Mark J. Ajluni, P.E.
Nuclear Licensing Director

Southern Nuclear Operating Company, Inc.
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7673
Fax 205.992.7885

SOUTHERN COMPANY

December 16, 2010

Docket Nos.: 50-321 50-424
50-366 50-425

NL-10-0794

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

Edwin I. Hatch Nuclear Plant – Units 1 & 2
Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request for Incorporation of Previously NRC Approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev. 1, "Delete Safety Limit Violation Notification Requirements"

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Southern Nuclear Operating Company (SNC) is submitting a request for an amendment to the Technical Specifications (TS) for Edwin I. Hatch Nuclear Plant (HNP) and Vogtle Electric Generating Plant (VEGP). The proposed amendments affect Section 2.0 "Safety Limits (SLs)."

This amendment request proposes to delete requirements from the Technical Specifications that duplicate requirements found in the regulations (10 CFR 50.36).

The proposed changes to TS 2.0, "Safety Limits (SLs)" are consistent with Nuclear Regulatory Commission (NRC) approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev. 1, "Delete Safety Limit Violation Notification Requirements."

In addition to the strict implementation of TSTF-5-A, Rev. 1, SNC proposes to delete the first two paragraphs of TS Bases Section B 2.1.2 "Safety Limit Violations" and to delete labels 2.2.2 and 2.2.2.1 of this section. These changes are being made for consistency with the Improved Standard Technical Specifications and clarity of the TSTF-5-A, Rev. 1 implementation.

SNC requests approval of the proposed license amendments by September 24, 2011. Once approved, the amendment would be implemented within 90 days of issuance of the amendment.

Enclosure 1 provides the basis for the proposed changes. Enclosure 2 contains TS markup pages. Enclosure 3 provides clean-typed TS pages. Enclosure 4 includes TS Bases markups for reference only.

ADD1
NRR

SNC has evaluated this request under the standards set forth in 10 CFR 50.92(c) and determined that a finding of "no significant hazards consideration" is justified.

Mr. M. J. Ajluni states he is Nuclear Licensing Director of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

This letter contains no NRC commitments.

Respectfully submitted,

Mark J Ajluni

M. J. Ajluni
Nuclear Licensing Director

MJA/GAL/emm

Sworn to and subscribed before me this 16th *day of* December, *2010.*

Charlotte A. Graham
Notary Public

My commission expires: 6/9/12

Enclosures
1. Basis for Proposed Changes
2. Technical Specification Markup Pages
3. Clean Typed Technical Specification Pages
4. Technical Specification Bases Markup Pages

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. D. R. Madison, Vice President – Hatch
Mr. T. E. Tynan, Vice President – Vogtle
Ms. P. M. Marino, Vice President – Engineering
RType: Hatch=CHA02.004; Vogtle=CVC7000

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Farley, Hatch and Vogtle
Mr. P. Boyle, NRR Project Manager
Mr. E.D Morris, Senior Resident Inspector – Hatch
Mr. M. Cain, Senior Resident Inspector – Vogtle

State of Georgia
Mr. Allen Barnes, Director – Environmental Protection Division

Edwin I. Hatch Nuclear Plant – Units 1 & 2
Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request For Incorporation Of Previously NRC Approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev. 1, "Delete Safety Limit Violation Notification Requirements"

Enclosure 1

Basis for Proposed Change

Edwin I. Hatch Nuclear Plant – Units 1 & 2
Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request For Incorporation Of Previously NRC Approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev. 1, "Delete Safety Limit Violation Notification Requirements"

Enclosure 1

Basis for Proposed Change

Table of Contents

1.0 Summary Description

2.0 Detailed Description

3.0 Technical Evaluation

4.0 Regulatory Evaluation

   4.1 No Significant Hazards Consideration

   4.2 Applicable Regulatory Requirements/Criteria

   4.3 Precedent

5.0 Environmental Consideration

6.0 References

Enclosure 1

Basis for Proposed Change

1.0 Summary Description

This amendment request proposes to delete requirements from the Technical Specifications (TS) that are duplicative or contained in other regulations or required to comply with requlations (10 CFR 50.36).

