July 29, 1999

Mr. W. R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
Seneca, SC 29679

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 RE: EXEMPTION FROM THE REQUIREMENTS OF 10 CFR PART 50, SECTION 50.60(a) (TAC NOS. MA5473, MA5474, AND MA5475)

Dear Mr. McCollum:

The Commission has approved the enclosed exemption from certain requirements of Title 10 of the Code of Federal Regulations, Part 50, Section 50.60(a). This action is in response to your application dated May 11, 1999, for the application at the Oconee Nuclear Station Units 1, 2, and 3, of the methodology contained in the American Society of Mechanical Engineers (ASME) Code Cases N-514 (as an alternate methodology for determining the low temperature overpressure protection system enable temperature), N-588 (for determining the reactor vessel pressure-temperature limits derived from postulating a circumferentially-oriented reference flaw in a circumferential weld), and N-626 (as an alternate reference fracture toughness for reactor vessel materials for use in determining the pressure-temperature limits). Note that the designation for Code Case N-626 has been changed to N-640 by the ASME code committee. During its review of your submittal, the staff determined that no exemption of Code Case N-514 was needed.

A copy of the exemption has been forwarded to the Office of the Federal Register for publication.

Sincerely,
ORIGINAL SIGNED BY:

David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure: Exemption

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Docket Nos. 50-269, 50-270, and 50-287

Enclosure: Exemption

cc w/encl: See next page

cc:

Ms. Lisa F. Vaughn
Legal Department (PBO5E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

Anne Cottington, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Mr. Rick N. Edwards
Framatome Technologies
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852-1631

Manager, LIS
NUS Corporation
2650 McCormick Drive, 3rd Floor
Clearwater, Florida 34619-1035

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
7812B Rochester Highway
Seneca, South Carolina 29672

Virgil R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201-1708

County Supervisor of Oconee County
Walhalla, South Carolina 29621

Mr. J. E. Burchfield
Compliance Manager
Duke Energy Corporation
Oconee Nuclear Site
P. O. Box 1439
Seneca, South Carolina 29679

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

L. A. Keller
Manager - Nuclear Regulatory Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Mr. Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of Environment, Health, and Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Mr. Steven P. Shaver
Senior Sales Engineer
Westinghouse Electric Company
5929 Carnegie Blvd.
Suite 500
Charlotte, North Carolina 28209

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
DUKE ENERGY CORPORATION	)	Docket Nos. 50-269, 50-270, and 50-287
	)	
(Oconee Nuclear Station, Units 1, 2, and 3)	)	

EXEMPTION

I.

The Duke Energy Corporation (Duke/the licensee) is the holder of Facility Operating License Nos. DPR-38, DPR-47, and DPR-55, that authorize operation of the Oconee Nuclear Station, Units 1, 2, and 3 (Oconee), respectively. The licenses provide, among other things, that the facilities are subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (the Commission) now or hereafter in effect.

The facilities consist of pressurized water reactors located on Duke's Oconee site in Seneca, Oconee County, South Carolina.

II.

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix G requires that pressure-temperature (P-T) limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR Part 50, Appendix G states that "[t]he appropriate requirements on...the pressure-temperature limits and minimum permissible temperature must be met for all conditions." Appendix G of 10 CFR Part 50 specifies that the requirements for these limits are the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G limits.

Pressurized water reactor licensees have installed cold overpressure mitigation systems/low temperature overpressure protection (LTOP) systems in order to protect the reactor coolant pressure boundary (RCPB) from being operated outside of the boundaries established by the P-T limit curves and to provide pressure relief of the RCPB during low temperature overpressurization events. The licensee is required by the Oconee Units 1, 2, and 3 Technical Specifications (TS) to update and submit the changes to its LTOP setpoints whenever the licensee is requesting approval for amendments to the P-T limit curves in the Oconee Units 1, 2, and 3 TS.

