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A Duke Energy Company

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

Subject: Duke Energy
Oconee Nuclear Station, Units 1, 2 and 3
Docket Nos. 50-269,-270,-287
Third Ten Year Inservice Inspection Interval
Requests for Relief No. 04-ON-012

Pursuant to 10 CFR 50.55a(a)(3)(ii), attached is a Request for Relief to use an alternative to the 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. 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.

The Oconee Reactor Coolant Pressure Boundary was originally designed and built to USAS B31.7 Nuclear Power Piping Code (February 1968) and Addenda (June 1968), which considered the Reactor Coolant Pressure Boundary to be through the first valve.

As a result, certain sections of pipe, specified in the attached request, were designed such that the system leakage test conditions specified for Class 1 cannot be attained without excessive hardship.

Request for Relief 04-ON-012 is to allow Duke Energy to credit testing performed to date on these sections in lieu of the required system leakage tests.

A047

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

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

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Duke Energy Corporation
Oconee Nuclear Station Units 1, 2, & 3
Request for Relief No. 04-ON-012

1.0 System / Component for Which Relief is Requested:

Relief is requested for sections of ASME Code Class 1 piping and components connected to the Reactor Coolant System (RCS) that are isolated from direct RCS pressure during normal plant operation. These sections of pipe are isolated from the RCS by their configuration, being either up stream of a check valve, between two or more check valves, or between two normally closed valves that remain closed when the unit is in Modes 3, 2, or 1. The normal operating pressure for the RCS is 2150 to 2160 psig.

At Oconee Nuclear Station, for each unit, there are five such sections of pipe that would represent a hardship to perform a Class 1 hydrostatic test. All five sections are totally contained within the reactor building. The five sections of pipe are as follows:

1. Between LP-1 and LP-2
2. Between LP-103 and LP-104
3. Between LP-46 and LP-131
4. Between LP-47 and CF-12
5. Between LP-48 and CF-14

This represents 15 sections of pipe between the three units. The section of pipe between these valves can not be hydrostatically tested as specified in ASME Section XI 1989 edition, no addenda.

This request is for all three units at Oconee Nuclear Station, Units 1, 2, & 3, since the pipe configuration for these sections are alike between the three units.

Section 1: Configuration for LP-1 and LP-2:

LP-1 and LP-2 are part of the decay heat removal system. They are placed in service after the reactor has been shutdown and depressurized to about 350 psig.

LP-1 and LP-2 are 12 inch gate valves separated by a piece of 12 inch schedule 140 stainless steel pipe about 30 inches long. LP-1 is the first valve off the reactor coolant system (RCS) on the decay heat removal line. This section of pipe is covered with metal reflective insulation. Attachment 1 shows the pipe configuration with applicable design parameters.

Section 2: Configuration for LP-103 and LP-104:

LP-103 and LP-104 are part of the boron dilution system. These valves would be used during a LOCA to prevent unacceptably high concentration of boron in the reactor vessel. Down stream of LP-104 this pipe is open ended into the reactor building basement. When this section of pipe is placed in service water from the RCS would fall into the basement where it could be recycled through the LPI pumps

LP-103 and LP-104 are 3 inch gate valves separated by a piece of 3 inch schedule 160 stainless steel pipe about 57 inches long. LP-103 is the first valve off the decay heat removal line upstream of LP-1. This section of pipe is not insulated. Attachment 1 shows the pipe configuration with applicable design parameters.

Section 3: Configuration for LP-46 and LP-131

LP-46 and LP-131 are part of the alternate pressurizer spray supply line. This line provides no event mitigation function.

LP-46 and LP-131 are 1-1/2 inch check valves separated by a piece of 1-1/2 inch schedule 160 stainless steel pipe about 100 feet long. This section of pipe is not insulated. Attachment 2 shows the pipe configuration with applicable design parameters.

Section 4: Configuration for LP-47 and CF-12 (and CF-11)

These three valves, LP-47, CF-12, and CF-11 form the section of pipe that directs water from the core flood tank or the LPI pump to the reactor vessel. During a LOCA LP-47 serves to dead head the LPI pumps until the RCS pressure drops below the discharge pressure of the LPI pump. At that time the LPI pumps start supplying cooling water to the core.