The proposed changes to TS 2.0, "Safety Limits (SLs)" are consistent with Nuclear Regulatory Commission (NRC) approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev. 1, "Delete Safety Limit Violation Notification Requirements."

2.0 Detailed Description

The proposed change is to revise the Technical Specifications as follows:

<u>Changes to VEGP Technical Specifications/Technical Specification Bases</u>

Affected Technical Specification	Change Description
2.2 SL Violations	Eliminate Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6
B 2.1.1 Reactor Core SLs Bases	Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, 2.2.6 and the label of 2.2.1
B 2.1.1 Reactor Core SLs Bases	Eliminate References 5 and 6
B 2.1.2 RCS Pressure SL Bases	Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, 2.2.6, the labels of 2.2.2.1 and 2.2.2.2. Eliminate fist two paragraphs of "Safety Limit Violations" and labels 2.2.2 and 2.2.2.1.
B 2.1.2 RCS Pressure SL Bases	Eliminate References 6 and 7

<u>Changes to HNP Unit 1 Technical Specification/Technical Specification Bases</u>

Affected Technical Specification	Change Description
2.2 SL Violations	Eliminate 2.2.1, 2.2.3, 2.2.4, and 2.2.5. Change paragraph number from 2.2.2.1 to 2.2.1. Change paragraph number from 2.2.2.2 to 2.2.2.
B 2.1.1 Reactor Core SLs Violations	Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, and the label on 2.2.2.
B 2.1.1 Reactor Core SLs	Eliminate References 3 and 5
B 2.1.2 RCS Pressure SLs Violations	Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, and the label on 2.2.2
B 2.1.2 RCS Pressure SL	Eliminate References 7 and 8

Changes to HNP Unit 2 Technical Specification/Technical Specification Bases

Affected Technical Specification	Change Description
2.2 SL Violations	Eliminate 2.2.1, 2.2.3, 2.2.4, and 2.2.5. Change paragraph number from 2.2.2.1 to 2.2.1. Change paragraph number from 2.2.2.2 to 2.2.2.
B 2.1.1 Reactor Core SLs Violations	Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, and the label on 2.2.2.
B 2.1.1 Reactor Core SLs	Eliminate References 3 and 5
B 2.1.2 RCS Pressure SLs Violations	Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, and the label on 2.2.2
B 2.1.2 RCS Pressure SL	Eliminate References 7 and 8

3.0 Technical Evaluation

SNC has reviewed the TSTF-5-A, Revision 1, and found it to be applicable as written. The proposed changes are to remove the duplicative requirements to report safety limit violations and requirements to preclude restart after a safety limit violation without NRC approval from the TS. These are considered an administrative action. These reporting and restart requirements are duplicative of what is already contained in the regulations (i.e., 10 CFR 50.36). The reporting requirements in 10 CFR 50.36 require that appropriate prompt notifications are made to the NRC and that Licensee Event Reports (LERs) are submitted to the NRC. 10 CFR 50.36 requires that these reports be performed in accordance with the requirements of 10 CFR 50.72 and 10 CFR 50.73. Therefore, if a TS safety limit is violated, appropriate reporting will be made to the NRC in accordance with the regulations. 10 CFR 50.36 also requires that operations must not be resumed until authorized by the Commission. Removal of duplicative reporting and restart requirements from the TS results in simplification of the TS and Bases and less administrative burden to track duplicative requirements. Adequate administrative controls exist in administrative programs at SNC for the identification and reporting of safety limit violations, and restart restrictions following safety limit violations, in accordance with 10 CFR 50.36, 10 CFR 50.72, and 10 CFR 50.73. Labels associated with the duplicative TSs listed above are also proposed to be deleted and are reflected in the markups.

4.0 Regulatory Evaluation

4.1 No Significant Hazards Consideration

The changes proposed by this license amendment application would revise the Safety Limits Section 2.0 of the VEGP and HNP TS to delete duplicative notification, reporting, and restart requirements from the TS. This proposed change is consistent with the Nuclear Regulatory Commission (NRC) approved Industry/Technical Specification Task Force (TSTF) Standard

Enclosure 1

Basis for Proposed Change

Technical Specification Change Traveler TSTF-5-A, Rev. 1, "Delete Safety Limit Violation Notification Requirements." This change facilitates improved content and presentation of Administrative Controls.

