**
	15.04.02.B	N/A
	15.04.02.B.01	N/A
	15.04.02.B.01.a	N/A
	15.04.02.B.03	N/A
M.01	CTS 15.6.8.4.A is proposed to be revised by the addition of "radioactive gases, and particulates in" before the words "containment atmosphere and in plant gaseous effluent samples . . " The addition of this text imposes additional requirements on unit operation and is more restrictive.	
	CTS:	**ITS:**
	15.06.08.04.A	SPEC 5.05.03

DOC Number	DOC Text	
M.02	The CTS has been revised by the addition of a requirement to establish, implement and maintain a Primary Coolant Sources Outside Containment Program. This program is required to provide controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practical. The program will be required to include preventive maintenance and periodic visual inspection requirements, and integrated leak test requirements for each system. This change imposes additional requirements for unit operation and is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.02
		SPEC 5.05.02.a
		SPEC 5.05.02.b
M.03	CTS 15.4.11.1 has been revised from requiring the pressure drop test across the combined HEPA filters and charcoal adsorber banks be demonstrated to be < 6 inches of water at "design Flow rate" to "4950 cfm +/- 10%." Stipulating the value of the design flow in the Technical Specifications imposes additional requirements and is therefore more restrictive.	
	CTS:	**ITS:**
	15.04.11.01	SPEC 5.05.10.d
M.04	CTS 15.7.8.3.b.4) has been modified by the addition of a requirement in the Radiological Effluent Program to provide limitations on the functional capability and use of the appropriate portions of the of the liquid and gaseous effluent treatment system. This revision imposes additional requirements on unit operation and is more restrictive.	
	CTS:	**ITS:**
	15.07.08.03.B.05	SPEC 5.05.04.F
M.05	CTS 15.7.8.3.c has been modified by the addition of the following requirements. In addition to the requirements to specify the annual doses to a member of the public from radioactive materials in liquid effluents and radioactivity and radiation from uranium fuel cycle sources released from the facility to unrestricted areas, the ODCM will be required to specify quarterly doses and dose commitments. This revision imposes additional requirements and is more restrictive.	
	CTS:	**ITS:**
	15.07.08.03.C	SPEC 5.05.04.D
		SPEC 5.05.04.J

DOC Number	DOC Text	
M.06	The CTS has been modified by the addition of the requirement to provide limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I. This revision imposes additional requirements and is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.04.H
M.07	The CTS has been revised by the addition of a requirement to establish, implement and maintain a Component Cyclic or Transient Limit Program. This program is required to provide controls to track the FSAR Section 4.1, cyclic and transient occurrences to ensure that components are maintained within design limits. The requirement to establish, implement and maintain a Component Cyclic or Transient Limit Program imposes additional requirements for unit operation and is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.05
M.08	The CTS has been revised by the addition of a requirement to establish, implement and maintain a Reactor Coolant Pump Flywheel Inspection Program. This program is required to provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1. However, in lieu of position c.4.b(1) and c.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and PT) of exposed surfaces of the removed flywheels may be conducted at approximately 10 year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI. The requirement to establish, implement and maintain a Reactor Coolant Pump Flywheel Inspection Program imposes additional requirements for unit operation and is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.06
M.09	CTS 15.4.2.B.3 has been modified by the adoption of a table that indicates the required frequencies for performing inservice testing activities as they relate to the testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. Also, statements requiring the provisions of SR 3.0.2 and SR 3.0.3 to be applicable to the inservice testing activities frequencies have been added to CTS 15.4.2.B.3. These changes impose additional requirements and are therefore more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.07.a SPEC 5.05.07.b SPEC 5.05.07.c

15-Mar-00

DOC Number	DOC Text	
M.10	A statement requiring the provisions of SR 3.0.2 to be applicable to the SG Tube Surveillance Testing Program test frequencies has been added to CTS 15.4.2.A. This change imposes additional requirements and is therefore more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.08
M.11	CTS 15.4.11.4.b and 15.4.11.4.c have been revised from requiring the DOP and the halogenated hydrocarbon testing at "design velocity +/- 20%" to "4950 cfm +/- 10%," to stipulate the actual design flowrate of the Control Room Emergency ventilation system. This change imposes additional requirements and is therefore more restrictive.	
	CTS:	**ITS:**
	15.04.11.04.b	SPEC 5.05.10.b
	15.04.11.04.c	SPEC 5.05.10.b
M.12	A statement requiring the provisions of SR 3.0.2 and SR 3.0.3 to be applicable to the Ventilation Filter Test Program test frequencies has been added to CTS 15.4.11. This change imposes additional requirements and is therefore more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.10
M.13	CTS 15.7.5 has been modified by the addition of a requirement to establish, implement and maintain an Explosive Gas Monitoring Program. This program is required to provide controls for potentially explosive gas mixtures contained in the on-service Gas Decay Tank. The program will include a limit for oxygen concentration in the on-service Gas Decay Tank and a surveillance program to ensure the limit is maintained. Additionally, the provisions of SR 3.0.2 and SR 3.0.3 will be applicable to the program surveillance frequencies. The requirement to establish, implement and maintain an Explosive Gas Monitoring Program imposes additional requirements and is therefore more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.11 SPEC 5.05.11.A

DOC Number	DOC Text
M.14	CTS 15.4.6.A.6 has been modified by specifying the diesel fuel oil program will establish acceptability of new fuel for use by: determining that the fuel has an API gravity or an absolute specific gravity within limits, a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and by determining the fuel has a clear and bright appearance with proper color; within 31 days of addition of the new fuel oil to storage tanks, the properties of the new fuel oil (other than API or absolute specific gravity, appearance, and flash point and kinematic viscosity) will be verified to be within limits for ASTM 2D fuel oil; and total particulate concentration of the fuel oil shall be < 10 mg/l, when tested every 92 days in accordance with the applicable ASTM standards. Adopting these requirements imposes additional requirements on unit operation and is therefore more restrictive.

CTS:	ITS:
NEW	SPEC 5.05.12.A
	SPEC 5.05.12.A.1
	SPEC 5.05.12.A.2
	SPEC 5.05.12.A.3
	SPEC 5.05.12.B
	SPEC 5.05.12.C

DOC Number	DOC Text
M.15	Two new programs are added in the ITS. These programs are:

ITS 5.5.13 Technical Specification (TS) Bases Control
ITS 5.5.14 Safety Function Determination Program (SFDP)

The TS Bases Control Program is provided to specifically delineate the appropriate methods and reviews necessary for a change to the Technical Specification Bases. The Safety Function Determination Program is included to support implementation of the support system OPERABILITY characteristics of the Technical Specifications.

Adopting these programs imposes additional requirements and is therefore more restrictive.

CTS:	ITS:
NEW	SPEC 5.05.13
	SPEC 5.05.13.A
	SPEC 5.05.13.B.1
	SPEC 5.05.13.B.2
	SPEC 5.05.13.C
	SPEC 5.05.13.D
	SPEC 5.05.14
	SPEC 5.05.14.01.A
	SPEC 5.05.14.01.B
	SPEC 5.05.14.01.C
	SPEC 5.05.14.01.D
	SPEC 5.05.14.02.A
	SPEC 5.05.14.02.B
	SPEC 5.05.14.02.C

DOC Number	DOC Text
M.16	Included in CTS 15.6.12 are the requirements for the Containment Leakage Rate Testing Program (CLRTP). These requirements will be retained in the proposed ITS in new section 5.5.15, with additional requirements for air lock testing being added.

NUREG-1431 SR 3.6.1.1 includes CLRTP acceptance criteria, which mirror those contained in CTS 15.6.12.D. However, these requirements were not adopted in proposed ITS SR 3.6.1.1. Proposed ITS SR 3.6.1.1 simply states "in accordance with the Containment Leakage Rate Testing Program" when describing the CLRTP acceptance criteria. Therefore, the PBNP CLRTP requirements are being added to section 5.5, "Programs and Manuals," of the proposed ITS so that the CLRTP requirements are included in the Technical Specifications.

NUREG-1431 SR 3.6.2.1 includes air lock leakage rate acceptance criteria. However, these requirements were not adopted in proposed ITS SR 3.6.2.1. Proposed ITS SR 3.6.2.1 simply states "in accordance with the Containment Leakage Rate Testing Program" when describing the air lock leakage rate acceptance criteria. Therefore, the PBNP air lock leakage rate acceptance criteria is being added to section 5.5.15 (CLRTP requirements) of the proposed ITS so that the requirements are included in the Technical Specifications.

This change is more restrictive, since it adds an additional section on CLRTP requirements to proposed ITS section 5.5.

CTS:	ITS:
NEW	SPEC 5.05.15.D.03
	SPEC 5.05.15.D.03.a
	SPEC 5.05.15.D.03.b

H. Wisconsin Electric 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 August 2, 1979 (and Supplements dated October 21, 1980, January 22, 1981, and July 27, 1988) and the safety evaluation issued January 8, 1997, for Technical Specification Amendment No. 170, subject to the following provision:

No Changes

The licensee 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.

ITS 5.5.9

I. Secondary Water Chemistry Monitoring Program

I. Deleted

The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;
2. Identification of the procedures used to quantify parameters that are critical to control points;
3. Identification of process sampling points
4. Procedure for the recording and management of data;
5. Procedures defining corrective actions for off control point chemistry condition; and
6. A procedure for identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

J. The licensee is authorized to repair Unit 1 steam generators by replacement of major components. Repairs shall be conducted in accordance with the licensee's commitments identified in the Commission approved Point Beach Nuclear Plant Unit No. 1 Steam Generator Repair Report dated August 9, 1982 and revised March 1, 1983 and additional commitments identified in the staff's related Safety Evaluation.

No Changes

K. Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 174, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

G. Safety Injection Logic

The licensee is authorized to modify the safety injection actuation logic and actuation power supplies and related changes as described in licensee's application for amendment dated April 27, 1979, as supplemented May 7, 1979. In the interim period until the power supply modification has been completed, should any DC powered safety injection actuation channel be in a failed condition for greater than one hour, the unit shall thereafter be shutdown using normal procedures and placed in a block-permissive condition for safety injection actuation.

No Changes

H. Wisconsin Electric 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 August 2, 1979 (and Supplements dated October 21, 1980, January 22, 1981, and July 27, 1988) and the safety evaluation issued January 8, 1997, for Technical Specification Amendment No. 174, subject to the following provision:

> The licensee 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.

ITS 5.5.9

I. Secondary Water Chemistry Monitoring Program

The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

I. Deleted

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;
2. Identification of the procedures used to quantify parameters that are critical to control points;
3. Identification of process sampling points
4. Procedure for the recording and management of data;
5. Procedures defining corrective actions for off control point chemistry condition; and
6. A procedure for identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

J. Additional Conditions

No Changes

The Additional Conditions contained in Appendix C, as revised through Amendment No. 178, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

~~15.3.9~~ Effluent Release

Radioactive Effluent ~~Release limits~~, effluent ~~sampling, and~~ effluent analyses requirements are contained in the Radiological Effluent and Materials Control and Accountability ~~Program Manual~~.

A.1

15.3.12 CONTROL ROOM EMERGENCY FILTRATION

Applicability

Applies to the operability of the control room emergency filtration.

Objective

To specify functional requirements of the control room emergency filtration during power operation and refueling operation.

Insert 5.5-1 A.4

< See 3.7.10 >

Specification

1. Except as specified in 15.3.12.3 below, the control room emergency filtration system shall be operable at all times during power operation and refueling operation of either unit.

2. a. The results of in-place ~~cold DOP and halogenated hydrocarbon~~ tests, conducted in accordance with ~~Specification 15.4.11~~, on HEPA filter and charcoal adsorber banks shall show ~~a minimum of 99% DOP removal and 99% halogenated hydrocarbon removal.~~ LA.6

   b. The results of laboratory charcoal adsorbent tests, conducted in accordance with ~~Specification 15.4.11~~, shall show ~~a minimum of 99% removal of methyl iodide~~. ~~If laboratory analysis results for in-place charcoal indicate less than 99% methyl iodide removal, this specification may be met by replacement with charcoal adsorbent which has been verified to achieve 99% minimum removal and which has been stored in sealed containers, and retesting the charcoal adsorber bank for halogenated hydrocarbon removal.~~ LA.6

< See 3.7.10 >

   c. The results of fan testing, conducted in accordance with specification 15.4.11, shall show operation within ±10% of design flow.

A.11 methyl iodide penetration ≤ 1.0%

the applicable portions of Regulatory Guide 1.52, Revision 2 and ASME N510-1989

the applicable portions of Regulatory Guide 1.52, Revision 2 and ASTM D3803-1989 LA.6

penetration and system bypass ≤ 1.0% A.11

15.4.2 INSERVICE INSPECTION AND TESTING OF SAFETY CLASS COMPONENTS

~~Applicability~~

~~Applies to inservice inspection and testing programs for Safety Class Components.~~

~~Objectives~~

~~To provide assurance of the continuing integrity and operability of the safety class systems through the establishment of the appropriate programmatic controls.~~

A.12

Specifications

A. Steam Generator Tube Inspection Requirements

1. Tube Inspection

   Entry from the hot-leg side with examination from the point of entry completely around the U-bend to the top support of the cold-leg is considered a tube inspection.

2. Sample Selection and Testing

   Selection and testing of steam generator tubes shall be made on the following basis:

   (a) One steam generator of each unit may be selected for inspection during inservice inspection in accordance with the following requirements:

   1. The inservice inspection may be limited to one steam generator on an alternating sequence basis. This examination shall include at least 6% of the tubes if the results of the first or a prior inspection indicate that both generators are performing in a comparable manner.

      A.4 → 5.5.8-1

   2. When both steam generators are required to be examined by Table 15.4.2-1 and if the condition of the tubes in one generator is found to be more severe than in the other steam generator of a unit, the steam generator sampling sequence at the subsequent inservice inspection shall be modified to examine the steam generator with the more severe condition.

      A.4 → 5.5.8-1

   (b) The minimum sample size, inspection result classification and the associated required action shall be in conformance with the requirements specified in Table 15.4.2-1. The results of each sampling examination of a steam generator shall be classified into the following three categories:

Category C-1: Less than 5% of the total number of tubes examined are degraded but none are defective.

Category C-2: Between 5% and 10% of the total number of tubes examined are degraded, but none are defective or one tube to not more than 1% of the sample is defective.

Category C-3: More than 10% of the total number of tubes examined are degraded, but none are defective or more than 1% of the sample is defective.

In the first sample of a given steam generator during any inservice inspection, degraded tubes not beyond the plugging limit detected by the prior examinations in that steam generator shall be included in the above percentage calculations, only if these tubes are demonstrated to have a further wall penetration of greater than 10% of the nominal tube wall thickness.

(c) Tubes shall be selected for examination primarily from those areas of the tube bundle where service experience has shown the most severe tube degradation.

5.5.8-1 A.4

(d) In addition to the sample size specified in Table 15.4.2-1, the tubes examined in a given steam generator during the first examination of any inservice inspection shall include all non-plugged tubes in that steam generator that from prior examination were degraded.

A.9

~~(e) In addition to the sample size required in Specifications 15.4.2.A.2(a) through (d), all F* tubes shall be inspected in the F* region. The results of F* tube inspections are not to be used as a basis for additional inspections per Table 15.4.2-1.[1]~~

(f) During the second and third sample examinations of any inservice inspection, the tube inspection may be limited to those sections of the tube lengths where imperfections were detected during the prior examination.

3. Examination Method and Requirements

The examination method shall meet the intent of the requirements in ASME Section XI Appendix IV. This includes equipment, personnel and procedure requirements, certification and calibration along with records and reports. The actual technique may be the latest industry accepted technique, provided the flaw detection capability is as good or better than the technique endorsed by the code in effect per ~~Technical Specification 15.4.2.B.1.~~ This allows the use of improvements in inspection techniques that were not included in the code in effect. However, it means that word-for-word compliance with Appendix IV of ASME Section XI may not be possible.

~~1 This requirement applies only to the Westinghouse Model 44 steam generators in Unit 2. Following steam generator replacement in Unit 2, this requirement is null and void.~~

10 CFR 50, Section 50.55a(g) A.10

A.9

A.4

4. Inspection Intervals

(a) Inservice inspections shall not be more than 24 calendar months apart.

(b) The inservice inspections may be scheduled to be coincident with refueling outages or any plant shutdown, provided the inspection intervals of 15.4.2.A.4(a) are not exceeded.

5.5.8.d.1 A.4

(c) If two consecutive inservice inspections covering a time span of at least 12 months yield results that fall in C-1 category, the inspection frequency may be extended to 40 month intervals.

(d) If the results of the inservice inspection of steam generator tubing conducted in accordance with Table 15.4.2-1 requires that a third sample examination must be performed, and the results of this fall in category C-3, the inspection frequency shall be reduced to not more than 20 month intervals. The reduction shall apply until a subsequent inspection demonstrates that a third sample examination is not required.

A.4 5.5.8-1 3.4.13.d

A.4 5.5.8.b

(e) Unscheduled inspections shall be conducted in accordance with Specifications 15.4.2.A.2 on any steam generator with primary-to-secondary tube leakage exceeding Specification 15.3.1.D.4. All steam generators shall be inspected in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineered safeguards, or a main steam line or feedwater line break.

5. Acceptance Limits

(a) Definitions:

Imperfection is an exception to the dimension, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

Degradation means a service induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

Degraded Tube is a tube that contains imperfections caused by degradation greater than 20% of the nominal tube wall thickness.

A.4

Defect is an imperfection of such severity that it exceeds the minimum acceptable tube wall thickness of 50%. A tube containing a defect is defective.

Plugging Limit is the imperfection depth beyond which the tube must be removed from service or repaired, because the tube may become defective prior to the next scheduled inspection. The plugging limit is 40% of the nominal tube wall thickness.

A.9

~~F* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is 1.12 inches (including eddy current uncertainty). The F* distance is measured from the bottom of the upper roll transition of the repair roll toward the bottom of the tubesheet.[2]~~

~~F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded within the F* distance.[2]~~

6. Corrective Measures

A.9

All tubes that leak or have degradation exceeding the plugging limit shall be plugged or repaired by a process such as sleeving[1] ~~or classification as an F* tube[2]~~ prior to return to power from a refueling or inservice inspection condition. Sleeved tubes having sleeve degradation exceeding 40% of the nominal sleeve wall thickness shall be plugged.

7. Reports

Insert 5.5-2 M.10

(a) After each inservice examination, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission as soon as practicable.

(b) The complete results of the steam generator tube inservice inspection shall be included in the Annual Results and Data Report for the period in which the inspection was completed.

Reports shall include:

< See 5.6 >

1. Number and extent of tubes inspected.

2. Location and percent of all thickness penetration for each indication.

3. Identification of tubes plugged or repaired.

(c) Reports required by Table 15.4.2-1 - Steam Generator Tube Inspection shall provide the information required by Specification 15.4.2.A.7(b) and a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. The report shall be submitted to the Commission prior to resumption of plant operation.

