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NUREG-1625

Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants

Draft Report for Comment

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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Division of Reactor Program Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

COMMENTS ON DRAFT REPORT

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ABSTRACT

This NUREG report describes the staff's proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants (STS PDW). The report includes a detailed discussion of the strategy followed for determining the contents of the STS PDW. The proposed STS PDW is being published to provide the general public and the nuclear community with an opportunity for comment.

The contents of the proposed STS PDW are based primarily on the Standard Technical Specifications, Westinghouse Plants (NUREG-1431, Revision 1, April 1995), which in turn were based on the criteria in the Nuclear Regulatory Commission (NRC) Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (SECY-93-067, 58 FR 39132, July 22, 1993). The proposed STS PDW reflect the experience gained in the development of the Permanently Defueled Technical Specifications (PDTS) for the Trojan Nuclear Plant, the first PDTS approved by the NRC that were based on the improved STS for Westinghouse Plants. Differences between the STS PDW and the PDTS for the Trojan Nuclear Plant were necessitated by the assumption that a licensee may wish to use the STS PDW in the conversion from existing Technical Specifications as early as 1 month after permanent shutdown of the reactor. At 1 month, the fission product inventory of irradiated fuel in the spent fuel storage pool would make it necessary to maintain the Fuel Building Air Cleaning System in an operable condition. As licensees begin to plan permanent shutdown of their nuclear power plants, they are encouraged to adopt the STS PDW to an extent that is practical and consistent with their licensing basis.

CONTENTS

Bases

SELECTED ABBREVIATIONS

COT	CHANNEL OPERATIONAL TEST
CFR	Code of Federal Regulations
CREFS	Control Room Emergency Filtration System
DBA	Design Basis Accident
DG	diesel generator
ESF	engineered safety feature
FBACS	Fuel Building Air Cleanup System
FSAR	Final Safety Analysis Report
GDC	General Design Criterion
HEPA	high-efficiency particulate air
IRAC	Independent Review and Audit Committee
LER	Licensee Event Report
LCO	Limiting Condition for Operation
LMITCO	Lockheed Martin Idaho Technologies Company
LOP	loss of power
MDR	maximum density racks
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
PDTS	Permanently Defueled Technical Specifications
PGE	Portland General Electric Company
PCP	Process Control Program
PWR	pressurized water reactor
QA	quality assurance
RWP	Radiation Work Permit
RTD	resistance temperature detector
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SL	Safety Limit
STS	Standard Technical Specifications
STS PDW	Standard Technical Specifications for Permanently Defueled Westinghouse Plants
SS	Shift Supervisor
SR	Surveillance Requirement
TS	Technical Specifications
TADOT	TRIP ACTUATING DEVICE OPERATIONAL TEST
VFTP	Ventilation Filter Testing Program
WOG STS	Standard Technical Specifications, Westinghouse Plants

DEVELOPMENT STRATEGY

Introduction

Given the number of nuclear power plants nearing end of life, the Nuclear Regulatory Commission (NRC) has recognized the need for generic guidance on appropriate Technical Specifications for permanently shutdown power reactors. Accordingly, in FY-1997 the NRC staff proposed a plan for developing such guidance based on recently approved Permanently Defueled Technical Specifications (PDTS) and on experience with the development and implementation of the PDTS.

The first task assigned under this project was to prepare the Standard Technical Specifications for Permanently Defueled Westinghouse Plants (STS PDW). The project team was directed to make full use of the improved Standard Technical Specifications, Westinghouse Plants (WOG STS); pertinent NRC regulations; and PDTS documents that have been approved by the NRC—particularly the PDTS developed by Portland General Electric Company (PGE) for the Trojan Nuclear Plant, the only PDTS currently approved by the NRC that has used the improved STS format. In addition, the project team was instructed to prepare an interview plan and conduct interviews with knowledgeable NRC staff members and licensee personnel to gain a perspective on the strengths and weaknesses of implemented PDTS. The results of this effort are published in this NUREG-series report as the proposed STS PDW.

Background

This section summarizes key considerations related to the design and preparation of the WOG STS, reviews implications of those considerations in the preparation of the proposed STS PDW, and addresses public safety implications associated with a power reactor's transition from operating status to a permanently shutdown and defueled condition.

On February 6, 1987, the NRC issued its Policy Statement on Technical Specification Improvements for interim use and public comment. The policy statement established a set of three objective criteria for use in defining requirements that should be controlled by Technical Specifications (see SECY-87-269). These criteria, plus a more recently developed fourth criterion, were reiterated in the Commission's Final Policy Statement on Technical Specification Improvements, SECY-93-067. The Commission recognized that—

> . . . the four criteria carried a theme of focusing on the technical requirements for features of controlling importance to safety. Since many of the requirements are of immediate concern to the health and safety of the public, this Policy Statement adopts, for the purpose of relocating requirements from Technical Specifications to licensee-controlled documents, the subjective statement of the purpose of Technical Specifications expressed by the Atomic Safety and Licensing Appeal Board

in Portland General Electric Company (Trojan Nuclear Plant), ALAB-531, 9 NRC 263 (1979). There the Appeal Board interpreted Technical Specifications as being reserved for those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

Thus, the Commission established the principle that Technical Specifications are intended to impose conditions or limitations on reactor operation necessary to avoid an "immediate threat to public health and safety." This principle is stated in the first paragraph of the Commission's Final Policy Statement.

In terms of the potential accidents considered when a reactor is licensed to operate, a permanently defueled reactor does not pose a threat of comparable magnitude. The sole function of a permanently defueled facility is to store irradiated fuel under water in a quiescent state. The stored fuel does not require the constant observation and attention so necessary for responsible reactor operations. Moreover, the potential fission product source term for an accidental release is orders of magnitude smaller than that of a reactor operating at a power level of approximately 3000 MW. With no pressurized reactor system or significant stored energy available to act as a driving force, the potential for dispersal of fission product in a fuel storage facility is greatly reduced. Because of differences in the threat posed to public health and safety by the two types of facilities, the reactor and its primary system must be housed within a containment building, whereas the spent reactor fuel can be safely stored in a confinement building that has air filtration capability.

With respect to criticality concerns in a fuel storage pool, there are few ways in which an accidental criticality could occur. The fuel array in the fuel storage pool is subcritical by at least 5% $\Delta K/K$. There are no credible means by which sufficient reactivity can be added to the storage pool to cause such an event. Criticality on prompt neutrons alone is even more remote. Where the potential exists for accidental criticality, however, protection is afforded by the required presence of dissolved boron in the fuel storage pool.

Once a reactor has been permanently defueled, public safety depends primarily on maintaining safe storage of the irradiated fuel, which in turn relies on the attentiveness of the facility staff to ensure that the fuel remains covered by an ample amount of water. This attention must include routine monitoring of key parameters (e.g., water level, water temperature, and boron concentration) that could indicate the need for action to avoid a threat to fuel integrity. The warning, either by observation or annunciation, must allow sufficient time to take the appropriate action.

The most probable action required will be the addition of makeup water to the fuel pool. Loss of water may be caused by accidental leakage from the pool system or by the loss of forced cooling, which would result in pool heatup and the loss of water by boiling and evaporation. A worst-case scenario would involve the loss of forced cooling for a fuel storage pool containing the entire core of a recently shutdown reactor and having a large inventory of previously stored fuel. For such an event, it is estimated that the water loss attributable to boiling and evaporation would reduce the pool water level to a height of 10 feet above the top of the stored fuel in about 50 hours. Hence, despite the fact that the sense of "immediacy" is removed from the threat, it is clear that monitoring is still required. The increased time available for detection and for taking corrective action means that surveillance intervals can be longer. Since very few parameters need to be monitored to maintain the water in the fuel pool at a specified level and to ensure availability of the cooling system, the staff workload for this activity is greatly reduced. For the workload remaining, periodic eyes-on surveillance, annunciation of off-normal conditions, and staff attention and corrective action can be used in lieu of constant and intense monitoring, automatic actions, and immediate operator actions required by emergency operating procedures during reactor operation.

Some relocation and movement of stored irradiated fuel may occasionally be necessary, creating the potential for an event involving the accidental dropping of a fuel assembly and the release of fission products to the storage pool and building atmosphere. For the first few months after final reactor shutdown, the fission product inventory of each recently stored fuel assembly will be sufficient to require continued OPERABILITY of the Fuel Building Air Cleanup System (FBACS). By 6 months after reactor shutdown, the decay of shorter lived fission products with the potential to become airborne (particularly radioiodine isotopes) will mean that FBACS OPERABILITY is no longer required.

It is probable that all fuel assemblies in the fuel storage pool will eventually be transferred to a dry storage facility, at which time there will no longer be a need for Section 2 or 3 of the STS PWD (i.e., Safety Limits and Limiting Conditions for Operation). The contents of the Technical Specifications will then depend on the status of the facility (i.e., whether the licensee has chosen the SAFSTOR or DECON option). Regardless of the option chosen, appropriate requirements for the Technical Specification can be specified within the ADMINISTRATIVE CONTROLS Section.

Technical Approach

The first task assigned under the current project required that each team member become familiar with several pertinent documents, including the following:

- NUREG-1431, "Standard Technical Specifications, Westinghouse Plants";
- 10 CFR 50.36, 10 CFR 50.75, and 10 CFR 50.82 (plus amendments related to decommissioning that were issued on July 29, 1996);
- NRC-approved PDTS and NRC Safety Evaluation Report for the Trojan Nuclear Plant; and
- Guidance concerning relevant contents for PDTS provided by the NRR Technical Specifications Branch (OTSB).

Based on the project team's understanding of these documents, a prototype for the draft STS PDW (Rev. 0) was assembled. This document included all sections of the WOG STS identified by the NRC staff as potentially relevant to the STS PDW and was supplemented by two sections of the Trojan PDTS. The prototype was then edited to remove subsections related solely to operating MODES, thereby producing Revision 1 of the draft.

The project team used the Trojan PDTS as a model for developing the proposed STS PDW based on recommendations by the NRC staff and on the favorable response of team members after reviewing the Trojan document. Accordingly, the team conducted a detailed comparison of the Trojan PDTS with the draft STS PDW (Rev. 1) to identify specific differences. Many sections under consideration in the prototype document had not been included in the Trojan PDTS. Also, a number of inconsistencies between the two documents were noted. These inconsistencies usually involved including restrictions specified in the Trojan PDTS through a Limiting Condition for Operation, a Required Action, or a Surveillance Requirement that was not found in the WOG STS. This information was used to develop an interview plan and to design interview questionnaires for each proposed class of interviewee (i.e., NRC Project Manager, NRC Regional Inspector, Licensee Developer, and Licensee Implementor). Questions were designed to accomplish the following:

- Gather specific information related to the scope and contents of a PDTS document;
- Determine how well the Trojan PDTS met the needs of NRC inspectors;
- Elicit the licensee's views on a wide range of issues, including but not limited to—
  - Strengths and weaknesses of the Trojan PDTS,
  - Technical rationale for PDTS requirements that were more conservative than the WOG STS,
  - The basis for the licensee's decision to adopt the WOG STS format for the Trojan PDTS, and
  - The potential value of an STS PDW document for licensees that have not converted to the improved STS, as well as the likelihood that such licensees would use the STS PDW.

Interviews were conducted over a 4-week period, with each interview documented immediately after its completion. The exception to this practice involved interviews at the Trojan Nuclear Plant, where the schedule did not facilitate development of separate interview records. In this case, a single composite record was prepared wherein recorded responses and observations were identified with specific interviewees. A report on the interview results was prepared for and transmitted to the NRC.

For the development of the Trojan PDTS, PGE used a modified form of the four criteria described in the Final Commission Policy Statement (published in SECY-93-067 and referred to in the Background section above). These criteria were used in determining which Trojan Technical Specifications should be retained in the PDTS. The project team regarded these criteria as logical and reasonable extensions of the criteria put forth in the Final Commission Policy Statement and adopted them for use on the STS PDW. A comparison of the criteria presented in the policy statement and the modified version used in the development of the Trojan PDTS—and then adopted for use in developing the proposed STS PDW—is presented in Table 1 below.

The information provided in Table 2 is intended to assist the reader in understanding the basis for decisions regarding the contents of the proposed STS PDW. The first column identifies all sections that were seriously considered for inclusion in the proposed STS PDW. The WOG STS document was the principal source of these sections. Exceptions were adopted from the Trojan PDTS and are so noted. The second column lists each of the Specifications included in the proposed STS PDW under its current number and identifies those Specifications not included. The third column specifies whether or not the Specification has been included in the Trojan PDTS, noting any substantial differences from the WOG STS. Also included here is a brief statement of the rational used whenever a WOG STS Specification was excluded from the proposed STS PDW or when a requirement that is substantially different from the WOG STS has been incorporated.