Southern Nuclear Operating Company (SNC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment", as discussed below:

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

Response: No.

The proposed change to remove the duplicative safety limit reporting, notification, and restart constraint requirements from the TSs does not affect the plant or operation of the plant. The change simply removes duplicative information from the TS that is covered in the NRC regulations. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change to remove the duplicative safety limit reporting, notification, and restart constraint requirements from the TS does not introduce any new accident scenarios, failure mechanisms, or limiting single failures. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed change has no adverse effect on any safety-related system or component and does not challenge the performance or integrity of any safety related system. This change is considered an administrative action to remove duplicative reporting, notification, and restart constraint requirements. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed changes are administrative and do not involve any reduction in a margin of safety. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed change has no adverse effect on any safety-related system or component and does not

4.2 Applicable Regulatory Requirements/Criteria

The proposed change to remove the notification, reporting, and restart requirements if a safety limit is violated from the TSs simply removes duplicative information from the TSs that is covered in the regulations (10 CFR 50.36). The reporting requirements in 10 CFR 50.36 require that appropriate prompt notifications are made to the NRC and that the Licensee Event Reports (LERs) are submitted to the NRC. 10 CFR 50.36 requires that theses reports be performed in accordance with the requirements of 10 CFR 50.72 and 10 CFR 50.73. Therefore, if a TS safety limit is violated, appropriate reporting will be made to the NRC in accordance with the regulations. Adequate administrative controls exist in administrative programs at SNC for the identification and necessary reporting of safety limit violations in accordance with 10 CFR 50.36, 10 CFR 50.72, and 10 CFR 50.73. This change is consistent with Nuclear Regulatory Commission (NRC) approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev 1, "Delete Safety Limit Violation Notification Requirements."

4.3 Precedent

The proposed change to remove the duplicative notification, reporting, and restart requirements if a safety limit is violated from the TSs is consistent with Nuclear Regulatory Commission (NRC) approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev. 1, "Delete Notification, Reporting, and Restart Requirements if a Safety Limit is Violated." This change is consistent with the license amendment application for Peach Bottom Atomic Power Station, Units 2 and 3. The NRC approved this license amendment request by letter dated May 10, 2006.

5.0 Environmental Consideration

The scope of the proposed amendment is limited to the categorical exclusion provided by 10 CFR 51.21(c)(10)(ii) "Changes recordkeeping, reporting, or administrative procedures or requirements." Therefore, no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 References

1. Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev.1 "Delete Safety Limit Violation Notification Requirements."

2. May 10, 2006 letter from Richard V. Guzman (NRC), Subject: Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments Re: Incorporation of Previously NRC Approved Generic Technical Specification Changes (TAC Nos. MC3683, ...)

6.0 References

1. Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev.1 "Delete Safety Limit Violation Notification Requirements."

2. May 10, 2006 letter from Richard V. Guzman (NRC), Subject: Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments Re: Incorporation of Previously NRC Approved Generic Technical Specification Changes (TAC Nos. MC3683, ...)

Edwin I. Hatch Nuclear Plant – Units 1 & 2
Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request For Incorporation Of Previously NRC Approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev. 1, "Delete Safety Limit Violation Notification Requirements"

Enclosure 2

Technical Specification Markup Pages

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System (RCS) Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

within 2 hours

2.2 SL Violations

With any SL violation, the following actions shall be completed:

~~2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.~~

~~2.2.2 Within 2 hours:~~

2.2.~~2.~~1 Restore compliance with all SLs; and

2.2.~~2.~~2 Insert all insertable control rods.

