FPL Energy
Seabrook Station

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February 3, 2003

Docket No. 50-443
NYN-03006

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Seabrook Station
License Amendment Request 02-08
"Changes To Technical Specification Associated With
Secondary Coolant Specific Activity"

FPL Energy Seabrook, LLC (FPLE Seabrook) has enclosed herein License Amendment Request (LAR) 02-08. License Amendment Request 02-08 is submitted pursuant to the requirements of 10 CFR 50.90 and 10 CFR 50.4.

LAR 02-08 proposes changes to the Seabrook Station Technical Specification (TS) 3/4.7.1.4, "Turbine Cycle – Specific Activity" and associated Bases. Part of the proposed change will adopt similar wording as presented in the improved Standard Technical Specifications for Westinghouse Plants, NUREG-1431, Revision 2. Also proposed is an exemption from the requirements of TS 4.0.4 when entering MODE 4, along with conditions for when the surveillance requirement must be satisfied in MODE 4.

The TS Index is revised as well to reflect the previously discussed changes.

The proposed change will afford FPLE Seabrook operational flexibility, particularly during and immediately after refueling outages, when sampling and analysis do not provide relevant information to the Technical Specifications.

The Station Operation Review Committee and the Nuclear Safety Audit Review Committee have reviewed LAR 02-08.

As discussed in the enclosed LAR Section IV, the proposed change does not involve a significant hazard consideration pursuant to 10 CFR 50.92. A copy of this letter and the enclosed LAR has been forwarded to the New Hampshire State Liaison Officer pursuant to 10 CFR 50.91(b).

FPLE Seabrook has determined that LAR 02-08 meets the criterion of 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

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FPLE Seabrook requests NRC Staff review of LAR 02-08, and issuance of a license amendment by October 1, 2003.

Should you have any questions regarding this letter, please contact Mr. James M. Peschel, Manager – Regulatory Programs, at (603) 773-7194.

Very truly yours,

FPL Energy Seabrook, LLC

Mark E. Warner

Mark E. Warner
Site Vice President

cc: H. J. Miller, NRC Region I Administrator
R. D. Starkey, NRC Project Manager, Project Directorate I-2
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FPL Energy
Seabrook Station

SEABROOK STATION UNIT 1

Facility Operating License NPF-86
Docket No. 50-443

License Amendment Request 02-08,
"Changes To Technical Specification Associated With Secondary Coolant Specific Activity"

FPL Energy Seabrook, LLC, pursuant to 10 CFR 50.90 submits License Amendment Request 02-08. The following information is enclosed in support of this License Amendment Request:

- Section I - Introduction and Safety Assessment for Proposed Changes
- Section II - Markup of Proposed Changes
- Section III - Retype of Proposed Changes
- Section IV - Determination of Significant Hazards for Proposed Changes
- Section V - Proposed Schedule for License Amendment Issuance And Effectiveness
- Section VI - Environmental Impact Assessment

I, Mark E. Warner, Site Vice President of FPL Energy Seabrook, LLC, hereby affirm that the information and statements contained within this License Amendment Request are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed
before me this
3rd day of February, 2003

Victoria Susan Robertson
Notary Public

Mark E. Warner
Mark E. Warner
Site Vice President

VICTORIA SUSAN ROBERTSON
MY COMMISSION EXPIRES JAN. 15, 2008
NOTARY PUBLIC
NEW HAMPSHIRE

SECTION I

INTRODUCTION AND SAFETY ASSESSMENT FOR PROPOSED CHANGE

I. INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

A. Introduction and Description of Change

Introduction

License Amendment Request (LAR) 02-08 proposes changes to the Seabrook Station Technical Specification (TS) 3/4.7.1.4, "Turbine Cycle – Specific Activity," and associated Bases. Part of the proposed changes will adopt similar wording as presented in the improved Standard Technical Specifications (ITS) for Westinghouse Plants, NUREG-1431, Revision 2. Some of the wording of the proposed changes is slightly different than that of ITS. However, these differences are of a non-technical nature and equivalent to the ITS wording. Also proposed is an exemption from the requirements of TS 4.0.4 when entering MODE 4, along with conditions for when the surveillance requirement must be satisfied in MODE 4.

