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See Proposed Change To Tech Specs

SUBJECT: Forwards missing pages of 960827 TS submittal requesting TS change conversion to improved STS consistent w/NUREG-1431, "STS Westinghouse Plants," Rev 1.

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10 CFR 50.90

CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
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RNP File No: 13510HA
Serial: RNP-RA/96-0206

DEC 18 1996

United States Nuclear Regulatory Commission
Attn: Document Control Desk
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE
CONVERSION TO IMPROVED STANDARD TECHNICAL
SPECIFICATIONS CONSISTENT WITH NUREG-1431, "STANDARD
TECHNICAL SPECIFICATIONS-WESTINGHOUSE PLANTS," REVISION 1
ERRATA

Gentlemen:

By letter dated August 27, 1996, we submitted a request for a change to the Technical Specifications (TS) for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, in accordance with 10 CFR 50.90. The proposed change will convert the HBRSEP, Unit No. 2 TS to be consistent with NUREG-1431, "Standard Technical Specifications-Westinghouse Plants," Revision 1.

During NRC review of our submittal, the NRC identified that certain pages of the submittal containing inserted information were missing. We have determined that the missing pages were inadvertently omitted during duplication of the submittal. As a result of the duplication errors, an audit of the submittal package was performed to verify which pages were not present in the submittal. The missing pages are provided in the attachment to this letter with instructions for inserting the pages into each affected enclosure to our letter dated August 27, 1996.

A001

Additionally, the NRC has recently identified discrepancies in the Documentation of Changes (DOCs) for the markup of the Current Technical Specifications (CTS) for Improved Technical Specifications (ITS) Section 3.9, "Refueling Operations." An electronic version of the DOCs provided separately to the NRC contains information missing from the printed copy of the submittal. We are investigating the discrepancies and are performing a review of the submittal

to verify that the identification of changes in markup material are in accordance with discussion information provided. Discrepancies noted during the review will be corrected in the form of revised pages to be inserted into our submittal. We plan to submit the revised pages within the next 30 days.

Enclosure 1 provides an affidavit as required by 10 CFR 50.30(b).

In accordance with 10 CFR 50.91(b), we are providing the State of South Carolina with a copy of this letter with the enclosure and attachment.

If you have any questions concerning this matter, please contact me at (803) 857-1437.

Very truly yours,

H. K. Chernoff
Supervisor - Licensing/Regulatory Programs

ALG/klb
Enclosures:
1. Affidavit

Attachment

c: Mr. Max K. Batavia, Chief, Bureau of Radiological Health (SC)
Mr. S. D. Ebneter, Regional Administrator, USNRC, Region II
Ms. B. L. Mozafari, USNRC Project Manager, HBRSEP (4 copies)
Mr. J. Zeiler, USNRC Resident Inspector, HBRSEP
Attorney General (SC) (w/out Enclosures)
Lockheed Idaho Technology, Inc.

Affidavit

State of South Carolina
County of Darlington

C. S. Hinnant, having been first duly sworn, did depose and say that the information contained in letter 96-0206 is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

C S Hinnant

Sworn to and subscribed before me

this 17th day of December 1996

(Seal) Albert L Garrison
Notary Public for South Carolina

My commission expires: March 22, 2005

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE CONVERSION TO IMPROVED STANDARD TECHNICAL SPECIFICATIONS CONSISTENT WITH NUREG-1431, "STANDARD TECHNICAL SPECIFICATIONS-WESTINGHOUSE PLANTS," REVISION 1 ERRATA

Instructions for Inserting Pages.

Attached pages are to be inserted into enclosures to our letter dated, August 27, 1996. There are no pages to be removed from the enclosures. To facilitate insertion of the attached pages into the affected enclosures to our letter dated, August 27, 1996, the attached pages have been divided into groups corresponding to the applicable enclosure, and separate insert instructions are provided for each group corresponding to the applicable enclosure, separated by a colored page.

1. Enclosure 6 to Serial RNP-RA/96-0141, "Conversion Package Section 1.0"

   a. Part 4, "Markup of NUREG-1431, Revision 1, `Standard Technical Specifications - Westinghouse Plants,' (ISTS)"

Insert Page	Insert After Page
1.1-1	Cover page to Part 4

   b. Part 8, "Proposed HBRSEP, Unit No. 2 ITS (Including Table of Contents)

Insert Page	Insert After Page
Cover Page to Part 8	Cover page to Part 7

1

BASES

BACKGROUND (continued)

The proper functioning of the Reactor Protection System (RPS) and ~~steam generator~~ main steam safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

Insert B 2.1.1-2

The Reactor Trip System setpoints ~~(Ref. 2)~~ specified in LCO 3.3.1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressurizer pressure, and THERMAL POWER level flow, core power distribution that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

3

~~a. High pressurizer pressure trip;~~

~~b. Low pressurizer pressure trip;~~

a. ~~c.~~ Overtemperature ΔT trip;

b. ~~d.~~ Overpower ΔT trip;

c. ~~e.~~ Power Range Neutron Flux trip; and

d. ~~f.~~ ~~Steam generator~~ main steam safety valves.

4

Maintaining the DNBR above the limit ensures ~~The limitation~~ that the average enthalpy in the hot leg is ~~be~~ less than or equal to the enthalpy of saturated liquid and also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

4

(continued)

Insert B2.1.1-2

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by maintaining the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB), and at this point there is a sharp reduction in the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters (i.e., thermal power, reactor coolant temperature and pressure) have been related to DNB through correlations. DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum DNB ratio, or DNBR, during normal operational and anticipated transients, is restricted to the safety limit. A DNBR at the safety limit corresponds to a 95% probability, at a 95% confidence level, that DNB will not occur, and is chosen as an appropriate margin to DNB for all operating conditions. The DNBR safety limit is a conservative design value which is used as a basis for setting core safety limits. Based on rod bundle tests, no fuel damage is expected at this DNBR or greater. For the standard mixing vane fuel, the Siemens Power Corporation XNB correlation has a DNBR safety limit of 1.17 (Ref. 2) and for the high thermal performance fuel the Siemens ANFP correlation has a DNBR safety limit of 1.154 (Ref. 3).