LP-47 is a 10 inch check valve and CF-12 and CF-11 are 12 inch check valves. This line is part of the emergency core cooling system, during a loss of coolant accident the water would be supplied to the core from the core flood tank through check valves CF-11 and CF-12. This would force closed check valve LP-47; after the core flood tank becomes empty the pressure continues to decrease and eventually cooling water is supplied to the core through check valves LP-47 and CF-12 and check valve CF-11 would be forced closed. Only the part of the pipe down stream of CF-12 is insulated. Attachment 3 shows the pipe configuration with applicable design parameters.

Section 5: Configuration for LP-48 and CF-14 (and CF-13)

These three valves, LP-48, CF-14, and CF-13 form the section of pipe that directs water from the core flood tank or the LPI pump to the reactor vessel. During a LOCA LP-48 serves to dead head the LPI pumps until the RCS pressure drops below the discharge pressure of the LPI pump. At that time the LPI pumps start supplying cooling water to the core.

LP-48 is a 10 inch check valve and CF-14 and CF-13 are 12 inch check valves. This line is part of the emergency core cooling system, during a loss of coolant accident the water would be supplied to the core from the core flood tank through check valves CF-13 and CF-14. This would force closed check valve LP-48; after the core flood tank becomes empty the pressure continues to decrease and eventually cooling water is supplied to the core through LP-48 and CF-14 and check valve CF-13 is forced closed. Only the part of the pipe down stream of CF-14 is insulated. Attachment 3 shows the pipe configuration with applicable design parameters

All these sections of pipe were installed as B31.7 Class B (ASME Class II). However, when preparing ISI boundaries these sections were identified as ASME Class I pipe. When Oconee was constructed the requirement was for the RCS boundary to extend through the first valve, which is why these sections of pipe were installed as ASME Class II. (Reference UFSAR Sections 3.1.1, General Criterion 1 and 3.2.2.1 System Classifications, enclosed as attachment 4.)

2.0 Code Requirement

<u>The 1989 edition with no addenda of ASME B&PV Code</u> Section XI, Table IWB-2500-1 Examination Category B-P, Item B15-51, Pressure Retaining Boundary, System Hydrostatic Test. Note # 2, in this table, states that the 'pressure retaining boundary during the system hydrostatic test shall include 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.

<u>Code Case N-498-1</u>: 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 VT-2 visual examination shall include all components within the boundary identified in (a)(2) above.

IWB-5221 System Leakage Test

(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 for Class 1 piping, that either at or near the end of each 10 year ISI interval, a system leakage test be performed that would extend a test pressure equal to the nominal operating pressure (2155psig) associated with 100 % reactor power to all Class 1 pressure retaining components connected to the RCS.

3.0 Code Requirement From Which Relief is Requested:

IWB-5221 System Leakage Test

(a) The system leakage test shall be conducted at a pressure not less than the nominal operating pressure associated with 100% rated reactor power.

In accordance with the requirements of 10 CFR 50.55a(a)(3)(ii), relief is requested from the requirements of the 1989 ASME B&PV Section XI Code, paragraph IWB-5221(a). This paragraph mandates 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 Class 1 piping.

Specifically, relief is requested from the requirement to extend 2155 psig as a test pressure (for holding time and VT-2 examination) to certain portions of the ASME Code Class 1 piping and components connected to the RCS, that are normally isolated from receiving direct RCS pressure during their normal operation. These sections of pipe are isolated from the RCS by their configuration, either up stream of a check valve, between two or more check valves, or between two normally closed valves that remain closed during unit operation in Modes 1, 2, or 3, when the RCS pressure is either at or approaching 2155 psig.

4.0 Basis for Relief:

The following discussion provides the basis for the request for relief and approval of the proposed alternate testing in accordance with the provisions in 10 CFR 50.55a(a)(3)(ii) due to the hardship that would incur complying with the ASME Section XI Code requirements.

Section 1: Between LP-1 and LP-2

There is an interlock in the open circuit for LP-1 which prevents this valve from being opened when the RCS pressure is above 400 psig. To be able to apply RCS pressure to this section of pipe would require defeating this interlock. If the interlock was defeated to allow the section of pipe down stream of LP-1 to be tested at RCS pressure then the test is depending on one valve,LP-2, to prevent overpressure of the pipe downstream of LP-2.

The stated design conditions for LP-2 are 2500 psig at 300 degrees F. To pressure test this section of pipe to RCS conditions would require either replacing LP-2 or as a minimum uprating the valve and associated piping.

Since the pipe down stream of LP-2 is rated for 388 psig, it would not be possible to apply RCS operating pressure to the section of pipe between LP-1 & LP-2 coming from the down stream side of LP-2.