1 Brazed joints shall not be employed. Tubes previously subject to explosive plugging shall not be sleeved.

~~2 Applicable only to the Westinghouse Model 44 steam generators in Unit 2. Following steam generator replacement in Unit 2, the definitions and F* repair option are null and void.~~

A.9

LB.5

A.4

B. In-Service ~~Inspection and~~ Testing of Safety Class Components Other than Steam Generator Tubes

< See 3.6.1 >

LB.5

1. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g) modified by Section 50.55a(b), except where specific written relief is granted by the NRC, pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

   a. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

2. Containment isolation valves will be tested in accordance with the Containment Leakage Rate Testing Program.

3. Inservice testing of ASME Code Class 1, 2, and 3 pumps, valves, ~~and snubbers~~ shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a.

   LB.5

   a. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

Insert 5.5-3

M.9

Basis

The steam generator tube inspection requirements are based on the guidance given in NRC Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." ASME Section XI Appendix IV is being used for defining the basic requirements or the inspection method. However, at the present time, changes and improvements in steam generator eddy current inspection are occurring faster than the code can be revised. Thus, in order to ensure that the best possible exam of the tubing and/or sleeves is being done, the technique utilized will, in general, be the latest industry-accepted technique. This means that complete word-for-word compliance with Appendix IV may not be possible. However, the basic requirements and intent will be met, to the extent practical.

Specification 15.4.2.B delineates programmatic requirements for establishing Inservice Inspection and Testing programs in accordance with the ASME Section XI Code and 10 CFR 50.55a requirements. The Code establishes criteria for system and component inspection and testing to ensure an appropriate level of reliability and detection of abnormal conditions. Failure to meet Code requirements is evaluated on an individual system or component bases to determine operability. Appropriate LCOs are entered if a system or component is determined to be inoperable.

As stated in 15.4.2.B.1, safety class components, other than the steam generator tubing, will be inspected in accordance with ASME Section XI. The code edition/addenda utilized for the inspection interval will be as defined in

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10 CFR 50. The same code is utilized for both Unit 1 and Unit 2. Safety-related components are classified as safety Class 1, 2, or 3. The code boundaries are defined based upon the following documents:

(a) Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants."

(b) American National Standard N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

(c) Point Beach Nuclear Plant Units 1 & 2 Final Safety Analysis Report.

Code classified components are tabulated showing each specific examination area and the examination requirements in an inspection interval long-term plan. This plan is completely revised for each ten-year inspection interval.

A sound roll expansion throughout the F* distance provides a tube to tubesheet interface that ensures the requirements of Regulatory Guide 1.121 are met regardless of the severity of any tube degradation below the F* distance. The F* distance of 1.12 inches is comprised of 0.88 inches of sound roll expansion that ensures tube integrity requirements are met plus 0.24 inches which allows for eddy current measurement uncertainty.

LA.5

TABLE 15.4.2-1
STEAM GENERATOR TUBE INSPECTION PER UNIT
POINT BEACH UNITS 1 & 2

1ST SAMPLE EXAMINATION			2ND SAMPLE EXAMINATION		3RD SAMPLE EXAMINATION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per Steam Generator (S.G.) S=3(N/n) % Where: N is the number of steam generators in the plant = 2 n is the number of steam generators inspected during an examination	C-1	Acceptable for continued service	N/A	N/A	N/A	N/A
	C-2	Plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator	C-1	Acceptable for continued service	N/A	N/A
			C-2	Plug or repair tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator	C-1	Acceptable for continued service
					C-2	Plug or repair tubes exceeding plug limit. Acceptable for continued service
					C-3	Perform action required under C-3 of 1st sample examination
			C-3	Perform action required under C-3 of 1st sample examination	N/A	N/A
	C-3	Inspect essentially all tubes in this S.G., plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in the other steam generator. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).	C-1 in other S.G.	Acceptable for continued service	N/A	N/A
			C-2 in other S.G.	Perform action required under C-2 of 2nd sample examination above	N/A	N/A
			C-3 in other S.G.	Inspect essentially all tubes in S.G. & plug or repair tubes exceeding the plugging limit. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).	N/A	N/A

15.4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage and structural integrity.

Objective

To verify that potential leakage from the containment and the pre-stressing tendon loads are maintained within acceptable values.

< See 3.6.1 >

Specification

I. Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

II. TENDON SURVEILLANCE

A. Object

In order to insure containment structural integrity, selected tendons shall be periodically inspected for symptoms of material deterioration or lift-off force reduction. The tendons for inspection shall be randomly but representatively selected from each group for each inspection; however, to develop a history and to correlate the observed data, one tendon from each group shall be kept unchanged after initial selection. Tendons selected for inspection will consist of five hoop tendons, three vertical tendons located approximately 120° apart, and three dome tendons, one from each of the three dome tendon groups.

B. Frequency

Tendon surveillance shall be conducted at five-year intervals in accordance with the following schedule:*

Unit	Year	Surveillance Required
1	1984	Physical
2	1984	Visual
1	1989	Visual
2	1989	Physical

LB.2

~~*Subsequent five-year interval inspections repeat this pattern.~~

C. Inspections

Tendon surveillance in accordance with 15.4.4.II.B shall consist of either a visual or physical inspection.

(1) Visual Inspection

a. Tendon anchorage assembly hardware of the randomly selected tendons shall be visually examined to the extent practicable without dismantling load bearing components of the anchorage. The immediate concrete area shall be checked visually for indications of abnormal material behavior.

(2) Physical Inspection

a. Tendons which are physically inspected shall first be visually inspected in accordance with C.(1).

b. All tendons which are physically inspected shall be subjected to a lift-off test to monitor their prestressing force.

(i) If the prestressing force of a selected tendon in a group lies above the predicted lower limit, the tendon is considered to be acceptable.

(ii) If the prestressing force of a selected tendon lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of the test tendon, shall be checked for their prestressing forces. If the prestressing forces for these tendons are above the predicted lower limit for the tendons, all three tendons shall be restored to the required level of integrity. A single deficiency shall be considered unique and acceptable. If the prestressing force of either of the adjacent tendons falls below the predicted lower limit of the tendon, additional lift-off testing should be done if necessary, so that the cause and extent of such occurrence can be determined and the condition shall be considered an abnormal degradation of the containment structure and the provisions of Specification 15.3.6.E are applicable.

(iii) If the prestressing force of the selected test tendon falls below 90% of the predicted lower limit, the tendon shall be completely detensioned and a determination shall be made as to the cause of the condition. Such a condition shall be considered an abnormal degradation of the containment structure and the provisions of Specification 15.3.6.E are applicable.

(iv) If the average of all measured tendon forces for each group (corrected for average condition) is found to be less than the minimum required prestress level at Anchorage location for that group, the condition should be considered as abnormal degradation of the containment structure and the provisions of 15.3.6.E are applicable. The average minimum design values adjusted for elastic losses are as follows:(6)

Hoop	134.5 ksi
Vertical	140.6 ksi
Dome	137.4 ksi

c. One randomly selected tendon from each group of tendons shall be subjected to complete detensioning in order to identify broken or damaged wires. During the retensioning of the detensioned tendon, simultaneous measurements of elongation and jacking force shall be made at a minimum of two levels of force between the required seating force and zero. During the detensioning and retensioning of the tendons tested, if the elongation corresponding to a specific load differs by more than 5% from that recorded during installation of the tendons, an investigation shall be made to ensure that such discrepancies are not related to wire failures or slippage of wires in anchorages.

d. A tendon wire shall be removed from the one tendon from each group which has been completely detensioned. The wire shall be inspected over its entire length to determine if evidence of corrosion or other deleterious effects are present. Tensile tests shall be made on three samples cut from each removed wire. The samples will be cut from the midsection and each end of the removed wire. Failure of the material to demonstrate the minimum required tensile strength of 240,000 psi shall be considered an abnormal condition of the containment structure and the engineering evaluation provisions of Specification 15.3.6.E.1 are applicable. If an acceptable justification for continued operation cannot be concluded from this evaluation, then the shutdown requirements of Specification 15.3.6.E.1 are applicable.

e. The sheathing filler grease will be sampled and inspected on each physically inspected tendon. The operability of the sheathing filler grease shall be verified by assuring:

1) There are no voids in the filler material in excess of 5% of net duct volume.

2) Complete grease coverage exists for the different parts of the Anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits specified by the manufacturer.

D. Reports

A final report documenting the results of each tendon surveillance will be prepared and maintained as a permanent plant record.

Abnormal conditions observed during testing will be evaluated to determine the effect of such conditions on containment structural integrity. This evaluation should be completed within 30 days of the identification of the condition. Any condition which is determined in this evaluation to have a significant adverse effect on containment structural integrity will be considered an abnormal degradation of the containment structure.

Any abnormal degradation of the containment structure identified during the engineering evaluation of abnormal conditions shall be reported to the Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

III. End Anchorage Concrete Surveillance

A. Specific locations for surveillance will be determined by information obtained from design calculations, as-built end anchorage concrete and prestressing records, observations of the end anchorage concrete during and after prestressing, and results of deformation measurements made during prestressing and the initial structural test.

B. The inspection intervals will be approximately one-half year and one year after the initial structural test and shall be chosen such that the inspection occurs during the warmest and coldest part of the year following the initial structural test.

LB.3

LB.3

C. The inspections made shall include:

(1) Visual inspection of the end anchorage concrete exterior surfaces.
(2) A determination of the temperatures of the liner plate area or containment interior surface in locations near the end anchorage concrete under surveillance.
(3) Measurement of concrete temperatures at specific end anchorage concrete surfaces being inspected.
(4) The mapping of the predominant visible concrete crack patterns.
(5) The measurement of the crack widths, by use of optical comparators or wire feeler gauges.
(6) The measurement of movements, if any, by use of demountable mechanical extensometers.

D. The measurements and observations shall be compared with those to which prestressed structures have been subjected in normal and abnormal load conditions and with those of preceding measurements and observations at the same location on the reactor containment.

E. The acceptance criteria shall be as follows:

If the inspections determine that the conditions are favorable in comparison with experience and predictions, the close inspections will be terminated by the last of the inspections stated in the schedule and a report will be prepared which documents the findings and recommends the schedule for future inspections, if any. If the inspections detect symptoms of greater than normal cracking or movements, an immediate investigation will be made to determine the cause.

IV. Liner Plate

A. The liner plate will be examined before the initial pressure test to determine the following:

(1) Locate areas which have inward deformations. The magnitude of the inward deformations will be measured and recorded. The areas will be permanently marked for future reference. The inward deformations will be measured between the angle stiffeners which are on 15-inch centers. The measurements will be accurate to ± .01 inch.

LB.4

(2) Try to locate areas having strain concentrations by visual examination paying particular attention to the condition of the liner surface. Record the location of any areas having strain concentrations.

B. Shortly after the initial pressure test and at about one year after initial start-up, reexamine the areas located in section (A). Measure and record inward deformations. Record observations pertaining to strain concentrations.

C. If the difference in the measured inward deformations exceeds 0.25 inch (for a particular location) and/or changes in strain concentration exist, then an investigation will be made. The investigation will determine the cause and any necessary corrective action.

D. The surveillance program will only be continued beyond the one year after initial start-up inspection if some corrective action was needed. If required, the frequency of inspection for a continued surveillance program will be determined shortly after the "one year after initial start-up inspection".

E. In addition to the preceding requirements, temperature readings will be obtained at the locations where inward deformations were measured. Temperature measurements will also be obtained on the outside of the containment building wall.

Basis

The containment is designed for an accident pressure of 60 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a temperature of about 105°F. With these initial conditions, the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 286°F.

Prior to initial operation, the containment was strength tested at 69 psig and then leak-tested. The design objective of this preoperational leakage rate test was established as 0.4% by weight per 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment,[2] which is equipped with independent leak-testable penetrations and contains channels over all containment liner welds, which were independently leak-tested during construction.

LB.4

Safety analyses have been performed on the basis of a leakage rate of 0.40% by weight per 24 hours at 60 psig. With this leakage rate and with minimum containment engineered safety systems for iodine removal in operation, i.e. one spray pump with sodium hydroxide addition, the public exposure would be well below 10 CFR 100 values in the event of the design basis accident.[3]

The safety analyses indicate that the containment leakage rates could be slightly in excess of 0.75% per day before a two-hour thyroid dose of 300R could be received at the site boundary.

The performance of periodic integrated leakage rate tests during plant life provide a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. These tests are performed in accordance with the Containment Leakage Rate Testing Program.

Periodic visual and physical inspection of the containment tendons is the method to be used to determine loss of load-carrying capability because of wire breakage or deterioration. The tendon surveillance program specified in 15.4.4.II is based on the recommendation of Regulatory Guide 1.35 Rev. 3. Containment tendon structural integrity was demonstrated for both units at the end of one, three and eight years following the initial containment structural integrity test.

The pre-stress lift-off test provides a direct measure of the load-carrying capability of the tendon. A deterioration of the corrosion preventive properties of the sheathing filler will be indicated by a change in the physical appearance of the filler. If the surveillance program indicates, by extensive wire breakage, tendon stress-strain relations, or other abnormal conditions, that the pre-stressing tendons are not behaving as expected, the abnormal conditions will be subjected to an engineering analysis and evaluation in accordance with Specification 15.4.4.II.D to determine whether the condition could result in a significant adverse impact on the containment structural integrity. The specified acceptance criteria are such as to alert attention to the situation well before the tendon load-carrying capability would deteriorate to a point that failure during a design basis accident might be possible. Thus, the cause of the incipient deterioration could be evaluated and corrective action studied without need to shut down the reactor. If the engineering evaluation determines that the abnormal condition could result in a significant adverse impact on the containment structural integrity, an abnormal degradation situation will be declared and a report submitted to the NRC in accordance with the specifications.

References

(1) FSAR Section 5.1.2.2
(2) FSAR Section 5.1.2
(3) FSAR Section 14.3.5
(4) FSAR Section 14.3.4
(5) Deleted
(6) FSAR pages 5.1-61 and 5.1-62

A.4

3. The proper operation of Emergency Lighting, including the automatic transfer switch for DC lights, will be demonstrated during each reactor shutdown for a major fuel reloading.
4. Each diesel generator shall be given an inspections following the manufacturer's recommendations for this class of stand-by service.
5. Operability of the diesel fuel oil system shall be verified monthly.

< See 3.8.1 >

6. A diesel fuel oil testing program shall be maintained to test both new fuel oil upon receipt and fuel oil stored in the fuel oil storage tanks which supply the emergency diesel generators ~~on a quarterly frequency~~ in accordance with applicable ASTM standards.

A.13

Insert 5.5-4 ← M.14

< See 3.8.1 >

The above tests will be considered satisfactory if all applicable equipment operates as designed.

A.13 → the program shall include sampling and testing requirements, and acceptance criteria

B. Safety-Related Station Batteries

These surveillance specifications are applicable to all four safety-related station batteries: D05, D06, D105, and D106; and the safety-related station swing battery D305.

< See 3.8.6 >

1. Every month the voltage of each cell (to the nearest 0.05 volt), the specific gravity and temperature of a pilot cell in each battery and each battery voltage shall be measured and recorded.
2. Every 3 months the specific gravity, the height of electrolyte, and the amount of water added, for each cell, and the temperature of every fifth cell, shall be measured and recorded.
3. At each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.
4. Each Safety-Related Station Battery shall be demonstrated OPERABLE:
   a. At least once per 18 months (SERVICE TEST) by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle.
   b. At least once per 60 months (PERFORMANCE TEST) by verifying that the battery capacity is at least 80% of the manufacturer's rating. This performance discharge test may be performed in lieu of the battery service test.

15.4.10 RADIOLOGICAL ENVIRONMENTAL MONITORING

Radioactive Environmental Monitoring Program (REMP) sampling and analyses requirements are addressed in the Radiological Effluent and Materials Control and Accountability Program Manual. The requirement for the REMP is specified in Section 15.7.8.3.

A.1

15.4.11 CONTROL ROOM EMERGENCY FILTRATION

< See 3.7.10 >

Applicability

Applies to periodic testing requirements of the control room emergency filtration equipment.

Objective

To verify the operability of the control room emergency filtration and its ability to remove radioactive contaminants when required.

LA.6

Specification

1. ~~At least once per year~~ the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at ~~design flow rate~~ 4950 cfm ± 10%. M.3
2. The control room emergency filtration automatic initiation shall be demonstrated once per year.
3. The control room emergency filtration shall be operated at least 10 hours every month.

< See 3.7.10 >

4. Components of the control room emergency filtration shall be tested as follows:
   a. HEPA filters and charcoal adsorbers shall be tested and analyzed ~~at least once per year, or after 720 hours of operation since the previous test, and following significant painting, fire or chemical release in the control room during filtration operation.~~
   b. Cold DOP testing of the HEPA filter bank shall be performed ~~after each complete or partial replacement of HEPA filters, or after any structural maintenance on the filter housing.~~ DOP testing shall be at design velocity ± 20%.

M.11 → 4950 cfm ± 10%

LA.6 → in accordance with the applicable portions of Regulatory Guide 1.52, Revision 2, ASME N510-1989 and ASTM D3803-1989

Insert 5.5-5 ← M.12

A.4

c. Halogenated hydrocarbon testing of the charcoal adsorber bank shall be performed after each complete or partial replacement of charcoal adsorbers or after any structural maintenance of the adsorber housing. Halogenated hydrocarbon testing shall be at design velocity ± 20%.

4950 cfm ± 10% ← M.11

d. Laboratory sample analysis of in-place charcoal adsorbent shall be performed at least once per year for standby service or after every 720 hours of system operation and, as a minimum, shall be conducted at velocities within 20% of design, 1.75 mg/m³ inlet iodide concentration, 95% relative humidity and 30°C (86°F).

e. Fans shall be tested at least once per year or after 720 hours of operation since the previous test, and following fan maintenance or repair.

< See 3.7.10 >

LA.6 → in accordance with the applicable portions of Regulatory Guide 1.52, Revision 2, ASME N510-1989 and ASTM D3803-1989

Basis

The control room emergency filtration system is designed to filter the control room atmosphere and makeup air to the control room during control room isolation conditions. The control room emergency filtration is normally isolated and not in operation and testing more frequently than that specified is not required to insure operability or performance. If the efficiencies of HEPA and charcoal adsorbers are as specified, the resulting control room doses during accident conditions will be less than allowable levels in Criterion 19 of Appendix A to 10 CFR 50. The charcoal adsorbent laboratory sample analysis is performed in accordance with ASTM D3803-89, "Standard Test Method for Nuclear-Grade Activated Carbon."

A.4

A.4

TABLE 15.4.16-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES[a][b]

System	Check Valve No.
Residual Heat Removal	
Line 1	853C
	853A
Line 2	853D
	853B
Safety Injection	
Loop A Cold Leg	867A
	845A
	845E
Loop B Cold Leg	867B
	845B
	845F
R.V. Hot Leg Line A	845C
R.V. Hot Leg Line B	845D

< See 3.4.14 >

A.4

~~Footnotes~~:

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Minimum differential test pressure shall not be less than 150 psid.