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. ~~2~~ 4) provide more restrictive limits to ensure that the SLs are not exceeded.

U

SAFETY LIMITS

The curves provided in Figure ~~B~~ 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and ~~average~~ reactor vessel inlet temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation. Insert B 2.1.1-3

8

~~The curves are based on enthalpy hot channel factor limits provided in the COLR. The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.~~

The SL is higher than the limit calculated when the Axial Flux Difference ~~AFD~~ is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature and overpower ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. ~~3 and 4~~ 4).

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam ~~steam generator~~ safety valves ~~or~~ and automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Protection ~~Trip~~ System (RPS) Instrumentation." In MODES 3, 4,

5

(continued)

Insert B2.1.1-3

Figure 2.1.1-1 shows the allowable power level decreasing with increasing reactor vessel inlet temperature at selected pressurizer pressures for constant flow (i.e., three loop operation, minimum flow 97.3×10^6 lbm/hr)

The area where clad integrity is assured is below these lines. The temperature limits at low power are considerably more conservative than would be required if they were based on the minimum allowable DNB ratio, but are set to preclude bulk boiling at the vessel exit. The safety limit curves given in Figure 2.1.1-1 are for constant flow conditions. These curves would not be applicable in cases where total reactor coolant flow is less than 97.3×10^6 lbm/hr. The evaluation of such an event would be based upon the analysis presented in Section 15.3 of the UFSAR.

1

BASES

APPLICABILITY (continued)	5, and 6. Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

TSTF-5

~~2.2.1~~

restore compliance and

6

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

in a safe condition and

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

~~2.2.3~~

~~If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).~~

~~2.2.4~~

~~If SL 2.1.1 is violated, the Plant Superintendent and the Vice President - Nuclear Operations shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.~~

~~2.2.5~~

~~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. 6). A copy of the report shall also be provided to the Plant Superintendent and the Vice President - Nuclear Operations.~~

TSTF-5

(continued)

1

BASES

TSTF-5

SAFETY LIMIT VIOLATIONS (continued)

~~2.2.6~~

~~If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~

REFERENCES

7

1. 10 CFR 50, Proposed Appendix A, GDC 10 32FR 10213, July 11, 1967
2. FSAR, Section [X.2] XN-NF-711(P) Rev. 0, "XNB Addendum for 26 Inch Spacer"
3. WCAP-8746-A, March 1977. ANF-1224(P) "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel"
4. WCAP-9273-NP-A, July 1985. 4 FSAR, Sections 3.1, 4.4, 7.2, and 15.0
5. ~~10 CFR 50.72.~~
6. ~~10 CFR 50.73.~~

TSTF-5

8

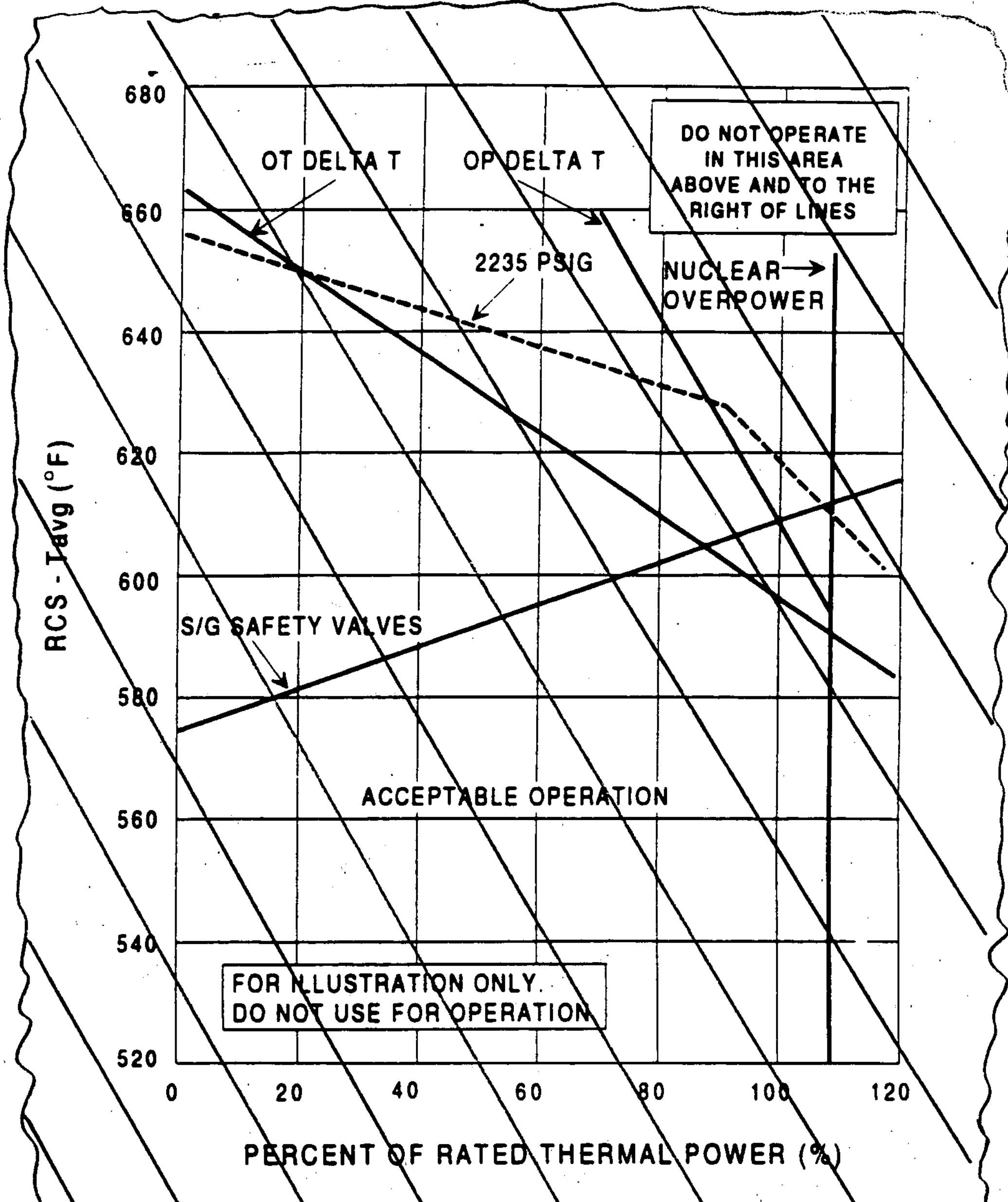


Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

1

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. ~~According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.~~