The most significant differences between the proposed STS PDW and the Trojan PDTS are directly attributable to differences in the time assumed to pass between final shutdown of a reactor and the effective date of the plant-specific PDTS. The contents of the proposed STS PDW were based on the assumption that conversion to a PDTS could occur as early as 1 month after shutdown, whereas the Trojan reactor had been shut down for more than 2 years at the time of PDTS conversion. In order to cover potential Design Basis Accidents that might require the use of mitigating equipment (e.g., a fuel handling accident requiring actuation of the FBACS), the proposed STS PDW adopted an Applicability time period of [6] months for several Limiting Conditions for Operation. (This time period was included in brackets because a plant-specific analysis could undoubtedly justify a substantially shorter Applicability timeframe.)

Revision 3 of the proposed STS PDW was prepared to reflect the project team's analysis of all information gathered to date. This draft was then subjected to an internal peer review, which resulted in a number of worthwhile suggestions that were incorporated into the proposed STS PDW presented in this report.

Table 1. Comparison of Final Commission Policy Statement Criteria with Trojan PDTS Criteria (and Adopted for the Proposed STS PDW)

Criterion	Final Commission Policy Statement	Trojan PDTS and Proposed STS PDW
1	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.	Installed instrumentation that is used to detect, and indicate in the control room, a significant degradation of the fuel storage pool integrity.
2	A process variable that is an initial condition of a design basis accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.	A process variable that is an initial condition of a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.	A structure, system, or component that functions or actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.	A structure, system, or component that operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

Table 2. Proposed STS PDW: Summary of Contents, Sources, and Rationale

WOG STS	Proposed STS PDW	Rationale
1.0 USE AND APPLICATION	**1.0 USE AND APPLICATION**	**Included in Trojan PDTS.**
1.1 Definitions	1.1 Definitions	A minimum number of definitions were included in Trojan PDTS. Definitions were retained only for those terms used in the proposed STS PDW.
1.2 Logical Connectors	1.2 Logical Connectors	A section on logical connectors was included in the Trojan PDTS.
1.3 Completion Times	1.3 Completion Times	A section on completion times was included in the Trojan PDTS.
1.4 Frequency	1.4 Frequency	A section on frequency was included in the Trojan PDTS.
2.0 SAFETY LIMITS (SLs)	**2.0 SAFETY LIMITS (SLs)**	**Not included in Trojan PDTS; however, the team concluded that the importance of keeping the fuel covered justified including Safety Limits.**
3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	**3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY**	**Included in Trojan PDTS.**
3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY	**3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY**	**Included in Trojan PDTS.**
3.3 INSTRUMENTATION	**3.1 INSTRUMENTATION**	**Not included in Trojan PDTS.**
3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	Not included in the proposed STS PDW.	Not included in Trojan PDTS. Reasons WOG STS Specification 3.3.5 was not included in the proposed STS PDW: - No actions identified that would require Control Room habitability at all times. Thus, the Control Room Emergency Filtration System (CREFS) is not required. - Should LOP occur, fuel movement within the spent fuel storage pool will be suspended, greatly reducing the likelihood of a fuel handling accident. Thus, OPERABILITY of the FBACS at all times is not imperative. - Consequently, DGs are not required to provide essential power to the CREFS or FBACS.

WOG STS	Proposed STS PDW	Rationale
3.3.7 Control Room Emergency Filtration System (CREFS) Actuation Instrumentation	Not included in the proposed STS PDW.	Reasons WOG STS Specification 3.3.7 was not retained: - OPERABILITY of the FBACS is required whenever fuel is being moved in the spent fuel storage pool or loads are being moved over fuel storage racks within [6] months after permanent shutdown of the reactor. Should a fuel handling accident occur, operation of the FBACS would ensure that the Control Room would remain habitable.
3.3.8 Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation	3.1.1 Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation	Not included in Trojan PDTS. Reasons WOG STS Specification 3.3.8 was included in the proposed STS PDW as Specification 3.1.1: - For several months after reactor shutdown, a fuel handling accident within the spent fuel storage pool could release radioactive fission products to the fuel building atmosphere sufficient to require operations of the FBACS. - Based on interview data, conversion to PDTS could be a soon as 30 days after reactor shutdown. - Thus, Applicability of this LCO is limited to [6] months after permanent shutdown of the reactor.
3.7 PLANT SYSTEMS	**3.2 PLANT SYSTEMS**	**Included in Trojan PDTS.**
3.7.10 Control Room Emergency Filtration System (CREFS)	Not included in the proposed STS PDW.	Not included in Trojan PDTS. Reasons WOG STS Specification 3.7.10 was not included in the proposed STS PDW: - No actions identified that would require the Control Room to be manned at all times. - OPERABILITY of the FBACS is required whenever fuel is being moved in the spent fuel storage pool or loads are being moved over fuel storage racks within [6] months after permanent shutdown of the reactor. Should a fuel handling accident occur, operation of the FBACS would ensure that the Control Room would remain habitable.
3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)	Not included in the proposed STS PDW.	Not included in Trojan PDTS. Reasons WOG STS Specification 3.7.11 was not included in the proposed STS PDW: - Control Room heat load is only a potential problem if its isolation is necessary. - Should a fuel handling accident occur, operation of the FBACS would avoid the need for isolation. - No actions identified that would require the Control Room to be manned at all times.

WOG STS	Proposed STS PDW	Rationale
3.7.13 Fuel Building Air Cleanup System (FBACS)	3.2.1 Fuel Building Air Cleanup System (FBACS)	Not included in Trojan PDTS. Reasons WOG STS Specification 3.7.13 was included in the proposed STS PDW as Specification 3.2.1: - For several months after reactor shutdown, a fuel handling accident within the spent fuel storage pool could release radioactive fission products to the fuel building atmosphere sufficient to require operations of the FBACS. - Based on interview data, conversion to PDTS could occur as soon as 30 days after reactor shutdown. - Thus, Applicability of this LCO is limited to [6] months after permanent shutdown of the reactor.
3.7.15 Fuel Storage Pool Water Level	3.2.2 Fuel Storage Pool Water Level	Trojan PDTS included an added restriction on moving loads over storage racks. Reason this restriction was included in Specification 3.2.2 of the proposed STS PDW: - Suspension of fuel movement is intended to avoid a fuel handling accident. - A dropped load on a fuel rack is also a potential cause of a fuel handling accident. - To be consistent, movement of loads over fuel racks should also be suspended.
Not included in the WOG STS.	3.2.3 Fuel Storage Pool Coolant Temperature	Included in Trojan PDTS. Including Specification 3.2.3 in the proposed STS PDW establishes a temperature limit and a Surveillance Frequency that will provide early indication of a major malfunction of the fuel storage pool water cooling system. This indication permits timely restoration of the cooling system and confirmation that alternate sources of makeup water are available should one be required.
Not included in the WOG STS.	3.2.4 Spent Fuel Storage Pool Load Restrictions	Included in Trojan PDTS. Including Specification 3.2.4 in the proposed STS PDW places a restriction on loads that can be carried over fuel storage racks, thus limiting the potential severity of a fuel handling accident as a result of an accidental load drop.
[3.7.16 Fuel Storage Pool Boron Concentration]	[3.2.5 Fuel Storage Pool Boron Concentration]	Included in Trojan PDTS. Including WOG STS Specification 3.7.16 in the proposed STS PDW as Specification 3.2.5 provides additional assurance that an accidental criticality will not occur in the fuel storage pool.
[3.7.17 Spent Fuel Assembly Storage]	[3.2.6 Spent Fuel Assembly Storage]	Included in Trojan PDTS. Including WOG STS Specification 3.7.17 in the proposed STS PDW as Specification 3.2.6 provides additional assurance that an accidental criticality will not occur in the fuel storage pool.

WOG STS	Proposed STS PDW	Rationale
3.8 ELECTRICAL POWER SYSTEMS	**3.3 ELECTRICAL POWER SYSTEMS**	**Not included in Trojan PDTS.**
3.8.2 AC Sources—Shutdown	3.3.1 AC Power Systems	Not included in Trojan PDTS. Including WOG STS Specification 3.3.1 in the proposed STS PDW as Specification 3.3.1 provides assurance that an AC source is available to power the FBACS should a fuel handling accident occur during movement of fuel or loads in the spent fuel storage pool. The OPERABILITY requirement applies during the first [6] months after final shutdown of the reactor.
3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air	Not included in the proposed STS PDW.	Not included in Trojan PDTS. Reasons WOG STS Specification 3.8.3 was not included in the proposed STS PDW: - See reasons provided for not including STS Specification 3.3.5, LOP DG Start Instrumentation. - No requirement for an emergency diesel generator was identified.
3.8.5 DC Sources—Shutdown	Not included in the proposed STS PDW.	Not included in Trojan PDTS. Reason WOG STS Specification 3.8.5 was not included in the proposed STS PDW: - The only requirement for a DC power source identified is for the FBACS control instrumentation, which is covered by proposed STS PDW Specification 3.3.2, Electrical Distribution Systems.
3.8.6 Battery Cell Parameters	Not included in the proposed STS PDW.	Not included in Trojan PDTS. Reason WOG STS Specification 3.8.6 was not included in the proposed STS PDW: - The only requirement for batteries identified is for the FBACS control instrumentation, which is covered by proposed STS PDW Specification 3.3.2, Electrical Distribution Systems.
3.8.8 Inverters—Shutdown	Not included in the proposed STS PDW.	Not included in Trojan PDTS. Reason WOG STS Specification 3.8.8 was not included in the proposed STS PDW: - The only requirement for inverters is for FBACS control instrumentation, which is covered by proposed STS PDW Specification 3.3.2, Electrical Distribution Systems.

WOG STS	Proposed STS PDW	Rationale
3.8.10 Distribution Systems—Shutdown	3.3.2 Electrical Distribution Systems	Not included in Trojan PDTS. Including WOG STS Specification 3.8.10 in the proposed STS PDW as Specification 3.3.2 provides assurance that the AC and DC power requirements for the FBACS are met during the Applicability period (the first [6] months after permanent shutdown of the reactor).
4.0 DESIGN FEATURES	**4.0 DESIGN FEATURES**	**Included in Trojan PDTS.**
4.1 Site Location	4.1 Site Location	Included in Trojan PDTS.
4.3 Fuel Storage	4.3 Fuel Storage	Included in Trojan PDTS.
5.0 ADMINISTRATIVE CONTROLS	**5.0 ADMINISTRATIVE CONTROLS**	**Included in Trojan PDTS.**
5.1 Responsibility	5.1 Responsibility	Included in Trojan PDTS.
5.2 Organization	5.2 Organization	Included in Trojan PDTS.
5.3 Unit Staff Qualifications	5.3 Facility Staff Qualifications	Included in Trojan PDTS.
	5.4 Training	Included in Trojan PDTS.
	5.5 Reviews and Audits	Included in Trojan PDTS.
5.4 Procedures	5.6 Procedures	Procedures included with Programs and Manuals in Trojan PDTS.
5.5 Programs and Manuals	5.7 Programs and Manuals	The Technical Specifications Bases Control from the Trojan PDTS 5.6 is included in proposed STS PDW Specification 5.7.
5.6 Reporting Requirements	5.8 Reporting Requirements	Included in Trojan PDTS.
	5.9 Record Retention	Included in Trojan PDTS.
[5.7 High Radiation Area]	[5.10 High Radiation Area]	Included in Trojan PDTS.

1.0 USE AND APPLICATION

1.1 Definitions

-----NOTE-----
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification prescribing Required Actions to be taken under designated Conditions within specified Completion Times.
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Specification 5.4.1.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, display, and trip functions. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

(continued)

1.1 Definitions (continued)

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
OPERABLE—OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe storage of irradiated fuel. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, provided the facility is in the specified Condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required Action and associated Completion Time not met.	A.1 Verify . . . AND A.2 Restore . . .	6 hours 36 hours

Condition A has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition A is entered.