The TS Index is revised as well to reflect the previously discussed changes.

The proposed change will afford FPLE Seabrook operational flexibility, particularly during and immediately after refueling outages when sampling and analysis do not provide relevant information to satisfy Technical Specification requirements.

Description of Proposed Changes

Index Pages

Index page vii is being revised to reflect the change associated with the deletion of Table 4.7-1. The title is being deleted and replaced with "This Table Number is not used."

Technical Specification 3.7.1.4

The Current Technical Specification (CTS) LCO and ACTION statement use the terms "microCurie/gram" and "Secondary Coolant System". These terms are being replaced with "µCi/gm" and "secondary coolant".

The CTS APPLICABILITY for MODE 4 is being conditioned with the following note:

> "The provisions of Specification 4.0.4 are not applicable for entry into MODE 4, however, once steam generator pressure exceeds 100 psig, the requirements of Specification 4.7.1.4 must be met within 12 hours if not performed within the past 31 days."

The CTS Surveillance Requirement (SR) states the following:

> "The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1."

The proposed TS SR states the following:

> "At least once every 31 days, verify the specific activity of the secondary coolant is less than or equal to 0.1 µCi/gm DOSE EQUIVALENT I-131."

The changes associated with the proposed SR consist of the following:

- Table 4.7-1, item 1, "Gross Radioactivity Determination" "At least once per 72 hours" is being deleted.
- Table 4.7-1, items 2a) and 2b) are being deleted and replaced with a sample and analysis frequency of "At least every 31 days" in the proposed SR.
- Table 4.7-1, "Secondary Coolant System Specific Activity Sample and Analysis Program" title and column headings are being deleted and replaced with "(This Table Number is not used.)."

Bases Changes

The associated TS Bases is being revised for the proposed changes to 3/4.7.1.4. The Bases for this TS is also being formatted similar to NUREG-1431, Revision 2, "Standard Technical Specifications, Westinghouse Plants."

B. Safety Assessment

Index Pages

The index pages require changing to reflect the proposed changes. This is an administrative change, which aids the user and public in identifying the location of specific information contained within the TS. The revision of the index to reflect the proposed changes does not adversely affect the safety of the plant nor does it adversely affect the health and safety of the public.

Technical Specification 3.7.1.4

The CTS LCO and ACTION use the terms "microCurie/gram" and "Secondary Coolant System". These terms are being replaced with "µCi/gm" and "secondary coolant". The proposed terms are equal to the existing terms. This is an administrative change in the wording. There is no change to the interpretation or the implementation of the TS. This wording is the same as the wording used in NUREG-1431, STS 3.7.18. Therefore, this change does not adversely affect the safety of the plant nor does it adversely affect the health and safety of the public.

The CTS SR references Table 4.7-1. The CTS requirements for sampling are once per 72 hours for "gross radioactivity", and once per 31 days for DEI if ^{131}I is greater than "10% of the allowable limit" (i.e., >0.01 microcurie/gram), or once per 6 months if DEI is <0.01 microcurie/gram. The proposed change eliminates Table 4.7-1 and incorporates ITS wording. The proposed SR will state: "At least every 31 days, verify the specific activity of the secondary coolant is less than or equal to 0.1 μCi/gm DOSE EQUIVALENT I-131." This proposed SR is justified as follows:

- Item 1 in Table 4.7-1, "Gross Radioactivity Determination," requires sampling the secondary coolant once every 72 hours for "gross activity." This measurement was intended as a trigger to determine if DEI had changed "above 10% of the allowable limit." The Updated Final Safety Analysis Report (UFSAR) dose analyses demonstrate that radioiodines and the resulting thyroid dose are limiting, not noble gases and whole body dose, thereby not requiring gross activity sampling from which the doses associated with noble gases and whole body dose can be determined. The primary to secondary leakage limits of TS 3.4.6.2 and the DOSE EQUIVALENT I-131 limits of TS 3.7.1.4 ensure the UFSAR dose analyses remain valid. Therefore, including this requirement in the proposed SR is not justified because, the secondary coolant is unnecessarily being analyzed for gross activity once every 72 hours. The elimination of measurement and analysis of gross activity is also consistent with the requirements specified in NUREG-1431, STS 3.7.18.