INSERT B2.1.2-1

2

The design pressure of the RCS is 2485 psig ~~2500 psia~~. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components were ~~are~~ hydrostatically tested at 3110 psig ~~125% of design pressure~~, according to the ASME Code requirements prior to initial operation with ~~when there is~~ no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

9

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

Insert B2.1.2-1

According to 10 CFR 50 Proposed Appendix A (Ref. 1), GDC 9 "Reactor Coolant System Pressure Boundary" and GDC 34 "Reactor Coolant Pressure Boundary (RCPB) Rapid Propagation Failure Prevention," the reactor coolant pressure boundary design conditions are not to be exceeded during normal operations and transients. Also, in accordance with proposed GDC 33, "Reactor Coolant Pressure Boundary Capability," reactivity accidents, including rod ejection and inadvertent and sudden releases of energy to the coolant, do not result in damage to the RCPB.

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

Insert B 2.1.2-2

10

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence safety valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the MSSVs ~~safety valves~~ on the secondary plant are assumed to open when the main steam pressure reaches the MSSV ~~secondary plant safety valve~~ settings. ~~and nominal feedwater supply is maintained.~~

The Reactor Protection ~~RSP~~ System setpoints ~~(Ref. 5)~~ specified in LCO 3.3.1, together with the settings of the RCS Pressurizer Safety valves and the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization ~~(Ref. 5)~~. The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. Main Steam ~~line~~ power operated relief valves;

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valves.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under ~~U~~SAS, Section B31.1 (Ref. ~~6~~ 5) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

(continued)

Insert B2.1.2-2

the reactor is assumed to trip when the RCS pressure reaches the high RCS pressurizer pressure trip setpoint, the RCS pressurizer safety valves are assumed to open when the RCS pressure reaches the RCS safety valve setpoint, and

BASES (continued)

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

~~The following SL violations are applicable to the RCS pressure SL.~~

~~2.2.2.1~~ TSTF-5

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

~~2.2.2.2~~ TSTF-5

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

2.2.4

If the RCS pressure SL is violated, the Plant Superintendent and the Vice President - Nuclear Operations shall be notified within 24 hours. The 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

TSTF 5

2.2.5

If the RCS pressure SL 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. 8). A copy of the report shall also be provided to the Plant Superintendent and the Vice President - Nuclear Operations.

2.2.6

If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

2

REFERENCES

1. 10 CFR 50, Proposed Appendix A, GDC 14, GDC 15, and GDC 28. 32FR10213, July 11, 1967
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 100.

(continued)

BASES

REFERENCES (continued)

~~5. FSAR, Section [X.2].~~

5 ~~6~~. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

~~7. 10 CFR 50.72.~~

~~8. 10 CFR 50.73.~~

TSTF-5

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 7

JUSTIFICATION FOR DIFFERENCES (JFDs) TO ISTS BASES

1. In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis. Such changes are considered to be administrative, as neither technical content nor overall intent has been altered, and therefore have no impact on safety.

2. HBRSEP was designed and licensed to the proposed Appendix A to 10 CFR 50, which was published in the Federal Register on July 11, 1967 (FR 32FR1083). Appendix A to 10 CFR 50 effective in 1971 and subsequently amended, is somewhat different from the proposed 1967 criteria. UFSAR section 3.1 includes an evaluation of HBRSEP with respect to the proposed 1967 criteria. The ISTS statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description in the UFSAR.

3. The "high pressurizer pressure trip" and "low pressurizer pressure trip are deleted. These functions do not bound the protective envelope within the safety limits of ITS Figure 2.1.1-1.

4. The phrase "Maintaining the DNBR above the limit ensures..." is inserted to clarify the reactor heat transfer parameter that is maintained in order to ensure that the core exit temperature is a true representation of core exit enthalpy. When the margin to DNB is maintained, single phase or subcooled conditions will be present at the core exit, or hot leg. By maintaining the hot leg conditions subcooled, proper operation of the OTΔT and OPΔT reactor trip functions can be assured.

5. The term "or" is changed to "and" because it is the combination of the Reactor Protection System (i.e., OPΔT and OTΔT trips) and main steam safety valves that provides the envelope of protection against exceeding the DNBR limits of ITS Figure 2.1.1-1.

6. The phrases "restore compliance and" and "in a safe condition and" are inserted to clarify that the intent of the requirement is not only to attain MODE 3 conditions, but also to establish compliance with the safety limits as required by 10 CFR 50.36(c)(i) and to maintain the plant in a safe condition.

7. The references are modified to be consistent with other marked changes and current licensing basis.

8. ITS Figure B 2.1.1-1 is deleted. Discussion of safety limits in the Bases references ITS Figure 2.1.1-1.

9. The design pressure of the RCS is changed to the plant specific value of "2485 psig," in accordance with UFSAR Table 5.1.0-1. The phrase "125% of design pressure" is changed to "3110 psig," which agrees with the initial testing conditions specified in UFSAR Table 5.1.0-1. The word "are" is changed to "were" to reflect previous performance of the 3110 psig initial hydrostatic test.