Section 2: Between LP-103 and LP-104

LP-103 and LP-104 are 3 inch gate valves. Since this line is the primary boron dilution flow path, it is entirely contained within the reactor building and the pipe is open ended down stream of LP-104. Unlike the other sections of pipe described in this request for relief this pipe does not directly feed another section of pipe.

Per the manufacturers documentation valve LP-104 is designed for 3600 psig at 100 degrees F. To pressure test this section of pipe to RCS conditions would require either replacing LP-104 or as a minimum uprating the valve and associated piping.

If a leak was to develop between LP-103 and LP-104 the water from this leak would end up in the reactor building basement which is where the discharge of LP-104 also ends.

Section 3: Between LP-46 and LP-131

LP-46 is a check valve in the alternate supply line to the Pressurizer Spray Nozzle in the top of the pressurizer. There is no way to prevent this valve from opening when pressurized except to have the reactor coolant system in a solid condition. Operation of the plant in a solid condition is a hardship on the operators as well as the plant.

In addition LP-131 is a 1 1/2 check valve design parameters 3050 psig at 250 degrees F with a ethylene propylene rubber seat good for 450 degrees F. To pressure test this section of pipe to RCS conditions would require replacing LP-131 and uprating the valve and associated piping.

Section 4: Between LP-47 and CF-12

CF-12 is a 14 inch pressure seal check valve. The only practical path to pressurize this section of pipe to the requirements of a Class 1 pressure test would be to disassemble the check valve and remove the valve disc. Removing and installing the disc would be a hardship on personnel as well as the reactor coolant system since it would require a heat up to perform the pressure test and cool down and depressurization to assemble the valve.

Section 5: Between LP-48 and CF-14

CF-14 is a 14 inch pressure seal check valve. The only practical path to pressurize this section of pipe to the requirements of a Class 1 pressure test would be to disassemble the check valve and remove the valve disc. Removing and installing the disc would be a hardship on personnel as well as the reactor coolant system since it would require a heat up to perform the pressure test and cool down and depressurization to assemble the valve.

5.0 Alternate Examinations or Tests:

It is clear that through wall leakage that would occur at higher pressure, such as RCS pressure, would also reveal itself at lower pressures. It may take longer for some leaks to propagate through the pipe wall at lower pressure and the amount of leakage from a lower pressure test would be reduced, but the leakage would still be visible for a VT-2 examination.

Assuming a crack or defect would behave like a fixed area orifice, it can be shown that the leakage varies proportional to the square root of the ratio of the differential pressure (ref. Crane Technical Paper # 410). For example, if a leakage of L was projected to be present at a pressure of 2155 psig that same leak would exist at a reduced pressure with a reduced magnitude of:

$$\sqrt{\frac{270}{2155}} \times L = 0.35L$$

Inspections that reveal no leakage at 270 psig (where 35% of the leakage produced by 2155 psig would be present for detection during a VT-2 inspection) gives a high confidence that no leakage would be present at 2155 psig.

The reduced pressure testing performed as an alternate pressure test for Units 1, 2, & 3 that are covered in this request for relief range from 270 psig to approximately 2155 psig except for the section of pipe between LP-103 and LP-104, which was not pressure tested. Pressure that range between 270 psig and 2155 psig are sufficient to provide for the detection of any through wall leakage in the tested piping and components during the performance of the alternative tests.

Therefore, pursuant to the requirements of 10CFR50.55a(a)(3)(ii), the pressure tests and VT-2 examinations performed at the lower pressure indicated in the following alternative pressure tests are determined to provide an acceptable level of assurance of the quality, safety, and structural integrity of the tested piping.

Section 1: Between LP-1 and LP-2

The pipe between LP-1 through LP-2 was pressurized to 270 psig or greater.

Section 2: Between LP-103 and LP-104

For LP-103 to LP-104 all test were made with both valves closed; therefore, this section was not pressurized.

During the second period of the third interval in Unit 3 a hydrostatic test was performed on this section of pipe but only documented the VT-2 examination of the new welds (pressure was 2400 psig). The old welds were not inspected.

Section 3: Between LP-46 and LP-131

The piping between LP-46 to LP-131 was pressurized to approx 2155 during the second period, but no test was performed during the third period.