  Elimination of the 72 hour sampling frequency does not affect the margin of safety, since the limit on secondary coolant DEI is still 0.1 microcuries per gram.

- Item 2 in Table 4.7-1, "Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration," regarding sampling frequency had a less frequent analysis (once per 6 months) for DEI if secondary coolant activity was less than or equal to 0.01 microcuries per gram. The sampling frequency was the same as the proposed change (once per 31 days) for DEI if secondary coolant activity exceeded 0.01 microcuries per gram. By performing monthly analyses FPLE Seabrook is actually increasing the frequency for this surveillance regardless of concentration limits, resulting in a more restrictive frequency for the isotopic analysis. This change is appropriate because the 31 day frequency is based on the detection of increasing trends in the DEI level, and allows for appropriate action to be taken to maintain levels below the limiting condition for operation limit. Therefore, the current SR is less conservative than the proposed SR. Furthermore, installed plant radiation monitoring equipment (e.g., condenser off-gas radiation monitor) is capable of indicating whether a change in the primary to secondary leak rate has occurred which might effect the DEI. This monitor is a very early warning indicator of leaks, and will identify such changes well before iodines are measurable in the secondary coolant.

Based upon the previous discussion, the elimination of the gross activity determination and the frequency change for the dose equivalent I-131 determination does not adversely affect the safety of the plant nor does it adversely affect the health and safety of the public.

Deletion of Table 4.7-1 and replacement with "(This Table Number is not used.)"is an administrative change and does not affect the TS, since the elements of the Table have been moved, changed or their deletion justified. Removal of the surveillance requirements from a table format is a format preference and is consistent with other TSs format for Seabrook Station. It does not change the intent or implementation of the surveillance requirements. Therefore, the format change does not adversely affect the safety of the plant nor does it adversely affect the health and safety of the public.

The addition of the conditional note to the MODE 4 APPLICABILITY ("The provisions of Specification 4.0.4 are not applicable for entry into MODE 4, however, once steam generator pressure exceeds 100 psig, the requirements of Specification 4.7.1.4 must be met within 12 hours if not performed within the past 31 days.") accurately reflects that sampling/analysis for DEI is not accurate when the plant has been shutdown for the following reasons:

- As the plant enters MODE 4 from MODE 5, the steam generator pressure, temperature, and levels are not conducive to obtaining a representative sample. MODE 4 entry from MODE 5 starts when temperature is greater than 200°F in the reactor coolant system. There is not enough pressure in the steam generators to obtain a sample through the normal sample point. Thus, the only means to obtain a steam generator sample is through a steam generator drain line. Station experience has shown that when steam generator pressure exceeds 100 psig, a representative sample is obtainable in a reasonable time frame. Allowing 12 hours before the surveillance is required provides sufficient time to obtain and analyze the four steam generators, based on station experience.

- MODE 4 is a transition MODE. Plant operations and Technical Specifications commonly use MODE 3 or MODE 5 for stabilizing plant conditions. Usually, the time spent in MODE 4 is only the amount of time to safely transition the plant to either MODE 3 or MODE 5. The likelihood of a release to the atmosphere with the plant between 200°F and 350°F due to a Main Steam Safety Valve lifting, an Atmospheric Dump Valve opening, or a main steam line break during this relatively short time is low.

- The reason for sampling the steam generators is to ensure that the steam generator radioactivity is known beforehand in the event of an analyzed accident (e.g. Steam Generator Tube Rupture, Main Steam Line Break). During MODES 1 through 3, steam generator activity will be determined to be within limits every 31 days. During refueling outages steam generators are commonly drained, sludge lanced, and refilled. This will remove a large amount of any activity that may have been present in the steam generator. Any leakage from the reactor coolant system into the steam generator will be limited by TS 3.4.6.2, "Reactor Coolant System Leakage, Operational Leakage," for entry into MODE 4 and will be detected prior to the ability to sample the steam generators being re-established.