10. Additional assumptions are added to the conditions defining the pressure excursion resulting from a complete loss of external load without a direct reactor trip. The initial conditions for this analyzed event are discussed in UFSAR Section 15.2.2.3.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 8

PROPOSED HBRSEP, UNIT NO. 2 ITS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest cold leg temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

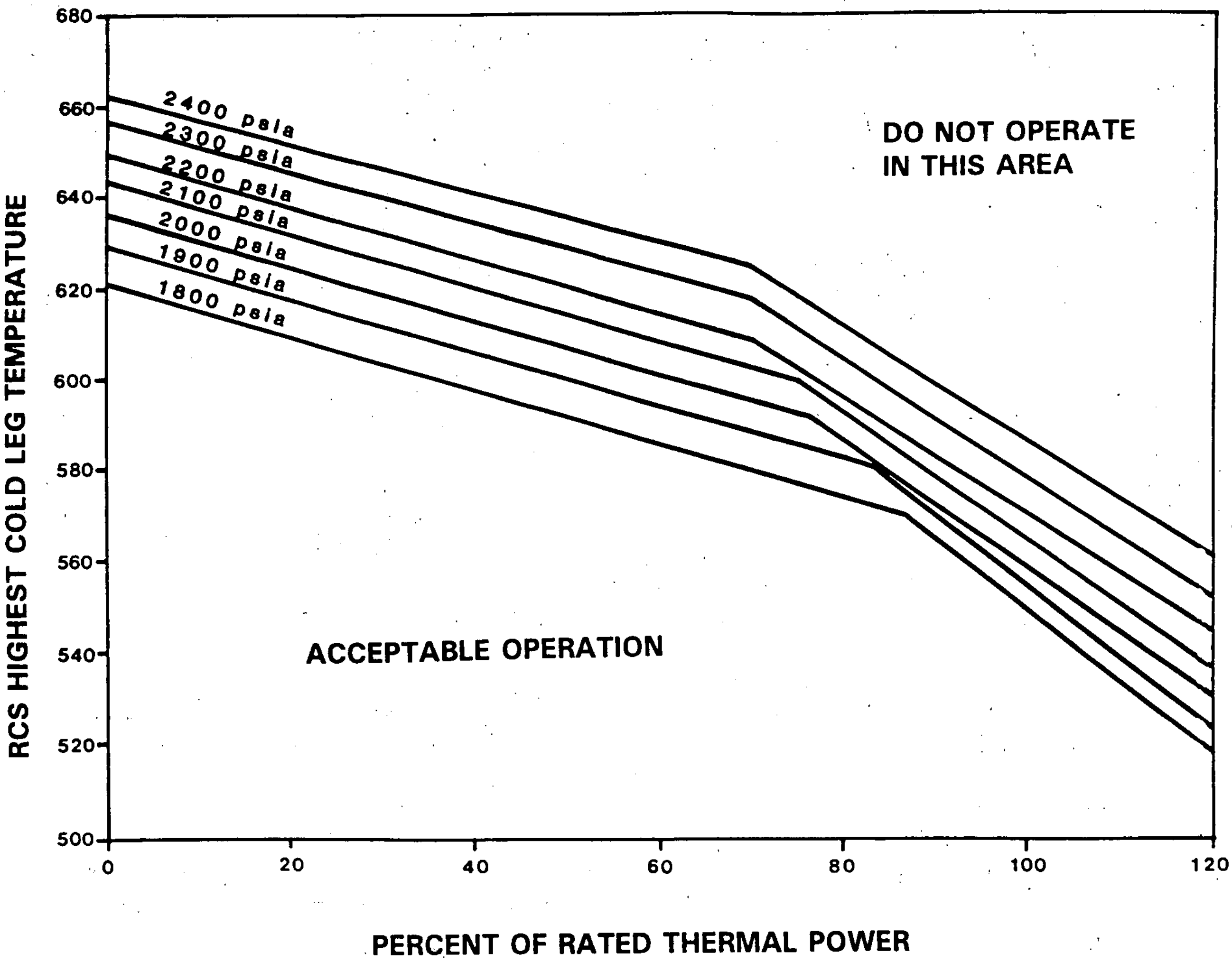
2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.



NOTE: BASED ON A MINIMUM RCS FLOW OF 97.3×10^6 lbm/hr

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety limits

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 9

PROPOSED BASES TO HBRSEP, UNIT NO. 2 ITS

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

The General Design Criteria (GDC) in existence at the time HBRSEP Unit No. 2 was licensed for operation (July 1970) were contained in the proposed Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," published in the Federal Register on July 11, 1967 (Ref. 1). Proposed GDC-6 required that the reactor core with its related controls and protection systems be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which had been stipulated and justified. The core and related auxiliary system designs provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat generation rate (LHGR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHGR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

(continued)

BASES

BACKGROUND (continued)

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

The proper functioning of the Reactor Protection System (RPS) and main steam safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by maintaining the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB), and at this point there is a sharp reduction in the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters (i.e., thermal power, reactor coolant temperature and pressure) have been related to DNB through correlations. DNB correlations have been developed to

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum DNB ratio, or DNBR, during normal operational and anticipated transients, is restricted to the safety limit. A DNBR at the safety limit corresponds to a 95% probability, at a 95% confidence level, that DNB will not occur, and is chosen as an appropriate margin to DNB for all operating conditions. The DNBR safety limit is a conservative design value which is used as a basis for setting core safety limits. Based on rod bundle tests, no fuel damage is expected at this DNBR or greater. For the standard mixing vane fuel, the Siemens Power Corporation XNB correlation has a DNBR safety limit of 1.17 (Ref. 2) and for the high thermal performance fuel the Siemens ANFP correlation has a DNBR safety limit of 1.154 (Ref. 3).

The Reactor Trip System setpoints specified in Limiting Condition for Operations (LCO) 3.3.1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressurizer pressure, flow, core power distribution, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. Overtemperature ΔT trip;

b. Overpower ΔT trip;

c. Power Range Neutron Flux trip; and

d. Main steam safety valves.