A program shall be established to verify the leakage from each RCS PIV is within the limits specified below, as ordered by Event V, issued April 20, 1981.

A.8

15.6.8 PLANT OPERATING PROCEDURES (Continued)

15.6.8.4 The following programs shall be established, implemented, and maintained.

A. Post-Accident Sampling* A.7

A program** which will ensure the capability to obtain and analyze reactor coolant, containment atmosphere, and in-plant gaseous effluent samples under accident conditions. The program shall include the following:

(i) Training of personnel; and

(ii) Procedures of sampling and analysis.

(iii) Provisions for maintenance of sampling and analysis equipment.

radioactive gases, and particulates in M.1

Insert 5.5-6, Primary Coolant Sources Outside Containment M.2

Insert 5.5-7, Component Cyclic or Transient Limit M.7

Insert 5.5-8, Reactor Coolant Pump Flywheel Inspection Program M.8

A.7

~~* Post-Accident Coolant Sampling and Post-Accident Containment Atmospheric Sampling Systems.~~

~~** It is acceptable if the licensee maintains details of the program in plant operation manuals.~~

15.6.12 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995.

B. The peak design containment internal accident pressure, P_a, is 60 psig.

C. The maximum allowable primary containment leakage rate, L_a at P_a, shall be 0.4% of containment air weight per day.

D. Leakage rate acceptance criteria are:

1. The containment leakage rate acceptance criterion is $\leq 1.0\ L_a$.

2. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.6\ L_a$ for the combined Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

SR 3.0.2

E. The provisions of ~~Specification 15.4.0.2~~ do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

F. The provisions of ~~Specification 15.4.0.3~~ are applicable to the Containment Leakage Rate Testing Program.

SR 3.0.3

Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.
2) For each door seal, leakage rate is $\leq 0.02\ L_a$ at $\geq P_a$ when tested at a differential pressure of $\geq$ 10 inches of Hg.

M.16

A.2

15.7 RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS (RETS)

RETS do not directly expand the responsibilities of the licensed operators of the Point Beach Nuclear Plant Units 1 and 2, and the material contained in this section of these Technical Specifications will not be the subject of SRO/RO licensing examinations.

15.7.1 DEFINITIONS

The definitions for frequently used terminology in these RETS are stated below. These definitions are supplemental to those definitions provided in Section 15.1.

A. Members of the Public

Members of the public include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

B. Offsite Dose Calculation Manual (ODCM)

The Offsite Dose Calculation Manual contains the methodology for the determination of gaseous and liquid effluent monitoring alarm or trip setpoints, the methodology for determining compliance with release limits, and the methodology used in the calculation of offsite doses due to radioactive gaseous and liquid effluents.

C. Radioactive Waste Handling

1. Process Control Program (PCP)

The Process Control Program contains the methodologies used to ensure that the processing and packaging of solid radioactive waste will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, and all other Federal and State regulations governing the disposal of the radioactive waste.

2. Solidification

The conversion of liquid wastes into a form that meets shipping and burial ground requirements.

D. Radiological Effluent and Materials Control and Accountability Program

The Radiological Effluent and Materials Control and Accountability Program (REMCAP) specified in 15.7.8.3 shall ensure that all radioactive effluent (materials released via liquid and atmospheric pathways as well as solid materials) from the Point Beach Nuclear Plant are controlled and accounted for in a manner which complies with the applicable regulations and shall ensure that control and accountability for atmospheric and liquid releases are supplemented by environmental sampling and dose calculations. The four major components of the REMCAP are the ODCM, the PCP, the Radiological Environmental Monitoring Program (REMP), and the Radiological Effluent Control Program (RECP). Other supporting guidance, procedures, manuals, or programs may be included or referenced but may not be subject to the program controls specified in 15.7.8.6 and 15.7.8.7.

LA.1

15.7.3 RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION OPERABILITY REQUIREMENTS

Operability requirements have been removed from the Technical Specifications and moved to the REMCAP Manual.

A.1

15.7.4 RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Surveillance requirements have been removed from the Technical Specifications and moved to the REMCAP Manual.

A.1

A.1

A.4

15.7.5 ~~RADIOACTIVE EFFLUENT RELEASE AND~~ EXPLOSIVE GAS CONCENTRATION LIMITS

Applicability

Applies to the explosive gas concentration in the radioactive gas decay tanks. Radioactive effluent release limits have been removed from Technical Specifications and placed in the REMCAP Manual.

Objective

To ensure explosive gas concentrations do not exceed the limits of Specification 15.7.5.A.

A.12

Specifications

LA.7

A. Explosive Gas Mixture

The concentration of oxygen in the on-service gas decay tank shall be limited to less than or equal to 4% by volume.

1. If the concentration of oxygen in the on-service gas decay tank is greater than 4% by volume, immediately suspend all additions of waste gases to the on-service gas decay tank.
2. Reduce the oxygen concentration to less than 4% oxygen by volume as soon as possible. If the on-service gas decay tank is at or near capacity and the tank must be isolated to permit the required decay time to conform with release limits, it will not be possible to immediately reduce the oxygen concentration. In this case, the tank will be isolated and the oxygen concentration reduced as soon as the gas decay requirements are satisfied.

Insert 5.5-9 ← M.13

~~15.7.6~~ ~~RADIOACTIVE EFFLUENT SAMPLING AND ANALYSIS REQUIREMENTS~~

~~Sampling and analysis requirements removed from the Technical Specifications and placed in the REMCAP Manual.~~

A.1

15.7.7 OPERATIONAL ENVIRONMENTAL MONITORING PROGRAM

The description of the environmental monitoring program has been removed from Technical Specifications and placed in the REMCAP Manual.

A.1

15.7.8 ADMINISTRATIVE CONTROLS

15.7.8.1 DELETED

LA.4

15.7.8.2 Audits

A. An audit of the activities encompassed by the Radioactive Effluent and Materials Control and Accountability Program (REMCAP), [the Offsite Dose Calculation Manual (ODCM), the Radiological Effluent Control Program (RECP), the Radiological Environmental Monitoring Program (REMP), and the Process Control Program (PCP)] and its implementing procedures shall be performed utilizing either offsite licensee personnel or a consulting firm.

B. The results of the audit above shall be transmitted to the Chief Nuclear Officer and the Chairman of the Offsite Review Committee.

15.7.8.3 Plant Operating Procedures and Programs

A.5

The Radioactive Effluent ~~and Materials~~ Control ~~and Accountability~~ Program ~~(REMCAP)~~ shall be established, implemented, and maintained in accordance with the provisions of Technical Specification 15.6.8. REMCAP shall assure that radioactive effluent and waste material from PBNP complies with applicable Federal, State, and burial ground regulations while keeping all exposures to members of the public as low as reasonably achievable (ALARA). This program shall conform to and implement the requirements of PBNP GDC 70 and of 10 CFR 50.34a and 10 CFR 50.36a for the control of radioactive effluents while maintaining doses from these effluents ALARA, shall implement the requirements of 10 CFR 50.34a and General Design Criterion 60 of Appendix A to 10 CFR 50 to suitably control the release and the processing of waste materials, shall conform to the guidance of Appendix I to 10 CFR 50 and PBNP GDC 17 for the assessment of radioactivity in the environs of PBNP and shall include remedial actions to be taken whenever the program limits are exceeded. REMCAP shall be implemented and maintained under the procedures ~~and methodologies specified in the four (4) manuals/programs listed below and supported, as required, by other procedures.~~ Effluent and environmental monitoring shall be addressed in the Quality Assurance Program.

A.5

The Radiological Effluent Control Program shall be contained in the ODCM,

by

A.5

< See 5.4.1 >

A.3 Replace with Insert 5.5-10

a. Environmental Manual (EM) - In order to assure conformance with PBNP GDC 17 and with the guidance of Appendix I to 10 CFR 50, this manual shall contain the Radiological Environmental Monitoring Program (REMP) methodology, parameters, and administrative functions required for monitoring, sampling, analyzing, and reporting of radiation levels and radionuclide concentrations in the environment around PBNP. ~~The REMP shall require an annual milk survey and shall require that radioanalyses to be conducted by a laboratory participating in an Interlaboratory Comparison Program.~~

LA.2

A.5

b. Radiological Effluent Control Program ~~Manual (RECM) - The RECM~~ shall contain the ~~Radiological Effluent Control Program (RECP)~~ parameters and methodologies needed in order to assure conformance with 10 CFR 50.36a for the control of effluents and for maintaining doses to the public ALARA and for compliance with 10 CFR 20.1302. The following ~~RECP~~ items are contained in the RECM: Radiological Effluent Control Program

A.5

A.5

liquid and gaseous

A.4 Monitoring

liquid and gaseous

1) radioactive effluent monitoring instrumentation operability, with actions to be taken if operability requirements are not met, and surveillance requirements.

2) radioactive effluent sampling and analyses requirements;

3) limitations on the concentrations of radioactive materials in liquid effluents to the unrestricted area to 10 times the concentration value in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402; A.5 and with the methodology and parameters in the ODCM

4) determination of the cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with ODCM methodology and parameters at least every 31 days;

Limitations on the functional capability and use of

M.4

5) assurance that the appropriate portions of the liquid and gaseous effluent treatment system ~~are used~~ whenever the projected doses in a period of 31 days would exceed 2 percent of the 10 CFR 50, Appendix I dose guidelines;

6) limitations to the release rate of radioactive effluent to the atmosphere in order to assure that the following dose rates would not be exceeded at or beyond the side boundary:

   a. 500 mrem/yr., total body, from noble gases,

   b. 3000 mrem/yr., skin, from noble gases,

   c. 1500 mrem/yr., any organ, from I-131, I-133, H-3, and all radioactive particulates with a half-life greater than 8 days;

7) limitations on the annual and quarterly doses to a member of the public at or beyond the site boundary from I-131, I-133, H-3, and radioactive particulates with a half life

M.5

Replace with Insert 5.5-11

> 8 days in effluents released to the atmosphere which conform to 10 CFR 50, Appendix I, dose guidelines, and

M.5 Replace with Insert 5.5-12

8) administrative functions and reporting requirements.

c. Offsite Dose Calculation Manual (ODCM) - The ODCM shall specify that the annual doses from PBNP effluent shall conform to the limits of Appendix I to 10 CFR 50 and 40 CFR 190, shall contain the parameters and methodology used to calculate the doses to members of the public from PBNP liquid and atmospheric releases ~~in order to demonstrate compliance with these dose limits~~, and shall contain the methodology to calculate setpoints ~~to ensure that the effluent radionuclide concentrations in unrestricted areas do not exceed the limits specified in the RECP.~~

A.5

The ODCM

A.5

~~d. Process Control program (PCP) - The PCP shall provide the methodologies for assuring that the processing and packaging of solid radioactive wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61 and 71, and all other applicable Federal and State regulations governing the disposal of the radioactive waste.~~

LB.1

The Radioactive Effluents Control Program shall include limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I.

M.6

15.7.8.5 Major Change to Radioactive Liquid, Gaseous and Solid Waste Treatment Systems

Licensee initiated major changes to the radioactive waste treatment systems (liquid, gaseous, and solid) shall be reported to the U.S. Nuclear Regulatory Commission with the annual update to the FSAR for the period in which the major change was complete. The discussion of each change shall include:

LA.3

A. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;

B. Information necessary to support the reason for the change;

C. A description of the equipment, components and processes involved and the interfaces with other plant systems;

D. An evaluation of the change, which shows how the predicted releases of radioactive materials in liquid effluents and gaseous effluents and/or quantity of solid waste will differ from those previously predicted in the license application and amendments thereto;

E. An evaluation of the change, which shows the expected maximum exposures to an individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

F. An estimate of the exposure to plant operating personnel as a result of the change.

15.7.8.6 Record Retention

Records of review performed for changes made to the REMCAP Manual and to the following REMCAP components; the EM, RECM, ODCM, and PCP; shall be kept for the duration of the operating licenses of Units 1 and 2 of the Point Beach Nuclear Plant.

< See 5.6 >

15.7.8.7 Revisions

A. Process Control Program

Revisions to the PCP shall be documented and records of reviews performed for the revision shall be retained as required by 15.7.8.6. The documentation shall contain:

1. Information sufficient to support the change together with the appropriate analyses or evaluations justifying the change(s), and
2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

LB.1

LB.1 ~~3. Revisions shall become effective after the approval of the Plant Manager.~~

B. Revisions to the ~~EM~~ ODCM, ~~RECM and REMCAP~~ A.5

1. Revisions to the ~~EM~~ ODCM, ~~RECM and REMCAP~~ shall be documented and reviews performed of the revision shall be retained ~~as required in~~

A.5 ~~15.7.8.6~~ The documentation shall contain:

a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the revision, and

b. A determination that the change will maintain the levels of radioactive effluent control required pursuant to 10 CFR 20.1302, 10 CFR 50.36a, Appendix I to 10 CFR 50, and 40 CFR 190.

2. Shall become effective after the approval of the Plant Manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire manual either as part of, or concurrent with, the Annual Monitoring Report for the period of the report in which the revision 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. Each copy shall indicate the date (e.g., month/year) the revision was implemented.

A.5 ODCM

A.6 ~~4. In addition to items 1-3 above, all changes regarding explosive gas must be made via the 50.59 process.~~

Insert 5.5-13, Technical Specification (TS) Bases Control Program M.15

Insert 5.5-14, Safety Function Determination Program (SFDP) M.15

Insert 5.5-1

A program shall be established to implement the following required testing of the Control Room Emergency Filtration system in accordance with the applicable portions of Regulatory Guide 1.52, Revision 2 and ASME N510-1989.

Insert 5.5-2

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test frequencies.

Insert 5.5-3

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

Insert 5.5-4

The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. a clear and bright appearance with proper color;

b. Within 31 days of addition of the new fuel oil to storage tanks verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil;

c. Total particulate concentration of the fuel oil is $\leq$ 10 mg/l when tested every 92 days in accordance with the applicable ASTM standards; and

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

Insert 5.5-5

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

Insert 5.5-6

Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Containment Spray, Safety Injection (High Head) and Safety Injection (Low Head) systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

Insert 5.5-7

Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section 4.1, cyclic and transient occurrences to ensure that components are maintained within design limits.

Insert 5.5-8

Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position c.4.b(1)and c.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at approximately 10 year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI.

Insert 5.5-9

The Explosive Gas Monitoring Program shall be established, implemented, and maintained.

This program shall provide controls for potentially explosive gas mixtures contained in the on-service Gas Decay Tank.

This program shall include a limit for oxygen concentration in the on-service Gas Decay Tank and a surveillance program to ensure the limit is maintained. This limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion.)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Monitoring Program surveillance test frequencies.

Insert 5.5-10

The ODCM shall contain the methodology and parameters used in the conduct of the radiological environmental monitoring program.

The ODCM shall also contain the radiological effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Monitoring Report required by Specification 5.6.2.

Insert 5.5-11

Radiological Effluents Control Program shall include limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the facility to unrestricted areas, conforming to 10 CFR 50, Appendix I.

Insert 5.5-12

Radiological Effluents Control Program shall include limitations on the annual dose or dose commitment to a 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.

Insert 5.5-13

5.5.13 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 involve 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 involves an unreviewed safety question as defined in 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 5.5.13b 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).

Insert 5.5-14

5.5.14 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

Justification For Deviations - NUREG-1431 Section 5.05

15-Mar-00

JFD Number	JFD Text	
01	NUREG-1431, specification 5.5.1 discusses the information required to be included in the "Annual Radiological Environmental Operating and Radioactive Effluent Release Reports." The Point Beach CTS requires one report be submitted to the NRC containing the same information. ITS specification 5.5.1 will retain the requirements of the CTS and require one report, the "Annual Monitoring Report."	
	ITS:	**NUREG:**
	SPEC 5.05.01.B	SPEC 5.05.01 01B
	SPEC 5.05.01.C.03	SPEC 5.05.01 02C
02	The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	N/A	SPEC 5.05.11.e
	SPEC 5.05.01.B	SPEC 5.05.01 01B
	SPEC 5.05.02	SPEC 5.05.02
	SPEC 5.05.05	SPEC 5.05.05
	SPEC 5.05.10	SPEC 5.05.11
	SPEC 5.05.10.a	SPEC 5.05.11.a
	SPEC 5.05.10.b	SPEC 5.05.11.b
	SPEC 5.05.10.c	SPEC 5.05.11.c
	SPEC 5.05.10.d	SPEC 5.05.11.d
	SPEC 5.05.11	SPEC 5.05.12
	SPEC 5.05.11.A	SPEC 5.05.12.A
03	NUREG-1431, specification 5.5.1, Offsite Dose Calculation Manual, has been renumbered to be consistent with the ITS format and for clarity.	
	ITS:	**NUREG:**
	SPEC 5.05.01.C	SPEC 5.05.01 02
	SPEC 5.05.01.C.01	SPEC 5.05.01 02A
	SPEC 5.05.01.C.01.i	SPEC 5.05.01 02A.1
	SPEC 5.05.01.C.01.ii	SPEC 5.05.01 02A.2
	SPEC 5.05.01.C.02	SPEC 5.05.01 02B
	SPEC 5.05.01.C.03	SPEC 5.05.01 02C

JFD Number	JFD Text	
04	NUREG-1431, Inservice Testing (IST) Program, has been modified to state that the IST Program provides controls for ASME Code Class 1, 2 and 3 "pumps and valves," in place of "components." 10 CFR 50.55.a(f) provides the regulatory requirements for an IST Program and specifies that ASME Code Class 1, 2 and 3 pumps and valves are the only components covered by an IST Program. 10 CFR 50.55a(g) provides regulatory requirements for an Inservice Inspection (ISI) Program and specifies that ASME Code Class 1, 2 and 3 components are covered by the ISI Program, and that pumps and valves are covered by the IST Program per 10 CFR 50.55.a(f). NUREG-1431 does not include ISI Program requirements as these requirements have been relocated to plant specific documents. Therefore, the components of the IST Program have been clarified to only include pumps and valves.	
	ITS:	**NUREG:**
	SPEC 5.05.07	SPEC 5.05.08
05	NUREG-1431, specification 5.5.6, Pre-Stressed Concrete Containment Tendon Surveillance Program, is not being retained in ITS. 10 CFR 50.55.a requires facilities to adopt the ASME Section XI, Subsection IWE and IWL programs by September 2001. Point Beach will adopt these Section XI programs prior to ITS implementation. Therefore, the Pre-Stressed Concrete Containment Tendon Surveillance Program will be duplicative of the requirements specified by ASME Section XI, as endorsed and required under 10 CFR 50.55a. Inclusion of these requirements via reference into 10 CFR 50.55a makes these requirement applicable to Point Beach without the need to duplicate these requirements in the Technical Specifications. Succeeding ITS program requirements have been renumbered accordingly.	
	ITS:	**NUREG:**
	N/A	SPEC 5.05.06
	SPEC 5.05.06	SPEC 5.05.07
	SPEC 5.05.07	SPEC 5.05.08
	SPEC 5.05.08	SPEC 5.05.09
	SPEC 5.05.09	SPEC 5.05.10
	SPEC 5.05.10	SPEC 5.05.11
	SPEC 5.05.11	SPEC 5.05.12
	SPEC 5.05.12	SPEC 5.05.13
	SPEC 5.05.13	SPEC 5.05.14
	SPEC 5.05.13.D	SPEC 5.05.14.D
	SPEC 5.05.14	SPEC 5.05.15

Justification For Deviations - NUREG-1431 Section 5.05

15-Mar-00

JFD Number	JFD Text
06	Point Beach current licensing basis steam generator tube surveillance requirements have been inserted, as indicated in NUREG-1431, specification 5.5.9.