- The reactor coolant activity will be limited by Technical Specification 3.4.8, "Reactor Coolant System, Specific Activity," in MODES 1 through 5. If the activity were to exceed the 1 microcuries per gram DOSE EQUIVALENT I-131 in the reactor coolant system, CTS 3.4.8 requires the plant to "be in at least HOT STANDBY with T_{avg} less than 500 °F within 6 hours" until the activity of the reactor coolant system is restored to within its limits. Once shutdown, one source for radioactivity in the reactor coolant system, a critical reactor, has been removed. Primary coolant activity begins to decrease due to radioactive decay and cleanup of the reactor coolant system through the letdown demineralizers. The other source of radioactivity, failed fuel, would have been detected through reactor coolant system monitoring at power, and fuel sipping to identify known leaking assemblies during outages. Monitoring of reactor coolant activity for entry into MODE 4 provides adequate indicators to alert operators to possible increases in steam generator activity which would require immediate attention.

To summarize the previous discussion:

1) A representative sample is only available from the normal sample point at the upper end of MODE 4,

2) MODE 4 is a transition MODE and the time spent in MODE 4 is relatively short,

3) During refueling outages, steam generators are usually cleaned removing most radioactive materials,

4) After outages, Technical Specifications 3.4.6.2, "Reactor Coolant System Leakage, Operational Leakage," and 3.4.8, "Reactor Coolant System, Specific Activity," provide adequate requirements to alert operators to increased activity in the steam generators,

5) Sources of radioactivity in MODE 4 have been either removed (reactor not critical) or detected by some other means (e.g., failed fuel detection, condenser off gas radiation monitor), and

6) During forced outages, monitoring reactor coolant activity (TS 3.4.8) and reactor coolant leakage (TS 3.4.6.2) provide adequate indicators to alert operators to possible increases in steam generator activity which would require immediate attention.

Therefore, based upon the previous discussion, allowing an exemption from the sampling requirements to enter MODE 4, allowing time for the steam generator to reach 100 psig (an adequate pressure for obtaining a sample in MODE 4 through the normal sample path), and allowing 12 hours to obtain and analyze the samples for four steam generators, is acceptable. Therefore, this change does not adversely affect the safety of the plant nor does it adversely affect the health and safety of the public.

C. Safety Assessment Conclusion

FPLE Seabrook concludes that based upon the previous discussion, the proposed changes contained herein do not adversely affect or endanger the health or safety of the general public or involve a significant safety hazard.

SECTION II

MARKUP OF PROPOSED CHANGES

Refer to the attached markup of the proposed changes to the Technical Specifications. The attached markup reflects the currently issued revision of the Technical Specifications listed below. Pending Technical Specifications or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed markup.

The following Technical Specification changes are included in the attached markup:

Technical Specification	Title	Page(s)
Index	Index	vii
3/4.7.1.4	Turbine Cycle – Specific Activity	3/4 7-7, 7-8
Bases 3/4.7.1.4	Specific Activity	B 3/4 7-2, B 3/4 7-2a

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the ~~S~~secondary ~~C~~coolant ~~System~~ shall be less than or equal to 0.1 ~~microcurie/gram~~ µCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.*

ACTION:

With the specific activity of the ~~S~~secondary ~~C~~coolant ~~System~~ greater than 0.1 ~~microcurie/gram~~ DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 ~~The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.~~

AT LEAST ONCE EVERY 31 DAYS, VERIFY THE SPECIFIC ACTIVITY OF THE SECONDARY COOLANT IS LESS THAN OR EQUAL TO 0.1 µCi/gm DOSE EQUIVALENT I-131.

* The provisions of Specification 4.0.4 are not applicable for entry into MODE 4, however, once steam generator pressure exceeds 100 psig, the requirements of Specification 4.7.1.4 must be met within 12 hours if not performed within the past 31 days.