The Required Actions of Condition A are to perform the verification required by ACTION A.1 within 6 hours AND to perform the verification required by ACTION A.2 within 36 hours. A total of 6 hours is allowed for performing ACTION A.1 and a total of 36 hours (not 42 hours) is allowed for performing ACTION A.2 from the time that Condition A was entered. If ACTION A.1 is completed within 3 hours, the time allowed for completing ACTION A.2 is the next 33 hours because the total time allowed for completing ACTION A.2 is 36 hours.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. System inoperable.	A.1 Verify affected system isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore system to OPERABLE status.	72 hours

Required Action A.1 has two Completion Times. The 1-hour Completion Time begins at the time the Condition is entered and the "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1. The 72-hour Completion Time for Required Action A.2 also begins at the time the Condition is entered and runs concurrently.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

EXAMPLES

The following examples illustrate the various ways in which Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) occurs when irradiated fuel is stored in the spent fuel pool.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify parameter is within limits.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in the specified Condition in the Applicability of the LCO, then SR 3.0.3 becomes applicable.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify parameter is within limits.	Within 24 hours prior to moving irradiated fuel AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one-time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified Condition has been met (i.e., the "prior to" performance in this example).

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Spent Fuel Cladding SL

The water level in the fuel storage pool shall be maintained above plant elevation [XX feet]. This will ensure that the water level will be maintained at least 10 feet above the irradiated fuel assemblies seated in the fuel storage racks, which in turn will protect the fuel cladding.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance within 1 hour.

2.2.2 Within 4 hours, notify the Nuclear Regulatory Commission (NRC) Operations Center.

2.2.3 Within 4 hours, notify the [Plant Superintendent and Vice President—Nuclear Operations].

2.2.4 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the [offsite review function], and the [Plant Superintendent and Vice President—Nuclear Operations].

2.2.5 Crane hoists or fuel handling activities shall not be resumed until authorized by the NRC.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the specified Conditions in the Applicability, except as provided in LCO 3.0.2.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Action(s) of the associated Condition(s) shall be met.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during specified Condition(s) in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be defined as failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be defined as failure to meet the LCO, except as provided in SR 3.0.3. Surveillances are not required on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified Condition of the Frequency is met.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare that the LCO has not been met may be delayed. Starting from the time of discovery, the delay period may continue up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

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

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

3.1 INSTRUMENTATION

3.1.1 Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation

LCO 3.1.1 The FBACS actuation instrumentation for each Function in Table 3.1.1-1 shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies or movement of loads over storage racks containing irradiated fuel within [6] months after the reactor has been permanently shut down.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. FBACS Actuation instrumentation inoperable.	A.1 Suspend movement of irradiated fuel assemblies and movement of loads over storage racks containing fuel.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
Refer to Table 3.1.1-1 to determine which SRs apply for each FBACS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Perform CHANNEL CHECK.	7 days
SR 3.1.1.2	Perform COT.	92 days
SR 3.1.1.3	-------------------NOTE----------------- Verification of setpoint is not required. --- Perform TADOT.	Within 7 days of conversion to PDTS
SR 3.1.1.4	Perform CHANNEL CALIBRATION.	Within 7 days of conversion to PDTS

Table 3.1.1-1 (page 1 of 1)
FBACS Actuation Instrumentation

FUNCTION	CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	(a)	1	SR 3.1.1.3	NA
2. Fuel Building Radiation				
a. Gaseous	(a)	[1]	SR 3.1.1.1 SR 3.1.1.2 SR 3.1.1.4	≤ [2] mR/hr
b. Particulate	(a)	[1]	SR 3.1.1.1 SR 3.1.1.2 SR 3.1.1.4	≤ [2] mR/hr

(a) During movement of irradiated fuel assemblies or movement of loads over storage racks containing irradiated fuel in the fuel building.

3.2 PLANT SYSTEMS

3.2.1 Fuel Building Air Cleanup System (FBACS)

LCO 3.2.1 An FBACS train shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies or movement of loads over storage racks containing irradiated fuel within [6] months after the reactor has been permanently shut down.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. FBACS inoperable.	A.1 Suspend movement of irradiated fuel assemblies and movement of loads over storage racks containing fuel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Operate the FBACS train for [≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days
SR 3.2.1.2	Perform required FBACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]
[SR 3.2.1.3	Verify that the FBACS train actuates on an actual or simulated actuation signal.	[92] days]
SR 3.2.1.4	Verify that the FBACS train can maintain a pressure of ≤ [−0.125] inches water gauge with respect to atmospheric pressure during the [post-accident] mode of operation at a flow rate of ≤ [20,000] cfm.	[92] days
[SR 3.2.1.5	Verify that each FBACS filter bypass damper can be closed.	[92] days]

3.2 PLANT SYSTEMS

3.2.2 Fuel Storage Pool Water Level

LCO 3.2.2 The fuel storage pool water level shall be ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies and movement of loads over storage racks containing fuel.	Immediately
	<u>AND</u>	
	A.2 Initiate makeup flow to the fuel storage pool.	Immediately
	<u>AND</u>	
	A.3 Restore water level to within limit.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify that the fuel storage pool water level is $\geq$ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	24 hours

3.2 PLANT SYSTEMS

3.2.3 Fuel Storage Pool Coolant Temperature

LCO 3.2.3 The fuel storage pool coolant temperature shall be maintained ≤ [140°F].

APPLICABILITY: Whenever irradiated fuel assemblies are stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool coolant temperature not within limit.	A.1 Initiate action to restore the fuel storage pool coolant temperature to within limit.	Immediately
	<u>AND</u>	
	A.2 Verify that a fuel storage pool makeup water source is available.	Immediately
	<u>AND</u>	
	A.3 Restore fuel storage pool coolant temperature to within limit.	[7] days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify that the fuel storage pool coolant temperature is ≤ [140°F].	24 hours

3.2 PLANT SYSTEMS

3.2.4 Fuel Storage Pool Load Restrictions

LCO 3.2.4 Loads carried over the fuel storage pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over [20,000] ft-lb if the loads are dropped.

APPLICABILITY: Whenever fuel assemblies are stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Load restriction not within limit.	A.1 Suspend movement of load. AND A.2 Evaluate and choose appropriate course of action. AND A.3 Place load in a safe position.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	Verify that the potential impact energy due to dropping the load is $\leq$ [20,000] ft-lb.	Prior to moving each load over storage racks containing fuel.

3.2 PLANT SYSTEMS

3.2.5 Fuel Storage Pool Boron Concentration

LCO 3.2.5 The fuel storage pool boron concentration shall be ≥ [2300] ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	A.1 Suspend movement of fuel assemblies and movement of loads over storage racks containing fuel.	Immediately
	<u>AND</u>	
	A.2 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately
	<u>AND</u>	
	A.3 Verify by administrative means [Region 2] that fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.5.1 Verify that the fuel storage pool boron concentration is within limit.	7 days

3.2 PLANT SYSTEMS

3.2.6 Spent Fuel Assembly Storage

LCO 3.2.6 The combination of initial enrichment and burnup of each spent fuel assembly stored in [Region 2] shall be within the Acceptable [Burnup Domain] of Figure [3.2.6-1] or in accordance with Specification 4.2.1.1.

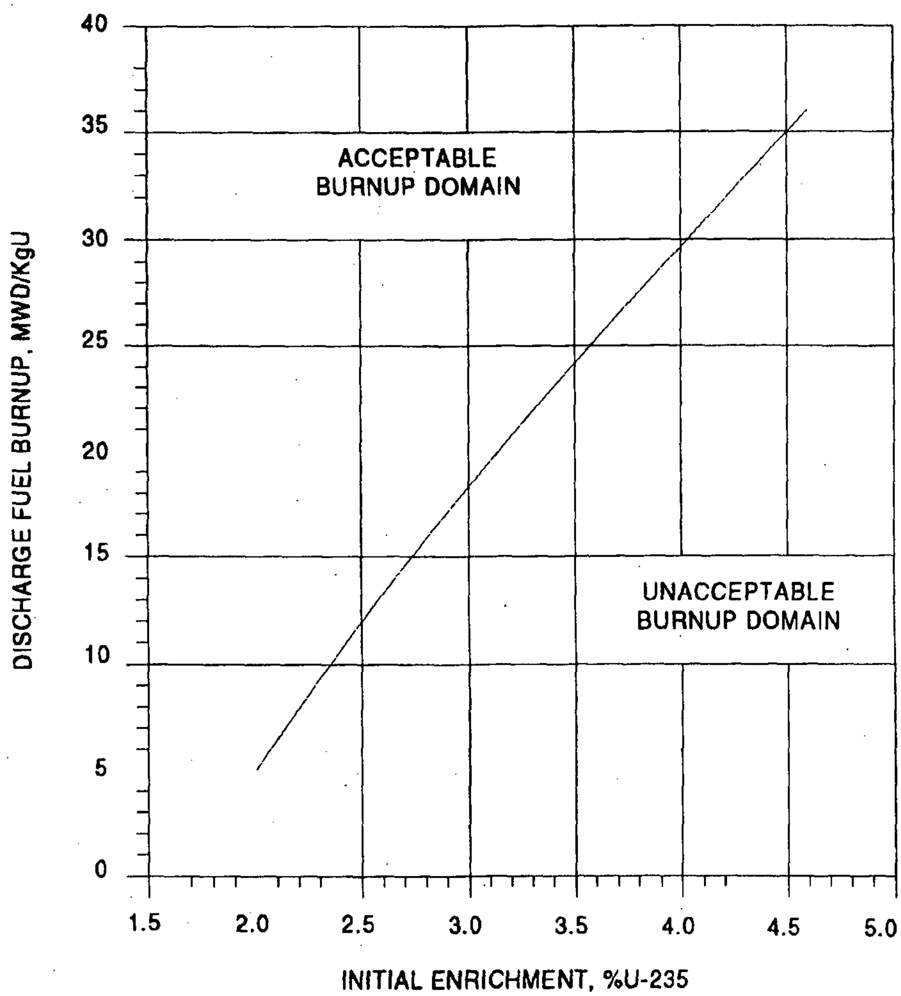
APPLICABILITY: Whenever any fuel assembly is stored in [Region 2] of the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly from [Region 2].	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.6.1	Verify by administrative means that the initial enrichment and burnup of the fuel assembly are in accordance with Figure [3.2.6-1] or Specification 4.2.1.1.	Prior to storing the fuel assembly in [Region 2]



Not to be used for Operation.
For illustration purposes only.

Figure [3.2.6-1] (page 1 of 1)
Fuel Assembly Burnup Limits in [Region 2]

3.3 ELECTRICAL POWER SYSTEMS

3.3.1 AC Power Systems

LCO 3.3.1 One circuit between the offsite transmission network and the onsite AC electrical power distribution subsystem(s) required by LCO 3.3.2, "Distribution Systems," shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies or during movement of loads over storage racks containing fuel within [6] months after the reactor has been permanently shut down.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required offsite circuit inoperable.	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	<u>AND</u>	
	A.2 Suspend movement of irradiated fuel assemblies or moving loads over storage racks containing fuel.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore offsite power circuit to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1	Verify that the breaker alignment is correct and that the indicated power is available for the offsite circuit.	7 days

3.3 ELECTRICAL POWER SYSTEMS

3.3.2 Electrical Distribution Systems

LCO 3.3.2 The electrical power distribution systems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies or during movement of loads over storage racks containing irradiated fuel within [6] months after the reactor has been permanently shut down.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Electrical power distribution system(s) inoperable.	A.1 Declare required feature(s) supported by the power source(s) and distribution system as inoperable.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies or movement of loads over storage racks containing fuel.	Immediately
	AND	
	A.3 Initiate actions to restore electrical power distribution system to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Verify correct breaker alignments and voltage to electrical power distribution system.	7 days

4.0 DESIGN FEATURES

4.1 Site Location [Text description of site location.]

4.2 Fuel Storage

4.2.1 Criticality

4.2.1.1 The spent fuel storage racks were designed and shall be maintained for:

a. Fuel assemblies having a maximum U-235 enrichment 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 Final Safety Analysis Report (FSAR)];

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

[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 [3.2.6-1] 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 [3.2.6-1] will be stored in compliance with the NRC-approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]

4.2.1.2 The new fuel storage racks were designed and shall be maintained for:

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

(continued)

4.0 DESIGN FEATURES

4.2 Fuel Storage (continued)

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];

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

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

4.2.2 Drainage

The spent fuel storage pool was designed for and shall be maintained to prevent inadvertent draining of the pool below elevation [XX feet].

4.2.3 Capacity

The spent fuel storage pool was designed for and shall be maintained with a storage capacity limited to [1737] fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The [Plant Superintendent] shall be responsible for the facility and shall delegate in writing the succession to this responsibility during any absence.

The [Plant Superintendent] or designee, in accordance with approved administrative procedures, shall approve each proposed test, fuel movement, load movement, or experiment prior to implementation and shall approve modification to structures, systems, or equipment that affect the safe storage of irradiated fuel.

