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ONS-2014-031

March 4, 2014


ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, Maryland 20852

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station
Docket Numbers 50-269, 50-270, and 50-287
Technical Specification (TS) Bases Change

Please find attached a change to the Oconee Nuclear Station (ONS) TS Bases. This change was processed in accordance with the provisions of Technical Specification 5.5.15, "Technical Specifications (TS) Bases Control Program." TSB Change 2014-01 revised TS Basis 2.1.2, Reactor Coolant System (RCS) Pressure Safety Limits, by adding additional information to provide discriminators for normal operation, anticipated transients (upset), emergency operation, and faulted operation.

Any questions regarding this information should be directed to Sandra N. Severance at (864) 873-3466.

Sincerely,


Scott L. Batson
Vice President
Oconee Nuclear Station

Attachment

ADD
NRR

U. S. Nuclear Regulatory Commission

March 4, 2014

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cc: Mr. Victor McCree, Regional Administrator
U.S. Nuclear Regulatory Commission, Region II
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, GA 30303-1257

Mr. Richard Guzman, Senior Project Manager (ONS)
(By electronic mail only)
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
11555 Rockville Pike
Mail Stop O-8C2
Rockville, MD 20852

Mr. Eddy Crowe
Senior Resident Inspector
Oconee Nuclear Station



February 17, 2014

Re: Oconee Nuclear Station
Technical Specification Bases Change

Please replace the corresponding pages in your copy of the Oconee Technical Specifications Bases Manual as follows:

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INSERT THESE PAGES

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If you have any questions concerning the contents of this Technical Specification Bases update, contact Sandra Severance at (864) 873-3466.



Chris Wasik
Regulatory Affairs Manager

Attachment

Oconee Nuclear Station

Revised Technical Specification Bases Pages

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B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND According to ONS Design Criteria (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation nor during anticipated transients. Normal Operation is defined as Service Level A and Anticipated Transient (or Upset Operation) is defined as Service Level B (Ref 7). ONS Design Criteria (Ref. 1), specifies that reactivity accidents including rod ejection (i.e., Service Level C) do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psig. During normal operation and anticipated transients, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with Section III of the ASME Code (Ref. 2). Hence, the safety limit is 2750 psig. Note that the Rod Ejection Accident is not an anticipated transient and, thus, has a different limit. To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure prior to initial operation, according to the ASME Code requirements. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

APPLICABLE SAFETY ANALYSES The RCS pressurizer safety valves, operating in conjunction with the Reactor Protection System trip settings, ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2).

The limiting peak pressure transient, as determined by the safety analyses (Ref. 5), is 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 valve (PORV);
- b. Steam line turbine bypass valves;

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APPLICABLE SAFETY ANALYSES (continued)

- c. Control system runback of reactor and turbine power; and
- d. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III (Ref. 2), is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.7 (Ref. 4), is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2750 psig.

Overpressurization of the RCS can 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 and steam generators, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 6).

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 during 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 significantly.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

2.2.2

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. 6).

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

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SAFETY LIMIT VIOLATIONS
(continued)

2.2.3

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 potentially 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. This 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.

REFERENCES

1. UFSAR, Section 3.1.
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. ASME USAS B31.7, Nuclear Power Piping, dated February 1968 with June 1968 Errata.
5. UFSAR, Chapters 5 and 15.
6. 10 CFR 100.
7. ASME Boiler and Pressure Vessel Code, 1983 Edition, Section III, Article NCA-2140.