(THIS TABLE NUMBER IS NOT USED)

TABLE 4.7-1

~~SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY~~

~~SAMPLE AND ANALYSIS PROGRAM~~

~~TYPE OF MEASUREMENT AND ANALYSIS~~	~~SAMPLE AND ANALYSIS FREQUENCY~~
~~1. Gross Radioactivity Determination~~	~~At least once per 72 hours.~~
~~2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration~~	~~a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines.~~ ~~b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.~~

~~91~~ 95

AMENDMENT NO. 246

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

The electric motor-driven emergency feedwater pump is capable of delivering a total feedwater flow of 650 gpm at a pressure of 1221 psig to the entrance of the steam generators. The steam-driven emergency feedwater pump is capable of delivering a total feedwater flow of 650 gpm at a pressure of 1221 psig to the entrance of the steam generators. The startup feedwater pump serves as the third auxiliary feedwater pump and can be manually aligned to be powered from an emergency bus (Bus 5). The startup feedwater pump is capable of taking suction on the dedicated emergency feedwater volume of water in the condensate storage tank and delivering a total feedwater flow of in excess of 650 gpm at a pressure of 1221 psig to the entrance of the steam generator via either the main feedwater header or with manual alignment to the emergency feedwater flow path. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the indicated minimum water volume ensures that sufficient water is available to cool the RCS to a temperature of 350°F. The OPERABILITY of the concrete enclosure ensures this availability of water following rupture of the condensate storage tank by a tornado generated missle. The contained water volume limit includes an allowance for water not usable because of instrument uncertainity, tank discharge line location, or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

~~The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.~~

INSERT A

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reac-tivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.6 ATMOSPHERIC RELIEF VALVES

The OPERABILITY of the Atmospheric Relief Valves (ARVs) ensures the controlled removal of reactor decay heat during reactor cooldown, plant startup, and after a turbine trip, when the condenser and/or the turbine bypass system are not available. When available, the ARVs can be used to reduce main steam pressure for both hot shutdown and cold shutdown conditions. The ARVs provide a method for cooling the plant to residual heat removal entry conditions should the turbine bypass system to the condenser be unavailable. This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST).

One ARV line for each of the four steam generators is provided. Each ARV line consists of one ARV and an associated block valve. The ARVs are provided with upstream block valves to provide an alternate means of isolation.

The ARVs are equipped with pneumatic controllers to permit control of the cooldown rate. The ARVs are provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ARVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the ARVs for the time required for Reactor Coolant System cooldown to RHR entry conditions. The ARVs are OPERABLE with only a DC power source available. In addition, handwheels are provided for local manual operation.

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3/4.7.1.4 SPECIFIC ACTIVITY

BACKGROUND

Activity in the secondary coolant results from Reactor Coolant System leakage through the steam generator tube(s). Under steady state conditions, the activity is primarily iodines with relatively short half-lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.6.2, "Reactor Coolant System Leakage - Operational Leakage") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.8, "Reactor Coolant System Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half-lives (i.e., < 20 hours).

With the specified activity limit, the resultant 2-hour thyroid dose to a person at the SITE BOUNDARY would be a small fraction of the 10 CFR 100 (Ref. 1) limits if the main steam safety valves (MSSVs) were open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour SITE BOUNDARY exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit SITE BOUNDARY limits (Ref. 1) for whole body and thyroid dose rates.

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With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Emergency Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LIMITING CONDITION FOR OPERATION (LCO)

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

MODE 4 is conditioned by a footnote to recognize that sampling in MODE 4 is limited by the steam generator conditions necessary to obtain a sample. Upon entering MODE 4 from MODE 5, there is not enough steam pressure in the steam generator to provide a sample through the normal sample point. Due to plant limitations, a representative sample can be obtained with greater than 100 psig steam pressure in the steam generator. By requiring the sample to be taken within 12 hours of achieving the 100 psig steam pressure, adequate time for obtaining and analyzing a sample are ensured.

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(continued)

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated completion time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least HOT STANDBY within 6 hours, and in COLD SHUTDOWN within the following 30 hours. The ACTION completion times are reasonable, based on operating experience, to reach the required unit shutdown conditions from full power in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS (SR)

SR 4.7.1.4

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31-day frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.
2. UFSAR, Chapter 15.