Maintaining the DNBR above the limit ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid and also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

(continued)

COLORED SEPARATION PAGE

3. Enclosure 8 to Serial RNP-RA/96-0141, "Conversion Package Section 3.0"

   a. Part 8, "Proposed HBRSEP, Unit No. 2 ITS"

Insert Page	Insert After Page
3.0-4, 3.0-5	3.0-3

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions 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 failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed 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.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

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 the LCO not met may be delayed, from the time of discovery, 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

(continued)

3.0 SR APPLICABILITY

SR 3.0.3 (continued)	declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

COLORED SEPARATION PAGE

4. Enclosure 9 to Serial RNP-RA/96-0141, "Conversion Package Section 3.1"

   a. Part 3, "No Significant Hazards Consideration (NSHC), And Basis For Categorical Exclusion From 10 CFR 51.22"

Insert Page	Insert After Page
8	7

   b. Part 4, "Markup of NUREG-1431, Revision 1, `Standard Technical Specifications - Westinghouse Plants,' (ISTS)"

Insert Page	Insert After Page
3.1-3	3.1-2

   c. Part 5, "Justification for Differences (JFDs) to ISTS

Insert Page	Insert After Page
3	2

preferable to mandating a unit shutdown with the increased risk for shutdown transients. Therefore, this change does not involve a significant reduction in a margin of safety.

L3 Change

Carolina Power & Light Company has evaluated the Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the change does not involve a significant hazards consideration are discussed below.

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

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The Frequency of performance of the surveillance does not increase the probability of occurrence of any analyzed event, since the function of the equipment, or limit for the parameter does not change. Further, the Frequency of performance of a surveillance does not increase the consequences of an accident because a change in surveillance frequency does not change the assumed response of the equipment to perform its specific mitigation functions. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The change will still ensure compliance with the limiting condition for operation is maintained. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   There are no margins of safety, related to any safety analysis, that are dependent upon the change. The change increases a surveillance interval. Exercising each rod every 92 days provides confidence that rods continue to be trippable. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Therefore, this change does not involve a significant reduction in a margin of safety.

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Core Reactivity

[M3] LCO 3.1.3 2 The measured core reactivity shall be within ± 1% Δk/k of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[M3] A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	72 hours
	AND	
[M3]	A.2 Establish appropriate operating restrictions and SRs.	72 hours
[M3] B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

ITS SR 3.1.8.1 requires a COT within 12 hours prior to initiation of PHYSICS TESTS regardless of whether the COT has been performed within its required Frequency. Initiation of PHYSICS TESTS does not impact the ability of the monitors to perform their required Function, and does not affect the Trip Setpoints or RPS trip capability, and does not invalidate previous surveillances. Therefore, an additional surveillance required to be performed "prior to" this event is an extraneous and unnecessary performance of a surveillance.

COLORED SEPARATION PAGE

5. Enclosure 12 to Serial RNP-RA/96-0141, "Conversion Package Section 3.4"

a. Part 1, "Markup of Current Technical Specifications (CTS)

Insert Page	Insert After Page
CTS Page 3.1-11 for ITS Specification 3.4.9	CTS Page 3.3-5a for ITS Specification 3.4.8

b. Part 5, "Discussion of Changes (DOCs) for CTS Markup"

Insert Page	Insert After Page
21	20

c. Part 4, "Markup of NUREG-1431, Revision 1, `Standard Technical Specifications - Westinghouse Plants,' (ISTS)"

Insert Page	Insert After Page
3.4-27	3.4-26
3.4-29	ITS Inserts 3.4.12-2 and 3.4.12-3
3.4-31	ITS Inserts 3.4.12-4 and 3.4.12-5

ITS

A1

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, at which the moderator temperature coefficient is outside the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limits shall be less than or equal to:

See 3.1.3

a) +5.0 pcm/°F at less than 50% of rated power, or

b) 0 pcm/°F at 50% of rated power and above.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1.

See 3.4.2

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is outside the limits specified in the COLR, the reactor shall be made subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

See 3.1.3

not enter MODES 1, 2, 3

M17

[LCO 3.4.9.a] [LCO 3.4.9.b] 3.1.3.4 The reactor shall ~~be maintained subcritical by at least 1%~~ until normal water level is established in the pressurizer.

≤ 63.3% in MODE 1,
≤ 92% in MODES 2 and 3

A13

Basis

~~During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator temperature coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator temperature coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator temperature coefficient outside the limits specified in the COLR has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase in moderator temperature or decrease in coolant pressure. This requirement is~~

A2

controlled documents. Relocation of the specific requirements for systems or variables contained in these Specifications to licensee documents will have no impact on the operability or maintenance of those systems or variables. The licensee will initially continue to meet the requirements contained in the relocated Specifications. The licensee is allowed to make changes to these requirements in accordance with the provisions of 10 CFR 50.59. Such changes can be made without prior NRC approval, if the change does not involve an unreviewed safety question, as defined in 10 CFR 50.59. These controls are considered adequate for assuring that structures, systems, and components in the relocated Specifications are maintained operable, and variables are maintained within limits. This change is consistent with the NRC Final Policy Statement on Technical Specification Improvements.

1

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

[3.1.2.1.d] [3.3.1.3] LCO 3.4.12

Insert 3.4.12-1

~~An LTOP System shall be OPERABLE with a maximum of [one] [high pressure injection (HPI)] pump [and one charging pump] capable of injecting into the RCS and the accumulators isolated and either a or b below:~~ 15

~~a. Two RCS relief valves, as follows:~~

~~1. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR or~~ 4

~~[2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ [436.5] psig and ≤ [463.5] psig, or]~~ 21

~~[3. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint ≥ [436.5] psig and ≤ [463.5] psig].~~

~~b. The RCS depressurized and an RCS vent of ≥ [2.07] square inches.~~ 15

[3.1.2.1.d] APPLICABILITY: MODE 4 ~~when all RCS cold leg temperature is ≤ [275]°F~~, 5 and 5 15
~~MODE 5~~
MODE 6 when the reactor vessel head is on.