ITS:	NUREG:
SPEC 5.05.08 T 5.05.08-01	SPEC 5.05.09
SPEC 5.05.08.a	SPEC 5.05.09
SPEC 5.05.08.a.01	SPEC 5.05.09
SPEC 5.05.08.a.02	SPEC 5.05.09
SPEC 5.05.08.a.03	SPEC 5.05.09
SPEC 5.05.08.a.04	SPEC 5.05.09
SPEC 5.05.08.a.05	SPEC 5.05.09
SPEC 5.05.08.a.06	SPEC 5.05.09
SPEC 5.05.08.b	SPEC 5.05.09
SPEC 5.05.08.b.01	SPEC 5.05.09
SPEC 5.05.08.b.01.i	SPEC 5.05.09
SPEC 5.05.08.b.01.ii	SPEC 5.05.09
SPEC 5.05.08.b.02	SPEC 5.05.09
SPEC 5.05.08.b.02.i	SPEC 5.05.09
SPEC 5.05.08.b.02.ii	SPEC 5.05.09
SPEC 5.05.08.b.02.iii	SPEC 5.05.09
SPEC 5.05.08.b.03	SPEC 5.05.09
SPEC 5.05.08.b.04	SPEC 5.05.09
SPEC 5.05.08.b.05	SPEC 5.05.09
SPEC 5.05.08.c	SPEC 5.05.09
SPEC 5.05.08.d	SPEC 5.05.09
SPEC 5.05.08.d.01	SPEC 5.05.09
SPEC 5.05.08.d.02	SPEC 5.05.09
SPEC 5.05.08.d.03	SPEC 5.05.09
SPEC 5.05.08.d.04	SPEC 5.05.09
SPEC 5.05.08.d.05	SPEC 5.05.09

JFD Number	JFD Text	
	SPEC 5.05.08.e	SPEC 5.05.09
07	NUREG-1431, specification 5.5.10, Secondary Water Chemistry Program, has been modified by the deletion of the requirement for the program to provide controls to monitor secondary water chemistry to inhibit low pressure turbine disc stress corrosion cracking. This requirement is not a part of the Point Beach current licensing basis.	
	ITS:	**NUREG:**
	SPEC 5.05.09	SPEC 5.05.10
08	NUREG-1431, specification 5.5.11, Ventilation Filter Testing Program (VFTP), has been modified. References to "Engineered Safety Feature (ESF) filter ventilation systems" have been replaced with "Control Room Emergency Filtration System (F-16)," because this is the only filter ventilation system at Point Beach maintained in accordance with the applicable portions of Regulatory Guide 1.52 and ASME N510-1989. This distinction has also resulted in reorganization of the specification.	
	ITS:	**NUREG:**
	SPEC 5.05.10	SPEC 5.05.11
	SPEC 5.05.10.a	SPEC 5.05.11.a
	SPEC 5.05.10.b	SPEC 5.05.11.b
	SPEC 5.05.10.c	SPEC 5.05.11.c
	SPEC 5.05.10.d	SPEC 5.05.11.d
09	NUREG-1431, specification 5.5.11.d requirement to demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers, has been modified by the deletion of the requirement to include "prefilters." The Point Beach current licensing basis (CTS 15.4.11.1) does not require "prefilters" to be included in the overall pressure drop surveillance.	
	ITS:	**NUREG:**
	SPEC 5.05.10.d	SPEC 5.05.11.d

JFD Number	JFD Text	
10	The liquid radwaste requirements of NUREG-1431, specification 5.5.12, Explosive Gas and Storage Tank Radioactivity Monitoring Program, have not been retained in ITS 5.5.11. This reflects the current licensing basis (CTS 15.7.5). The requirements associated with Radiological Effluent Technical Specifications were relocated from the Point Beach CTS to the Radiological Effluents and Materials Control and Accountability Program (REMCAP) using the guidance of Generic Letters 89-01 and 95-10. Removal of these requirements were approved in the NRC Safety Evaluation for Amendments 184 (Unit 1) and 188 (Unit 2). The specification has been reformatted as appropriate due to the deletion of the radwaste related requirements.	
	ITS:	**NUREG:**
	N/A	SPEC 5.05.12.B
		SPEC 5.05.12.C
	SPEC 5.05.11	SPEC 5.05.12
11	NUREG-1431, specification 5.5.12.a has been modified by the deletion of the requirement for a hydrogen concentration limit in the Gas Decay Tank. Point Beach current licensing basis (CTS 15.7.5) only requires a limit on the concentration of oxygen in the Gas Decay Tank.	
	ITS:	**NUREG:**
	SPEC 5.05.11.A	SPEC 5.05.12.A
12	NUREG-1431, specification 5.5.13, Diesel Fuel Oil Testing Program, has been modified. The requirement to verify the other properties of ASTM 2D fuel oil within 31 days "following" addition of new fuel oil to the storage tanks, has been modified to within 31 days "of" addition of new fuel oil to the storage tanks. This change is necessary due to the configuration of the diesel fuel oil storage system at Point Beach. The Point Beach diesel fuel oil system includes a fill tank in addition to the storage tanks. The new fuel oil is received in the fill tank, where the new fuel oil is sampled and stored until the test results are obtained. Once satisfactory test results are obtained, the fuel oil is transferred to the storage tanks. Therefore, the requirement to verify "the other properties" of ASTM 2D fuel within 31 days "of" addition to the storage tanks is necessary to prevent redundant testing of new fuel oil (tested upon receipt in the fill tank), upon transfer to the storage tanks.	
	ITS:	**NUREG:**
	SPEC 5.05.12.B	SPEC 5.05.13.B
13	NUREG-1431, specification 5.5.13, Diesel Fuel Oil Testing Program, has been revised from requiring the total particulate concentration of the fuel oil to be tested "every 31 days" to "every 92 days", consistent with CTS 15.4.6.A.6.	
	ITS:	**NUREG:**
	SPEC 5.05.12.C	SPEC 5.05.13.C

JFD Number	JFD Text	
14	NUREG-1431, specification 5.5.13, Diesel Fuel Oil Testing Program, has been revised from requiring the total particulate concentration of the fuel oil to be tested in accordance with "ASTM D-2276, Method A-2 or A-3" to "the applicable ASTM standards." This change is necessary because ASTM D-2276 provides testing requirements for a field monitor system. Point Beach does not utilize a field monitor, but rather uses the laboratory analysis method. Specifying the total particulate concentration of the fuel will be tested in accordance with the applicable ASTM standards is consistent with CTS 15.4.6.A.6.	
	ITS:	**NUREG:**
	SPEC 5.05.12.C	SPEC 5.05.13.C
15	NUREG-1431 has been modified by the addition of a Containment Leakage Rate Testing Program, based on the current licensing basis. This additional program is based on CTS 15.6.12, approved exemptions to 10 CFR 50, Appendix J, and the requirements of 10 CFR 50 Appendix J, Option B.	
	ITS:	**NUREG:**
	SPEC 5.05.15	N/A
	SPEC 5.05.15.A	N/A
	SPEC 5.05.15.B	N/A
	SPEC 5.05.15.C	N/A
	SPEC 5.05.15.D	N/A
	SPEC 5.05.15.D.01	N/A
	SPEC 5.05.15.D.02	N/A
	SPEC 5.05.15.D.03	N/A
	SPEC 5.05.15.D.03.a	N/A
	SPEC 5.05.15.D.03.b	N/A
	SPEC 5.05.15.E	N/A
	SPEC 5.05.15.F	N/A

15-Mar-00

JFD Number	JFD Text	
16	NUREG-1431 has been modified by the addition of a Reactor Coolant System Pressure Isolation Valve Leakage Program, based on CTS 15.4.16.	
	ITS:	**NUREG:**
	SPEC 5.05.16	N/A
	SPEC 5.05.16.01	N/A
	SPEC 5.05.16.02	N/A
	SPEC 5.05.16.03	N/A
	SPEC 5.05.16.04	N/A

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 <u>Offsite Dose Calculation Manual (ODCM)</u>

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~~ 1 Monitoring ~~Operating, and Radioactive Effluent Release~~ Report~~s~~ required by Specification [5.6.2] 2 5.6.2 ~~and Specification [5.6.3]~~. 1

3 c. Licensee initiated changes to the ODCM:

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

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

ii. 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;

Approved
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Plant Manager

3

2. b. Shall become effective after the approval of the ~~[Plant Superintendent]~~; 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 Annual Monitoring ~~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

1

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

2

Containment Spray, Safety Injection (High Head) and Safety Injection (Low Head) systems.

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [Recirculation Spray, Safety Injection, Chemical and Volume Control, gas stripper, and Hydrogen Recombiner]. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

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

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

be taken whenever the program limits are exceeded. T he program shall include the following elements:

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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;

ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ~~10 CFR 20, Appendix B, Table 2, Column 2;~~

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 from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

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;

Approved TSTF-258, R4

from the site

at or

Insert 5.0-06

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;~~

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

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

Approved TSTF-258

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 from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

, beyond the site boundary,

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

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

2 4.1

5.5.5 Component Cyclic or Transient Limit

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

5

~~5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program~~

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.~~

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

5 6

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

Insert 5.0-04

Approved TSTF-237

Approved TSTF-118

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test frequencies.

5.5 Programs and Manuals

5.5.8 Inservice Testing Program

5 → 7

Approved TSTF-279

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components ~~including applicable supports.~~ The program shall include the following:

pumps and valves

4

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

6 → Replace with Insert 5.0-14

5.5.9 Steam Generator (SG) Tube Surveillance Program

5 → 8

~~Reviewer's Note: The Licensee's current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~

5.5 Programs and Manuals

5.5.$\boxed{10}$ 9 5 7

Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include:

7

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

2

5.5.$\boxed{11}$ 10 5

Ventilation Filter Testing Program (VFTP)

Regulatory Guide 1.52, Revision 2

A program shall be established to implement the following required testing of ~~Engineered Safety Feature (ESF) filter ventilation systems~~ at the frequencies specified in ~~[Regulatory Guide]~~, and in accordance with ~~[Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1]~~.

the Control Room Emergency Filtration System (F-16)

8

ANSI N510-1980.

the Control Room Emergency Filtration System (F-16) 8

a. Demonstrate for ~~each of the ESF systems~~ that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ~~< [0.05]~~% when tested in accordance with ~~[Regulatory Guide 1.52, Revision 2, and ASME N510-1989]~~ at ~~the system flowrate specified below [± 10%]~~.

≤ 1.0

2

ANSI N510-1980, Section 10,

2

~~ESF Ventilation System~~	~~Flowrate~~
~~[ ]~~	~~[ ]~~

a system flowrate of 4950 cfm ± 10%

8

5.5 Programs and Manuals

5.5.~~11~~ 10 Ventilation Filter Testing Program (VFTP) (continued)

5 → 10

b. Demonstrate for ~~each of the ESF systems~~ the Control Room Emergency Filtration System (F-16) that an inplace test of the charcoal adsorber shows a penetration and system bypass ~~< [0.05]~~ ≤ 1.0% when tested in accordance with ~~[Regulatory Guide 1.52, Revision 2, and ASME N510-1989]~~ ANSI N510-1980, Section 12. at ~~the system flowrate specified below [± 10%]~~ a system flowrate of 4950 cfm ± 10%.

8 2

~~ESF Ventilation System~~	~~Flowrate~~
~~[ ]~~	~~[ ]~~

c. Demonstrate for ~~each of the ESF systems~~ the Control Room Emergency Filtration System (F-16) that a laboratory test of a sample of the charcoal adsorber, when obtained as described in ~~[Regulatory Guide 1.52, Revision 2]~~ ANSI N510-1980, Section 13, shows the methyl iodide penetration ~~less than the value specified below~~ ≤ 1.0 % when tested in accordance with ~~[ASTM D3803-1989]~~ ASTM D3803-1989 at a temperature of ≤ ~~[30°C]~~ 30 °C and greater than or equal to the relative humidity ~~specified below~~ of 95%.

2 8 2 8 2 8

~~ESF Ventilation System~~	~~Penetration~~	~~RH~~
~~[ ]~~	~~[ ]~~	~~[ ]~~

2

~~Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/ (safety factor).~~

~~Safety factor = [5] for systems with heaters.~~
~~= [7] for systems without heaters.~~

8 9

d. Demonstrate for ~~each of the ESF systems~~ the Control Room Emergency Filtration System (F-16) that the pressure drop across the combined HEPA filters ~~, the prefilters,~~ and the charcoal adsorbers is less than ~~the value specified below~~ 6 inches of water when tested in accordance with ~~[Regulatory Guide 1.52,~~ ANSI N510-1980, Sections 10 and 12

8 2 8

5.5 Programs and Manuals

5.5.~~11~~ 10 Ventilation Filter Testing Program (VFTP) (continued)

~~Revision 2, and ASME N510-1989]~~ at ~~the system flowrate specified below [± 10%]~~ a system flowrate of 4950 cfm ± 10%.

~~ESF Ventilation System~~	~~Delta P~~	~~Flowrate~~
~~[ ]~~	~~[ ]~~	~~[ ]~~

~~e. [Demonstrate that the heaters for each of the ESF systems dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989].]~~

~~ESF Ventilation System~~	~~Wattage~~
~~[ ]~~	~~[ ]~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.~~12~~ 11 Explosive Gas ~~and Storage Tank Radioactivity~~ Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the ~~[Waste Gas Holdup System]~~ on-service Gas Decay Tank. ~~[the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"]~~.

The program shall include:

a. ~~The limits for concentrations of hydrogen and oxygen~~ a limit for oxygen concentration in the ~~[Waste Gas Holdup System]~~ on-service Gas Decay Tank and a surveillance program to ensure the ~~limits are maintained. Such limits~~ limit is maintained. This limit shall be

5.5 Programs and Manuals

10

5.5.12 Explosive Gas ~~and Storage Tank Radioactivity~~ Monitoring Program (continued)

5 → 11

appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); .

10

10

~~b. A surveillance program to ensure that the quantity of radioactivity contained in [each gas storage tank and fed into the offgas treatment system] is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of [an uncontrolled release of the tanks' contents]; and~~

~~c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas ~~and Storage Tank Radioactivity~~ Monitoring Program surveillance frequencies.

10

5.5.13 Diesel Fuel Oil Testing Program

5 → 12

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. an API gravity or an absolute specific gravity wit hin limits,

Approved TSTF-118, R0

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5 Programs and Manuals

12

5.5.~~13~~ 12 — 5 Diesel Fuel Oil Testing Program (continued)

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. a clear and bright appearance with proper color;

b. ~~Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks~~; and

Within 31 days following addition of the new fuel oil to storage tanks verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil

Approved TSTF-106, R1

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every ~~31~~ 92 days in accordance with ~~ASTM D-2276~~, ~~Method A-2 or A-3~~.

14 13 14 the applicable ASTM standard

5.5.~~14~~ 13 — 5 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 involve 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 inv olves an unreviewed safety question as defined in 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.~~14~~ 13b 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

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP)

5 14

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s),

Approved TSTF-273, R2

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

15

Insert 5.0-01 CLRTP

16 Insert 5.0-02 PIV Leakage Program

Section 5.0 Inserts

Insert 5.0-01:

5.5.15 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995.

b. The peak design containment internal accident pressure, P_a, is 60 psig.

c. The maximum allowable containment leakage rate, L_a at P_a, shall be 0.4% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$.

2. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $\leq 0.6\ L_a$ for the combined Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests.

3. Air lock testing acceptance criteria are:

   i. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   ii. For each door seal, leakage rate is equivalent to $\leq 0.02\ L_a$ at $\geq P_a$ when tested at a differential pressure of $\geq$ to 10 inches of Hg.

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Section 5.0 Inserts

Insert 5.0-02:

5.5.16 Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) Leakage Program

A program shall be established to verify the leakage from each RCS PIV is within the limits specified below, in accordance with the Event V Order, issued April 20, 1981.

Minimum differential test pressure shall not be less than 150 psid.

Leakage rate acceptance criteria are:

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

Section 5.0 Inserts

2

2

Insert 5.0-04:

In lieu of Position C.4.b(1)and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at approximately 10 year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI.

~~-------------------------------Reviewers Note--------------------------------------~~

~~1. Licensees shall confirm that the flywheels are made of SA 533B material. Further, licensees having Group-15 flywheels (as determined in WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination") need to demonstrate that material properties of their A516 material is equivalent to SA 533B material, and its reference temperature, RT is less than 30 degrees F.~~

~~2. For flywheels not made of SA 533B or A516 material, licensees need to either demonstrate that the flywheel material properties are bounded by those of SA 533B material, or provide the minimum specified ultimate tensile stress, the fracture toughness, and the reference temperature, RT_{NDT}, for that material. For the latter, the licensees should employ these material properties, and use the methodology in the topical report, as extended in the two responses to the staff's RAI, to provide as assessment to justify a change in inspection schedule for their plants.~~

~~---~~

2

Section 5.0 Inserts

Insert 5.0-06:

shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500mrem/yr to the whole body and a dose rate ≤ 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: a dose rate ≤ 1500 mrem/yr to any organ;

Section 5.0 Inserts

Insert 5.0-14:

This program provides controls for inservice inspection and testing of steam generator tubing.

a. Definitions.

1. Tube Inspection - Entry from the hot-leg side with examination from the point of entry completely around the U-bend to the top support of the cold-leg is considered a tube inspection.

2. Imperfection - An exception to the dimension, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

3. Degradation - A service induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.

4. Degraded Tube - A tube that contains imperfections caused by degradation greater than 20% of the nominal tube wall thickness.

5. Defect - An imperfection of such severity that it exceeds the minimum acceptable tube wall thickness of 50%. A tube containing a defect is defective.

6. Plugging Limit - The imperfection depth beyond which the tube must be removed from service or repaired, because the tube may become defective prior to the next scheduled inspection. The plugging limit is 40% of the nominal wall thickness.

b. Sample Selection and Testing

Selection and testing of steam generator tubes shall be made on the following basis:

1. One steam generator of each unit may be selected for inspection during inservice inspection in accordance with the following requirements:

   i. The inservice inspection may be limited to one steam generator on an alternating sequence basis. This examination shall include at least 6% of the tubes if the results of the first or a prior inspection indicate that both generators are performing in a comparable manner.

   ii. When both steam generators are required to be examined by Table 5.5.8-1 and if the condition of tubes in one steam generator is found to be more severe than in the other steam generator of a unit, the steam generator sampling sequence at the subsequent inservice inspection shall be modified to examine the steam generator with the more severe condition.

