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A Duke Energy Company

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December 21, 2004

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: Duke Energy
Oconee Nuclear Station, Units 1 and 2
Docket Nos. 50-269,-270
Third Ten Year Inservice Inspection Interval
Requests for Relief No. 04-ON-013

Pursuant to 10 CFR 50.55a(a)(3)(i), attached is a Request for Relief from certain requirements of ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition with no Addenda.

The ASME Code considers components which comprise the Reactor Coolant Pressure Boundary to be ASME Class 1 through the second valve. Table IWB-2500-1 Examination Category B-P, Pressure Retaining Components addresses all Class 1 components within the system boundary. Paragraph IWB-5221(a) mandated performance of a system leakage test at a test pressure equivalent to the normal operating pressure associated with 100% rated reactor power (i.e. 2155 psig) for ASME Class 1 piping. Further, this test is to be conducted once during the ten year interval, at or near the end of the interval.

Four specific sections of pipe, in equivalent locations on both Unit 1 and Unit 2, were inadvertently omitted from testing during the Class 1 test during the third ten year interval. That interval ended on these Units on January 1, 2004.

Request for Relief 04-ON-013 is to allow Duke Energy to credit testing to be performed on these sections during the fourth ten year interval in lieu of the required third interval tests. In this case, the test on these piping sections would be performed in the first refueling outage in the fourth interval as follows:

A047

Unit 1 - 1EOC22, Scheduled to begin 04/09/05
Unit 2 - 2EOC21, Scheduled to begin 10/22/05.

Such a relief would be in accordance with Oconee Technical Specification SR 3.0.3 which allows a missed surveillance to be delayed up to the limit of the specified frequency. However, although Section 5.5 of the Oconee Technical Specifications address programs such as the Inservice Test Program and the Containment Tendon Surveillance Program which are mandated by 10 CFR 50.55(a) and impose ASME Code related activities, the Inservice Inspection Program is not specifically invoked by Technical Specifications. Therefore, the guidance of SR 3.0.3 is not directly applicable to the Inservice Inspection Program.

Therefore, Duke Energy requests that the NRC grant relief as authorized under 10 CFR 50.55a(a)(3)(i).

If there are any questions or further information is needed you may contact R. P. Todd at (864) 885-3418.

Very truly yours,

R. A. Jones
Site Vice President

Attachment

xc w/att: Mr. William D. Travers
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Atlanta, GA 30303

L. N. Olshan, Project Manager, Section 1
Project Directorate II
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xc(w/o attch):

M. C. Shannon
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Mr. Henry Porter
Division of Radioactive Waste Management
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1.0 System / Component for Which Relief is Requested:

Relief is requested for portions of ASME Code Class 1 piping and components connected to the Reactor Coolant System (RCS) that are normally isolated from direct RCS pressure during normal operation. These areas are isolated from the RCS by their configuration between two normally closed valves that remain closed when the unit is in Modes 3, 2, or 1.

This relief is requested for the third 10-year interval that started on July 15, 1994 and ended January 1, 2004.

2.0 Applicable Code Edition and Addenda:

ASME Boiler & Pressure Vessel Code, Section XI, 1989 Edition with no addenda.

3.0 Applicable Code Requirement:

Section XI, Table IWB-2500-1 Examination Category B-P, Pressure Retaining Components, System Hydrostatic Test, all Class 1 components within the system boundary.

This test is to be conducted once during the 10 year interval, either at or near the end of the interval.

Code Case N-498-1: Alternative Rules for 10-year system hydrostatic testing for Class 1, 2, and 3 systems, Section XI, Division 1.

a. It is the opinion of the committee that as an alternative to the 10 year system Hydrostatic test required by Table IWB-2500-1 Category B-P, the following rules shall be used:

   (1) A system leakage test (IWB-5221) shall be conducted at or near the end of each inspection interval, prior to reactor startup.

   (2) The boundary subject to test pressurization during the system leakage test shall extend to all Class 1 pressure retaining components within the system boundary.

   (3) Prior to performing the VT-2 visual examination, the system shall be pressurized to nominal operating pressure for at least 4 hours for insulated systems and 10 minutes for non-insulated systems. The system shall be maintained at nominal operating pressure during the performance of the VT-2 visual examination.