5.1.2 The [Shift Supervisor (SS)] shall be responsible for the operational command function. During any absence of the [SS] from the Control Room, another qualified operator shall be designated to assume the command function. The individual maintaining the command function shall remain in the Control Room.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 General Organizational Requirements

Onsite and offsite organizations shall be established for facility staff and corporate management, respectively. These organizations shall include positions for activities affecting the safe storage of irradiated fuel.

a. Lines of authority, responsibility, and communication shall be defined and established throughout the organization. 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.

b. The [Plant Superintendent] shall have overall responsibility for the facility.

c. The [specified corporate executive position] shall have corporate responsibility for overall nuclear safety of the facility and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to ensure the safe storage of irradiated fuel.

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

5.2.2 Facility Staff

The facility staff organization shall be as follows:

a. Each onduty shift shall include, as a minimum, the crew composition shown in Table 5.2.2-1;

(continued)

5.2 Organization

5.2.2 Facility Staff (continued)

b. At least one qualified person (non-certified operator or CERTIFIED FUEL HANDLER) shall be present in the Control Room when irradiated fuel is stored in the spent fuel pool;

c. An individual qualified in radiation protection procedures shall be on site during fuel handling operations or movement of loads over storage racks containing fuel;

d. Fuel handling operations or movement of loads over storage racks containing fuel shall be directly supervised by a CERTIFIED FUEL HANDLER;

e. Administrative procedures shall be developed and implemented to limit the working hours of shift personnel who perform functions important to the safe storage of irradiated fuel assemblies (e.g., CERTIFIED FUEL HANDLERS, non-certified operators, radiation protection personnel, and key maintenance personnel);

f. The [Shift Supervisor] shall be a CERTIFIED FUEL HANDLER; and

g. [Shift Supervisors] shall report to an individual who is a CERTIFIED FUEL HANDLER.

(continued)

Table 5.2.2-1 (page 1 of 1)
Minimum Shift Crew Composition[a]

POSITION	MINIMUM CREW NUMBER
Shift Supervisor	1
Non-Certified Operator	1
Total	2

[a] To accommodate unexpected absences of onduty shift crew members, the shift crew composition may be one less than the minimum requirements depicted in Table 5.2.2-1 for no more than 2 hours, provided immediate action is taken to restore the shift crew composition to within minimum requirements. During such absences, no fuel movement or movement of loads over storage racks containing fuel shall be permitted. This provision does not permit any shift crew position to be unmanned upon shift change due to the absence or tardiness of an oncoming shift crew member.

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 [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].

5.0 ADMINISTRATIVE CONTROLS

5.4 Training

5.4.1 An NRC-approved retraining and replacement training program for CERTIFIED FUEL HANDLERS shall be maintained under the direction of the [Plant Superintendent].

5.0 ADMINISTRATIVE CONTROLS

5.5 Reviews and Audits

5.5.1 Independent Safety Review

An Independent Safety Review shall be a thorough review conducted by a qualified Independent Safety Reviewer. Persons performing these reviews shall be knowledgeable in the subject area being reviewed. Independent Safety Reviews must be completed prior to implementation of proposed activities.

a. Independent Safety Reviewers shall be individuals without direct responsibility for the performance of the activities under review; these reviewers may be from the same functionally cognizant organization as the individual or group performing the original work.

b. Independent Safety Reviewers shall have at least 5 years of professional experience and either a Bachelor's Degree in Engineering or the Physical Sciences or shall have equivalent qualifications in accordance with ANSI/ANS-3.1-1981. The [Plant Superintendent] shall document the appointment of Independent Safety Reviewers.

c. The following subjects shall be independently reviewed by a qualified Independent Safety Reviewer:

   1. safety evaluations for changes in the facility as described in the Safety Analysis Report (SAR), changes in procedures as described in the SAR, and tests or experiments not described in the SAR to verify that such actions do not involve a change to the Technical Specifications or will not involve an unreviewed safety question as defined in 10 CFR 50.59;

   2. proposed changes to the programs required by Specification 5.7.2, to verify that such changes do not involve a change to the Technical Specifications and will not involve an unreviewed safety question as defined in 10 CFR 50.59; and

   3. proposed changes to the Technical Specification Bases.

(continued)

5.5 Reviews and Audits (continued)

5.5.2 Independent Review and Audit Committee (IRAC)

The IRAC is responsible for reviewing, auditing, and advising the [Plant Superintendent] on matters related to the safe storage of irradiated fuel. This review and audit function is independent of line organization responsibilities.

a. The IRAC shall include a minimum of five members. Alternates may be substituted for regular members. The licensee shall designate in writing the chairman, the members, and alternates for the IRAC. The chairman shall not have management responsibilities for, or report to, the line organizations responsible for operation or maintenance of the fuel storage facility.

b. The IRAC shall collectively have experience and knowledge in the following functional areas:

   1. fuel handling and storage (including the potential for criticality),
   2. chemistry and radiochemistry,
   3. engineering,
   4. radiation protection, and
   5. quality assurance.

c. The IRAC shall hold at least one meeting per quarter.

d. A quorum shall consist of three regular members or their duly appointed alternates. Those members representing the line organizations responsible for the operation and maintenance of the facility shall not constitute a majority of the quorum. At least one member of the quorum shall be the chairman or the chairman's designated alternate.

e. As a minimum, the IRAC shall perform the following functions:

   1. advise the [Plant Superintendent] on all matters related to safe storage of irradiated fuel;

   2. advise the management of the audited organization and the [Plant Superintendent] of audit results as they relate to safe storage of irradiated fuel;

(continued)

5.5 Reviews and Audits

5.5.2 Independent Review and Audit Committee (continued)

3. recommend to the management of the audited organization, and its management, any corrective action to improve the safe storage of irradiated fuel; and

4. notify [a specified corporate executive position] of any safety significant disagreement between the IRAC and the [Plant Superintendent] within 24 hours.

f. The IRAC shall be responsible for reviewing:

1. the safety evaluations for procedures, and changes thereto, completed under the provisions of 10 CFR 50.59 to verify that such actions do not involve an unreviewed safety question as defined in 10 CFR 50.59. This review may be completed after implementation of the affected procedure;

2. changes to structures, systems, or components important to the safe storage of irradiated fuel to verify that such changes do not involve an unreviewed safety question as defined in 10 CFR 50.59. This review may be completed after implementation of the change;

3. tests or experiments involving the safe storage of irradiated fuel to verify that such tests or experiments do not involve an unreviewed safety question as defined in 10 CFR 50.59. This review may be completed after performance of the test or experiment;

4. proposed changes to these Technical Specifications or the license;

5. violations of codes, regulations, orders, license requirements, or internal procedures/instructions having nuclear safety significance;

6. indications of unanticipated deficiencies in any aspect of design or operation of structures, systems, or components that could affect safe storage of irradiated fuel;

(continued)

5.5 Reviews and Audits

5.5.2 Independent Review and Audit Committee (continued)

7. significant accidental, unplanned, or uncontrolled radioactive releases, including corrective action(s) to prevent recurrence;

8. significant operating abnormalities or deviations from normal and expected performance of equipment that affect safe storage of irradiated fuel;

9. the performance of the corrective action system; and

10. internal and external experience information related to the safe storage of irradiated fuel that may indicate areas for improving facility safety.

Reports or records of these reviews shall be forwarded to the [Plant Superintendent] within 30 days after completion of the review.

g. The IRAC's audit responsibilities shall encompass:

1. conformance of irradiated fuel storage to provisions contained within the Technical Specifications and applicable license conditions;

2. the training and qualifications of facility staff;

3. implementation of all programs required by Specification 5.7;

4. actions taken to correct deficiencies occurring in structures, systems, components, or methods of operation that affect safe storage of irradiated fuel;

5. facility operations, modifications, maintenance, and Surveillance related to the safe storage of irradiated fuel to verify independently that these activities are performed safely and correctly; and

6. other activities and documents as requested by the [Plant Superintendent].

Reports or records of these audits, including any recommendations for improving the safe storage of irradiated fuel, shall be forwarded to the [Plant Superintendent] within 30 days after completion of the audit.

(continued)

5.5 Reviews and Audits (continued)

5.5.3 Records

Written records of reviews and audits shall be maintained. As a minimum, these records shall include:

a. Results of the activities conducted under the provisions of Specifications 5.5.1 and 5.5.2;

b. Recommendations to the management of the audited organization;

c. An assessment of the safety significance of review or audit findings;

d. Documentation of reviews conducted under Specification 5.5.1.c; and

e. Determination of whether each item considered under Specifications 5.5.2.f.1 through 5.5.2.f.3 involves an unreviewed safety question as defined in 10 CFR 50.59.

5.0 ADMINISTRATIVE CONTROLS

5.6 Procedures

5.6.1 Scope

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

a. The safe storage of irradiated fuel, including procedures recommended in Regulatory Guide 1.33, Appendix A (Revision 2, February 1978);

b. Defueled facility security plan;

c. Defueled facility emergency plan;

d. Quality assurance program; and

e. All programs stipulated in Specification 5.7.

5.6.2 Review and Approval

Each procedure required by Specification 5.6.1, and changes thereto, shall be independently reviewed in accordance with established administrative procedures and approved by the [Plant Superintendent] or designee prior to implementation.

5.6.3 Temporary Changes

Temporary changes to procedures of Specification 5.6.1 may be made, provided they meet the following conditions:

a. The intent of the existing procedure is not altered;

b. The change is approved by a member of the facility management staff and by a CERTIFIED FUEL HANDLER; and

c. The change is documented, reviewed, and approved by a responsible manager in accordance with approved administrative procedures within 14 days of implementation.

5.0 ADMINISTRATIVE CONTROLS

5.7 Programs and Manuals

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

5.7.1 Radiation Protection Program

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

5.7.2 Process Control Program (PCP)

The PCP shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes will be accomplished to ensure compliance with 10 CFR Parts 20, 61, and 71; State regulations; burial ground requirements; and other requirements governing the disposal of solid radioactive waste.

Licensee-initiated changes to the PCP:

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

   1. sufficient information to support each change, together with the appropriate analyses or evaluations to justify the change; and

   2. a determination that each change maintains the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations; and

b. Shall become effective after review and approval by an Independent Safety Reviewer and the approval of the [Plant Superintendent] or designee.

5.7.3 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

(continued)

5.7 Programs and Manuals

5.7.3 <u>Offsite Dose Calculation Manual</u> (continued)

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 Program required by Specifications 5.7.4 and 5.7.5 respectively, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required under Specifications [5.8.2] and [5.8.3].

Licensee-initiated changes to the ODCM:

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

   1. sufficient information to support each change together with the appropriate analyses or evaluations to justify the change; and

   2. a determination that each change maintains the levels of radioactive effluent control required by 10 CFR 20.1302 and 40 CFR Part 190 and that the change will not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the review and approval of an Independent Safety Reviewer and the approval of the [Plant Superintendent]; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change was made to the ODCM.

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.

5.7.4 <u>Radioactive Effluent Controls Program</u>

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of

(continued)

5.7 Programs and Manuals

5.7.4 Radioactive Effluent Controls Program (continued)

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 determinations, in accordance with the methodology described in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR Part 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 described 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 the facility to unrestricted areas, conforming to 10 CFR Part 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 described in the ODCM (performed at least every 92 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 Part 50, Appendix I;

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 as follows:

    1. for noble gases: less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose of 3000 mrem/yr to the skin; and

(continued)

5.7 Programs and Manuals

5.7.4 Radioactive Effluent Controls Program (continued)

2. for iodine-131, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses to a member of the public from tritium and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas beyond the site boundary, conforming to 10 CFR Part 50, Appendix I;

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

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

5.7.5 Radiological Environmental Monitoring Program

This program monitors the radiation and radionuclides in the environs of the facility. The program shall provide representative measurements of radioactive materials in the highest potential exposure pathways, verification of the accuracy of the effluent monitoring program, and modeling of environmental exposure pathways. The program shall be contained in the ODCM and shall include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters described in the ODCM; and

b. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

(continued)

5.7 Programs and Manuals (continued)

5.7.6 Storage Tank Radioactivity Monitoring Program

This program applies to any outdoor liquid radwaste tank that is not surrounded by a liner, dike, or wall capable of holding the tank's contents and that does not have an overflow to an area drain connected to the liquid radwaste treatment system. In the event of an uncontrolled release of the contents of such a tank, this program shall ensure that limits specified in 10 CFR Part 20, Appendix B, are not exceeded at the nearest boundary of the restricted area. For temporary storage tanks, a limit of [10] curies (excluding tritium and dissolved or entrained noble gases) may be used in lieu of the above criteria.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.7.7 Fire Protection Program

The principal objective of this program is to ensure that appropriate fire protection measures are maintained to protect the facility from fires that could affect the safe storage of irradiated fuel or cause the release of radioactive materials.