[M25]

------------------------------NOTE------------------------------
Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in ~~the PTLR~~ Figures 3.4.3-1 and 3.4.3-2. 4

1

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
[M25]	~~D.~~ E. Required Action and associated Completion Time of Condition ~~[D]~~ E not met.	~~D.1~~ E.1	Increase RCS cold leg temperature to > ~~[275]~~ 350 °F.	12 hours
		OR		
		~~D.2~~ F.2	Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in ~~the PTLR~~ Figures 3.4.3-1 and 3.4.3-2. 4	12 hours
[3.1.2.1.d]	~~E.~~ G. One required ~~RCS relief valve~~ PORV 21 inoperable in MODE 4.	~~E.1~~ G.1	Restore required ~~RCS relief valve~~ PORV 21 to OPERABLE status.	7 days
[3.1.2.1.d]	~~G.~~ H. One required ~~RCS relief valve~~ PORV 21 inoperable in MODE 5 or 6.	~~F.1~~ H.1	Restore required ~~RCS relief valve~~ PORV 21 to OPERABLE status.	24 hours

(continued)

CTS

1

SURVEILLANCE REQUIREMENTS (continued)

CTS	SURVEILLANCE		FREQUENCY
	[SR 3.4.12.4	Verify RHR suction valve is open for each required RHR suction relief valve.	12 hours] 21
[3.1.2.1.4]	SR 3.4.12.5 6	------NOTE------ Only required to be performed met 28 when complying with LCO 3.4.12.b. ------------ Verify RCS vent ≥ [2.07] 3 square inches open.	12 hours for unlocked open vent valve(s) AND 31 days for locked open vent valve(s)
[4.2.5.1.c]	SR 3.4.12.6 7	Verify PORV block valve is open for each required PORV.	72 hours
	[SR 3.4.12.7	Verify associated RHR suction isolation valve is locked open with operator power removed for each required RHR suction relief valve.	31 days] 21
[4.2.5.1.a] [T4.1-1 (31)]	SR 3.4.12.8	------NOTE------ ~~Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ [275]°F.~~ 29 ------------ Perform a COT on each required PORV, excluding actuation.	Once within 31 days prior to entering MODE 4, 5, or 6 when the reactor vessel head is on AND 29 31 days thereafter 29

(continued)

COLORED SEPARATION PAGE

6. Enclosure 13 to Serial RNP-RA/96-0141, "Conversion Package Section 3.5"

   a. Part 1, "Markup of Current Technical Specifications (CTS)

Insert Page	Insert After Page
CTS Page 3.3-1 for ITS Specification 3.5.4	CTS Page 3.3-5 for ITS Specification 3.5.3

   b. Part 4, "Markup of NUREG-1431, Revision 1, `Standard Technical Specifications - Westinghouse Plants,' (ISTS)"

Insert Page	Insert After Page
3.5.2-4a	3.5-4

Specification 3.5.4

ITS

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3.3 EMERGENCY CORE COOLING SYSTEM, AUXILIARY COOLING SYSTEMS, AIR RECIRCULATION FAN COOLERS, CONTAINMENT SPRAY, POST ACCIDENT CONTAINMENT VENTING SYSTEM, AND ISOLATION SEAL WATER SYSTEM

~~Applicability~~

~~Applies to the operating status of the Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, Containment Spray, Post Accident Containment Venting System, and Isolation Seal Water System.~~

~~Objective~~

~~To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment and critical components in normal operating and emergency situations, and (3) to remove airborne iodine from the containment atmosphere following a postulated Design Basis Accident.~~

A13

Specification

IN MODES 1, 2, 3 + 4

[Applicability] 3.3.1 Safety Injection and Residual Heat Removal Systems

3.3.1.1 ~~The reactor shall not be made critical unless~~ the following conditions are met:

Shall be OPERABLE

M19

[LCO 3.5.4] [SR 3.5.4.2] [SR 3.5.4.3] a. The refueling water tank ~~contains not less than 300,000 gallons of water with a boron concentration of at least 1950 ppm.~~

M19

Add RA A.1

L7

M19

Add RA B.1

RA C.1 + C.2

M18

Add SR 3.5.4.1

M19

ITS Insert 3.5.2-1

----------------------------------NOTE----------------------------------

Entry into Conditions and Required Actions may be delayed in MODE 3, when one SI pump flow path is isolated for up to 24 hours to perform pressure isolation valve testing per SR 3.4.14.1.

COLORED SEPARATION PAGE

7. Enclosure 14 to Serial RNP-RA/96-0141, "Conversion Package Section 3.6"

   a. Part 6, "Markup of ISTS Bases"

Insert Page	Insert After Page
B 3.6-9a	B 3.6-9
B 3.6-27a	B 3.6-27
B 3.6-28a	B 3.6-28
B 3.6-30a	B 3.6-30
B 3.6-31a	B 3.6-31
B 3.6-33a	B 3.6-33
B 3.6-35a	B 3.6-35

Insert B3.6.1-2A

Air lock leakage is not acceptable if its

Insert B3.6.1-2

the Containment Leakage Rate Testing Program.

Insert B3.6.2-2

used for entry and exit (procedures require strict adherence to single door opening),

Insert 3.6.2-3

every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

Insert B3.6.2-4

given that the interlock is not challenged during use of the interlock.

Insert B3.6.3-2

Inboard purge supply and exhaust valves are restricted from exceeding 70 degrees open. This restriction assures proper valve closure under dynamic conditions and consequently limits offsite dose consequences resulting from a DBA which occurs when the valves are open.

Insert B3.6.3-3

They may be opened during plant operation when needed for safety related reasons (both equipment and personnel) to support plant operations and maintenance activities within the containment.