Section 4: Between LP-47 and CF-12

The piping between LP-47 and CF-12 was pressurized to 600 plus or minus (+/-) 25 psig, (Mode 3)

Section 5: Between LP-48 and CF-14

The piping between LP-48 and CF-14 was pressurized to 600 plus or minus (+/-) 25 psig. (Mode 3)

6.0 Justification for the Granting of Relief

If a leak were to develop in any of the piping described in the request for relief it could be detected by various means available to the operators. There are RIA's which monitor the reactor building atmosphere and alarm when the radiation levels reach pre-set limits.

In addition, plant Technical Specifications 3.4.13 requires that at least once every 72 hours when above MODE 5 a reactor coolant system water inventory balance is 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 in MODE 3 in 12 hours and MODE 5 in 36 hours.

Beside the RIA's and the water inventory monitoring, other leakage detection methods available include frequency of having to pump the reactor building normal sump. Increase 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 properly dispositioned.

Except for the section of pipe between LP-103 and LP-104, each alternate test indicated in this RFR was performed using a reduced pressure for the specified hold time and then VT-2 inspected to verify no through wall leaks exist in the tested pipe. The test boundaries included all piping described in Part I of this RFR. Additionally these sections of pipe are with in the Class 1 ISI boundary and thus undergo both volumetric and surface examinations as required by the ASME Section XI Code.

The section of pipe between LP-103 and LP-104 was not pressure tested. However, a failure of this section of pipe would not prevent the line from performing its intended function. The weld on the upstream side of LP-103 is tested at the end of each refueling outage during the leak check of the RCS. Additionally this section of pipe is with in the Class 1 ISI boundary and thus undergoes both volumetric and surface examinations as required by the ASME Section XI Code.

Finally, the Nuclear Regulatory Commission has granted similar relief to Catawba Nuclear Station, Kewaunee Nuclear Power Station, McGuire Nuclear Station, and Sequoyah Nuclear Station in response to that station's request for relief from certain ASME Section XI Class 1 pressure testing.

Based on the hardships without a compensating increase in quality and safety discussed above, the proposed alternatives and other inservice inspections for the applicable piping systems covered by this request will provide an acceptable level of assurance of the piping integrity in lieu of fully complying with the ASME Section

XI Code, and supporting Code Case N-498-1, requirements, pursuant to the provision in 10CFR50.55a(a)(3)(ii).

7.0 Implementation Schedule:

Relief is requested for the third ten-year interval of the ISI schedule. The third interval is scheduled to end as follows:

Unit 1	January 1, 2004
Unit 2	September 9, 2004
Unit 3	December 16, 2004

8.0 Other Information:

The following individuals reviewed this request for relief:

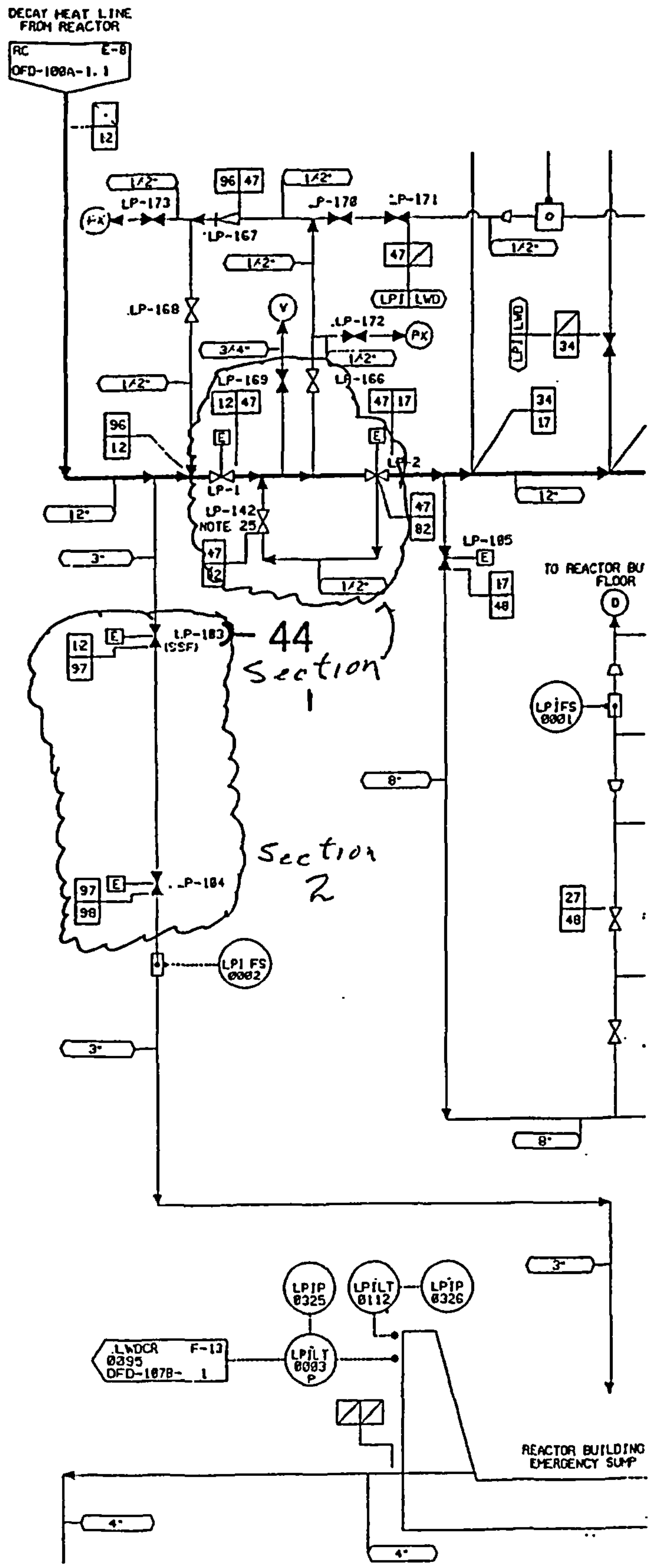
Mary Jo Clarkson	RCS System Engineer
Aaron Best	Core Flood System Engineer
Frank Eppler	Low Pressure Injection System Engineer
Vick Dixon	ISI Pressure Test Coordinator
Basil Carney	Repair and Replacement Engineer