5.7.8 Fuel Storage Pool Water Chemistry Program

This program prescribes requirements and methodologies to be used for monitoring the water chemistry of the fuel storage pool. The objective of the program is to reduce the potential effects of corrosion affecting the safe storage of irradiated fuel to an acceptable level. The program shall include identification of critical variables and control points for these variables. The program shall also include sampling frequencies, analytical procedures, and the identification of corrective actions to be taken for off-normal conditions. The NRC shall be notified prior to elimination of or changes to the acceptance criteria for monitored variables.

5.7.9 Fuel Storage Pool Cooling and Makeup Monitoring Program

This program requires that the primary method for spent fuel cooling and for makeup capability be monitored and maintained.

(continued)

5.7 Programs and Manuals

5.7.9 Fuel Storage Pool Cooling and Makeup Monitoring Program (continued)

[Facilities located in areas where the potential exists for very low temperatures shall ensure that equipment, components, systems, and water sources are protected against freezing.] This program shall provide reasonable assurance that the equipment, components, systems, and water sources used for cooling the spent fuel pool and maintaining the water level are capable of fulfilling their intended functions.

5.7.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required tests of the Fuel Building Air Cleaning System (FBACS) at the Frequencies specified in [Regulatory Guide XX] and in accordance with [Regulatory Guide 1.52 (Revision 2), ASME N510-1989, and AG-1]. During the post-shutdown period, the VFTP is required only when the FBACS must be maintained in an OPERABLE condition.

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

FBACS Ventilation System	Flowrate
[ ]	[ ]

b. Demonstrate for the FBACS that an inplace test of the charcoal adsorber shows a penetration and system bypass of < [0.05%] when tested in accordance with [Regulatory Guide 1.52 (Revision 2) and ASME N510-1989] at the system flowrate specified below [±10%].

FBACS Ventilation System	Flowrate
[ ]	[ ]

(continued)

5.7 Programs and Manuals

[

5.7.10 Ventilation Filter Testing Program (continued)

c. Demonstrate for the FBACS that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52 (Revision 2)], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of ≤ [30°C] and greater than or equal to the relative humidity specified below.

FBACS Ventilation System	Penetration	RH
[ ]	[ ]	[ ]

[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.]

d. Demonstrate for the FBACS that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52 (Revision 2) and ASME N510-1989] at the system flowrate specified below [± 10%].

FBACS Ventilation System	Delta P	Flowrate
[ ]	[ ]	[ ]

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

]

(continued)

5.7 Programs and Manuals

[5.7.10 Ventilation Filter Testing Program (continued)

FBACS Ventilation System	Wattage
[ ]	[ ]

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

5.7.11 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 in a manner that is consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.7.11.b 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 that is consistent with 10 CFR 50.71(e).

5.0 ADMINISTRATIVE CONTROLS

5.8 Reporting Requirements

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

5.8.1 Occupational Radiation Exposure Report

An Occupational Radiation Exposure Report covering the activities of the facility as described below for the previous calendar year shall be submitted by [April 30] of each year.

---------------------------------NOTE---------------------------------
A single submittal may be made for a site with multiple units or fuel storage facilities. The submittal should combine sections common to all units or fuel storage facilities at the site.

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 person-rem exposure according to work and job functions (e.g., fuel handling, surveillance, maintenance, and waste processing). 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 totaling < 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.

5.8.2 Annual Radiological Environmental Monitoring Report

---------------------------------NOTE---------------------------------
A single submittal may be made for a site with multiple units or fuel storage facilities. The submittal should combine sections common to all units or fuel storage facilities at the site.

The Annual Radiological Environmental Monitoring Report covering the activities during the previous calendar year shall be submitted by [May 15] of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM).

(continued)

5.8 Reporting Requirements

5.8.2 Annual Radiological Environmental Monitoring Report (continued)

The Annual Radiological Environmental 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 tables 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 submitted report shall note and explain the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

The Annual Radiological Environmental Monitoring Report shall include licensee-initiated changes to the ODCM during the period of the report as described in Specification 5.7.3, or these changes shall be submitted concurrently.

5.8.3 Radioactive Effluent Release Report

NOTE

A single submittal may be made for a site with multiple units or fuel storage facilities. The submittal should combine sections common to all units or fuel storage facilities at the site; however, for units and fuel storage facilities with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit or facility.

The Radioactive Effluent Release Report covering the operation of the facility 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 facility. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program.

5.0 ADMINISTRATIVE CONTROLS

5.9 Record Retention

5.9.1 The following records shall be retained for at least 3 years:

a. All Licensee Event Reports required by 10 CFR 50.73;

b. Records of changes made to the procedures required by Specification 5.6.1.

5.9.2 The following records shall be retained for at least 5 years:

a. Records and logs of activities related to the safe storage of irradiated fuel;

b. Records and logs of principal maintenance activities, inspections, and repair and replacement of principal items of equipment related to safe storage of irradiated fuel;

c. Records of Surveillance activities, inspections, and calibrations required by the Technical Specifications (TS);

d. Records of sealed source and fission detector leak tests and results; and

e. Records of annual physical inventory of all sealed source material of record.

5.9.3 The following records shall be retained for the duration of the Possession Only License:

a. Records and drawing changes reflecting design modifications made to structures, systems, and components needed for the safe storage of irradiated fuel as described in the Safety Analysis Report;

b. Records of irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

(continued)

5.9 Record Retention

5.9.3 (continued)

e. Records of radioactive waste disposal in accordance with 10 CFR 20.2108;

f. Records of training and qualification for members of the facility staff;

g. Records of quality assurance (QA) activities required by the QA program and classified as permanent records by applicable regulations, codes, and standards;

h. Records of reviews performed for changes made to procedures, equipment, or reviews of tests and experiments pursuant to 10 CFR 50.59;

i. Records of the reviews and audits required by Specifications 5.5.1 and 5.5.2;

j. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date (these records should include procedures effective at specified times and records showing that these procedures were followed); and

k. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.

5.0 ADMINISTRATIVE CONTROLS

[5.10 High Radiation Area]

5.10.1 Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR Part 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. Entrance to each such area 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 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 and 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.

(continued)

[5.10 High Radiation Area (continued)]

5.10.2 In addition to the requirements of Specification 5.10.1, areas with radiation levels of ≥ 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 specifies the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.10.3 Areas with radiation levels of > 1000 mrem/hr that are accessible to personnel, that are located within large areas such as reactor containment (i.e., where no enclosure exists or can reasonably be constructed for purposes of locking), or that cannot be continuously guarded shall be barricaded and conspicuously posted. A flashing light shall be activated as a warning device.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Spent Fuel Cladding SL

BASES

BACKGROUND

The Spent Fuel Cladding SL protects the integrity of the fuel cladding against overheating by ensuring that the fuel remains covered by water. In the event of a loss of water from the fuel storage pool to a level that permits exposure of fuel assemblies to air, the cladding could overheat and cause the release of fission products into the atmosphere of the fuel storage building. The spent fuel cladding then serves as the final barrier in preventing the release of fission products into the atmosphere. By establishing a lower limit on the fuel storage pool water level, the continued integrity of the fuel cladding is ensured. According to 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 61 (Fuel storage and handling and radioactivity control) and GDC 63 (Monitoring fuel and waste storage) (Ref. 1), fuel storage systems must be designed to prevent significant reduction in fuel storage coolant inventory under accident conditions. Appropriate systems shall be provided (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

APPLICABLE SAFETY ANALYSES

In the event of an extended loss of forced cooling, the water temperature in the spent fuel storage pool could increase to the boiling point. The heat removed by boiling and evaporation provides an adequate heat sink for the irradiated fuel stored in the pool as long as the fuel assemblies remain covered by water. The rate of water level decrease due to boiling and evaporation is primarily dependent upon the volume of water in the spent fuel storage pool, the operating history of the final reactor core, and the decay heat load, which is determined on the basis of the time elapsed since the facility was shut down and permanently defueled.

Additional details on this safety analysis can be found in the Bases for B 3.2.2, Fuel Storage Pool Water Level.

(continued)

BASES (continued)

SAFETY LIMITS

The normal water level in the fuel storage pool is maintained at 23 feet above the irradiated fuel seated in the fuel storage racks. A water level at least 10 feet above the stored fuel is still ample to protect the fuel cladding and will provide shielding sufficient for the protection of facility personnel.

APPLICABILITY

SL 2.1.1 applies whenever irradiated fuel assemblies are stored in the fuel storage pool.

SAFETY LIMIT VIOLATIONS

The following SL violations are applicable to SL 2.1.1, Spent Fuel Cladding.

2.2.1

If SL 2.1.1 is violated, restore compliance within 1 hour. The allowable Completion Time of 1 hour recognizes the importance of maintaining the fuel storage pool to a level where the potential for challenges to the fuel cladding is minimized.

2.2.2

If SL 2.1.1 is violated, notify the Nuclear Regulatory Commission (NRC) Operations Center within 4 hours, in accordance with 10 CFR 50.72 (Ref. 2).

2.2.3

If SL 2.1.1 is violated, notify the [Plant Superintendent and Vice President—Nuclear Operations] within 4 hours, . The 4-hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the facility before reporting to senior management.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.4

If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 3). A copy of the report shall also be provided to the [Plant Superintendent and the Vice President—Nuclear Operations].

2.2.5

If SL 2.1.1 is violated, crane hoists or fuel handling activities shall not be resumed until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the normal fuel handling operations within the fuel storage building are reinitiated.

REFERENCES

1. 10 CFR Part 50, Appendix A, GDC 61 and GDC 63.
2. 10 CFR 50.72.
3. 10 CFR 50.73.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 and LCO 3.0.2 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each Specification as the requirement for when the LCO shall be met (i.e., when the facility is in a specified Condition of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not necessary when an LCO is met within the specified Completion Time, unless otherwise specified.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The Completion Times of the Required Actions are also applicable when a specified Condition in the Applicability is entered intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of surveillances, preventive maintenance, corrective maintenance, or investigation of problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise the safe storage of irradiated fuel. Intentional entry into ACTIONS should not be made for convenience.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.3 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the specified Conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is required to ensure that Surveillances are performed in order to verify that facility conditions are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Surveillances do not have to be performed when the facility is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

SR 3.0.2

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., other ongoing surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any Surveillance is the verification of conformance with the SRs. Any exceptions to SR 3.0.2 are stated in the individual Specifications.

The provisions of SR 3.0.2 are not intended for repeated use merely as a convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

(continued)

BASES (continued)

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay of up to 24 hours applies from the time of discovery that the Surveillance has not been performed in accordance with SR 3.0.2, not from the time the specified Frequency was not met.

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

The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. On discovery that a Surveillance with a Frequency based on facility conditions or operational situations (rather than time) is not performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility that is not intended as a convenience to extend Surveillance intervals.

B 3.1 INSTRUMENTATION

B 3.1.1 Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation

BASES

BACKGROUND

The FBACS is designed to remove gaseous and particulate radioactive material from the atmosphere of the fuel storage building in the event of a fuel handling accident. The system is described in the Bases for Limiting Condition for Operation (LCO) 3.2.1, Fuel Building Air Cleanup System (FBACS). The system automatically initiates filtered ventilation of the fuel building on the activation of a high radiation signal or by means of actuation from the main Control Room.

High levels of gaseous or particulate radioactive material will actuate the FBACS. Each FBACS train contains a gaseous monitor and a particulate monitor. High radiation detected by either monitor actuates fuel building isolation dampers and starts the FBACS. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the fuel building. Since the radiation monitors include an air sampling system, various components (e.g., such as sample line valves, sample line heaters, sample pumps, and filter motors) are required to ensure OPERABILITY of the monitors and ultimately of the FBACS.

APPLICABLE SAFETY ANALYSES

In the event of a fuel handling accident, the FBACS reduces the amount of radioactive material from the fuel building air before it is discharged to the environment. This action ensures that, in the event of a fuel handling accident, offsite doses remain within the limits cited in 10 CFR Part 100 (Ref. 1).

The FBACS Actuation Instrumentation LCO satisfies Criterion 3 of the Standard Technical Specifications for Permanently Defueled Westinghouse Plants (STS PDW).