Insert B3.6.3-4

Containment Pressure and Vacuum Relief Valves

The containment pressure and vacuum relief valves are provided to control variations in containment pressure with respect to atmospheric pressure which may result from air temperature changes, barometric pressure changes or air in-leakage. These valves are normally maintained closed, however they may be opened as needed in MODES 1, 2, 3 and 4 to equalize internal and external pressure, provided that they are not open simultaneously with the containment purge valves.

Insert B3.6.3-5

Isolation of containment ventilation isolation valves is complete within approximately two seconds following generation of the phase A containment isolation signal. Isolation of the remaining containment isolation valves is complete within approximately ten seconds following generation of either the phase A or phase B containment isolation signal. Upon completion of containment isolation, leakage is

Insert B3.6.3-6

air-cylinder operators, with spring assisted closure capable of closing valves in two seconds. These valves fail to the closed position on a loss of a control signal or instrument air.

Insert B3.6.3-7A

In the event required IVSW supply is isolated to a penetration flowpath, Note 5 directs entry into applicable Conditions and Required Actions of LCO 3.6.8.

Insert B3.6.3-8

A check valve may not be used to isolate the penetration. The device used to isolate the penetration should be the one closest to the containment.

Insert B3.6.3-9

For affected penetration flow paths that cannot be restored to OPERABLE status within the specified Completion Time and that have been

Insert B3.6.3-10

- Not used. -

COLORED SEPARATION PAGE

8. Enclosure 16 to Serial RNP-RA/96-0141, "Conversion Package Section 3.8"

   a. Part 1, "Markup of Current Technical Specifications (CTS)

Insert Page	Insert After Page
CTS Page 3.7-2a for ITS Specification 3.8.1	CTS Page 3.7-2 for ITS Specification 3.8.1

   b. Part 4, "Markup of NUREG-1431, Revision 1, `Standard Technical Specifications - Westinghouse Plants,' (ISTS)"

Insert Page	Insert After Page
3.8-13a	3.8-13

   c. Part 6, "Markup of ISTS Bases"

Insert Page	Insert After Page
B 3.8-26a	B 3.8-26

Specification 3.8.1

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ITS

e) During periods when a diesel generator is being operated for testing purposes, its protective trips ~~listed in Specification 3.7.1.d~~ need not be bypassed after the diesel generator has properly assumed the load on its bus.

LA1

[SR 3.8.1.3 Note 5]

[SR 3.8.1.8 Note 3]

[SR 3.8.1.9 Note 3]

[SR 3.8.1.11 Note 3]

[SR 3.8.1.14 Note 3]

[SR 3.8.1.15 Note 2]

Insert 3.8.1-7A

3. During periods when a diesel generator is being operated for testing purposes, its protective trips need not be bypassed after the diesel generator has properly assumed the load on its bus.

Insert 3.8.1-7

and frequency ≥ 58.8 Hz, and after steady state conditions are reached, maintains voltage ≥ 467 V and ≤ 493 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.

Insert B3.8.1-7

Note 3 to this SR permits removal of the bypass for protective trips after the DG has properly assumed its loads on the bus. This reduces exposure of the DG to undue risk of damage that might render it inoperable.

COLORED SEPARATION PAGE

9. Enclosure 21 to Serial RNP-RA/96-0141, "Compilation of CTS Pages"

Insert Page	Insert After Page
CTS License page 4 for ITS Specification 5.5	At beginning of Enclosure
CTS Page 4.1-10 for ITS Specification 3.5.1	CTS Page 4.1-10 for ITS Specification 3.4.16
CTS Page 4.1-11 for ITS Specification 3.7.15	CTS Page 4.1-11 for ITS Specification 3.4.16
CTS Page 4.1-12 for ITS Specification 3.4.13	CTS Page 4.1-12 for ITS Specification 3.4.10
CTS Page 6.9-4 for ITS Specification 5.6	CTS Page 6.9-3 for ITS Specification 5.6
CTS Page 6.16-1 for ITS Specification 5.5	CTS Page 6.15-1 for ITS Chapter 5.0 Relocated Specifications

Additionally, CTS pages 3.3-1 through 3.3-4 were issued out of order after page 3.3-10. Rearrange the pages in page number order by ascending ITS Specification as annotated in the upper right hand corner.

(FOL DPR-23)

Specification 5.5

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ITS

F. Physical Protection

NO CHANGE

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "H. B. Robinson Steam Electric Plant Industrial Security Plan," with revisions submitted through October 21, 1987; "H. B. Robinson Steam Electric Plant Security Personnel Training and Qualification Plan," with revisions submitted through January 16, 1987; and "H. B. Robinson Steam Electric Plant Safeguards Contingency Plan," with revisions submitted through March 27, 1986. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

G. The following programs shall be implemented and maintained by the licensee:

(1) A secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include: the identification of critical parameters, their sampling frequency, sampling points and control band limits; requirements for the documentation and review of sample results; the identification of the authority responsible for the interpretation of sample results; the procedures used to measure the critical parameters; and the procedures which identify the administrative events and corrective actions required to return the secondary chemistry to its normal control band following an out of control band condition.

See 5.5.10

[5.5.2]

(2) A program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include: provisions for preventive maintenance and periodic visual inspection requirements, and integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

(3) ~~A program to determine the airborne iodine concentration in vital areas under accident conditions. This program shall include: training of personnel, procedures for monitoring, and provisions for maintenance of sampling and analysis equipment.~~

LA5

(4) A program to ensure the capability to obtain and analyze reactor coolant, radioactive iodines, and particulates in plant gaseous effluents, and containment atmosphere samples

See 5.5.3

The systems include RHR, SIS, CSS, Post Accident Containment Ventilation; and portions of CVCS, Liquid and gaseous waste Disposal, and Sampling.