Section 5.0 Inserts

2. The minimum sample size, inspection result classification and the associated required action shall be in conformance with the requirements of Table 5.5.8-1. The results of each sampling examination of a steam generator shall be classified in the following three categories:

   i. Category C-1: Less than 5% of the total tubes examined are degraded but none are defective.

   ii. Category C-2: Between 5% and 10% of the total number of tubes examined are degraded, but none are defective <u>or</u> one tube to not more than 1% of the sample is defective.

   iii. Category C-3: More than 10% of the total number of tubes examined are degraded, but none are defective <u>or</u> more than 1% of the sample is defective.

   If the first sample of a given steam generator during any inservice inspection, degraded tubes not beyond the plugging limit detected by prior examinations in that steam generator shall be included in the above percentage calculations, only if these tubes are demonstrated to have a further wall penetration of greater than 10% of the tube nominal wall thickness.

3. Tubes shall be selected for examination primarily from those areas of the tube bundle where service experience has shown the most severe tube degradation.

4. In addition to the sample size specified in Table 5.5.8-1, the tubes examined in a given steam generator during the first examination of any inservice inspection shall include all non-plugged tubes in that steam generator that from prior examination were degraded.

5. During the second and third sample examinations of any inservice inspection, the tube inspection may be limited to those sections of the tube lengths where imperfections were detected during the prior examination.

c. Examination Method and Requirements.

   The examination method shall meet the intent of the requirements in ASME Section XI Appendix IV. This includes equipment, personnel, and procedure requirements, certification and calibration along with records and reports. The actual technique may be the latest industry accepted technique, provided the flaw detection capability is as good or better than the technique endorsed by the code in effect per 10 CFR 50, Section 50.55a(g). This allows the use of improvements in inspection techniques that were not included in the code in effect. However, it means that word-for-word compliance with Appendix IV of ASME Section XI may not be possible.

Section 5.0 Inserts

d. Inspection Intervals

1. Inservice inspections shall not be more than 24 calendar months apart.

2. The inservice inspections may be scheduled to be coincident with refueling outages or any plant shutdown, provided the inspection intervals of 5.5.8.d.1 are not exceeded.

3. If two consecutive inservice inspections covering a time span of at least 12 months yield results that fall in C-1 category, the inspection frequency may be extended to 40 month intervals.

4. If the results of the inservice inspection of steam generator tubing conducted in accordance with Table 5.5.8-1 requires that a third sample examination must be performed, and the results of this fall in the category C-3, the inspection frequency shall be reduced to not more than 20 month intervals. The reduction shall apply until a subsequent inspection demonstrates that a third sample examination is not required.

5. Unscheduled inspections shall be conducted in accordance with Specification 5.5.8.b on any steam generator with primary-to-secondary tube leakage exceeding Specification 3.4.13.d. All steam generators shall be inspected in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineered safeguards, or a main steam line or feedwater line break.

e. Corrective Measures

All tubes that leak or have degradation exceeding the plugging limit shall be plugged or repaired by a process such as sleeving prior to return to power from a refueling or inservice inspection condition. Sleeved tubes having sleeve degradation exceeding 40% of the nominal sleeve wall thickness shall be plugged. (Brazed joints shall not be employed. Tubes previously subject to explosive plugging shall not be sleeved.)

TABLE 5.5.8-1
STEAM GENERATOR TUBE INSPECTION PER UNIT
POINT BEACH UNITS 1 & 2

<table>
<tr><th></th><th colspan="2">1ST SAMPLE EXAMINATION</th><th colspan="2">2ND SAMPLE EXAMINATION</th><th colspan="2">3RD SAMPLE EXAMINATION</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S tubes per Steam Generator (S.G.)

S=3(N/n) %

Where:

N is the number of steam generators in the plant = 2

n is the number of steam generators inspected during an examination</td><td>C-1</td><td>Acceptable for continued service</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="5">C-2</td><td rowspan="5">Plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator</td><td>C-1</td><td>Acceptable for continued service</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-2</td><td rowspan="3">Plug or repair tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator</td><td>C-1</td><td>Acceptable for continued service</td></tr>
<tr><td>C-2</td><td>Plug or repair tubes exceeding plug limit. Acceptable for continued service</td></tr>
<tr><td>C-3</td><td>Perform action required under C-3 of 1st sample examination</td></tr>
<tr><td>C-3</td><td>Perform action required under C-3 of 1st sample examination</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-3</td><td rowspan="3">Inspect essentially all tubes in this S.G., plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in the other steam generator. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).</td><td>C-1 in other S.G.</td><td>Acceptable for continued service</td><td>N/A</td><td>N/A</td></tr>
<tr><td>C-2 in other S.G.</td><td>Perform action required under C-2 of 2nd sample examination above</td><td>N/A</td><td>N/A</td></tr>
<tr><td>C-3 in other S.G.</td><td>Inspect essentially all tubes in S.G. & plug or repair tubes exceeding the plugging limit. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).</td><td>N/A</td><td>N/A</td></tr>
</table>

No Significant Hazards Considerations - NUREG-1431 Section 5.05

15-Mar-00

NSHC Number	NSHC Text
A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
LA	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

15-Mar-00

NSHC Number	NSHC Text
LB	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change involves deletion of a Specifications/information which is duplicative of information contained in the Code of Federal Regulations (CFRs). This information is more appropriately addressed by the CFRs and serves no purpose in the Technical Specifications. Deletion of this information will not result in an increase in the probability of an accident. Regulatory requirements do not alter plant design or configuration; therefore, this does not alter any event precursor. Accordingly, there will be no effect on the consequences of any accident. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change deletes materials from the Technical Specifications which are adequately addressed in the CFRs. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change deletes materials from the Technical Specifications which are duplicative of requirements contained in the CFRs. These items are not an input to any accident analysis and, therefore, have no impact on margin of safety.

NSHC Number	NSHC Text
M	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not affect any assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

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 Monitoring Report required by Specification 5.6.2.

c. Licensee initiated changes to the ODCM:

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

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

    ii. 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 Annual Monitoring 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.

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Containment Spray, Safety Injection (High Head) and Safety Injection (Low Head) systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5 Programs and Manuals

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 from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

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;

5.5 Programs and Manuals

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 to areas at or beyond the site boundary shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500mrem/yr to the whole body and a dose rate ≤ 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: a dose rate ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit 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 from each unit 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.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5 Programs and Manuals

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section 4.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position c.4.b(1)and c.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at approximately 10 year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI.

5.5 Programs and Manuals

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program

This program provides controls for inservice inspection and testing of steam generator tubing.

a. Definitions.

1. Tube Inspection – Entry from the hot-leg side with examination from the point of entry completely around the U-bend to the top support of the cold-leg is considered a tube inspection.

2. Imperfection - An exception to the dimension, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

3. Degradation – A service induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.

4. Degraded Tube – A tube that contains imperfections caused by degradation greater than 20% of the nominal tube wall thickness.

5. Defect – An imperfection of such severity that it exceeds the minimum acceptable tube wall thickness of 50%. A tube containing a defect is defective.

6. Plugging Limit – The imperfection depth beyond which the tube must be removed from service or repaired, because the tube may become defective prior to the next scheduled inspection. The plugging limit is 40% of the nominal wall thickness.

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

b. Sample Selection and Testing

Selection and testing of steam generator tubes shall be made on the following basis:

1. One steam generator of each unit may be selected for inspection during inservice inspection in accordance with the following requirements:

    i. The inservice inspection may be limited to one steam generator on an alternating sequence basis. This examination shall include at least 6% of the tubes if the results of the first or a prior inspection indicate that both generators are performing in a comparable manner.

    ii. When both steam generators are required to be examined by Table 5.5.8-1 and if the condition of tubes in one steam generator is found to be more severe than in the other steam generator of a unit, the steam generator sampling sequence at the subsequent inservice inspection shall be modified to examine the steam generator with the more severe condition.

2. The minimum sample size, inspection result classification and the associated required action shall be in conformance with the requirements of Table 5.5.8-1. The results of each sampling examination of a steam generator shall be classified in the following three categories:

    i. Category C-1: Less than 5% of the total tubes examined are degraded but none are defective.

    ii. Category C-2: Between 5% and 10% of the total number of tubes examined are degraded, but none are defective or one tube to not more than 1% of the sample is defective.

    iii. Category C-3: More than 10% of the total number of tubes examined are degraded, but none are defective or more than 1% of the sample is defective.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

If the first sample of a given steam generator during any inservice inspection, degraded tubes not beyond the plugging limit detected by prior examinations in that steam generator shall be included in the above percentage calculations, only if these tubes are demonstrated to have a further wall penetration of greater than 10% of the tube nominal wall thickness.

3. Tubes shall be selected for examination primarily from those areas of the tube bundle where service experience has shown the most severe tube degradation.

4. In addition to the sample size specified in Table 5.5.8-1, the tubes examined in a given steam generator during the first examination of any inservice inspection shall include all non-plugged tubes in that steam generator that from prior examination were degraded.

5. During the second and third sample examinations of any inservice inspection, the tube inspection may be limited to those sections of the tube lengths where imperfections were detected during the prior examination.

c. Examination Method and Requirements.

The examination method shall meet the intent of the requirements in ASME Section XI Appendix IV. This includes equipment, personnel, and procedure requirements, certification and calibration along with records and reports. The actual technique may be the latest industry accepted technique, provided the flaw detection capability is as good or better than the technique endorsed by the code in effect per 10 CFR 50, Section 50.55a(g). This allows the use of improvements in inspection techniques that were not included in the code in effect. However, it means that word-for-word compliance with Appendix IV of ASME Section XI may not be possible.

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

d. Inspection Intervals

1. Inservice inspections shall not be more than 24 calendar months apart.

2. The inservice inspections may be scheduled to be coincident with refueling outages or any plant shutdown, provided the inspection intervals of 5.5.8.d.1 are not exceeded.

3. If two consecutive inservice inspections covering a time span of at least 12 months yield results that fall in C-1 category, the inspection frequency may be extended to 40 month intervals.

4. If the results of the inservice inspection of steam generator tubing conducted in accordance with Table 5.5.8-1 requires that a third sample examination must be performed, and the results of this fall in the category C-3, the inspection frequency shall be reduced to not more than 20 month intervals. The reduction shall apply until a subsequent inspection demonstrates that a third sample examination is not required.

5. Unscheduled inspections shall be conducted in accordance with Specification 5.5.8.b on any steam generator with primary-to-secondary tube leakage exceeding Specification 3.4.13.d. All steam generators shall be inspected in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineered safeguards, or a main steam line or feedwater line break.

e. Corrective Measures

All tubes that leak or have degradation exceeding the plugging limit shall be plugged or repaired by a process such as sleeving prior to return to power from a refueling or inservice inspection condition. Sleeved tubes having sleeve degradation exceeding 40% of the nominal sleeve wall thickness shall be plugged. (Brazed joints shall not be employed.

Tubes previously subject to explosive plugging shall not be sleeved)

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test frequencies.

TABLE 5.5.8-1
STEAM GENERATOR TUBE INSPECTION PER UNIT
POINT BEACH UNITS 1 & 2

	1ST SAMPLE EXAMINATION		2ND SAMPLE EXAMINATION		3RD SAMPLE EXAMINATION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per Steam Generator (S.G.)	C-1	Acceptable for continued service	N/A	N/A	N/A	N/A
S=3(N/n) %	C-2	Plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator	C-1	Acceptable for continued service	N/A	N/A
Where:			C-2	Plug or repair tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator	C-1	Acceptable for continued service
N is the number of steam generators in the plant = 2					C-2	Plug or repair tubes exceeding plug limit. Acceptable for continued service
					C-3	Perform action required under C-3 of 1st sample examination
			C-3	Perform action required under C-3 of 1st sample examination	N/A	N/A
n is the number of steam generators inspected during an examination	C-3	Inspect essentially all tubes in this S.G., plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in the other steam generator. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).	C-1 in other S.G.	Acceptable for continued service	N/A	N/A
			C-2 in other S.G.	Perform action required under C-2 of 2nd sample examination above	N/A	N/A
			C-3 in other S.G.	Inspect essentially all tubes in S.G. & plug or repair tubes exceeding the plugging limit. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).	N/A	N/A

5.5 Programs and Manuals

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of the Control Room Emergency Filtration System (F-16) at the frequencies specified in Regulatory Guide 1.52, Revision 2, and in accordance with ANSI N510-1980.

a. Demonstrate for the Control Room Emergency Filtration System (F-16) that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass $\leq 1.0\%$ when tested in accordance with ANSI N510-1980, Section 10, at a system flowrate of 4950 cfm $\pm$ 10%.

b. Demonstrate for the Control Room Emergency Filtration System (F-16) that an inplace test of the charcoal adsorber shows a penetration and system bypass $\leq 1.0\%$ when tested in accordance with ANSI N510-1980, Section 12, at a system flowrate of 4950 cfm $\pm$ 10%.

5.5.10 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for the Control Room Emergency Filtration System (F-16) that a laboratory test of a sample of the charcoal adsorber, when obtained as described in ANSI N510-1980, Section 13, shows the methyl iodide penetration ≤ 1.0%, when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and greater than or equal to a relative humidity of 95%.

d. Demonstrate for the Control Room Emergency Filtration System (F-16) that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than 6 inches of water when tested in accordance with ANSI N510-1980, Sections 10 and 12, at a system flowrate of 4950 cfm ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.11 Explosive Gas Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the on-service Gas Decay Tank.

The program shall include a limit for oxygen concentration in the on-service Gas Decay Tank and a surveillance program to ensure the limit is maintained. This limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion).

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Monitoring Program surveillance frequencies.

5.5 Programs and Manuals

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. a clear and bright appearance with proper color;

b. Within 31 days of addition of the new fuel oil to storage tanks verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with the applicable ASTM standard.

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5.13 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 involve 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 involves an unreviewed safety question as defined in 10 CFR 50.59.

5.5.13 Technical Specifications (TS) Bases Control 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 5.5.13b 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).

5.5.14 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995.

b. The peak design containment internal accident pressure, P_a, is 60 psig.

c. The maximum allowable containment leakage rate, L_a at P_a, shall be 0.4% of containment air weight per day.

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$.

2. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $\leq 0.6\ L_a$ for the combined Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests.

3. Air lock testing acceptance criteria are:

i. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

ii. For each door seal, leakage rate is equivalent to $\leq 0.02\ L_a$ at $\geq P_a$ when tested at a differential pressure of $\geq$ to 10 inches of Hg.

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5 Programs and Manuals

5.5.16 Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) Leakage Program

A program shall be established to verify the leakage from each RCS PIV is within the limits specified below, in accordance with the Event V Order, issued April 20, 1981.

a. Minimum differential test pressure shall not be less than 150 psid.

b. Leakage rate acceptance criteria are:

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

Cross-Reference Report - NUREG-1431 Section 5.06

ITS to CTS

15-Mar-00

ITS	CTS	DOC
SPEC 5.05.01	15.07.08.04.A.06	A.06
SPEC 5.06	15.06.09	A.01
	15.06.09.01.B.01	A.01
SPEC 5.06.01	15.06.09.01.B.02.B	A.03
SPEC 5.06.02	15.07.08.04	A.01
	15.07.08.04.A.01	A.04
	15.07.08.04.A.03	A.05
SPEC 5.06.03	15.06.09.01.C.01	LA.01
SPEC 5.06.04	NEW	M.03
SPEC 5.06.05	NEW	M.04
SPEC 5.06.06	15.06.09.02.D	M.02
	NEW	M.05
SPEC 5.06.07	15.04.04.II.D	A.01
SPEC 5.06.08	15.04.02.A.07	A.02
	15.06.09.01.B.02.A	A.01

Cross-Reference Report - NUREG-1431 Section 5.06
CTS to ITS

15-Mar-00

CTS	ITS	DOC
15.04.02.A.07	SPEC 5.06.08	A.02
15.04.04.II.D	N/A	LA.03
	SPEC 5.06.07	A.01
15.06.09	SPEC 5.06	A.01
15.06.09.01.A	N/A	LB.01
15.06.09.01.B	N/A	A.01
15.06.09.01.B.01	SPEC 5.06	A.01
15.06.09.01.B.02.A	SPEC 5.06.08	A.01
15.06.09.01.B.02.B	SPEC 5.06.01	A.03
15.06.09.01.B.02.C	N/A	LB.02
15.06.09.01.B.02.D	N/A	A.07
15.06.09.01.B.02.E	N/A	M.01
15.06.09.01.C.01	SPEC 5.06.03	LA.01
15.06.09.02	N/A	A.01
15.06.09.02.B	N/A	LA.02
15.06.09.02.C	N/A	L.04
15.06.09.02.D	SPEC 5.06.06	M.02
15.06.09.02.E	N/A	L.01
15.06.10	N/A	LA.03
15.07.08.04	SPEC 5.06.02	A.01
15.07.08.04.A.01	SPEC 5.06.02	A.04
15.07.08.04.A.02	N/A	L.02
15.07.08.04.A.03	SPEC 5.06.02	A.05
15.07.08.04.A.04	N/A	LB.03
15.07.08.04.A.05	N/A	L.03
15.07.08.04.A.06	SPEC 5.05.01	A.06
15.07.08.04.A.07	N/A	LA.04
15.07.08.04.B	N/A	LA.04
15.07.08.04.C	N/A	LA.04
15.07.08.04.D	N/A	LA.04
15.07.08.06	N/A	LA.03

Description of Changes - NUREG-1431 Section 5.06

15-Mar-00

DOC Number	DOC Text
A.01	In the conversion of Point Beach Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.04.04.II.D	SPEC 5.06.07
15.06.09	SPEC 5.06
15.06.09.01.B	N/A
15.06.09.01.B.01	SPEC 5.06
15.06.09.01.B.02.A	SPEC 5.06.08
15.06.09.02	N/A
15.07.08.04	SPEC 5.06.02

A.02 Included in CTS 15.4.2.A.7 are the reporting requirements for steam generator tube inservice inspection results. These reporting requirements will be retained in proposed ITS 5.6.8 (NUREG-1431 item 5.6.10). NUREG-1431, item 5.6.10, "Steam Generator Tube Inspector Report," has a bracketed reviewer's note stating "Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used."

Therefore, the CTS requirements have simply been transferred to proposed ITS 5.6.8, with certain wording preferences changed to reflect the nomenclature and numbering used in the ITS. The term "Annual Results and Data Report" has been changed to "a Report", because the "Annual Results and Data Report" is not being maintained in the proposed ITS. This change is administrative.