Therefore, in order to address provisions of amendments to the TS P-T limits and LTOP curves, the licensee requested in its submittal dated May 11, 1999, that the staff exempt Oconee Units 1, 2, and 3 from application of specific requirements of 10 CFR Part 50, Section 50.60(a) and 10 CFR Part 50, Appendix G, and substitute use of three ASME Code Cases as follows:

1. N-514 as an alternate methodology for determining the low temperature overpressure protection system enable temperature,
2. N-588 for determining the reactor vessel P-T limits derived from postulating a circumferentially-oriented reference flaw in a circumferential weld, and
3. N-626 as an alternate reference fracture toughness for reactor vessel materials for use in determining the P-T limits. (As a result of recent ASME code committee action, the designation for Code Case N-626 was changed to N-640. Therefore, Code Case N-640 will be discussed below rather than Code Case N-626, the designation referenced in the submittal.)

The proposed action is in accordance with the licensee's application for exemption contained in a submittal dated May 11, 1999, and is needed to support the TS amendments that are contained in the same submittal and are being processed separately. The proposed amendments will revise the P-T limits of TS 3.4.3 for Oconee Units 1, 2, and 3 related to the heatup, cooldown, and inservice test limitations for the Reactor Coolant System of each unit to a maximum of 33 Effective Full Power Years (EFPY). It will also revise TS 3.4.12, Low Temperature Overpressure Protection System, to reflect the revised P-T limits of the reactor vessels.

Code Case N-514

During staff review of this submittal, the staff determined that granting of an exemption to use Code Case N-514 to redefine the LTOP enable temperature as RT_{NDT} +50°F was not necessary. Since the prior definition of the enable temperature as RT_{NDT} +90°F is found only in an NRC Branch Technical Position, an exemption is not required.

Code Case N-588

This requested exemption will allow the use of ASME Code Case N-588 to determine stress intensity factors for postulated defects in circumferential welds. Appendix G of 10 CFR Part 50 requires, in part, that Article G-2120 of ASME Section XI, Appendix G, be used to determine the maximum postulated defects in reactor pressure vessels (RPV) when determining the P-T limits for the vessel. Article G-2120 specifies that the postulated defect be in the surface of the vessel material and normal (perpendicular in the plane of the material) to the direction of maximum stress. ASME Section XI, Appendix G, also provides methodology to determine the stress intensity factors for a maximum postulated defect normal to the maximum stress. The purpose of this article is to prevent non-ductile failure of the RPV by providing procedures to identify the most limiting postulated fractures to be considered in the development of P-T limits.

Per Article G-2120 of ASME Section XI, Appendix G, the postulated flaw "normal to the direction of maximum stress" would be an axially-oriented flaw for each reactor vessel beltline material. This postulated reference flaw is intended to be a conservative, bounding defect when compared to those defects that may have gone undetected during the fabrication process.

Engineering experience and non-destructive examinations over the course of the last thirty years have shown this to be a valid assumption and have shown that no service-induced degradation mechanism exists in pressurized water reactors that would cause significant growth of preservice flaws.

However, for a circumferential weld, it is extremely unlikely that axial flaws of appreciable size would be introduced perpendicular to the weld seam during fabrication since the nature of the welding process leads to any extended flaws being introduced parallel to the direction of travel of the welding head. In addition, the size of flaw required to be postulated by the ASME Code, if oriented axially, would extend across the entire nominal width of the circumferential weld and into the base material on either side. Given the strict procedure controls required during the fabrication of ASME Code Class 1 reactor vessels and the extensive amount of preservice and inservice non-destructive examination to which their welded regions have been subjected, it has been confirmed that any remaining defects are small and do not cross transverse to the weld bead orientation. Therefore, the NRC staff finds that the application of this degree of non-physical conservatism is not necessary to achieve the underlying intent of 10 CFR Part 50, Appendix G.

In summary, the underlying purpose of 10 CFR Part 50, Appendix G and ASME Section XI, Appendix G, is to satisfy the requirement that: (1) the reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure that when stressed the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized, and (2) P-T operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

Application of Code Case N-588 to determine P-T operating and test limit curves per ASME Section XI, Appendix G, provides appropriate, conservative procedures to determine limiting maximum postulated defects and to consider those defects in the P-T limits. This application of the code case maintains the margin of safety for circumferential welds equivalent to that originally contemplated for plates/forgings and axial welds.