~~2.2.3 Within 24 hours, notify the Plant Manager, the Vice President - Hatch and the offsite review committee.~~

(continued)

2.0 SAFETY LIMITS (SLs)

~~2.2 SL Violations (continued)~~

~~2.2.4 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the offsite review committee, the Plant Manager, and the Vice President - Hatch.~~

~~2.2.5 Operation of the unit shall not be resumed until authorized by the NRC.~~

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System (RCS) Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

within 2 hours

2.2 SL Violations

With any SL violation, the following actions shall be completed:

~~2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.~~

~~2.2.2 Within 2 hours:~~

2.2.~~2.~~1 Restore compliance with all SLs; and

2.2.~~2.~~2 Insert all insertable control rods.

~~2.2.3 Within 24 hours, notify the Plant Manager, the Vice President – Hatch, and the offsite review committee.~~

(continued)

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 loop average 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.

~~2.2.3 Within 1 hour notify the NRC Operations Center, in accordance with 10 CFR 50.72.~~

~~2.2.4 Within 24 hours, notify the Plant Manager and Vice President - Vogtle.~~

~~2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared and submitted to the NRC pursuant to 10 CFR 50.73.~~

~~2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.~~

Edwin I. Hatch Nuclear Plant – Units 1 & 2
Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request For Incorporation Of Previously NRC Approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev. 1, "Delete Safety Limit Violation Notification Requirements"

Enclosure 3

Clean Typed Technical Specification Pages

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System (RCS) Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System (RCS) Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

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 loop average 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.

Edwin I. Hatch Nuclear Plant – Units 1 & 2
Vogtle Electric Generating Plant – Units 1 & 2
License Amendment Request For Incorporation Of Previously NRC Approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-5-A, Rev. 1, "Delete Safety Limit Violation Notification Requirements"

Enclosure 4

Technical Specification Bases Markup Pages

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

active fuel must be adjusted for assemblies with a fuel length not 150 inches. For example, the top of the active fuel for GE13 fuel is 162.44 inches below instrument zero since the fuel length for this fuel type is 146 inches. The Core Operating Limits Report identifies fuel types and fuel lengths used in the current operating cycle.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

~~2.2.1~~

~~If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3).~~

~~2.2.2~~

3

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

~~2.2.3~~

~~If any SL is violated, the senior management of the nuclear plant and the utility, and the Safety Review Board (SRB) shall be notified within 24 hours. The 24 hour period provides time for plant operators and~~

(continued)

BASES

~~SAFETY LIMIT VIOLATIONS~~ ~~2.2.3 (continued)~~

~~staff to take the immediate action and assess the condition of the unit before reporting to the senior management.~~

~~2.2.4~~

~~If any 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. 5). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility, and the SRB.~~

~~2.2.5~~

~~If any 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~~.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuels" (revision specified in the COLR).

~~3. 10 CFR 50.72.~~

3

~~4~~. 10 CFR 100.

~~5. 10 CFR 50.73.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

Addenda through the Winter of 1966 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to the USAS Nuclear Power Piping Code, Section B31.1, 1967 Edition, including Addenda A, C, and D (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1150 psig for suction piping and 1325 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 120% of design pressures of 1150 psig for suction piping and 1325 psig for discharge piping. The most limiting of these two allowances is the 110% of the reactor vessel design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

~~2.2.1~~

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

~~2.2.2~~

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). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action.

6

(continued)

BASES

~~SAFETY LIMIT VIOLATIONS (continued)~~

~~2.2.3~~

~~If any SL is violated, the senior management of the nuclear plant and the utility, and the SRB shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the immediate action and assess the condition of the unit before reporting to the senior management.~~

~~2.2.4~~

~~If any 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 senior management of the nuclear plant and the utility, and the SRB.~~

~~2.2.5~~

~~If any 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.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14 and GDC 15.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.

4. 10 CFR 100.

5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda Winter of 1966.