SECTION III

RETYPE OF PROPOSED CHANGES

Refer to the attached retype of the proposed changes to the Technical Specifications. The attached retype reflects the currently issued version of the Technical Specifications. Pending Technical Specification changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

PLANT SYSTEMS

TURBINE CYCLE

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant shall be less than or equal to 0.1 µCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

With the specific activity of the secondary coolant greater than 0.1 µCi/gm DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 At least once every 31 days, verify the specific activity of the secondary coolant is less than or equal to 0.1 µCi/gm DOSE EQUIVALENT I-131.

* The provisions of Specification 4.0.4 are not applicable for entry into MODE 4, however, once steam generator pressure exceeds 100 psig, the requirements of Specification 4.7.1.4 must be met within 12 hours if not performed within the past 31 days.

TABLE 4.7-1

(THIS TABLE NUMBER IS NOT USED)

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

The electric motor-driven emergency feedwater pump is capable of delivering a total feedwater flow of 650 gpm at a pressure of 1221 psig to the entrance of the steam generators. The steam-driven emergency feedwater pump is capable of delivering a total feedwater flow of 650 gpm at a pressure of 1221 psig to the entrance of the steam generators. The startup feedwater pump serves as the third auxiliary feedwater pump and can be manually aligned to be powered from an emergency bus (Bus 5). The startup feedwater pump is capable of taking suction on the dedicated emergency feedwater volume of water in the condensate storage tank and delivering a total feedwater flow of in excess of 650 gpm at a pressure of 1221 psig to the entrance of the steam generator via either the main feedwater header or with manual alignment to the emergency feedwater flow path. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the indicated minimum water volume ensures that sufficient water is available to cool the RCS to a temperature of 350°F. The OPERABILITY of the concrete enclosure ensures this availability of water following rupture of the condensate storage tank by a tornado generated missile. The contained water volume limit includes an allowance for water not usable because of instrument uncertainity, tank discharge line location, or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

BACKGROUND

Activity in the secondary coolant results from Reactor Coolant System leakage through the steam generator tube(s). Under steady state conditions, the activity is primarily iodines with relatively short half-lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.6.2, "Reactor Coolant System Leakage - Operational Leakage") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.8, "Reactor Coolant System Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half-lives (i.e., < 20 hours).

3/4.7.1 TURBINE CYCLE (Continued)

With the specified activity limit, the resultant 2-hour thyroid dose to a person at the SITE BOUNDARY would be a small fraction of the 10 CFR 100 (Ref. 1) limits if the main steam safety valves (MSSVs) were open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour SITE BOUNDARY exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 µCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit SITE BOUNDARY limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Emergency Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LIMITING CONDITION FOR OPERATION (LCO)

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be $\leq$ 0.10 µCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

3/4.7.1 TURBINE CYCLE (Continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

MODE 4 is conditioned by a footnote to recognize that sampling in MODE 4 is limited by the steam generator conditions necessary to obtain a sample. Upon entering MODE 4 from MODE 5, there is not enough steam pressure in the steam generator to provide a sample through the normal sample point. Due to plant limitations, a representative sample can be obtained with greater than 100 psig steam pressure in the steam generator. By requiring the sample to be taken within 12 hours of achieving the 100 psig steam pressure, adequate time for obtaining and analyzing a sample are ensured.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated completion time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least HOT STANDBYwithin 6 hours, and in COLD SHUTDOWN within the following 30 hours. The ACTION completion times are reasonable, based on operating experience, to reach the required unit shutdown conditions from full power in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS (SR)

SR 4.7.1.4

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31-day frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.
2. UFSAR, Chapter 15.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.6 ATMOSPHERIC RELIEF VALVES

The OPERABILITY of the Atmospheric Relief Valves (ARVs) ensures the controlled removal of reactor decay heat during reactor cooldown, plant startup, and after a turbine trip, when the condenser and/or the turbine bypass system are not available. When available, the ARVs can be used to reduce main steam pressure for both hot shutdown and cold shutdown conditions. The ARVs provide a method for cooling the plant to residual heat removal entry conditions should the turbine bypass system to the condenser be unavailable. This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST).