(4) The test temperatures and pressures shall not exceed limiting conditions for the hydrostatic test curve as contained in the plant Technical Specifications.

(5) The VT-2 visual examination shall include all components within the boundary identified in (a)(2) above.

(6) Test instrumentation requirements of IWA-5260 are not applicable.

<u>IWB-5221 System Leakage Test</u>

(a) The system leakage test shall be conducted at a test pressure not less than the nominal operating pressure associated with 100% rated reactor power.
(b) The system test pressure and temperature shall be attained at a rate in accordance with the heat-up limitations specified for the system.

The application of the above ASME Code requirements along with those of Code Case N-498-1 would require all Class 1 pressure retaining components within the system boundary to be pressurized and VT-2 visually examined.

Relief is requested to defer the 10-year interval pressure test on the four identified areas in Units 1 and 2 from the end of the third interval to the first refueling outage in the fourth interval.

In accordance with the requirements of 10 CFR 50.55a(a)(3)(i), relief is requested from the requirements of the 1989 ASME B&PV Code, Section XI, Category B-P, Table IWB-2500-1, footnote (6). This footnote mandates performance of the 10-year pressure test to be conducted at or near the end of each inspection interval.

4.0 Reason for Request:

There are four areas in Unit 1 and four identical areas in Unit 2 at Oconee Nuclear Station that were inadvertently not pressure tested during the performance of the Class 1 10-year pressure test. These four sections of pipe are identical configurations in each unit and are as follows:

Section 1: 1" pipe between isolation valves HP-490 and HP-497
Section 2: 1" pipe between isolation valves HP-491 and HP-498
Section 3: 1" pipe between isolation valves HP-492 and HP-499
Section 4: 1" pipe between isolation valves HP-493 and HP-500

5.0 Proposed Alternative and Basis for Use:

The four identified areas in Unit 1 and Unit 2 were inadvertently omitted from the Class 1, 10-year pressure test, performed in the third interval but will be 10-year pressure tested during the next refueling outage which will be during the fourth interval.

Per the requirements of the 1989 ASME B&PV Code, Section XI, Category B-P, Table IWB-2500-1, footnote (1), the four identified areas in Unit 1 and Unit 2 have received a VT-2 visual exam each refueling outage without being pressurized and have had no evidence of leakage. The next opportunity to pressurize these areas and perform a VT-2 visual exam will be during the next refueling outage.

If a leak were to develop in any of the four Unit 1 or Unit 2 areas described in this relief request, it could be detected by various means available to the operators. There are area monitors which monitor the reactor building atmosphere and will alarm when the radiation levels reach set limits.

In addition, plant Technical Specification 3.4.13 requires that at least once every 72 hours, when above Mode 5, the reactor coolant system water inventory balance be performed. This Technical Specification limits the amount of unknown leakage to 1 gpm. If this limit is exceeded, then the source must be identified or the reactor must be placed in Mode 3 within 12 hours and Mode 5 in 36 hours.

Besides the area radiation monitors and the water inventory monitoring, other leakage detection methods available include frequency of having to pump the reactor building normal sump. Increased frequency for pumping the normal sump would be an indication that there is a leak in the reactor building.