LCO

LCO requirements ensure that instrumentation necessary to initiate the FBACS is OPERABLE during movement of irradiated fuel assemblies or the movement of loads over storage racks

(continued)

BASES

LCO (continued)

containing irradiated fuel within [6] months after the reactor has been permanently shut down.

a. Manual Initiation

This LCO requires one OPERABLE channel, a relaxation from the two-channel requirement for an operating pressurized water reactor (PWR). The CERTIFIED FUEL HANDLER can initiate the FBACS from the Control Room at any time. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures that the proper amount of redundancy is maintained in the manual actuation circuitry to provide the CERTIFIED FUEL HANDLER with the capability for manual initiation.

The manual actuation circuitry for each channel consists of one push button and interconnected wiring to the actuation logic cabinet.

b. Automatic Actuation Logic and Actuation Relays

The FBACS Actuation Instrumentation LCO requires one OPERABLE train of Actuation Logic and Relays.

APPLICABILITY

To ensure that the FBACS is available for the removal of fission products resulting from a fuel handling accident, both the manual and high radiation FBACS initiation functions must be OPERABLE during the movement of irradiated fuel assemblies or movement of loads over storage racks containing irradiated fuel. OPERABILITY of this system shall be maintained for the first [6] months after permanent shutdown of the reactor. After [6] months, the fission product inventory of the fuel will have decayed to the point that OPERABILITY of the accident mitigation equipment is no longer required.

ACTIONS

Channel inoperability is usually caused either by outright failure or by drift of the bistable or process module that is sufficient to exceed the tolerance allowed by calibration

(continued)

BASES

ACTIONS (continued)

procedures. Typically, the drift is small and produces an actuation delay rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification.

A.1

Condition A applies to the actuation logic train function of the Solid State Protection System (SSPS), to the radiation monitor functions, and to the manual function. Condition A applies to the failure of a single actuation logic train, radiation monitor channel, or manual channel. If the FBACS is inoperable, movement of irradiated fuel assemblies and movement of loads over storage racks containing irradiated fuel must be suspended immediately.

SURVEILLANCE REQUIREMENTS

A Note indicates that Table 3.1.1-1 should be used to determine which SRs apply to various FBACS Actuation Functions.

SR 3.1.1.1

Performance of the CHANNEL CHECK once every 7 days is necessary to detect a gross failure of the FBACS circuitry. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the facility staff, based on a combination of the channel instrument uncertainties, including those related to indication and

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted beyond its limit.

The operating experience on which Frequency is based demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.1.1.2

A CHANNEL OPERATIONAL TEST (COT) is performed once every 92 days on each required channel to ensure that the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide FBACS actuation. The setpoints shall be left consistent with the facility-specific calibration procedure tolerance (Ref. 2). The 92-day Frequency is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.1.1.3

SR 3.1.1.3 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT). This test is a check of the manual Actuation Functions and is performed within 7 days of conversion to facility-specific Permanently Defueled Technical Specifications (PDTS). Each manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles). The SR is modified by a Note that excludes verification of setpoints during the TADOT. The functions tested have no setpoints associated with them.

SR 3.1.1.4

A CHANNEL CALIBRATION is performed within 7 days of conversion to PDTS. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.4 (continued)

with the required accuracy within the prescribed range. This SR requires a single CHANNEL CALIBRATION test during the Applicability period of [6] months after conversion to the PDTS. This Frequency is sufficient given the 18-month Frequency required for operating plants.

REFERENCES

1. 10 CFR 100.11.

2. Setpoint Calibration Procedure.

B 3.2 PLANT SYSTEMS

B 3.2.1 Fuel Building Air Cleanup System (FBACS)

BASES

BACKGROUND

The FBACS is designed to remove gaseous and particulate radioactive material from the atmosphere of the fuel storage building in the event of a fuel handling accident. In conjunction with other normally operating systems, the FBACS also controls temperature and humidity in the fuel pool area.

The FBACS train consists of a heater, a prefilter or demister, a high-efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally isotopes of iodine), and a fan. Ductwork, valves or dampers, and instrumentation are also included in the train. A second bank of HEPA filters is located downstream from the adsorber section. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines and to back up the primary HEPA filter should it develop a leak. The system initiates filtered ventilation of the fuel building in the event of a high radiation signal and maintains the building at a negative pressure of 1/8 inch of water to preclude exfiltration.

The FBACS is a standby system, parts of which may also support normal facility operations. Upon receipt of the actuating signal, normal air discharges from the fuel building are discontinued, the building is isolated, and the stream of ventilation air begins to discharge through the FBACS. The prefilters or demisters remove large particles or entrained water droplets from the airstream to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The FBACS is discussed in the Final Safety Analysis Report (FSAR), Sections [6.5.1], [9.4.5], and [15.7.4] (Refs. 1, 2, and 3, respectively) because it may be used for atmospheric cleanup functions under both normal and post-accident conditions.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The FBACS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The fission product inventory of the fuel is determined based on the irradiation history assumed for the fuel assembly. The fraction of these fission products potentially available for release to the fuel building atmosphere is determined for the fuel handling accident. The accident analysis accounts for the reduction in airborne radioactive material provided by a single train of this filtration system. The FBACS is tested periodically in accordance with the Ventilation Filter Testing Program (VFTP) to demonstrate that the system has the required removal efficiencies for particulate and gaseous material and that a single train is capable of maintaining the building at a negative pressure of 1/8 inch of water to preclude exfiltration. Assumptions and analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).

The FBACS LCO satisfies Criterion 3 of the STS PDW.

LCO

One OPERABLE train of the FBACS is required. In the event of a fuel handling accident, total system failure could result in an atmospheric release from the fuel handling building in excess of the reference values prescribed by 10 CFR Part 100 (Ref. 5).

The FBACS is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE. The FBACS is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber do not excessively restrict flow and are capable of performing their filtration function; and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

(continued)

BASES (continued)

APPLICABILITY

An OPERABLE FBACS is required to alleviate the consequences of a fuel handling accident during movement of irradiated fuel assemblies or movement of loads over storage racks containing irradiated fuel within [6] months following permanent shutdown of the reactor. After [6] months, the fission product inventory of the fuel will have decayed to the point that OPERABILITY of the accident mitigation equipment is no longer required.

ACTIONS

A.1

When the FBACS is inoperable, action must be taken immediately to suspend movement of irradiated fuel assemblies or movement of loads over storage racks containing irradiated fuel. This does not preclude the movement of fuel or loads to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

Standby systems should be checked periodically to ensure that they function properly. Given that environmental and normal operating conditions on this system are not severe, testing each train on a 31-day Frequency provides an adequate check.

Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for at least 10 continuous hours with the heaters energized, while systems without heaters need only be operated 15 minutes to demonstrate the system functionality.] The 31-day Frequency is based on the known reliability of the equipment.

[SR 3.2.1.2

This SR verifies that the required FBACS testing is performed in accordance with the [VFTP]. The FBACS filter tests are conducted in accordance with Regulatory Guide 1.52 (Ref. 6). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated]

(continued)

BASES

SURVEILLANCE REQUIREMENTS

[SR 3.2.1.2 (continued)]

charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the [VFTP].

SR 3.2.1.3

This SR verifies that each FBACS train starts and operates on an actual or simulated actuation signal. The [92]-day Frequency is based on the requirement for only a single FBACS train and an Applicability period of [6] months after permanent shutdown of the reactor.

SR 3.2.1.4

This SR verifies the integrity of the fuel building enclosure. The ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify that the FBACS functions properly. During the [post-accident] MODE, the FBACS is designed to maintain a slight negative pressure in the fuel building to prevent unfiltered leakage. The FBACS is designed to maintain a $\leq$ [-0.125] inches water gauge with respect to atmospheric pressure at a flow rate of [20,000] cfm to the fuel building. The Frequency of [92] days is based on the requirement for only a single FBACS train and an Applicability period of [6] months after permanent shutdown of the reactor.

SR 3.2.1.5

Operating the FBACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FBACS filter bypass damper is verified when it can be closed. The [92]-day Frequency is based on the requirement for a single FBACS train and an Applicability period of [6] months after permanent shutdown of the reactor.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section [6.5.1].
2. FSAR, Section [9.4.5].
3. FSAR, Section [15.7.4].
4. Regulatory Guide 1.25.
5. 10 CFR Part 100.
6. Regulatory Guide 1.52 (Rev. 2).

B 3.2 PLANT SYSTEMS

B 3.2.2 Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the fuel storage pool required by this Limiting Condition for Operation (LCO) meets the assumptions of iodine decontamination factors after a fuel handling accident. The specified water level shields the stored fuel and thereby minimizes the general area dose rate. The water also provides shielding during the movement of spent fuel. A general description of the fuel storage pool design is given in the FSAR, Section [9.1.2] (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section [9.1.3] (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section [15.7.4] (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the fuel storage pool meets the assumptions of the reference values prescribed in fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2-hour thyroid dose per person at the exclusion area boundary is a small fraction of the reference values prescribed in 10 CFR Part 100 (Ref. 5). In Reference 4, it is assumed that there will be 23 feet of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 feet of water, the assumptions of Reference 4 apply. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 feet of water between the top of the fuel bundle and the surface. To offset this small nonconservatism, the analysis assumes the failure of all fuel rods in the dropped assembly, although analysis shows that only the first few rows fail from a hypothetical maximum drop. The minimum water level is also consistent with the assumptions of the analysis of a loss of forced cooling in the fuel storage pool as described in the FSAR. In the event of an extended loss of forced cooling, the water

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	temperature in the spent fuel storage pool could increase to the point that boiling begins. The heat removed by boiling and evaporation provides an adequate heat sink for the irradiated fuel stored in the pool as long as the fuel assemblies remain covered by water. The rate of water level decrease due to boiling and evaporation is primarily dependent upon the volume of water in the spent fuel storage pool and the decay heat load. For the first several months after permanent shutdown, the principal heat load is attributable to the heat generation rate of the final reactor core, which in turn is determined by the operating history of the core and the time elapsed since the reactor was shut down.

Conservative calculations for the [Trojan Nuclear Plant] indicate that, based on a 1-year decay time with no remedial action, more than [10 days] would elapse after the temperature in the spent fuel pool exceeded [140°F] before the water level decreased from 23 feet above the fuel assemblies to 10 feet above the fuel assemblies. A makeup rate of [< 8 gpm] is sufficient to offset the loss of water due to boiling and evaporation. A water level 10 feet above the top of the irradiated fuel assemblies would continue to provide an adequate heat sink for the fuel and sufficient shielding for personnel working in the area.

Based on data from Trojan Nuclear Plant, values were estimated for the principal parameters required to evaluate the loss of water from the spent fuel pool caused by boiling and evaporation. Cases were evaluated for assumed fission product decay times of 6 months and 1 month for a reactor with a similar operating history and a spent fuel pool of similar design. These estimates are presented in Table 3.2.2-1.

Values of heat generation and water loss rates presented in Table 3.2.2-1 demonstrate that, in the event forced cooling for the spent fuel pool is lost, ample time is available for restoring the failed cooling system or providing alternative sources of makeup water.

The Fuel Storage Pool Water Level LCO satisfies Criterion 2 of the STS PDW.

(continued)

BASES (continued)

LCO

The minimum water level for the fuel storage pool shall be ≥ 23 feet above the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3) and, thus, is the minimum water level required for fuel storage and movement within the facility.

APPLICABILITY

This LCO applies whenever irradiated fuel assemblies are stored in the fuel storage pool.

ACTIONS

A.1

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the water level of the fuel storage pool is lower than required, movement of irradiated fuel assemblies and loads over storage racks containing irradiated fuel is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This requirement does not preclude movement of a fuel assembly or load to a safe position.

A.2

When the minimum water level assumed in the loss of forced cooling analysis for the spent fuel pool is not met, steps should be taken to restore the water level within the specified limit. A number of sources of water are available to provide the necessary makeup water.

Methods of establishing makeup flow to the spent fuel pool are delineated in facility procedures.

A.3

Action A.3 requires that the water level for the spent fuel pool be restored within the specified limit within 24 hours. This time period is based on the fact that Action A.1 precludes events such as a fuel handling event that could result in the immediate release of radioactive material in

(continued)

BASES

ACTIONS

A.3 (continued)

the pool. Action A.3 requires that the level be restored in a reasonable time to ensure that the initial conditions assumed in the analysis of a loss of forced cooling are maintained.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

This SR verifies that sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 24-hour Frequency is appropriate inasmuch as the volume in the pool is normally stable and the interval is short enough to provide early detection of a level change. Changes in water level are specified by plant procedures and are acceptable based on operating experience.

REFERENCES

1. FSAR, Section [9.1.2].
2. FSAR, Section [9.1.3].
3. FSAR, Section [15.7.4].
4. Regulatory Guide 1.25, [Rev. 0].
5. 10 CFR 100.11.