One ARV line for each of the four steam generators is provided. Each ARV line consists of one ARV and an associated block valve. The ARVs are provided with upstream block valves to provide an alternate means of isolation.

The ARVs are equipped with pneumatic controllers to permit control of the cooldown rate. The ARVs are provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ARVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the ARVs for the time required for Reactor Coolant System cooldown to RHR entry conditions. The ARVs are OPERABLE with only a DC power source available. In addition, handwheels are provided for local manual operation.

SECTION IV

DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGES

IV. DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

License Amendment Request (LAR) 02-08 propose changes to the Seabrook Station Technical Specification (TS) 3/4.7.1.4, "Turbine Cycle – Specific Activity" and associated Bases. Part of the proposed change will adopt similar wording as presented in the improved Standard Technical Specifications for Westinghouse Plants, NUREG-1431, Revision 2. Also proposed is an exemption from the requirements of TS 4.0.4 when entering MODE 4, along with conditions for when the surveillance requirement must be satisfied in MODE 4..

The TS Index is revised as well to reflect the previously discussed changes.

The proposed changes will afford FPLE Seabrook operational flexibility, particularly during and immediately after refueling outages, when sampling and analysis does not provide relevant information to the Technical Specifications.

In accordance with 10 CFR 50.92, FPLE Seabrook has concluded that the proposed changes do not involve a no significant hazards consideration (NSHC). The basis for the conclusion that the proposed change does not involve a NSHC is as follows:

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

The proposed changes, in part, modify the modes of applicability by stating that TS 4.0.4 is not applicable for Mode 4 entry. For the surveillance requirement, the change specifies the conditions in Mode 4 that are necessary to obtain a representative sample from the steam generators. Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. The level of specific activity contained in the reactor coolant is germane to the consequences of an accident and is not related in any way to the probability of failure of a plant structure, system or component which would result in the occurrence of an analyzed event. Because the probability of failure of plant equipment is not affected, there is no impact on the probability of occurrence of a previously analyzed accident.

The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event. The proposed changes do not alter the initial conditions assumed in the analyses of interest. The plant parameters assumed for the analyses are maintained within assumed limits through compliance with the Technical Specifications and plant procedures. Additionally, the proposed changes do not impose any new safety analyses limits. Any deviation from the allowable activity limits will require the plant to be placed in a condition where the specification does not apply. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

   The proposed changes do not involve a physical alteration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated, or to the setpoints at which protective or mitigative actions are initiated. No alteration in the procedures that ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. These changes have no physical affect on any plant equipment. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The margin of safety is established through equipment design, limitations on operating parameters, and the setpoints at which automatic actions are initiated. No equipment design features are impacted by these changes, no operating parameters are revised, and no changes are proposed to the actuation setpoints. The limit on secondary coolant Dose Equivalent Iodine remains at the current value of 0.1 microcuries per gram. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Therefore, based upon the evaluation presented above and the previous discussion of the amendment request, FPLE Seabrook concludes that the proposed revisions to the Seabrook Station Technical Specifications do not constitute a significant hazard as defined by the criteria in 10 CFR 50.92(c).

SECTIONS V AND VI

PROPOSED SCHEDULE FOR LICENSE AMENDMENT ISSUANCE AND EFFECTIVENESS

AND

ENVIRONMENTAL IMPACT ASSESSMENT

V. PROPOSED SCHEDULE FOR LICENSE AMENDMENT ISSUANCE AND EFFECTIVENESS

FPLE Seabrook requests NRC review of License Amendment Request 02-08, and issuance of a license amendment by October 1, 2003, having immediate effectiveness and implementation within 90 days. Issuance of a license amendment by the requested date would afford FPLE Seabrook the flexibility for planning of technical resources in support of Seabrook Station's 9th refueling outage scheduled to commence in October of 2003.

VI. ENVIRONMENTAL IMPACT ASSESSMENT

FPLE Seabrook has reviewed the proposed license amendment against the criteria of 10 CFR 51.22 for environmental considerations. The proposed change does not involve a no significant hazards consideration, nor increase the types and amounts of effluent that may be released off-site, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, FPLE Seabrook concludes that the proposed changes meet the criteria delineated in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.