CTS:	ITS:
15.04.02.A.07	SPEC 5.06.08

A.03 CTS 15.06.09.01.B.02.B describes an attribute of the "Annual Results and Data Report". This attribute's content is being included in the "Occupational Radiation Exposure Report" in proposed ITS section 5.6.1. Minor administrative wording changes to this attribute were necessary to reflect the wording in NUREG-1431. This change is administrative.

CTS:	ITS:
15.06.09.01.B.02.B	SPEC 5.06.01

DOC Number	DOC Text	
A.04	Included in CTS 15.7.8.4.A are the requirements for the contents of the "Annual Monitoring Report (AMR)." CTS 15.7.8.4.A.1 contains the requirements for a subsection of the AMR referred to as the "Radiological Effluent Control Program (RECP)." The required contents of this subsection are consistent with the required contents of NUREG-1431, item 5.6.3 "Radioactive Effluent Release Report," which is proposed for adoption in whole as ITS 5.6.2 "Annual Monitoring Report." This change is administrative.	
	CTS:	**ITS:**
	15.07.08.04.A.01	SPEC 5.06.02
A.05	Included in CTS 15.7.8.4.A are the requirements for the contents of the "Annual Monitoring Report. (AMR)." CTS 15.7.8.4.A.3 contains the requirements for a subsection of the AMR referred to as the "Radiological Effluent Monitoring Program (REMP)." The required contents of this subsection are consistent with the required contents of NUREG-1431, item 5.6.2 "Annual Radiological Environmental Operating Report," which is proposed for adoption in whole as ITS 5.6.2 "Annual Monitoring Report." This change is administrative.	
	CTS:	**ITS:**
	15.07.08.04.A.03	SPEC 5.06.02
A.06	CTS 15.7.8.4.A.6 states the following, "A description of changes to the REMCAP, ODCM, EM, RECM or PCP which were implemented and became effective during the reporting period shall be submitted pursuant to Specification 15.7.8.7." This requirement is duplicative of the requirements contained in CTS 15.7.8.7. Therefore, this requirement will not be retained in the proposed ITS. Change requirements for the ODCM are described in proposed ITS 5.5.1. This change is administrative.	
	CTS:	**ITS:**
	15.07.08.04.A.06	SPEC 5.05.01
A.07	CTS 15.06.09.01.B.02.D states that the Annual Results and Data Report shall include a tabulation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves. This attribute will not be retained in the proposed ITS because the NUREG-1431 (STS) requirement in item 5.6.4, to include documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves in the monthly operating reports, was deleted from the STS in accordance with NRC approved TSTF-258, revision 4. The basis for this deletion (in TSTF-258) was that these attributes were not included in the guidance contained in NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report". Therefore, this reporting requirement will not be retained in the proposed ITS.	
	CTS:	**ITS:**
	15.06.09.01.B.02.D	N/A

Description of Changes - NUREG-1431 Section 5.06

15-Mar-00

DOC Number	DOC Text	
L.01	CTS 15.6.9.2.E states "if a main steam line radiation monitor (SA-11) fails and cannot be restored to operability in seven days, prepare a special report within thirty days of the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the channel to operable status." This requirement is not contained in NUREG-1431 and will not be retained in the proposed ITS. During the conversion of NUREG-1431 section 3.3.3 (Post Accident Monitoring Instrumentation) to the proposed ITS for PBNP, operability requirements for the steam line radiation monitor (SA-11) were not retained. This was based on the fact that this monitor is not identified as Type A or Category I in the PBNP Regulatory Guide 1.97 analyses, and therefore does not need to be included in the ITS. Not retaining this variable was less restrictive, but is acceptable because it does not result in a reduction in the margin of safety. Therefore, because this monitor was not retained in proposed ITS 3.3.3, the report requirements for its inoperability will not be retained in proposed ITS 5.6. This change is less restrictive.	
	CTS:	**ITS:**
	15.06.09.02.E	N/A
L.02	CTS 15.7.8.4.2 contains new and spent fuel receipts and shipment reporting requirements as a subsection of the "Annual Monitoring Report (AMR)." This attribute will not be retained in the proposed ITS. This reporting attribute is not contained in NUREG-1431. This reporting attribute was contained in the original Technical Specifications for PBNP (early 1970's). This administrative reporting requirement is not required to be in the ITS to provide adequate protection to the public health and safety. A search of the current licensing basis (CLB) database did not reveal any regulatory commitments associated with this reporting criteria. Therefore, this requirement will not be retained in the proposed ITS.	
	CTS:	**ITS:**
	15.07.08.04.A.02	N/A
L.03	CTS 15.7.8.4.5 contains administrative requirements for having meteorogical data kept on file on site as a subsection of the "Annual Monitoring Report (AMR)." This attribute will not be retained in the proposed ITS. This administrative requirement is not contained in NUREG-1431. This administrative requirement is not required to be in the ITS to provide adequate protection to the public health and safety. A search of the current licensing basis (CLB) database did not reveal any regulatory commitments associated with this administrative requirement. Therefore, this requirement will not be retained in the proposed ITS.	
	CTS:	**ITS:**
	15.07.08.04.A.05	N/A

DOC Number	DOC Text
L.04	CTS 15.6.9.2.C states "In the event the low temperature overpressure protection (LTOP) system (power operated relief valves in the low temperature overpressure protection mode) or residual heat removal system relief valves are operated to relieve a pressure transient which, by licensee's evaluation, could have resulted in an overpressurization incident had the system not been operable, a special report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the system on the transient and any corrective action necessary to prevent recurrence." This reporting requirement is a plant specific requirement that is not included in NUREG-1431 and will not be retained in the proposed ITS.

This reporting requirement was added in a Wisconsin Electric (WE) license amendment application (TSCR 56) dated 11/02/78, which was an application for adding numerous Technical Specification requirements for "overpressure mitigating system operation." This application was approved by the NRC via License Amendment No. 45 (DPR-24) and No. 50 (DPR-27), and SER dated 5/20/80. This reporting requirement was voluntarily added by WE probably because of the high number of overpressurization events that were occurring in the industry at that time (prior to the industry adding overpressurization protection). The NRC did not mention the addition of this reporting requirement in the SER dated 5/20/80; therefore, it was not used by the NRC as a basis for the approval of the license amendment.

This reporting requirement is not required to provide adequate protection to the public health and safety. Any occurrence of a reactor vessel overpressurization event at PBNP will be evaluated in accordance with the applicable regulations, 10 CFR 50.72 and 10 CFR 50.73, and reported to the NRC if appropriate.

It should be noted that the NUREG-1431 (STS) requirement in item 5.6.4, to include documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves in the monthly operating reports, was deleted from the STS in accordance with NRC approved TSTF-258, revision 4. The basis for this deletion (in TSTF-258) was that these attributes were not included in the guidance contained in NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report".

CTS:	ITS:
15.06.09.02.C	N/A

Description of Changes - NUREG-1431 Section 5.06

15-Mar-00

DOC Number	DOC Text
LA.01	CTS 15.6.9.1.C contains details that are to be included in the Monthly Operating Report. These unnecessary details are being deleted from the Technical Specifications and will be relocated to licensee control. NUREG-1431 item 5.6.3 contains the necessary details that are to be included in the monthly operating report, and the proposed ITS will adopt these in whole. These details included in the CTS are not required to be in the ITS and are better suited for procedural control. These details are not required to provide adequate protection to the public health and safety. Changes to plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, the level of safety is unaffected by the change. **CTS:** 15.06.09.01.C.01 **ITS:** SPEC 5.06.03
LA.02	CTS 15.6.9.2.B contains reporting requirements to notify the NRC within 14 days of any plans for removal of any poison assemblies from the spent fuel storage racks. This reporting requirement is being deleted from the Technical Specifications and will be relocated to licensee control. This requirement was added as a result of License Amendment No. 35 (DPR-24) and No. 41 (DPR-27), approved by the NRC via SER dated April 4, 1979, which allowed for re-racking of the spent fuel pool (to increase available size). This reporting requirement is a plant specific requirement that is not included in NUREG-1431. This reporting requirement included in the CTS is not required to be in the ITS and is better suited for procedural control. This reporting requirement is not required to provide adequate protection to the public health and safety. Changes to plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, the level of safety is unaffected by the change. **CTS:** 15.06.09.02.B **ITS:** N/A

DOC Number	DOC Text
LA.03	Wisconsin Electric Power Company has concluded that CTS 15.4.4.II.D, 15.6.10, and 15.7.8.6 can be relocated to licensee control. The basis for this conclusion is as follows: CTS section 15.4.4.II.D, 15.6.10, and 15.7.8.6 describes the administrative requirements for records and records retension. These requirements will not be maintained in the proposed ITS and will be relocated to licensee control. NUREG-1431 does not contain these administrative requirements. PBNP proposes to relocate the requirements in these CTS sections to Section 1.4, "Quality Assurance Program," of the PBNP Final Safety Analysis Report (FSAR). This section of the FSAR describes the PBNP Quality Assurance Program (QAP) in detail. Changes to this section of the FSAR are controlled in accordance with the requirements of 10 CFR 50.54(a). Relocating these requirements out of the CTS and into FSAR §1.4 is consistent with the guidance contained in NRC Administrative Letter (AL) 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995. NRC AL 95-06 states that the NRC encourages relocation of the record and record retention requirements out of the licensee's Technical Specifications and into QAP descriptions as long as future revisions to said functions are controlled in accordance with the requirements of 10 CFR 50.54. Conclusion: Based on the above information, the requirements in CTS 15.4.4.II.D, 15.6.10, and 15.7.8.6 can be relocated to licensee control.

CTS:	ITS:
15.04.04.II.D	N/A
15.06.10	N/A
15.07.08.06	N/A

DOC Number	DOC Text
LA.04	CTS 15.7.8.4.A.7, and CTS 15.7.8.4.B through D contain numerous details and requirements for the radioactive effluent control program. These requirements will not be maintained in the proposed ITS and will be relocated to licensee control. NUREG-1431 does not contain these administrative requirements. This detail is not required to be in the technical specifications to provide adequate protection to the public health and safety. These CTS requirements are already contained in the Point Beach Offsite Dose Calculation Manual (ODCM) and/or its implementing procedures. Changes to these requirements will be controlled in accordance with the 10 CFR 50.59 process. Therefore, the level of safety is unaffected by the change.

CTS:	ITS:
15.07.08.04.A.07	N/A
15.07.08.04.B	N/A
15.07.08.04.C	N/A
15.07.08.04.D	N/A

DOC Number	DOC Text
LB.01	CTS 15.6.9.1.A requires submission of a summary report to the NRC of plant startup and power escalation testing under the following conditions: a. Receipt of an operating license, b. Amendment to the license involving a planned increase in power level, c. Installation of fuel that has a different design or has been manufactured by a different fuel supplier, and d. Modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. These reporting requirements will not be maintained in the proposed ITS, and are not included in NUREG-1431. The conditions describing the above reporting requirements are either no longer applicable or would already require NRC review and approval under the applicable CFRs as follows. Item 15.6.9.1.A.a is no longer applicable since the receipt of the operating licenses for both units occurred in the early 1970s. Item 15.6.9.1.A.b would require submission of a license amendment request package to the NRC, in accordance with the requirements of 10 CFR 50.4 and 10 CFR 50.90, and review and approval from the NRC via a license amendment (the licensed power level is included in the Technical Specifications). Similiarily, item 15.6.9.1.A.c would require submission of a license amendment request package to the NRC, in accordance with the requirements of 10 CFR 50.4 and 10 CFR 50.90, and review and approval from the NRC via a license amendment (fuel design is included in the Technical Specifications). Item 15.6.9.1.A.d (modifications to the plant) is covered under the requirements of 10 CFR 50.59, and would require NRC review and approval if a unreviewed safety question (USQ) would exist as a result of the proposed modification. Based on the above, this change removes CTS details that are duplicative of other regulatory requirements, or that are no longer applicable to PBNP.

CTS:	ITS:
15.06.09.01.A	N/A

DOC Number	DOC Text
LB.02	CTS 15.6.9.1.B.2.C states that the Annual Results and Data Report shall include a description of facility changes, tests or experiments as required pursuant to 10 CFR 50.59(b). This attribute will not be maintained in the proposed ITS. This item is not included in NUREG-1431 because it is duplicative of code requirements. 10 CFR 50.59(b)(2) requires that a report be submitted annually in accordance with 10 CFR 50.4 containing a brief description of any changes, tests, and experiments, including a summary of the safety evaluation of each. Therefore, CTS 15.6.9.1.B.2.C is duplicative of code requirements and will not be retained. **CTS:** 15.06.09.01.B.02.C **ITS:** N/A
LB.03	Included in CTS 15.7.8.4.A are the requirements for the contents of the "Annual Monitoring Report (AMR)." CTS 15.7.8.4.A.4 contains reporting requirements (as a subsection of the AMR) for leak testing of sealed sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination. This attribute will not be retained in the proposed ITS. This item is not included in NUREG-1431 because it is duplicative of code requirements (reporting requirements contained in 10 CFR 31.5). Therefore, CTS 15.7.8.4.A is duplicative of code requirements and will not be retained. **CTS:** 15.07.08.04.A.04 **ITS:** N/A

DOC Number	DOC Text
M.01	CTS 15.6.9.1.B.2.E requires the submission (as part of the Annual Results and Data Report) of the results of specific activity analysis only when the primary coolant exceeds the limits of CTS 15.3.1.C "Maximum Coolant Activity", and lists the specific information that is to be included in CTS 15.6.9.1.B.2.E.1 through CTS 15.6.9.1.B.2.E.5. This relaxation is not included in NUREG-1431 and will not be retained in the proposed ITS. This relaxation was added to the PBNP CTS as a result of NRC Generic Letter 85-19. The NRC determined (in GL 85-19) that the reporting requirements for iodine spiking could be reduced from short-term reports (Special Reports or Licensee Event Reports) to a report item that could be included in the annual report and encouraged licensees to amend their Technical Specifications to include this requirement. This was due to the fact that poor fuel reliability in the late 1970's and early 1980's lead to increased primary coolant activity levels, which lead to high volumes of special reports and LERs by licensees. PBNP has concluded that this reporting requirement relaxation is no longer required, due to the extremely low incidence of fuel cladding failures at Point Beach and hence extremely low probability of PBNP exceeding reactor coolant activity limits. Any incidences of high reactor coolant activity at PBNP will be reported to the NRC in accordance with the requirements of 10 CFR 50.72 and 10 CFR 50.73, as appropriate. Because the CTS requirement to report high reactor coolant activity in the annual report is a relaxation of the existing potential reportability requirements in 10 CFR 50.72 and 10 CFR 50.73, elimination of this relaxation is more restrictive. **CTS:** 15.06.09.01.B.02.E **ITS:** N/A
M.02	CTS 15.6.9.2.D requires that a special report be submitted to the NRC within 30 days if the minimum number of channels for the containment high-range radiation monitor is not restored within the allowed outage time. Similarily, proposed ITS LCO 3.3.3, Condition B requires submission of a report in accordance with ITS 5.6.6 (NUREG-1431 item 5.6.8) if the minimum number of channels for the containment high-range radiation monitor is not restored within the allowed outage time. The report requirements are similar in that they require outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. The proposed ITS 5.6.6 will adopt NUREG item 5.6.8 in whole. However, the ITS is more restrictive in that ITS 5.6.6 requires a report to be submitted within 14 days, whereas the CTS requirement is 30 days. Therefore, this change is more restrictive. **CTS:** 15.06.09.02.D **ITS:** SPEC 5.06.06

Description of Changes - NUREG-1431 Section 5.06

15-Mar-00

DOC Number	DOC Text	
M.03	NUREG-1431 item 5.6.5 and proposed ITS 5.6.4 requires that a Core Operating Limits Report (COLR) be established that documents the core operating limits prior to each reload cycle, and specifies the required contents of this report. This COLR report is not required by the CTS, therefore adopting this requirement is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.06.04
M.04	NUREG-1431 item 5.6.6 and proposed ITS 5.6.5 requires that a Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) be established that documents the RCS pressure and temperature limits, and specifies the required contents of this report. This PTLR is not required by the CTS, therefore adopting this requirement is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.06.05
M.05	NUREG-1431 item 5.6.8 and proposed ITS 5.6.6 requires that a Post Accident Monitoring (PAM) Instrumentation report be submitted within 14 days as required by condition B or G of (proposed ITS) LCO 3.3.3. Condition B or G of LCO 3.3.3 basically requires submission of a PAM report when any of the PAM instrumentation listed in TABLE 3.3.3-1 is inoperable for greater than 30 days. This requirement is more restrictive than the CTS, because TABLE 3.3.3-1 includes PAM instrumentation that is not currently required to be reported under the CTS special reporting requirements. Therefore, adopting this requirement is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.06.06

A.01 → See ITS 5.6 → The following reports shall be submitted in accordance with 10 CFR 50.4

15.6.9 PLANT REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following program for reporting of operating information shall be followed. Reports should be addressed to the Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4.

15.6.9.1 Routine Reports

A. Startup Report

LB.01

1. A summary report of plant startup and power escalation testing which addresses each of the tests identified in the FSAR and includes a general description of the measured values obtained during the test program and a comparison of these values with design predictions and specifications must be submitted under the following conditions:

   a. Receipt of an operating license.
   b. Amendment to the license involving a planned increase in power level.
   c. Installation of fuel that has a different design or has been manufactured by a different fuel supplier.
   d. Modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

   Any corrective actions that were required to obtain satisfactory operations shall also be described.

2. This report shall be submitted within the earliest time frame of the following:

A.01

LB.01

a. 90 days following completion of the startup tests.
b. 90 days following resumption or commencement of commercial power operation.
c. 9 months following initial criticality.

B. Annual Results and Data Report

See ITS 5.6

1. A results and data report covering the period of the previous calendar year shall be submitted prior to March 1 of each year.

2. This report shall include:

See ITS 5.6.8

a. Complete results of steam generator tube inservice inspection completed during the calendar year as required by specification 15.4.2.A.7

b. A tabulation on an annual basis of the number of station, utility, and other personnel receiving exposures greater than 100 mrem/year and their associated man-rem exposure according to work and job functions. The dose assignments to various duty functions may be estimates based on pocket dosimeter, TLD or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

LB.02

c. A description of facility changes, tests or experiments as required pursuant to 10 CFR 50.59(b).

A.07

d. A tabulation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves.

Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person – rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

A.03

e. Reactor coolant activity

M.01

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 15.3.1.C. The following information shall be included:

1. Reactor power history starting 48 hours prior to the first sample in which the activity limit was exceeded;
2. Results of the last isotopic analysis for radioiodine analysis prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include the date and time of sampling and the radioiodine concentrations;
3. Clean-up flow history starting 48 hours prior to the first sample in which the activity limit was exceeded,
4. Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady state level; and
5. The time duration when the specific activity of the primary coolant exceeded 0.8 microcuries per gram DOSE EQUIVALENT I-131.