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ITS

M8

and once within six hours after each solution volume increase of ≥ 270 Gallons that is not the result of addition from the RWST

TABLE 4.1.2

FREQUENCIES FOR SAMPLING TESTS

	Check	Frequency	Maximum Time Between Tests	
1. Reactor Coolant Samples	- Gross Activity[1]	Minimum 1 Per 72 hrs.	3 days	see 3.4.1
	- Radiochemical[2]	Monthly	45 days	
	- Radiochemical for $\bar{E}$ Determination	1 per 6 mos.[6][7]	6 months	
	- Isotopic Analysis for Dose Equivalent I-131 Concentration	1 per 14 days[7]	14 days	
	- Isotopic Analysis for Iodine Including I-131, I-133 and I-135	a) Once per 4 hours[8] b) One sample[9]		
	- Tritium Activity	Weekly	10 days	
	- Cl & O_2	5 day/week	3 days	
2. Reactor Coolant Boron	Boron concentration	Twice/week	5 days	
3. Refueling Water Storage Tank Water Sample	Boron concentration	Weekly	10 days	see 3.5.4
4. Boric Acid Tank	Boron concentration	Twice/week	5 days	CTS 3.4
5. Spray Additive Tank	NaOH concentration	Monthly	45 days	see 3.6
[SR 3.5.1.4] 6. Accumulator	Verify Boron concentration M9 in each accumulator is ≥ 1950 ppm + ≤ 2400 ppm	~~Monthly~~ 31 days	M10 ~~45 days~~	
7. Spent Fuel Pit	Boron concentration	Prior to Refueling or New Fuel Movement in the Spent Fuel Pit	NA*	see 3.7
8. Secondary Coolant	Gross activity Isotopic Analysis for Dose Equivalent I-131 Concentration	Minimum 1 Per 72 hrs. a) 1 per 31 days[10] b) 1 per 6 months[11]	3 days	see 3.7.1:
9. Stack Gas Iodine & Particulate Samples	I-131 and particulate radioactivity releases	Weekly[3]	10 days	see 3.4.16
10. Steam Generator Samples	Primary to secondary tube leakage	5 days/week	3 days	see 3.4.13

A1

NOTES TO TABLE 4.1-2

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant of units of μCi/gram.

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

See 3.4.16

(3) When iodine or particulate radioactivity levels exceed 10% of the limit in Specification 3.9.2.1. the sampling frequency shall be increased to a minimum of once each day.

(5) Deleted.

(6) Sample to be taken after a minimum of 2EFPD and 20 days of power operation have elapsed since the reactor was last subcritical for 48 hours or longer.

(7) Samples are to be taken in the power operating condition.

See 3.4.16

(8) Sample taken at all operating conditions whenever the specific activity exceed 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/E μCi/gram. These samples are to be taken until the specific activity of the reactor coolant system is restored within its limits.

(9) One sample between 2 and 6 hours following a thermal power change exceeding 15 percent of the rated thermal power within a one-hour period. Samples are required when in the hot shutdown or power operating modes.

~~(10) Sample whenever that gross activity determination indicates iodine concentrations are greater than 10% of the allowable limit.~~

~~(11) Sample whenever the gross activity determination indicates iodine concentrations are below 10 percent of the allowable limit.~~

L18

~~NA - Not applicable.~~

Specification 3.4.13

(A1)

TABLE 4.1-3

FREQUENCIES FOR EQUIPMENT TESTS

		Check	Frequency	Maximum Time Between Tests	
1.	Control Rods	Rod drop times of all full length rods	Each refueling shutdown	NA	See 3.1.4
2.	Control Rod	Partial movement of all full length rods	Every 2 weeks during reactor critical operations	20 days	
3.	Pressurizer Safety Valves	Set point	Each refueling shutdown	NA	See 3.4.10
4.	Main Steam Safety Valves	Verify each required MSSV lift setpoint per Table 4.1-4 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within +/- 1%.	In accordance with the Inservice Testing Program	NA	See 3.7.1
5.	Containment Isolation Trip	Functioning	Each refueling shutdown	NA	See 3.6.3 + 3.3.2
6.	Refueling System Interlocks	Functioning	Prior to each refueling shutdown	NA	See 3.9.1
7.	Service Water System	Functioning	Each refueling shutdown	NA	See 3.7.7
~~8.~~	~~DELETED~~				
[SR 3.4.13.1] 9.	Primary System ~~Leakage~~ water inventory balance	Evaluate	~~Daily~~ when reactor coolant system is above cold shutdown condition	~~NA~~ MODES 1,2,3+4	
10.	Diesel Fuel Supply	Fuel Inventory	Weekly	10 days	See 3.8.3
~~11.~~	~~DELETED~~				
12.	Turbine Steam Stop, Control, Reheat Stop, and Interceptor Valves	Closure	Quarterly during power operation and prior to startup	115 days	See 3.7.1

Once within 12 hours after reaching steady state operation conditions
AND
72 hours thereafter during steady state operation

(L11)

LAR dated 1/29/96

ITS [5.6.3]

Specification 5.6

A1

frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figure 1.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. [For ORs: approximate and conservative approximate methods are acceptable.] The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).

3. The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. LA22

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* In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request. LA22

Specification 5.5

ITS A.I

A1

[5.5.1] 6.16 OFFSITE DOSE CALCULATION MANUAL

6.16.1 ~~The ODCM shall be approved by the Commission prior to implementation.~~ A12

6.16.2 C, Licensee initiated changes to the ODCM: documented and records of reviews performed shall be retained.

1. Shall be ~~submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective.~~ This ~~submittal~~ documentation shall contain: A13

(a) Sufficient~~ly detailed~~ information to ~~totally~~ support the ~~rationale for the~~ change ~~without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box,~~ together with appropriate analyses or evaluations justifying the change(s);

(b) A determination that the change will not reduce the accuracy or reliability of effluent dose calculations or setpoint determinations ~~and~~ maintains the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and A14

~~3. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.~~ LA21

2. Shall become effective upon ~~review and acceptance by the PNSC~~ approval of the plant manager. M4

Add 5.5.1.b M12

Add 5.5.1.c.3 A13