Duke Power

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May 16, 2005

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Request for Relief Number 04-CN-004
Request for Relief from Pressure Requirement at Certain Class 1 Inservice Inspection Pressure Test Boundaries

References: Letters from Duke Energy Corporation to NRC, dated May 18, 2004 and January 27, 2005

Pursuant to 10 CFR 50.4, please find attached our reply to the enclosed request for additional information. This request for additional information was discussed in a telephone conference call between Duke Energy Corporation and the NRC on May 11, 2005. The attachment to this letter contains the request for additional information, followed by our response.

There are no regulatory commitments contained in this letter.

If you have any questions concerning this material, please call L.J. Rudy at (803) 831-3084.

Very truly yours,

D.M. Jamil

LJR/s

Attachment

A047

Catawba Nuclear Station 20th Anniversary 1985-2005

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xc (with attachment):

W.D. Travers, Regional Administrator
U.S. Nuclear Regulatory Commission, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

E.F. Guthrie, Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

S.E. Peters, Project Manager (addressee only)
U.S. Nuclear Regulatory Commission
Mail Stop O-8 G9
Washington, D.C. 20555-0001

Attachment

Response to Request for Additional Information

Request for Additional Information:

Given that Portions 6 and 7 could be at atmospheric pressure, show that your alternative provides reasonable assurance of structural integrity and leak-tightness of these portions of RR 04-CN-004. For example, discuss the material type, NDE performed, fluid type (borated), etc.

Duke Energy Corporation Response:

Portion 6:
A VT-2 qualified QC Inspector will examine the piping after a four hour hold time with the Reactor Coolant System (NC) at RCS pressure and temperature (2235 psig @ 557°F) with all affected double isolation valves and test/drain valves in their normal operating position (closed). These portions of piping could see a range of pressures dependent on valve leakage during the pressure test. With leakage through valves NC4, NC13, NC19 and NC94 for Portion 6, the high end of the range is RCS pressure. For the unlikely case of no leakage through these valves, the pressure in these portions of piping could be near atmospheric.

This alternative provides reasonable assurance of structural integrity because these lines are water filled and it is expected that any leak on these portions of piping would be identified by active leakage or boric acid deposits observed during the pressure test. Any leakage during the previous operating cycle would be indicated by boron deposits on the outside surface of the uninsulated piping regardless of valve leakage during the pressure test inspection. Furthermore, the pipe and fitting material is stainless steel as indicated in the response to question 7 in the January 27, 2005 letter. The piping and valve connections are socket welded and have been qualified for all design conditions including pressure, thermal, and seismic loadings. There is a significant margin in the required wall thickness for this 2" schedule 160 piping based on pressure loadings. (The required wall thickness for 2" schedule 160, SA376, TP304 pipe at a pressure of 2235 psig and a temperature of 650°F is 0.155"; the provided wall thickness is 0.343".) There are no common degradation mechanisms associated with Grade 304 stainless steel in the pressurized water reactor primary water environment as indicated by the response to question 7. The segment lengths of piping within the scope of this request are small (an average of approximately 3 feet of piping for each loop) and the associated number of socket welded joints are also small (an average of approximately 8 welds per loop). Past surface examinations of a subset of these welds have not identified any reportable indications. Based on these reasons, this alternative provides reasonable assurance of the structural integrity of this piping.

Portion 7:
A VT-2 qualified QC Inspector will examine the piping after a four hour hold time with the NC system at RCS pressure and temperature (2235 psig @ 557°F) with all affected double isolation valves and test/drain valves in their normal operating position (closed). These portions of piping could see a range of pressures dependent on valve leakage during the pressure test. With leakage through valves NC298, NC311, NC250A and NC252B for Portion 7, the high end of the range is RCS pressure. For the unlikely case of no leakage through these valves, the pressure in these portions of piping could be near atmospheric.

This alternative provides reasonable assurance of structural integrity because these uninsulated line lengths are small, the number of welded joints is few, the margin in pipe wall thickness is large for the operating pressure, and the piping and fitting materials have been very reliable for applications under pressurized water reactor primary system conditions.

During operation, the piping segments associated with NC298, NC311, NC250A, and NC252B are air filled. However, any leakage through these valves and subsequent leakage of the pressure boundary would quickly purge the relatively small air volumes associated with these segments. A four hour hold time at operating temperature and pressure prior to the pressure test further ensures that any leakage would be identified.

It is expected that any leak on these segments of piping would be identified during the pressure test in the form of boric acid deposits. Even with extremely low valve leakage, these segments are expected to pressurize over the operational cycle. If a pressure boundary leak occurs, any air would be subsequently purged from the volume, followed by the leak out of primary system water and the accumulation of boron residue as the water evaporates in the ambient containment environment. Thus, any leakage during the previous operating cycle would be indicated by boron deposits on the outside surface of the uninsulated piping and identified during plant walkdowns or the alternative pressure test.

Table 1

Segment (1)	Initial Operational Fluid	Pipe Size	t_n (provided pipe wall thickness, in)	t_{req} for 2235 psig @ 650°F, in	Piping Segment Length	Number of Welded Conn	Weld Type
NC298	air	3" sch 160	0.437	0.229	6"	2	butt
NC311	air	3/4" sch 160	0.218	0.069	6"	2	socket
NC250A & NC252B	air	1" sch 160	0.250	0.086	4'-6"	10 total	socket

(1) only the inboard reactor coolant loop valve has been listed here to identify the Portion 7 segment

The Portion 7 piping and valve connections are welded joints as indicated in Table 1 above and have been qualified for all design conditions including pressure, thermal, and seismic loadings. For pressure loads, there is a significant margin in the required wall thickness for these piping segments as indicated above. There are no common degradation mechanisms associated with Grade 304 stainless steel in the pressurized water reactor primary water environment as indicated by the response to question 7 in the January 27, 2005 letter. The segment lengths of piping within the scope of this request are small and the associated number of welded joints is also small. Past surface and volumetric examinations of a subset of these welds have not identified any reportable indications. Based on these reasons, this alternative provides reasonable assurance of the structural integrity of this piping.