C. Monthly Operating Reports

LA.01

ITS 5.6.3

1. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis ~~under the titles "Operating Data Report", "Average Daily Power Levels" and "Unit Shutdowns" and "Power Reduction". In addition, the report shall contain a narrative summary of operating experience that describes the operation of the facility, including major safety-related maintenance for the monthly report period.~~

2. ~~Completed reports shall be sent~~ by the tenth of each month following the calendar month covered by the report.

A.01

15.6.9.2 Unique Reporting Requirements

The following written reports shall be submitted to the Director, Office of Nuclear Reactor Regulation, USNRC.

A. Deleted

B. Poison Assembly Removal From Spent Fuel Storage Racks

LA.02 Plans for removal of any poison assemblies from the spent fuel storage racks shall be reported and described at least 14 days prior to the planned activity. Such report shall describe neutron attenuation testing for any replacement poison assemblies, if applicable, to confirm the presence of boron material.

C. Low Temperature Overpressure Protection System Operation

L.04 In the event the low temperature overpressure protection system (power operated relief valves in the low temperature overpressure protection mode) or residual heat removal system relief valves are operated to relieve a pressure transient which, by licensee's evaluation, could have resulted in an overpressurization incident had the system not been operable, a special report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the system on the transient and any corrective action necessary to prevent recurrence.

M.02

See ITS 5.6.6

D. Failure of Containment High-Range Radiation Monitor

A minimum of two in-containment radiation-level monitors with a maximum range of 10^8 rad/hr (10^7/hr for photons only) should be operable at all times except for cold shutdown and refueling outages. This is specified in Table 15.3.5-5, item 7. If the minimum number of operable channels are not restored to operable condition within seven days after failure, a special report shall be submitted to the NRC within thirty days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

E. Failure of Main Steam Line Radiation Monitors

L.01

If a main steam line radiation monitor (SA-11) fails and cannot be restored to operability in seven days, prepare a special report within thirty days of the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the channel to operable status.

M.03

Insert 5.6-1 (ITS 5.6.4 (COLR))

M.04

Insert 5.6-2 (ITS 5.6.5 (PTLR))

M.05

Insert 5.6-3 (ITS 5.6.6 (PAM Report))

A.01

2) Complete grease coverage exists for the different parts of the Anchorage system, and
3) The chemical properties of the filler material are within the tolerance limits specified by the manufacturer.

D. Reports

LA.03

A final report documenting the results of each tendon ~~surveillance will be~~ prepared and ~~maintained as a permanent~~ plant record.

See ITS 5.6.7 (Tendon Surveillance Report)

Abnormal conditions observed during testing will be evaluated to determine the effect of such conditions on containment structural integrity. This evaluation should be completed within 30 days of the identification of the condition. Any condition which is determined in this evaluation to have a significant adverse effect on containment structural integrity will be considered an abnormal degradation of the containment structure.

Any abnormal degradation of the containment structure identified during the engineering evaluation of abnormal conditions shall be reported to the Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

III. End Anchorage Concrete Surveillance

A. Specific locations for surveillance will be determined by information obtained from design calculations, as-built end anchorage concrete and prestressing records, observations of the end anchorage concrete during and after prestressing, and results of deformation measurements made during prestressing and the initial structural test.

B. The inspection intervals will be approximately one-half year and one year after the initial structural test and shall be chosen such that the inspection occurs during the warmest and coldest part of the year following the initial structural test.

Defect is an imperfection of such severity that it exceeds the minimum acceptable tube wall thickness of 50%. A tube containing a defect is defective.

Plugging Limit is the imperfection depth beyond which the tube must be removed from service or repaired, because the tube may become defective prior to the next scheduled inspection. The plugging limit is 40% of the nominal tube wall thickness.

F* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is 1.12 inches (including eddy current uncertainty). The F* distance is measured from the bottom of the upper roll transition of the repair roll toward the bottom of the tubesheet.[2]

F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded within the F* distance.[2]

6. Corrective Measures

All tubes that leak or have degradation exceeding the plugging limit shall be plugged or repaired by a process such as sleeving[1] or classification as an F* tube[2] prior to return to power from a refueling or inservice inspection condition. Sleeved tubes having sleeve degradation exceeding 40% of the nominal sleeve wall thickness shall be plugged.

7. Reports

A.02

a

ITS
5.6.8

Table 5.5.8-1

5.6.8.b

(a) After each inservice examination, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission as soon as practicable.

(b) The complete results of the steam generator tube inservice inspection shall be included in ~~the Annual Results and Data~~ Report for the period in which the inspection was completed.

Reports shall include:

1. Number and extent of tubes inspected.

2. Location and percent of all thickness penetration for each indication.

3. Identification of tubes plugged or repaired.

(c) Reports required by ~~Table 15.4.2-1~~ - Steam Generator Tube Inspection shall provide the information required by Specification ~~15.4.2.A.7(b)~~ and a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. The report shall be submitted to the Commission prior to resumption of plant operation.

1 Brazed joints shall not be employed. Tubes previously subject to explosive plugging shall not be sleeved.

2 Applicable only to the Westinghouse Model 44 steam generators in Unit 2. Following steam generator replacement in Unit 2, the definitions and F* repair option are null and void.

15.6.10 PLANT OPERATING RECORDS LA.03

Specification

Records and logs relative to the following items shall be retained for five (5) years unless a longer period is required by applicable regulations.

A. Records of normal plant operation, including power levels and periods of operation at each power level shall be retained for 5 years except those records of transient or operational cycles for reactor coolant system (RCS) components having a limited number of design transients, shall be retained for the duration of the operating license.

B. Records of principal maintenance activities, including inspection, repair, substitution, or replacement of items of equipment pertaining to nuclear safety shall be retained for a period of 5 years where these requirements do not conflict with requirements of 10 CFR 50.49(j), 10 CFR 50.59, and surveillance requirements of these Technical Specifications. The quality assurance, environmental qualification, installation, and service life records of components covered by these requirements shall be retained for the duration of the Operating License.

C. Records of Licensee Event Reports.

D. Records of installation, environmental qualification, periodic checks, inspections, and calibrations of equipment pertaining to nuclear safety to verify that surveillance requirements are being met will be retained for the duration of the Operating License. All other records of this type will be retained for 5 years.

E. Records of new and spent fuel inventory and assembly histories. (5 years following transfer)

F.* Records of design modifications made to systems and equipment, including drawings, as described in the FSAR.

G.* Records of plant radiation and contamination surveys.

H.* Records of off-site environmental surveys.

I.* Records of radiation exposure of all individuals entering radiation controlled areas of the plant, including records for preparation of NRC-4 forms, bioassay and whole body counting results; and records of

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*Items will be retained for the duration of the Operating License individual exposures exceeding 40-MPC hour limits, including evaluations and actions taken.

J.* Records of gaseous and liquid radioactive material released to the environment and dilution of these wastes.

K.* Records of any special reactor tests or experiments.

L. Records of changes made in the Operating Procedures.

M. Records of sealed source and fission detector leak tests and results performed pursuant to Specification 15.4.12, including annual physical inventory results verifying accountability of sources.

N. Records of training, qualification and requalification for NRC licensed personnel shall be retained for the duration of the operator's license per 10 CFR 55 requirements. Records of fire brigade member training, including drill critiques shall be maintained for 3 years in accordance with 10 CFR 50 Appendix R, Section I.4 requirements.

O.* Records of in-service inspections performed pursuant to these technical specifications.

P.* Records of Quality Assurance activities required by the QA Manual shall be maintained for the duration of the Operating License except those QA records relating to radioactive materials shipping packages, which shall be maintained for the lifetime of the packaging per 10 CFR 71.91(c) requirements.

Q.* Records of reviews performed for changes made to procedures or equipment, or reviews of tests and experiments pursuant to 10 CFR 50.59 and as required per Specification 15.6.5.1.6.

R.* Records of meetings of the Manager's Supervisory Staff and the Off-Site Review Committee.

S.* Records of Analyses for radiological environmental monitoring.

T. Records of radioactive material shipments having a specific activity of greater than 0.002 microcurie/gram shall be retained for a period of 2 years in accordance with 10 CFR 71.91(a).

U. Records concerning the Security Plan, procedures, testing, maintenance, and audit shall be maintained in accordance with the Commission-approved PBNP Modified Amended Security Plan.

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*Items will be retained for the duration of the Operating License.

A.01

15.7.8.4

See ITS 5.6.2

RETS Reporting Requirements

The following written reports shall be submitted to the Administrator, U.S. Nuclear Regulatory Commission Region III with a copy to the Director, Office of Inspection and Enforcement, USNRC, Washington, D.C. 20555 within the time periods specified.

A. Annual Monitoring Report

A report covering the operation of PBNP for the previous twelve (12) month period or fraction thereof, ending December 31, shall be submitted by May 1 of each year containing:

A.04 See ITS 5.6.2

1. RECP

Information relative to the quantities of radioactive liquid and gaseous effluents and solid radioactive waste released from PBNP during the reporting period. The information shall be consistent with the objectives outlined in the ODCM and RECM as well as Section IV.B.1 of Appendix I to 10 CFR 50 and 10 CFR 50.36a.

L.02

2. New and Spent Fuel Receipts and Shipments

a. Number and type of new fuel assemblies received during the reporting period, if any.

b. Number of spent fuel assemblies shipped off site during the reporting period, if any.

A.05 See ITS 5.6.2

3. REMP

Summaries and results from the REMP for the reporting period. The information shall be consistent with the objectives outlined in the EM and in sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

LB.03

4. Leak Testing of Sealed Sources

Results of required leak tests performed on sealed sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

L.03

5. Meteorological Data

Meteorological data shall be kept in file on site for review by the NRC upon request. The data available will include wind speed, wind direction and atmospheric stability. The data will be in the form of strip charts or hour-by-hour averages stored in electronic form for each of the parameters.

A.06

6. REMCAP Changes

A description of changes to the REMCAP, ODCM, ~~EM~~, RECM or PCP which were implemented and became ~~effective~~ during the reporting period shall be submitted ~~pursuant~~ to Specification 15.7.8.7.

LA.04

7. Special Circumstance Reports

   a. In accordance with note 7 to RECM Table 3-2, if the Waste Gas Holdup System Explosive Gas Monitor is out of service for greater than 14 days.

   b. In accordance with the EM, factors which render the LLDs unachievable.

   c. In accordance with the EM, failure of the analytical laboratory to participate in an Interlaboratory Comparison Program.

B. Measured Radioactivity Above Notification Levels

If the confirmed level of radioactivity remains above the notification levels specified in the EM, a written report describing the circumstance shall be prepared and submitted within thirty days of the confirmation that a notification level was exceeded.

C. Radioactive Liquid Effluent Treatment

If the radioactive liquid or gaseous effluent treatment system is inoperable and liquid or gaseous effluents are being discharged for 31 days without the treatment required to meet the release limits specified in the RECM, a special report shall be prepared and submitted to the Commission within thirty days which includes the following information:

1. Identification of the inoperable equipment or subsystem and the reason for inoperability.
2. Actions taken to restore the inoperable equipment to operable status.
3. Summary description of actions taken to prevent a recurrence.

D. Radioactive Effluent Releases

If the quantity of radioactive material actually released in liquid or gaseous effluents during any calendar quarter exceeds twice the quarterly limit as specified in the RECM, a special report shall be prepared and submitted to the Commission within thirty days of determination of the release quantity.

15.7.8.5 Major Change to Radioactive Liquid, Gaseous and Solid Waste Treatment Systems

Licensee initiated major changes to the radioactive waste treatment systems (liquid, gaseous, and solid) shall be reported to the U.S. Nuclear Regulatory Commission with the annual update to the FSAR for the period in which the major change was complete. The discussion of each change shall include:

See Spec 5.5

A. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;
B. Information necessary to support the reason for the change;
C. A description of the equipment, components and processes involved and the interfaces with other plant systems;
D. An evaluation of the change, which shows how the predicted releases of radioactive materials in liquid effluents and gaseous effluents and/or quantity of solid waste will differ from those previously predicted in the license application and amendments thereto;
E. An evaluation of the change, which shows the expected maximum exposures to an individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;
F. An estimate of the exposure to plant operating personnel as a result of the change.

15.7.8.6 Record Retention

LA.03

Record of reviews performed for changes made to the REMCAP Manual and to the following REMCAP components; the EM, RECM, ODCM, and PCP; shall be kept for the duration of the operating licenses of Units 1 and 2 of the Point Beach Nuclear Plant.

15.7.8.7 Revisions

See Spec 5.5

A. Process Control Program

Revisions to the PCP shall be documented and records of reviews performed for the revision shall be retained as required by 15.7.8.6. The documentation shall contain:

1. Information sufficient to support the change together with the appropriate analyses or evaluations justifying the changes(s), and
2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

Insert 5.6-1:

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)	LCO 2.1.1,	"Safety Limits (SLs)"
(2)	LCO 3.1.1,	"Shutdown Margin (SDM)"
(3)	LCO 3.1.3,	"Moderator Temperature Coefficient (MTC)"
(4)	LCO 3.1.5,	"Shutdown Bank Insertion Limits"
(5)	LCO 3.1.6,	"Control Bank Insertion Limits"
(6)	LCO 3.2.1,	"Heat Flux Hot Channel Factor ($F_Q(Z)$)"
(7)	LCO 3.2.2,	"Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)"
(8)	LCO 3.2.3,	"Axial Flux Difference (AFD)"
(9)	LCO 3.3.1,	"Reactor Protection System (RPS) Instrumentation - Overtemperature ΔT"
(10)	LCO 3.3.1,	"Reactor Protection System (RPS) Instrumentation - Overpower ΔT"
(11)	LCO 3.4.1,	"RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"
(12)	LCO 3.9.1	"Boron Concentration"

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) WCAP-14449-P-A, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWR's with Upper Plenum Injection," Revision 1, October 1999. (cores containing 422V+ fuel)

(2) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

(3) WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

(4) WCAP-14787-P, "Westinghouse Revised Thermal Design Procedures Instrument Uncertainty Methodology, Wisconsin Electric Power Company, Point Beach Unit 1 and 2," April 1999 (approved by NRC Safety Evaluation, February 8, 2000).

(5) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985.

(6) WCAP-10054-P-A, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Addendum 2, Revision 1, July 1997.

(7) WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

(8) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control," Revision 1A, February 1994.

(9) WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addenda, December 1988. (cores not containing 422 V+ fuel)

(10) WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions," Volume 1, Addendum 4, August 1990. (cores not containing 422 V+ fuel)

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 SDM, 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.

Insert 5.6-2:

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

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, LTOP enabling, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(1)	LCO 3.4.3,	"RCS Pressure and Temperature (P/T) Limits"
(2)	LCO 3.4.6,	"RCS Loops-MODE 4"
(3)	LCO 3.4.7,	"RCS Loops-MODE 5, Loops Filled"
(4)	LCO 3.4.10,	"Pressurizer Safety Valves"
(5)	LCO 3.4.12,	"Low Temperature Overpressure Protection (LTOP)"

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 documents: **This information will be submitted to the NRC for approval under Technical Specification Change Request (TSCR) 219. After NRC approval of TSCR 219, a supplement to this section will be submitted to identify the necessary approval amendments.**

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

Insert 5.6-3:

PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

Justification For Deviations - NUREG-1431 Section 5.06

15-Mar-00

JFD Number	JFD Text	
01	The brackets have been removed and the proper plant specific information has been provided. Point Beach is a two (2) Unit site, so these notes apply to Point Beach and are adopted in the proposed ITS.	
	ITS:	**NUREG:**
	SPEC 5.06.01	SPEC 5.06.01
	SPEC 5.06.02	SPEC 5.06.02
02	The proposed ITS title of section 5.6.2 was changed from "Annual Radiological Environmental Operating Report" to "Annual Monitoring Report" to be consistent with the current licensing basis title of this report and station procedures.	
	ITS:	**NUREG:**
	SPEC 5.06.02	SPEC 5.06.02
03	The submittal due date in proposed ITS 5.6.2 was changed from "May 15" to "April 30" to be consistent with the current licensing basis of this report and station procedures.	
	ITS:	**NUREG:**
	SPEC 5.06.02	SPEC 5.06.02
04	NUREG-1431 item 5.6.2 has a backeted item which states "[in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]." This bracketed item will not be adopted in the proposed Point Beach ITS. The format of this report is not part of the CTS. This report will follow the formating specified in the applicable Point Beach station procedures and will not be described in the ITS.	
	ITS:	**NUREG:**
	SPEC 5.06.02	SPEC 5.06.02

Justification For Deviations - NUREG-1431 Section 5.06

15-Mar-00

JFD Number	JFD Text
05	NUREG-1431 item 5.6.3 "Radioactive Effluent Report" is being combined into proposed ITS 5.6.2 "Annual Monitoring Report". Therefore, NUREG-1431 item 5.6.3 was reworded so that it now becomes an attribute that is to be included in proposed ITS 5.6.2. The note in NUREG-1431 item 5.6.3 was also engrained into the wording of the requirements of the "Radioactive Effluent Release Report." This change was necessary so that the attributes in NUREG-1431 item 5.6.3 could be included in one report (proposed ITS 5.6.2). Accordingly, to reflect the deletion of NUREG-1431 item 5.6.3, renumbering of the subsequent sections was necessary. This change is administrative.

ITS:	NUREG:
SPEC 5.06.02	SPEC 5.06.03
SPEC 5.06.03	SPEC 5.06.04
SPEC 5.06.04	SPEC 5.06.05
SPEC 5.06.05	SPEC 5.06.06
SPEC 5.06.06	SPEC 5.06.08
SPEC 5.06.07	SPEC 5.06.09
SPEC 5.06.08	SPEC 5.06.10

JFD Number	JFD Text
06	Brackets have been removed and the appropriate plant specific information has been provided.

ITS:	NUREG:
SPEC 5.06.04.A	SPEC 5.06.05.A
SPEC 5.06.04.B	SPEC 5.06.05.B
SPEC 5.06.05.A	SPEC 5.06.06.A
SPEC 5.06.05.B	SPEC 5.06.06.B
SPEC 5.06.06	SPEC 5.06.08
SPEC 5.06.08	SPEC 5.06.10

JFD Number	JFD Text
07	LTOP "arming" in proposed ITS section 5.6.5 (PTLR) was changed to LTOP "enabling", for consistency with Point Beach current licensing basis (CLB) terminology.

ITS:	NUREG:
SPEC 5.06.05.A	SPEC 5.06.06.A

Justification For Deviations - NUREG-1431 Section 5.06

15-Mar-00

JFD Number	JFD Text	
08	The reviewer notes in NUREG-1431 item 5.6.6 have not been adopted in the proposed ITS.	
	ITS:	**NUREG:**
	SPEC 5.06.05	SPEC 5.06.06
09	Included in CTS 15.4.4.II.D are the requirements for containment tendon surveillance reports. These current licensing basis requirements will be retained in whole in the proposed ITS 5.6.7, and the requirements in NUREG-1431 item 5.6.9 will not be adopted.	
	ITS:	**NUREG:**
	SPEC 5.06.07	SPEC 5.06.09
10	The sumittal date for the monthly report was changed to the 10th from the NUREG-1431 requirement of the 15th to be consistent with the current licensing basis (CTS 15.06.09.01.C.02).	
	ITS:	**NUREG:**
	SPEC 5.06.03	SPEC 5.06.04

5.0 ADMINISTRATIVE CONTROLS

1 → A single submittal may be made that combines sections common to Units 1 and 2.