Table 3.2.2-1. (page 1 of 1)
Estimated Values for Parameters Related to
Heatup of Spent Fuel Storage Pool

PARAMETER	1 YEAR	6 MONTHS	1 MONTH
Decay Heat Generated	1.05 MW	1.5 MW	2.8 MW
Spent Fuel Storage Pool Heatup Rate	1.68°F/hr	2.35°F/hr	4.4°F/hr
Time to Boil	43 hours	31 hours	16 hours
Boil-off Rate	7.5 gpm	10.5 gpm	20 gpm
Time to Reach 10 Feet Above Fuel	262 hours	187 hours	99 hours

B 3.2 PLANT SYSTEMS

B 3.2.3 Fuel Storage Pool Coolant Temperature

BASES

BACKGROUND

The water temperature in the fuel storage pool is normally controlled by a fuel storage pool cooling system. This system is designed to maintain the pool temperature [≤ 140°F]. In the unlikely event that cooling is interrupted for an extended period of time, the volume of water in the fuel storage pool provides an adequate heat sink for the heat generated by the irradiated fuel. The only requirement for the removal of heat from the irradiated fuel is that the fuel assemblies remain covered with water.

A general description of the fuel storage pool design and the Fuel Storage Pool Cooling and Demineralizer System is given in the FSAR, Section [9.1.2] (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section [9.1.3] (Ref. 2). Assumptions and analysis of a loss of forced cooling for the fuel storage pool are also given in the FSAR.

APPLICABLE SAFETY ANALYSIS

The fuel storage pool cooling system is used to prevent heatup of the water in the pool, which in turn could lead to a loss of coolant inventory due to boiling and evaporation. The fuel storage cooling system is designed to maintain the water in the fuel storage pool at a temperature [≤ 140°F]. The potential for an extended loss of forced cooling has been evaluated and is described in the FSAR. Conservative calculations performed for the Trojan Nuclear Plant were based on a decay time of 1 year after shutdown. The results indicate that more than 42 hours would elapse before the temperature of the water in the fuel storage pool rises from an initial temperature of 140°F to boiling at 212°F. Boiling and evaporation at the surface of the fuel storage pool would continue to provide an adequate heat sink for the irradiated fuel assemblies stored in the pool as long as the fuel assemblies remain covered with water. The rate of water loss due to boiling and evaporation is low; about 9 days would be available from the onset of boiling before the water level in the fuel storage pool drops from 23 feet above the top of the irradiated fuel to 10 feet above the fuel. Ten feet of water above the top of the irradiated fuel would continue to provide an adequate heat sink for the

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

fuel and would protect personnel working in the area from radiation. It would also provide sufficient time to establish makeup flow from a number of facility systems or to arrange for makeup using portable or temporary sources. Methods for providing cooling and makeup water to the fuel storage pool are prescribed by facility procedures.

The initial water inventory of the spent fuel pool is consistent with the assumptions of the analysis of the loss of forced cooling for the spent fuel storage pool described in the FSAR. In the event of an extended loss of forced cooling, the water temperature in the spent fuel storage pool could increase to the point that boiling begins. The heat removed by boiling and evaporation provides an adequate heat sink for the irradiated fuel stored in the pool as long as the fuel assemblies remain covered with water. The rate of water level decrease due to boiling and evaporation is primarily dependent on the volume of water in the spent fuel storage pool and the decay heat load. For the first several months after permanent shutdown, the heat load is primarily attributable to the final reactor core and is determined based on its operating history and the time elapsed since the reactor was shut down.

Based on the data for the Trojan Nuclear Plant, values were estimated for the principal parameters required to evaluate the loss of water from the spent fuel pool caused by boiling and evaporation. Cases were evaluated for assumed fission product decay times of 6 months and 1 month for a reactor with a similar operating history and a spent fuel pool of similar design. These estimates are presented in Table 3.2.2-1.

Values of heat generation and water loss rates presented in Table 3.2.2-1 demonstrate that, in the event forced cooling for the spent fuel pool is lost, ample time is available for restoring the failed cooling system or providing alternative sources of makeup water. The availability of alternative water sources and of time to route these sources into the spent fuel pool ensures the safety of the stored fuel until forced cooling can be restored and the temperature of the fuel storage pool is returned within specified limits.

The Fuel Storage Pool Coolant Temperature LCO satisfies Criterion 2 of the STS PDW.

(continued)

BASES (continued)

LCO

The LCO requires that the coolant temperature in the fuel storage pool be maintained [≤ 140°F] to provide early warning in the event of a loss of forced cooling for the fuel storage pool, thereby facilitating prompt corrective action to avoid uncovering the fuel.

APPLICABILITY

Irradiated fuel stored in the fuel storage pool produces heat due to radioactive decay. Water covering the irradiated fuel provides the required heat sink. This LCO establishes a requirement to monitor abnormal temperature increases in the fuel storage pool, which in turn could lead to a loss of water inventory due to boiling and evaporation. The LCO is applicable whenever irradiated fuel assemblies are stored in the fuel storage pool.

ACTIONS

<u>A.1</u>

An increase in the temperature of water in the fuel storage pool above the specified limit could indicate that the cooling system is not in service. Therefore, Action A.1 requires that immediate action be taken to restore the fuel storage pool temperature to within limit.

<u>A.2</u>

Since a continued increase in the temperature of water in the fuel storage pool could eventually lead to a loss of water inventory due to increased evaporation and subsequent boiling, Action A.2 requires that action be taken to confirm that a source of makeup water is available.

<u>A.3</u>

Action A.3 requires that the temperature be restored to within the limit within [7] days. The [7]-day Frequency allows time to make necessary repairs to restore the effectiveness of the cooling system for the fuel storage pool. Should the temperature increase to the point that the water level in the fuel storage pool decreases to < 23 feet due to increased evaporation and subsequent boiling, LCO 3.2.2 requires that immediate action be taken to initiate makeup flow to the fuel storage pool and restore the water level to within the specified limit.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The temperature of the spent fuel pool coolant is verified to be within the limit at a specified Frequency of 24 hours using a National Institute of Standards Technology (NIST) traceable instrument. The Surveillance requirement ensures that an abnormal increase in the temperature of the spent fuel pool coolant is detected so that appropriate action can be initiated. The 24-hour Frequency is appropriate since the temperature of the spent fuel pool coolant is not subject to rapid changes. This Frequency ensures that an increasing temperature in the spent fuel pool is detected prior to the beginning of a loss of coolant inventory due to boiling and evaporation of the coolant.

REFERENCES

1. FSAR, Section [9.1.2].

2. FSAR, Section [9.1.3].

B 3.2 PLANT SYSTEMS

B 3.2.4 Fuel Storage Pool Load Restrictions

BASES

BACKGROUND	The possibility of a fuel handling accident is very remote because of the many administrative controls and physical limitations imposed on the handling of irradiated fuel. Nonetheless, a potential fuel handling accident caused by dropping one irradiated fuel assembly onto another such assembly stored in the fuel storage racks has been analyzed in the FSAR, Section [15.7.4] (Ref. 1). This analysis evaluated the potential impact loads due to dropping a fuel assembly onto the storage racks during fuel handling operations. The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies ensures that the analyzed fuel handling accident remains bounding.
APPLICABLE SAFETY ANALYSES	A potential fuel handling accident caused by dropping one irradiated fuel assembly onto another such assembly stored in the fuel storage racks has been analyzed in the FSAR. This analysis evaluated the potential impact loads due to dropping a fuel assembly during fuel handling operations. The analysis concluded that the damage to the fuel rod cladding in one fuel assembly would be a conservative bounding assumption for this type of accident. Restricting movement of loads such that impact energies on the storage racks do not exceed [20,000] ft-lb ensures that no more than one fuel assembly will be ruptured in the event of a load-drop accident. The Fuel Storage Pool Load Restrictions LCO satisfies Criterion 2 of the STS PDW.
LCO	This LCO is intended to limit the consequences of a fuel handling or load-drop accident to those that would result from dropping a single fuel assembly or its equivalent load onto stored fuel.
APPLICABILITY	This LCO is applicable whenever irradiated fuel assemblies are stored in the fuel storage pool.

(continued)

BASES (continued)

ACTIONS

A.1

Should the load limitations of the LCO be violated, Action A.1 requires that the load movement cease, thereby providing time to evaluate the potential hazard and select the proper corrective action.

A.2

Should the load limitations of the LCO be violated, Action A.2 requires a deliberate evaluation and selection of an appropriate course of action to return the load to a safe Condition.

A.3

Once an appropriate course of action has been chosen, Action A.3 requires that the load be placed in a safe Condition. This can be accomplished either by reducing the height of the load or by moving the load to a location that is not over storage racks containing irradiated fuel. This action results in a Condition that is acceptable under the LCO, and no further actions are necessary.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

The SR requires that the potential impact energy of a load be determined to be $\leq$ 20,000 ft-lb prior to moving the load over the racks containing irradiated fuel and verifying that the impact energy is below the specified limit.

REFERENCES

1. FSAR, Section [15.7.4].

B 3.2 PLANT SYSTEMS

B 3.2.5 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

For spent fuel storage pools with Maximum Density Racks (MDR) [(Refs. 1 and 2)], the spent fuel storage pool is divided into two separate and distinct regions, which, for the purpose of criticality considerations, are considered as separate pools. [Region 1], with [336] storage positions, is designed to accommodate either new fuel (with a maximum enrichment of [4.65] wt% U-235) or spent fuel (regardless of the discharge fuel burnup). [Region 2], with [2670] storage positions, is designed to accommodate fuel of various initial enrichments that have accumulated minimum burnup within the acceptable domain according to Figure [3.2.6-1], in the accompanying Limiting Condition for Operation (LCO). Fuel assemblies not meeting the criteria of Figure [3.2.6-1] shall be stored in accordance with Specification 4.2.1.1.

The water in the spent fuel storage pool normally contains dissolved boron, which results in large subcriticality margins under actual operating conditions. Based on the accident condition in which all dissolved poison is assumed to have been lost, however, NRC guidelines specify that the limiting k_{eff} of 0.95 be evaluated in the absence of dissolved boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle, discussed in ANSI N-16.1-1975 and in an NRC letter dated April 14, 1978 (Ref. 3), allows credit for dissolved boron under other abnormal or accident conditions because only a single accident need be considered at any one time. For example, the most severe accident scenario is associated either with the movement of fuel from [Region 1 to Region 2] or the accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the k_{eff} of [Region 2]. To prevent these postulated criticality-related accidents, boron is dissolved in the pool water. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.2.6 (Spent Fuel Assembly Storage). Prior to movement of an assembly, it is necessary to perform SR 3.2.6.1.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

For potential accident occurrences, the presence of dissolved boron in the storage pool prevents criticality in both regions. The postulated accidents are basically of two types. The first involves a fuel assembly being incorrectly transferred from [Region 1 to Region 2] (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). The second is associated with a fuel assembly being dropped adjacent to the fully loaded [Region 2] storage rack. Either accident could produce a small positive reactivity effect on [Region 2]. However, the negative reactivity effect of the dissolved boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. These accident analyses are provided in the FSAR, Section [15.7.4] (Ref. 4).

The Fuel Storage Pool Boron Concentration LCO satisfies Criterion 2 of the STS PDW.

LCO

A boron concentration of $\geq$ [2300] ppm is required in the fuel storage pool. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in Reference 4. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and for either movement of fuel within the fuel storage pool or movement of loads over the fuel racks.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool or until a complete spent fuel storage pool verification has been performed after the last movement of fuel assemblies in the spent fuel storage pool. With no further fuel assembly movements in progress, the potential for a misloaded fuel assembly or a dropped fuel assembly does not exist.

(continued)

BASES (continued)

ACTIONS

A.1, A.2, and A.3

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies or loads over the fuel pool. Restoring the concentration of boron and suspending movement of fuel assemblies or loads occur simultaneously. In addition, there is a requirement to verify by administrative means that the fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool. Prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly or load to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7-day Frequency is appropriate because no major replenishment of pool water is expected to take place during such a short timeframe.

REFERENCES

1. Callaway FSAR, Appendix 9.1A, "The Maximum Density Rack (MDR) Design Concept."
2. Description and Evaluation for Proposed Changes to Facility Operating Licenses DPR-39 and DPR-48 (Zion Power Station).
3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
4. FSAR, Section [15.7.4].