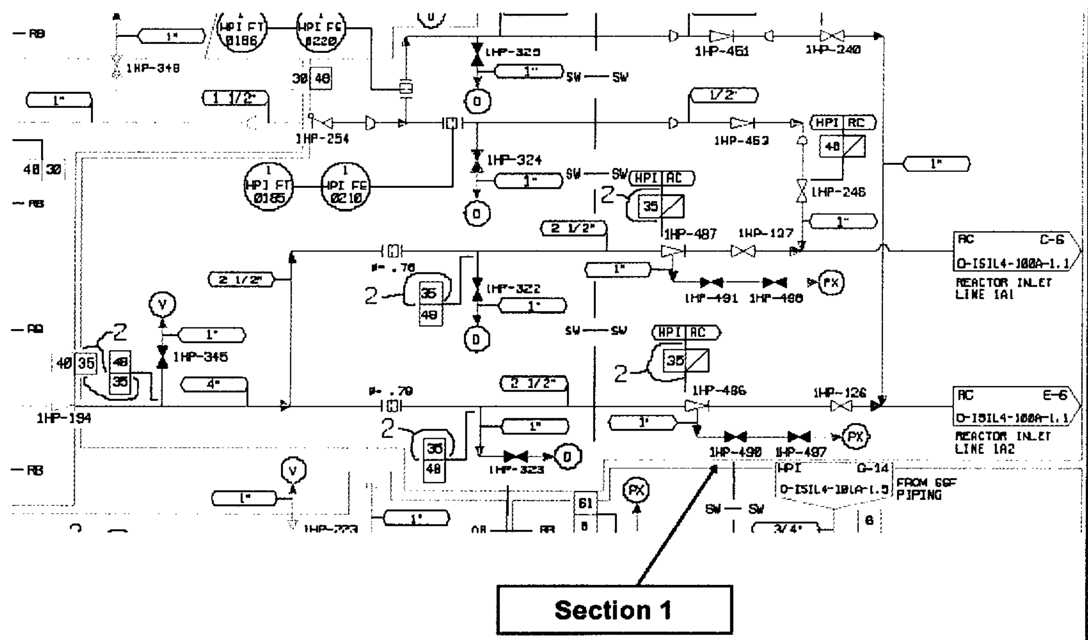
Starting at Mode 3, during reactor shutdown for refueling outages, numerous inspections are made in the reactor building looking for indications of leakage. Each leak is evaluated and repaired.

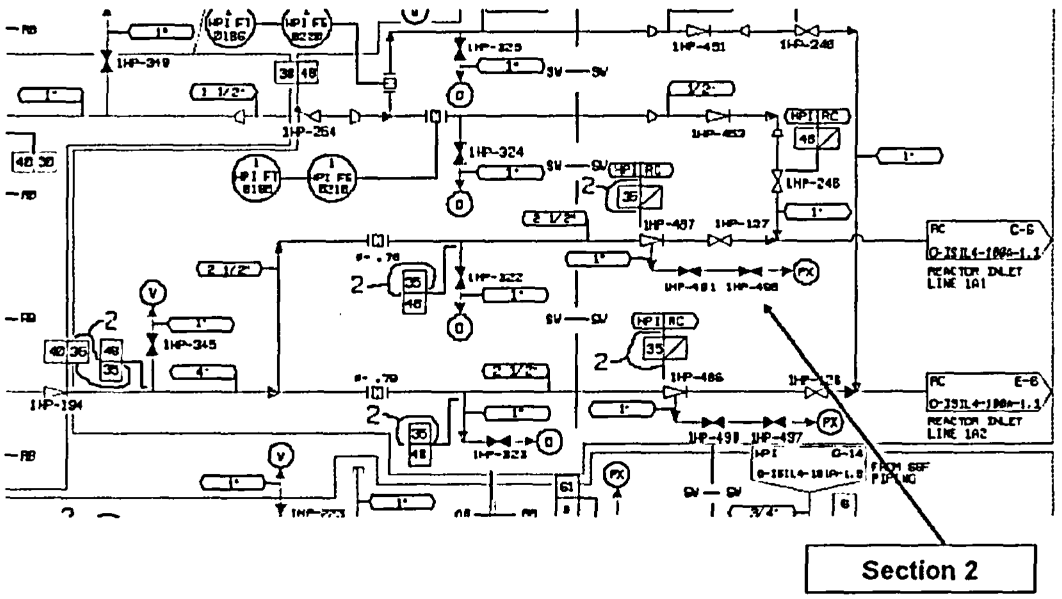
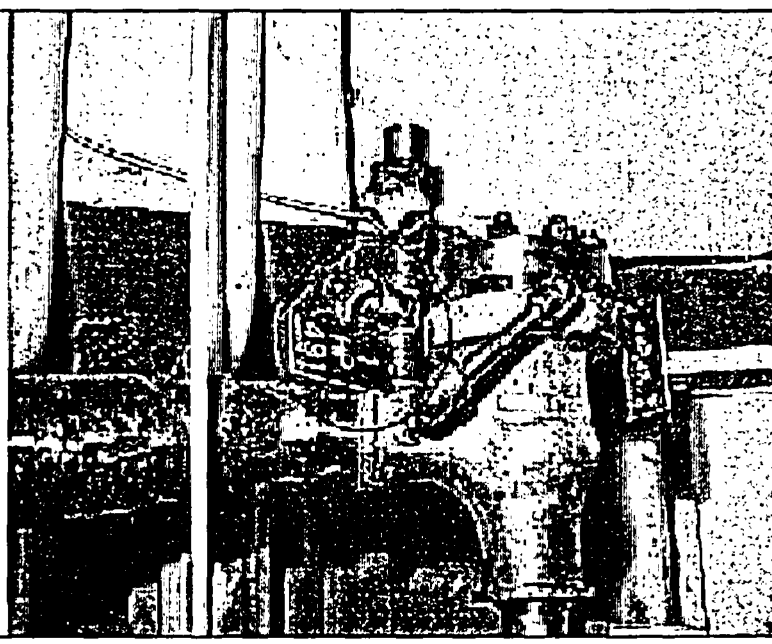
6.0 Implementation Schedule:

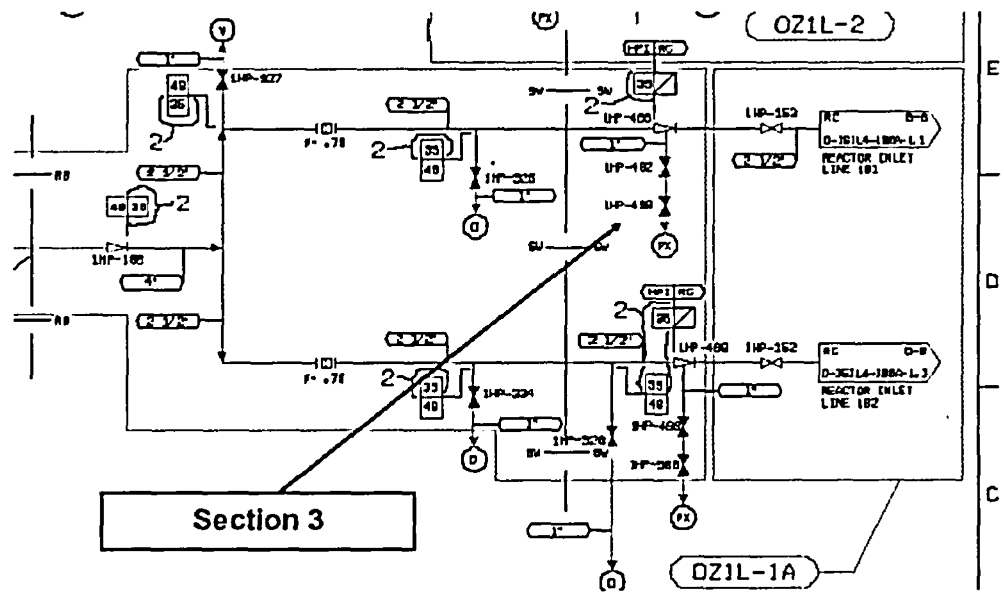
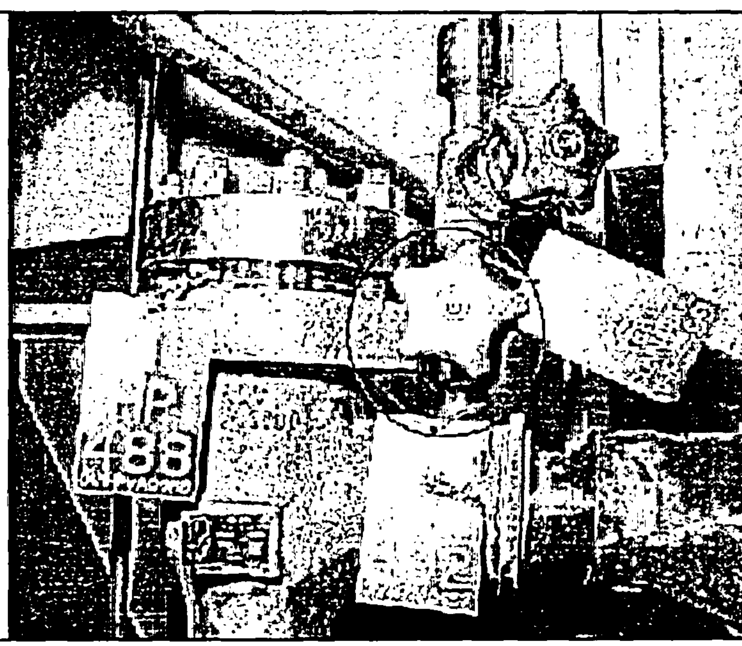
Relief is requested to defer the 10-year interval pressure test on the four identified areas in Units 1 and 2 from the end of the third interval to the first refueling outage in the fourth interval as follows:

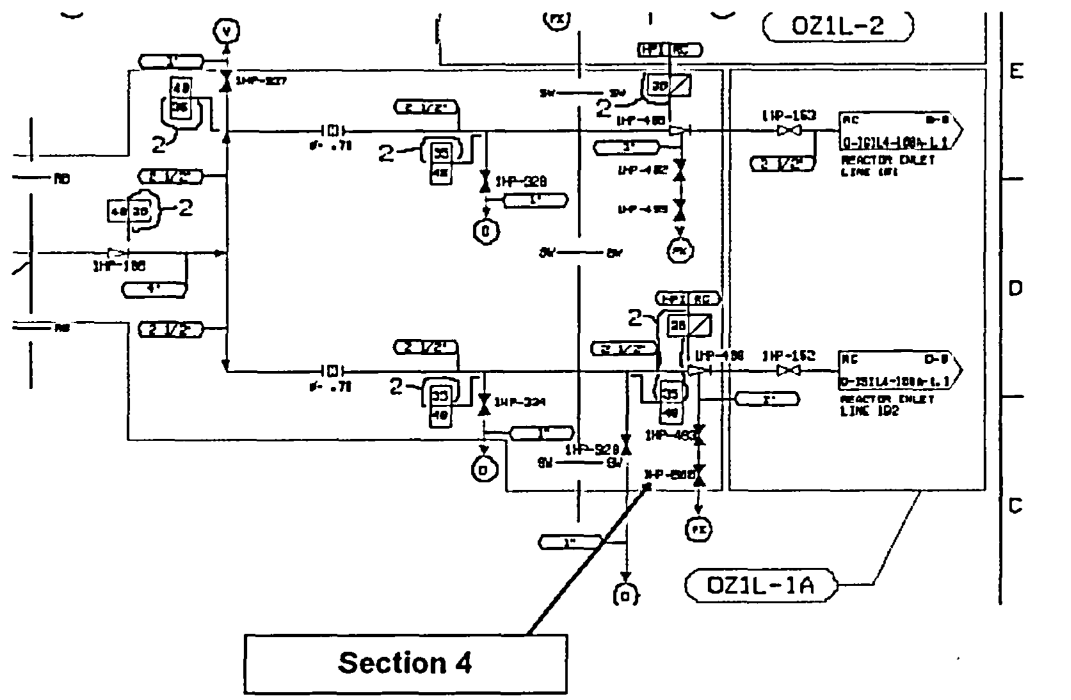
Unit 1 - 1EOC22, scheduled for 04/09/05
Unit 2 - 2EOC21, scheduled for 10/22/05

7.0 Other Information:

Section 1	Valves HP-490 through HP-497 Flow Diagram: OFD-101A-1.4 (Unit 1 Shown – Unit 2 Is Identical)
	
	No Digital Images Available For Section 1 – See Digital Image For Section 2 (Same Configuration)

Section 2	Valves HP-491 through HP-498 Flow Diagram: OFD-101A-1.4 (Unit 1 Shown – Unit 2 Is Identical)
	
Valve HP-498 on top & HP-491 on bottom	

Section 3	Valves HP-492 through HP-499 Flow Diagram: OFD-101A-1.4 (Unit 1 Shown – Unit 2 Is Identical)
	
Valve HP-499 on top & HP-492 on bottom	

Section 4	Valves HP-493 through HP-500 Flow Diagram: OFD-101A-1.4 (Unit 1 Shown – Unit 2 Is Identical)
	
Valve HP-500 on top & HP-493 on bottom	

Sponsored By:	Jim Baughman	Date	11/17/04
Approved By:	R. Kevin Rhyne	Date:	11/17/04