6. ASME, USAS, Nuclear Power Piping Code, Section B31.1, 1967 Edition, Addenda A, C, and D.

~~7. 10 CFR 50.72.~~

~~8. 10 CFR 50.73.~~

7

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

active fuel must be adjusted for assemblies with a fuel length not 150 inches. For example, the top of the active fuel for GE13 fuel is 162.44 inches below instrument zero since the fuel length for this fuel type is 146 inches. The Core Operating Limits Report identifies fuel types and fuel lengths used in the current operating cycle.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

~~2.2.1~~

~~If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3).~~

3

~~2.2.2~~

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. ~~4~~). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

~~2.2.3~~

~~If any SL is violated, the senior management of the nuclear plant and the utility, and the Safety Review Board (SRB) shall be notified within 24 hours. The 24 hour period provides time for plant operators and~~

(continued)

BASES

~~SAFETY LIMIT VIOLATIONS~~ ~~2.2.3 (continued)~~

~~staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.~~

~~2.2.4~~

~~If any 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. 5). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility, and the SRB.~~

~~2.2.5~~

~~If any 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.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuels," (revision specified in the COLR).

~~3. 10 CFR 50.72.~~

4. 10 CFR 100.

~~5. 10 CFR 50.73.~~

3

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects 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. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. Per 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

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 are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure - High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1968 Edition, including

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Addenda through the Summer of 1970 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1980 Edition, including addenda through Winter 1981 (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 1250 psig for suction piping and 1450 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 110% of design pressures of 1250 psig for suction piping and 1450 psig for discharge piping. The most limiting of these two allowances is the 110% of the reactor vessel and recirculation suction piping design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

~~2.2.1~~

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

~~2.2.2~~

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). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action.

(continued)

BASES

~~SAFETY LIMIT VIOLATIONS (continued)~~

~~2.2.3~~

~~If any SL is violated, the senior management of the nuclear plant and the utility, and the SRB shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the immediate action and assess the condition of the unit before reporting to the senior management.~~

~~2.2.4~~

~~If any 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 senior management of the nuclear plant and the utility, and the SRB.~~

~~2.2.5~~

~~If any 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.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14 and GDC 15.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. 10 CFR 100.
5. ASME, Boiler and Pressure Vessel Code, Section III, 1968 Edition, Addenda Summer of 1970.
6. ASME, Boiler and Pressure Vessel Code, Section III, 1980 Edition, Addenda Winter of 1981.
7. ~~10 CFR 50.72.~~
8. ~~10 CFR 50.73.~~

BASES

SAFETY LIMIT VIOLATIONS (continued)

~~2.2.1~~

If the reactor core 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.

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 the reactor core 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 the reactor core SL 2.1.1 is violated, the Plant Manager and the Vice President-Vogtle 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 the senior management.~~

~~2.2.5~~

~~If the reactor core SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC. This requirement is in accordance with 10 CFR 50.73 (Ref. 6).~~

~~2.2.6~~

~~If the reactor core 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.~~

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 7.2.
3. WCAP-8746-A, March 1977.
4. WCAP-9272-P-A, July 1985.
5. ~~10 CFR 50.72.~~
6. ~~10 CFR 50.73.~~

BASES

SAFETY LIMITS (continued)

Code, Section III, is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2735 psig.

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

~~Section 2.2, SL Violations, provides the Required Actions to be taken in response to a violation of a Safety Limit. The bases for the Required Actions of Section 2.2 applicable to a violation of the RCS pressure SL are discussed below.~~

~~2.2.2~~

~~The Required Actions of this subsection state the specific status in which the unit must be placed if the RCS pressure SL is violated. Separate Required Actions and Completion Times are provided for MODES 1 or 2 (Subsection 2.2.2.1) and for MODES 3, 4, or 5 (Subsection 2.2.2.2).~~

~~2.2.2.1~~

If the RCS pressure SL 2.2.2 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.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

~~2.2.2.2~~

If the RCS pressure SL 2.2.2 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.

~~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. 6).~~

~~2.2.4~~

~~If the RCS pressure SL 2.2.2 is violated, the Plant Manager and the Vice President-Vogtle 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.~~

~~2.2.5~~

~~If the RCS pressure SL 2.2.2 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC. This requirement is in accordance with 10 CFR 50.73 (Ref. 7).~~

~~2.2.6~~

~~If the RCS pressure SL 2.2.2 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews,~~

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

~~2.2.6~~

~~analyses, and actions are completed before the unit begins its restart to normal operation.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWB-5000.
4. 10 CFR 100.
5. FSAR, Section 7.2.
6. ~~10 CFR 50.72.~~
7. ~~10 CFR 50.73.~~