Therefore, pursuant to 10 CFR 50.12(a)(2)(ii), application of the code case would continue to achieve the underlying purpose of the rule.

Code Case N-640 (formerly Code Case N-626)

The licensee has proposed an exemption to allow use of ASME Code Case N-626 (which is now Code Case N-640) in conjunction with ASME Section XI, 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G, to determine that the P-T limits meet the underlying intent of the NRC regulations.

The proposed amendment to revise the P-T limits for Oconee Units 1, 2, and 3 rely in part on the requested exemption. These revised P-T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{Ia} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME Section XI, Appendix G process of determining P-T limit curves remain unchanged.

Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limits curve is more technically correct than the K_{Ia} curve. The K_{Ic} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heat-up and cooldown process of a reactor vessel. The licensee has determined that the use of the initial conservatism of the K_{Ia} curve when the curve was codified in 1974 was justified. This initial conservatism was necessary due to the limited knowledge of reactor pressure vessel materials. Since 1974, additional knowledge has been gained about reactor pressure vessel materials, which demonstrates that the lower bound on fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect the public health and safety from potential reactor pressure vessel failure. In addition, P-T curves based on the K_{Ic} curve will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operations. The two primary safety benefits in opening the low temperature operating window are a reduction in the challenges to RCS power operated relief valves and elimination of RCP impeller cavitation wear.

Since the RCS P-T operating window is defined by the P-T operating and test limit curves developed in accordance with the ASME Section XI, Appendix G procedure, continued operation of Oconee with these P-T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the P-T operating window. This restriction requires, under certain low temperature conditions, that only one reactor coolant pump in a reactor coolant loop be operated. The licensee has found from experience that the effect of this restriction is undesirable degradation of reactor coolant pump impellers that results from cavitation sustained when either one pump or one pump in each loop is operating. Implementation of the proposed P-T curves as allowed by ASME Code Case N-640 does not significantly reduce the margin of safety. Thus, pursuant to10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served.

In summary, the ASME Section XI, Appendix G procedure was conservatively developed based on the level of knowledge existing in 1974 concerning reactor pressure vessel materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. The NRC staff concurs that this increased knowledge permits relaxation of the ASME Section XI, Appendix G requirements by application of ASME Code Case N-640, while maintaining, pursuit to 10 CFR50.12(a)(2)(ii), the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

III.

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50, when (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. The staff accepts the licensee's determination that an exemption would be required to approve the use of Code Cases N-588 and N-626 (now Code Case N-640). The staff examined the licensee's rationale to support the exemption request and concurred that the use of the code cases would also meet the underlying intent of these regulations. Based upon a consideration of the conservatism that is explicitly incorporated into the methodologies of 10 CFR Part 50, Appendix G; Appendix G of the Code; and RG 1.99, Revision 2, the staff concluded that application of the code cases as described would provide an adequate margin of safety against brittle failure of the RPVs. This is also consistent with the determination that the staff has reached for other licensees under similar conditions based on the same considerations. Therefore, the staff concludes that requesting the exemption under the special circumstances of 10 CFR 50.12(a)(2)(ii) is appropriate and that the methodology of Code Cases N-588 and N-626 may be used to revise the LTOP setpoints and P-T limits for the Oconee Units 1, 2, and 3 reactor coolant system.

IV.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the exemption is authorized by law, will not endanger life or property or common defense and security, and is, otherwise, in the public interest. Therefore, the Commission hereby grants Duke an exemption from the requirements of 10 CFR Part 50, Section 50.60(a) and 10 CFR Part 50, Appendix G, for the Oconee Nuclear Station, Units 1, 2, and 3.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not result in any significant effect on the quality of the human environment (64 FR 40901).

This exemption is effective upon issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY:

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland, this 29th day of July 1999.

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Division of Licensing Project Management
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