B 3.2 PLANT SYSTEMS

B 3.2.6 Spent Fuel Assembly Storage

BASES

BACKGROUND

For spent fuel storage pools with maximum density racks (MDRs) [(Refs. 1 and 2)], the spent fuel storage pool is divided into two separate and distinct regions, which, for the purpose of criticality considerations, are considered as separate pools. [Region 1], with [336] storage positions, is designed to accommodate either new fuel (with a maximum enrichment of [4.65] wt% U-235) or spent fuel (regardless of the discharge fuel burnup). [Region 2], with [2670] storage positions, is designed to accommodate fuel of various initial enrichments that have accumulated minimum burnup within the acceptable domain according to Figure [3.2.6-1], in the accompanying Limiting Condition for Operation (LCO). Fuel assemblies not meeting the criteria of Figure [3.2.6-1] shall be stored in accordance with Specification 4.2.1.1.

The water in the spent fuel storage pool normally contains dissolved boron, which results in large subcriticality margins under actual operating conditions. Based on the accident condition in which all dissolved poison is assumed to have been lost, however, NRC guidelines specify that the limiting k_{eff} of 0.95 be evaluated in the absence of dissolved boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle, discussed in ANSI N-16.1-1975 and in an NRC letter dated April 14, 1978 (Ref. 3), allows credit for dissolved boron under other abnormal or accident conditions because only a single accident need be considered at any one time. For example, the most severe accident scenario is associated either with the movement of fuel from [Region 1 to Region 2] or the accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the k_{eff} of [Region 2]. To prevent these postulated criticality-related accidents, boron is dissolved in the pool water. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.2.6.1.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

These hypothetical accidents can only take place during or as a result of the movement of an assembly (Ref. 4). For these occurrences, the presence of dissolved boron in the spent fuel storage pool (controlled by LCO 3.2.5, Fuel Storage Pool Boron Concentration) and design of the spent fuel storage racks to a $k_{eff} \leq 0.95$ prevents criticality in both regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the likelihood of an accident will be diminished.

The Spent Fuel Assembly Storage LCO pool satisfies Criterion 2 of the STS PDW.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figure [3.2.6-1] in the accompanying LCO, ensure that the k_{eff} of the spent fuel storage pool will always remain < 0.95, assuming that the pool is flooded with unborated water. Fuel assemblies not meeting the criteria of Figure [3.2.6-1] shall be stored in accordance with Specification 4.2.1.1.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in [Region 2] of the fuel storage pool.

ACTIONS

A.1

When the configuration of fuel assemblies stored in [Region 2] of the spent fuel storage pool is not in accordance with Figure [3.2.6-1] or Specification 4.2.1.1, the necessary fuel assembly movement(s) should be initiated to bring the configuration into compliance with Figure [3.2.6-1] or Specification 4.2.1.1.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.6.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure [3.2.6-1] in the accompanying LCO. For fuel assemblies in the unacceptable range of Figure [3.2.6-1], performance of this SR will ensure compliance with Specification 4.2.1.1.

REFERENCES

1. Callaway FSAR, Appendix 9.1A, "The Maximum Density Rack (MDR) Design Concept."

2. Description and Evaluation for Proposed Changes to Facility Operating Licenses DPR-39 and DPR-48 (Zion Power Station).

3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

4. FSAR, Section [15.7.4].

B 3.3 ELECTRICAL POWER SYSTEMS

B 3.3.1 AC Power Systems

BASES

BACKGROUND

The AC electrical power distribution system consists of offsite power sources (preferred power sources, normal, and alternate(s)).

Offsite power is supplied to facility switchyard(s) from the transmission network by [two] transmission lines. From the switchyard(s), two circuits that are separated electrically and physically provide AC power through [stepdown station auxiliary transformers] to the 4.16 kV engineered safety feature (ESF) bus(es). Detailed descriptions of the offsite power network and the circuits to the ESF bus(es) are found in the FSAR, Section [8] (Ref. 1).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the required onsite distribution systems.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC power systems required during movement of fuel assemblies or movement of loads over storage racks containing irradiated fuel ensures that adequate AC electrical power is available to mitigate events postulated during fuel storage (e.g., fuel handling accidents).

Activities in plants that have been permanently defueled are generally well planned and are covered by administrative procedures and programs. Limiting Condition for Operation (LCO) requirements governing power systems in permanently defueled reactors are based on the following:

a. Protection of the fuel cladding, the principal concern for a plant that has been permanently defueled;

b. The ability to reduce significantly the likelihood of a Design Basis Accident (DBA) by taking administrative action (e.g., suspending the movement of fuel assemblies to avoid the possibility of a fuel handling accident);

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. The relatively long period of time available for restoring power to the fuel storage pool cooling system before the fuel cladding could be endangered;

d. The ability to use compensatory measures that may be appropriate under certain conditions. These may include reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses. For example, should electrical power be lost to the fuel storage pool cooling system and not be readily restorable, boiling or evaporation may result in excessive loss of water from the pool. Under these conditions, it may be appropriate to use a nonsafety-grade gasoline powered pump to restore the water level in the fuel storage pool.

In the event of a fuel handling accident, this LCO ensures the capability to supply power for support systems required to mitigate the consequences of a fuel handling accident.

The AC power systems LCO satisfies Criterion 3 of the STS PDW.

LCO

One offsite circuit capable of supplying the onsite power distribution system(s) of LCO 3.3.2, (AC Distribution Systems), ensures that all required loads are powered from offsite power. OPERABILITY of the required offsite circuit ensures the availability of a reliable AC source to mitigate the consequences of a fuel handling accident.

The offsite circuit must be capable of maintaining rated frequency and voltage and accepting required loads during all accidents analyzed for a permanently defueled reactor. Cross-tied trains supplied by a single offsite power circuit are acceptable in permanently defueled facilities.

(continued)

BASES (continued)

APPLICABILITY

The AC power systems required to be OPERABLE during movement of fuel assemblies or movement of loads over storage racks containing fuel within the first [6] months after permanent shutdown of the reactor provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the spent fuel storage pool;

b. Systems needed to mitigate the consequences of a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to fuel damage during the period that fuel is stored in the spent fuel storage pool are available; and

d. Instrumentation and control capabilities are available for monitoring and maintaining safe fuel storage.

ACTIONS

A.1, A.2, and A.3

An offsite circuit would be considered inoperable if it were not available to mitigate the consequences of a fuel handling accident while fuel is being moved. The availability of one train with offsite power is sufficient to support required features allowing continuation of fuel movement within the fuel storage pool. In the event that offsite power is unavailable and required features are not OPERABLE, fuel handling operations are ceased in order to preclude a fuel handling accident.

Further, immediate action must be initiated and continued until the necessary AC power is restored to the required systems.

The Completion Time "Immediately" is consistent with the times specified for actions requiring prompt attention. The restoration of the required AC electrical power source should be completed as quickly as practical, thereby minimizing the time during which required systems are without power and fuel movement is suspended.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.3.1.1 This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution bus(es) and loads are connected to their preferred power source and that appropriate independence of offsite circuits is maintained. The 7-day Frequency is adequate since the status of the breaker position is displayed in the Control Room and thus is not likely to change without the knowledge of the CERTIFIED FUEL HANDLER.
REFERENCES	1. FSAR, Section [8].

B 3.3 ELECTRICAL POWER SYSTEMS

B 3.3.2 Electrical Distribution Systems

BASES

BACKGROUND

The onsite electrical power distribution system for AC, DC, and vital buses is divided by train into [two] redundant and independent AC, DC, and AC vital bus electrical power distribution subsystems. Each train has connections to two preferred offsite power sources.

The AC electrical power subsystem for each train consists of a primary engineered safety feature (ESF) 4.16 kV bus and secondary [480 and 120] V buses, distribution panels, motor control centers, and load centers. Each [4.16 kV ESF bus] has at least [one separate and independent offsite source of power]. Each [4.16 kV ESF bus] is normally connected to a preferred offsite source. After a loss of the preferred offsite power source to a 4.16 kV ESF bus, a transfer to the alternate offsite source is accomplished by using a time-delayed bus undervoltage relay.

The secondary AC electrical power distribution system for each train includes the safety-related load centers, motor control centers, and distribution panels.

The 120 VAC vital buses are arranged in two load groups per train and are normally powered from the inverters. The alternate power supply for the vital buses are constant voltage source transformers powered from the same train as the associated inverter.

There are two independent 125/250 V DC electrical power distribution subsystems (one for each train).

APPLICABLE SAFETY ANALYSES

The initial conditions of the design basis fuel handling accident in the FSAR, Sections [6] and [15] (Refs. 1 and 2), assume that the Fuel Building Air Cleaning System (FBACS) is OPERABLE. The electrical power distribution systems for AC, DC, and AC vital buses are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the FBACS and, thus, mitigate the consequences of a fuel handling accident.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The requirement for OPERABILITY of the electrical power distribution system AC, DC, and AC vital buses is consistent with the initial assumptions of the accident analyses and the requirements for the OPERABILITY supported systems.

The OPERABILITY of the minimum electrical power distribution subsystems for AC, DC, and AC vital buses during movement of irradiated fuel assemblies or movement of loads over fuel storage racks containing irradiated fuel ensures that adequate power is provided to mitigate the consequences of a fuel handling accident.

The Electrical Distribution Systems LCO satisfies Criterion 3 of the STS PDW.

LCO

This LCO explicitly requires energization of the portions of the electrical distribution systems necessary to support OPERABILITY of the FBACS and its required systems, equipment, and components.

Maintaining energized distribution systems as described ensures the availability of sufficient power to operate the fuel storage pool equipment and to mitigate the consequences of postulated fuel handling accidents.

APPLICABILITY

The AC and DC electrical power distribution subsystems are required to be OPERABLE for a period of [6 months] after permanent shutdown of the reactor whenever fuel assemblies are moved in the spent fuel storage pool or loads are moved over fuel storage racks containing irradiated fuel. This provides assurance that electrical power for:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the spent fuel storage pool;

b. Systems needed to mitigate the consequences of a fuel handling accident are available;

(continued)

BASES

APPLICABILITY (continued)

c. Systems necessary to mitigate the effects of events that can lead to fuel damage are available during the period that fuel is in the fuel storage pool; and

d. Instrumentation and control capabilities are available for monitoring and maintaining safe fuel storage.

After [6] months, the fission product inventory of the fuel will have decayed to the point that OPERABILITY of the accident mitigation equipment is no longer required.

ACTIONS

A.1, A.2, and A.3

Although redundant trains of electrical power distribution subsystems may be available and OPERABLE, a single OPERABLE distribution subsystem train will be sufficient to support required features allowing the continuation of irradiated fuel movement or the movement of loads over racks containing irradiated fuel. This stipulation assumes that effective administrative ACTIONS are available to provide a comparable degree of protection to the stored fuel should the single train become inoperable. If the required subsystem train is inoperable, appropriate ACTIONS are promptly implemented to preclude a fuel handling accident by immediately stopping fuel and load handling over racks.

Suspension of these activities does not preclude completion of ACTIONS to place the fuel assembly or load in a safe location. These ACTIONS minimize the probability of a fuel handling accident. Further, immediate action must be initiated and continued until the necessary AC power is restored to the fuel storage building safety systems.

The Completion Time "Immediately" is consistent with the times specified for ACTIONS requiring prompt attention. The restoration of the required distribution systems should be completed expeditiously in order to minimize the time the systems may be without power.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

This Surveillance verifies that the electrical power distribution subsystems for AC, DC, and AC vital bus are functioning properly, with all the buses energized. Verification of the availability of proper voltage on the buses ensures that the required power is readily available for motive as well as control functions to support critical system loads connected to these buses. The 7-day Frequency takes into account the reliability of the electrical power distribution subsystems, and indications available in the Control Room to alert the CERTIFIED FUEL HANDLER to system malfunctions.

REFERENCES

1. FSAR, Chapter [6].
2. FSAR, Chapter [15].

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10. SUPPLEMENTARY NOTES

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11. ABSTRACT *(200 words or less)*

This NUREG report describes the staff's proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants (STS PDW). The report includes a detailed discussion of the strategy followed for determining the contents of the STS PDW. The proposed STS PDW is being published to provide the general public and the nuclear community with an opportunity for comment.

The contents of the proposed STS PDW are based primarily on the Standard Technical specifications, Westinghouse Plants (NUREG-1431, Revision 1, April 1995), which in turn were based on the criteria in the Nuclear Regulatory commission (NRC) Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (SECY-93-067, 58 FR 39132, July 22, 1993). The proposed STS PDW reflect the experience gained in the development of the Permanently defueled Technical specifications (PDTS) for the Trojan Nuclear Plant, the first PDTS approved by the NRC that were based on the improved STS for Westinghouse Plants. As licensees begin to plan permanent shutdown of their nuclear power plants, they are encouraged to adopt the STS PDW to an extent that is practical and consistent with their licensing basis.

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