5.6 Reporting Requirements

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

5.6.1 Occupational Radiation Exposure Report

~~------------------------------NOTE------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.
--~~

Replace with Insert 5.0-03

Approved TSTF-152, R0

~~A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year. [The initial report shall be submitted by April 30 of the year following the initial criticality.]~~

2 → Monitoring

5.6.2 Annual ~~Radiological Environmental Operating~~ Report

~~------------------------------NOTE------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.
--~~

The Annual ~~Radiological Environmental Operating~~ Report covering the operation of the unit 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

April 30

3

5.6 Reporting Requirements

5.6.2 Annual ~~Radiological Environmental Operating~~ Report (continued)

2

Monitoring

(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 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].~~ ~~[The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.]~~ 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.

4

Approved TSTF-348, R0

~~5.6.3 Radioactive Effluent Release Report~~

Approved TSTF-152, R0

shall

in the previous year

prior to May 1 of each year

~~------------------------------NOTE------------------------------~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~
~~--~~

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

Replace with Insert 5.0-13

5

5.6 Reporting Requirements

Approved TSTF-258, R4

5.6.~~4~~ 3 5 4 **Monthly Operating Reports**

Routine reports of operating statistics and shutdown experience~~[including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves,]~~ shall be submitted on a monthly basis ~~no later than the 15th of~~ each month following the calendar month covered by the report.

by the 10th 10

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:

6 Insert 5.0-08

[The individual specifications that address core operating limits must be referenced here.]

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:

Insert 5.0-09

[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

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 SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

5 5

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

5.6.~~6~~ **Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)**

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic

5.6 Reporting Requirements

7 enabling

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

6 Insert 5.0-10

, LTOP arming, and PORV ~~lift settings~~

testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following: [The individual specifications that address RCS pressure and temperature limits must be referenced here.]

Approved TSTF-233, R0

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 documents: [Identify the NRC staff approval document by date.]

6 Insert 5.0-11

6. LTOP ~~arming~~ temperature limit development methodology.

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

8

[Reviewers' Notes: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) S ystem lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.]

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

8

[6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value (2s_) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase RT_{NDT} + 2s_), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.]

Approved
TSTF-37, R2

5.6.7 EDG Failure Report

[If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement.]

5.6.~~8~~ 7 PAM Report

3 6

When a report is required by Condition B or G of LCO 3.3.[3], "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

6

5

5.6 Reporting Requirements (continued)

9 Insert 5.0-15

7 ~~8~~

Approved TSTF-37, R2

5.6.~~9~~ Tendon Surveillance Report

~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.~~

8 ~~9~~

5

5.6.~~10~~ [Steam Generator Tube Inspection Report

Reviewer's Note: Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.]

[Reviewer's Note: These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.]

Insert 5.0-12

6

Section 5.0 Inserts

Insert 5.0-03:

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person - rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

Section 5.0 Inserts

Insert 5.0-8:

(1)	LCO 2.1.1,	"Safety Limits (SLs)"
(2)	LCO 3.1.1,	"Shutdown Margin (SDM)"
(3)	LCO 3.1.3,	"Moderator Temperature Coefficient (MTC)"
(4)	LCO 3.1.5,	"Shutdown Bank Insertion Limits"
(5)	LCO 3.1.6,	"Control Bank Insertion Limits"
(6)	LCO 3.2.1,	"Heat Flux Hot Channel Factor ($F_Q(Z)$)"
(7)	LCO 3.2.2,	"Nuclear Enthalpy Rise Hot Channel Factor($F^N_{\Delta H}$)"
(8)	LCO 3.2.3,	"Axial Flux Difference (AFD)"
(9)	LCO 3.3.1,	"Reactor Protection System (RPS) Instrumentation - Overtemperature ΔT"
(10)	LCO 3.3.1,	"Reactor Protection System (RPS) Instrumentation - Overpower ΔT"
(11)	LCO 3.4.1,	"RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"
(12)	LCO 3.9.1	"Boron Concentration"

Section 5.0 Inserts

Insert 5.0-9:

(1) WCAP-14449-P-A, " Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWR's with Upper Plenum Injection," Revision 1, October 1999. (cores containing 422V+ fuel)

(2) WCAP-9272-P-A, " Westinghouse Reload Safety Evaluation Methodology," July 1985.

(3) WCAP-11397-P-A, " Revised Thermal Design Procedure," April 1989.

(4) WCAP-14787-P, " Westinghouse Revised Thermal Design Procedures Instrument Uncertainty Methodology, Wisconsin Electric Power Company, Point Beach Unit 1 and 2," April 1999 (approved by NRC Safety Evaluation, February 8, 2000).

(5) WCAP-10054-P-A, " Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985.

(6) WCAP-10054-P-A, " Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Addendum 2, Revision 1, July 1997.

(7) WCAP-8745-P-A, " Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

(8) WCAP-10216-P-A, " Relaxation of Constant Axial Offset Control," Revision 1A, February 1994.

(9) WCAP-10924-P-A, " Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addenda, December 1988. (cores not containing 422 V+ fuel)

(10) WCAP-10924-P-A, " LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions," Volume 1, Addendum 4, August 1990. (cores not containing 422 V+ fuel)

Section 5.0 Inserts

Insert 5.0-10:

(1) LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"
(2) LCO 3.4.6, "RCS Loops-MODE 4"
(3) LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled"
(4) LCO 3.4.10, "Pressurizer Safety Valves"
(5) LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)"

Section 5.0 Inserts

Insert 5.0-11:

This information will be submitted to the NRC for approval under Technical Specification Change Request (TSCR) 219. After NRC approval of TSCR 219, a supplement to this section will be submitted to identify the necessary approval amendments.

Section 5.0 Inserts

Insert 5.0-12:

Steam Generator Tube Inspector Report

(a) After each inservice examination, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission as soon as practicable.

(b) The complete results of the steam generator tube inservice inspection shall be included in a Report for the period in which the inspection was completed.

Reports shall include:

1. Number and extent of tubes inspected.
2. Location and percent of all thickness penetration for each indication.
3. Identification of tubes plugged or repaired.

(c) Reports required by Table 5.5.8-1, "Steam Generator Tube Inspection," shall provide the information required by Specification 5.6.8.(b) and a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. The report shall be submitted to the Commission prior to resumption of plant operation.

Section 5.0 Inserts

Insert 5.0-13:

The Annual Monitoring Report shall also include The Radioactive Effluent Release Report covering the operation of the units in the previous year and submitted in accordance with 10 CFR 50.36a.

The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the units. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

Section 5.0 Inserts

Insert 5.0-15:

Abnormal conditions observed during testing will be evaluated to determine the effect of such conditions on containment structural integrity. This evaluation should be completed within 30 days of the identification of the condition. Any condition which is determined in this evaluation to have a significant adverse effect on containment structural integrity will be considered an abnormal degradation of the containment structure.

Any abnormal degradation of the containment structure identified during the engineering evaluation of abnormal conditions shall be reported to the Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

NSHC Number	NSHC Text
A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
L.01	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change results in the deletion of an administrative reporting requirement for the main steam line radiation monitor (SA-11). During the conversion of NUREG-1431 section 3.3.3 (Post Accident Monitoring Instrumentation) to the proposed ITS for PBNP, operability requirements for the steam line radiation monitor (SA-11) were not retained. This was based on the fact that this monitor is not identified as Type A or Category I in the PBNP Regulatory Guide 1.97 analyses, and therefore does not need to be included in the ITS. Therefore, because this monitor was not retained in proposed ITS 3.3.3, the report requirements for its inoperability will not be retained in proposed ITS 5.6. Deleting administrative special report requirements does not involve a significant increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change of deleting administrative reporting requirements does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? Deleting administrative reporting requirements has no bearing on any margin of safety. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.02	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change results in the deletion of an administrative reporting requirement for new and spent fuel receipts and shipments. The deletion of an administrative reporting requirement does not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revision. The proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to this administrative reporting requirement. All Limiting Conditions of Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, this change does not increase the probability of previously evaluated accidents. This change does not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed change. Existing system and component redundancy and operation is not being changed by this proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, this change does not affect the consequences of previously evaluated accidents. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change of deleting administrative reporting requirements does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? Deleting administrative reporting requirements has no bearing on any margin of safety. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.03	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change results in the deletion of administrative requirements for having meteorological data kept on file on site as a subsection of the "Annual Monitoring Report (AMR)." The deletion of an administrative reporting requirement does not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revision. The proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to this administrative requirement. All Limiting Conditions of Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, this change does not increase the probability of previously evaluated accidents. This change does not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed change. Existing system and component redundancy and operation is not being changed by this proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, this change does not affect the consequences of previously evaluated accidents. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change of deleting this administrative requirement does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? Deleting this administrative requirement has no bearing on any margin of safety. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.04	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change results in the deletion of an administrative reporting requirement for low temperature overpressure protection system operation. The deletion of an administrative reporting requirement does not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revision. The proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to this administrative reporting requirement. All Limiting Conditions of Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, this change does not increase the probability of previously evaluated accidents. This change does not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed change. Existing system and component redundancy and operation is not being changed by this proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, this change does not affect the consequences of previously evaluated accidents. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change of deleting administrative reporting requirements does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? Deleting administrative reporting requirements has no bearing on any margin of safety. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
LA	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

NSHC Number	NSHC Text
LB	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change involves deletion of a Specifications/information which is duplicative of information contained in the Code of Federal Regulations (CFRs). This information is more appropriately addressed by the CFRs and serves no purpose in the Technical Specifications. Deletion of this information will not result in an increase in the probability of an accident. Regulatory requirements do not alter plant design or configuration; therefore, this does not alter any event precursor. Accordingly, there will be no effect on the consequences of any accident. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change deletes materials from the Technical Specifications which are adequately addressed in the CFRs. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change deletes materials from the Technical Specifications which are duplicative of requirements contained in the CFRs. These items are not an input to any accident analysis and, therefore, have no impact on margin of safety.

NSHC Number	NSHC Text
M	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not affect any assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 Occupational Radiation Exposure Report

-----NOTE-----
A single submittal may be made that combines sections common to Units 1 and 2.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person – rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Monitoring Report

-----NOTE-----
A single submittal may be made that combines sections common to Units 1 and 2.

The Annual Monitoring Report covering the operation of the units during the previous calendar year shall be submitted by April 30 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.

5.6 Reporting Requirements

5.6.2 Annual Monitoring Report (continued)

The Annual Monitoring Report shall include 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 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.

The Annual Monitoring Report shall also include The Radioactive Effluent Release Report covering the operation of the units in the previous year and submitted in accordance with 10 CFR 50.36a.

The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the units. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.3 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis by the 10th of each month following the calendar month covered by the report.

5.6.4 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) LCO 2.1.1, "Safety Limits (SLs)"
(2) LCO 3.1.1,"Shutdown Margin (SDM)"
(3) LCO 3.1.3, "Moderator Temperature Coefficient (MTC)"
(4) LCO 3.1.5,"Shutdown Bank Insertion Limits"
(5) LCO 3.1.6,"Control Bank Insertion Limits"
(6) LCO 3.2.1,"Heat Flux Hot Channel Factor ($F_Q(Z)$)"

5.6 Reporting Requirements

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) LCO 3.2.2,"Nuclear Enthalpy Rise Hot Channel Factor($F^N_{\Delta H}$)"
(8) LCO 3.2.3, "Axial Flux Difference (AFD)"
(9) LCO 3.3.1,"Reactor Protection System (RPS) Instrumentation - Overtemperature ΔT"
(10) LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation - Overpower ΔT"
(11) LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"
(12) LCO 3.9.1 "Boron Concentration"

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) WCAP-14449-P-A, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWR's with Upper Plenum Injection," Revision 1, October 1999. (cores containing 422V+ fuel)
(2) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
(3) WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.
(4) WCAP-14787-P, "Westinghouse Revised Thermal Design Procedures Instrument Uncertainty Methodology, Wisconsin Electric Power Company, Point Beach Unit 1 and 2," April 1999 (approved by NRC Safety Evaluation, February 8, 2000).
(5) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985.
(6) WCAP-10054-P-A, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Addendum 2, Revision 1, July 1997.
(7) WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.
(8) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control," Revision 1A, February 1994.
(9) WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addenda, December 1988. (cores not containing 422 V+ fuel)
(10) WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions," Volume 1, Addendum 4, August 1990. (cores not containing 422 V+ fuel)

5.6 Reporting Requirements

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (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 SDM, 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.

5.6.5 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, LTOP enabling, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(1) LCO 3.4.3, “RCS Pressure and Temperature (P/T) Limits”
(2) LCO 3.4.6,“RCS Loops-MODE 4”
(3) LCO 3.4.7,“RCS Loops-MODE 5, Loops Filled”
(4) LCO 3.4.10,“Pressurizer Safety Valves”
(5) LCO 3.4.12,“Low Temperature Overpressure Protection (LTOP)”

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 documents:
[This information will be submitted to the NRC for approval under Technical Specification Change Request (TSCR) 219. After NRC approval of TSCR 219, a supplement to this section will be submitted to identify the necessary approval amendments.]

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

5.6 Reporting Requirements

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Abnormal conditions observed during testing will be evaluated to determine the effect of such conditions on containment structural integrity. This evaluation should be completed within 30 days of the identification of the condition. Any condition which is determined in this evaluation to have a significant adverse effect on containment structural integrity will be considered an abnormal degradation of the containment structure.

Any abnormal degradation of the containment structure identified during the engineering evaluation of abnormal conditions shall be reported to the Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

(a) After each inservice examination, the number of tubes plugged or repaired in each steam generator shall be reported to the commission as soon as practicable.

(b) The complete results of the steam generator tube inservice inspection shall be included in a report for the period in which the inspection was completed.

Reports shall include:

1. Number and extent of tubes inspected.
2. Location and percent of all thickness penetration for each indication.
3. Identification of tubes plugged or repaired.

5.6 Reporting Requirements

5.6.8 Steam Generator Tube Inspection Report (continued)

(c) Reports required by Table 5.5.8-1, "Steam Generator Tube Inspection," shall provide the information required by Specification 5.6.8.(b) and a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. The report shall be submitted to the Commission prior to resumption of plant operation.

Cross-Reference Report - NUREG-1431 Section 5.07

ITS to CTS

15-Mar-00

ITS	CTS	DOC
SPEC 5.04.01.A	15.06.11	LB.02
SPEC 5.07	15.06.11	LB.01
SPEC 5.07.01	15.06.11	A.01
SPEC 5.07.02	15.06.11	A.01

Cross-Reference Report - NUREG-1431 Section 5.07
CTS to ITS

15-Mar-00

CTS	ITS	DOC
15.06.11	SPEC 5.04.01.A	LB.02
	SPEC 5.07	LB.01
	SPEC 5.07.01	A.01
	SPEC 5.07.02	A.01

Description of Changes - NUREG-1431 Section 5.07

15-Mar-00

DOC Number	DOC Text
A.01	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.06.11	SPEC 5.07.01
	SPEC 5.07.02

DOC Number	DOC Text
LB.01	CTS 15.6.11 states that the radiation protection program shall meet the requirements of 10 CFR 20. This statement will not be retained in the proposed ITS because the requirement is duplicative of the code of federal regulations (10 CFR Part 20), which all licensees are required to meet. Accordingly, this change removes CTS details that are duplicative of other regulatory requirements.

CTS:	ITS:
15.06.11	SPEC 5.07

DOC Number	DOC Text
LB.02	CTS 15.6.11 states that radiological control procedures shall be written and made available to all station personnel, and shall state permissible radiation exposure levels. This statement will not be retained in the proposed ITS because the requirement to have radiological control procedures that state permissible exposure levels is duplicative of the code of federal regulations (10 CFR Part 20), which all licensees are required to meet. Accordingly, this change removes CTS details that are duplicative of other regulatory requirements. The statement "and made available to all station personnel" is inherent to all PBNP station procedures and is, therefore, unnecessary in the proposed ITS.

CTS:	ITS:
15.06.11	SPEC 5.04.01.A

15.6.11 RADIATION PROTECTION PROGRAM

Specification

LB.02 LB.01

Radiological control procedures shall be written ~~and made available~~ to all station personnel, ~~and shall state permissible~~ radiation exposure levels. ~~The radiation protection program shall meet the requirements of 10 CFR 20.~~

Paragraph 20.1601 - Control of Access to High Radiation Areas

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from Any Surface Penetrated by the Radiation;

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures (e.g., health physics technicians) and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are following plant radiation protection procedures for entry to, exit from, and work in such area.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or
2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
3. A radiation monitoring device that continuously transmits dose rate and cumulative dose to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

See ITS 5.7.2
See ITS 5.7.1

A.01

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from Any Surface Penetrated by the Radiation, but Less than 500 rads/hour at 1 Meter from the Radiation Source or from Any Surface Penetrated by the Radiation:

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door or gate that prevents unauthorized entry, and, in addition:
   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.
   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,
   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area, or

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individual qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded and conspicuous, clearly visible flashing light shall be activated at the area as a warning device.

Justification For Deviations - NUREG-1431 Section 5.07

15-Mar-00

JFD Number	JFD Text	
01	The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	SPEC 5.07	SPEC 5.07

01

High Radiation Area 5.7

5.0 ADMINISTRATIVE CONTROLS

Approved TSTF-258, R4

Replace with Insert 5.0-05

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., [Health Physics Technicians]) 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 plant 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 Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in

01

[High Radiation Area]

[5.7]

Approved TSTF-258, R4

Replace with Insert 5.0-05

5.7.2 (continued)

the immediate work areas and the maximum allow able 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.

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.

Section 5.0 Inserts

Insert 5.0-05:

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 <u>High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation</u>

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation</u>

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

Section 5.0 Inserts

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where option (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

No Significant Hazards Considerations - NUREG-1431 Section 5.07

15-Mar-00

NSHC Number	NSHC Text
A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 5.07

15-Mar-00

NSHC Number	NSHC Text
LB	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change involves deletion of a Specifications/information which is duplicative of information contained in the Code of Federal Regulations (CFRs). This information is more appropriately addressed by the CFRs and serves no purpose in the Technical Specifications. Deletion of this information will not result in an increase in the probability of an accident. Regulatory requirements do not alter plant design or configuration; therefore, this does not alter any event precursor. Accordingly, there will be no effect on the consequences of any accident. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change deletes materials from the Technical Specifications which are adequately addressed in the CFRs. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change deletes materials from the Technical Specifications which are duplicative of requirements contained in the CFRs. These items are not an input to any accident analysis and, therefore, have no impact on margin of safety.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where option (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.