Prepared By: Basil W. Carney Date: 12/13/2004

Approved By: [illegible] Date: 12/13/04

Attachment # 1
Request For Relief 04-ON-012

OFD 102A-x.1



DESIGN PARAMETERS

LINE NO.	DUKE CLASS	DESIGN PRESSURE	DESIGN TEMP.	MATERIAL	PIPE SPEC. NO.	PIPE SCH. NO.	ISI CLASS
1	B	100 PSIG	200°F	SS	151.2	NOTE 10	B
2	C	3 PSIG	150°F	SS	151.3	NOTE 8	B
3	B	BLDG.	300°F	SS	151.2	NOTE 8	B
4	C	100 PSIG	200°F	SS	151.3	NOTE 10	B
5	B	200 PSIG	300°F	SS	151.2	NOTE 10	B
6	B	370 PSIG	300°F	SS	301.2	NOTE 10	B
8	B	470 PSIG/505 PSIG	300°F/250°F	SS	301.2	NOTE 9	B
12	A	2500 PSIG	650°F	SS	1501.1	NOTE 12	A
14	B	200 PSIG	300°F	SS	301.2	NOTE 11	B
15	E	470 PSIG/505 PSIG	300°F/250°F	SS	301.4	NOTE 9	-
17	B	388 PSIG	300°F	SS	301.2	NOTE 10	B
23	E	BLDG.	300°F	SS	151.4	NOTE 10	-
27	H	ATM.	300°F	SS	151.4	NOTE 10	-
28	BC	200 PSIG	300°F	SS	151.3	NOTE 10	B
32	BC	470 PSIG/505 PSIG	300°F/250°F	SS	301.3	NOTE 9	B
33	BC	370 PSIG	300°F	SS	301.3	NOTE 10	B
34	BC	388 PSIG	300°F	SS	301.3	NOTE 10	B
35	BC	BLDG.	300°F	SS	151.3	NOTE 8	B
41	B	388 PSIG	300°F	SS	301.2	NOTE 13	B
47	B	2500 PSIG	300°F	SS	1501.2	NOTE 12	A
48	B	BLDG.	300°F	SS	151.2	NOTE 10	B
49	C	3 PSIG	200°F	SS	151.3	NOTE 8	C
50	E	3 PSIG	200°F	SS	151.4	NOTE 8	-
56	E	BLDG.	300°F	SS	151.3	NOTE 16	-
58	C	3 PSIG	150°F	SS	151.3	NOTE 10	B
59	E	ATM.	150°F	SS	151.4	NOTE 10	-
60	B	370 PSIG	300°F	SS	301.2	NOTE 23	B
67	B	2500 PSIG	650°F	SS	1501.2	NOTE 16	B

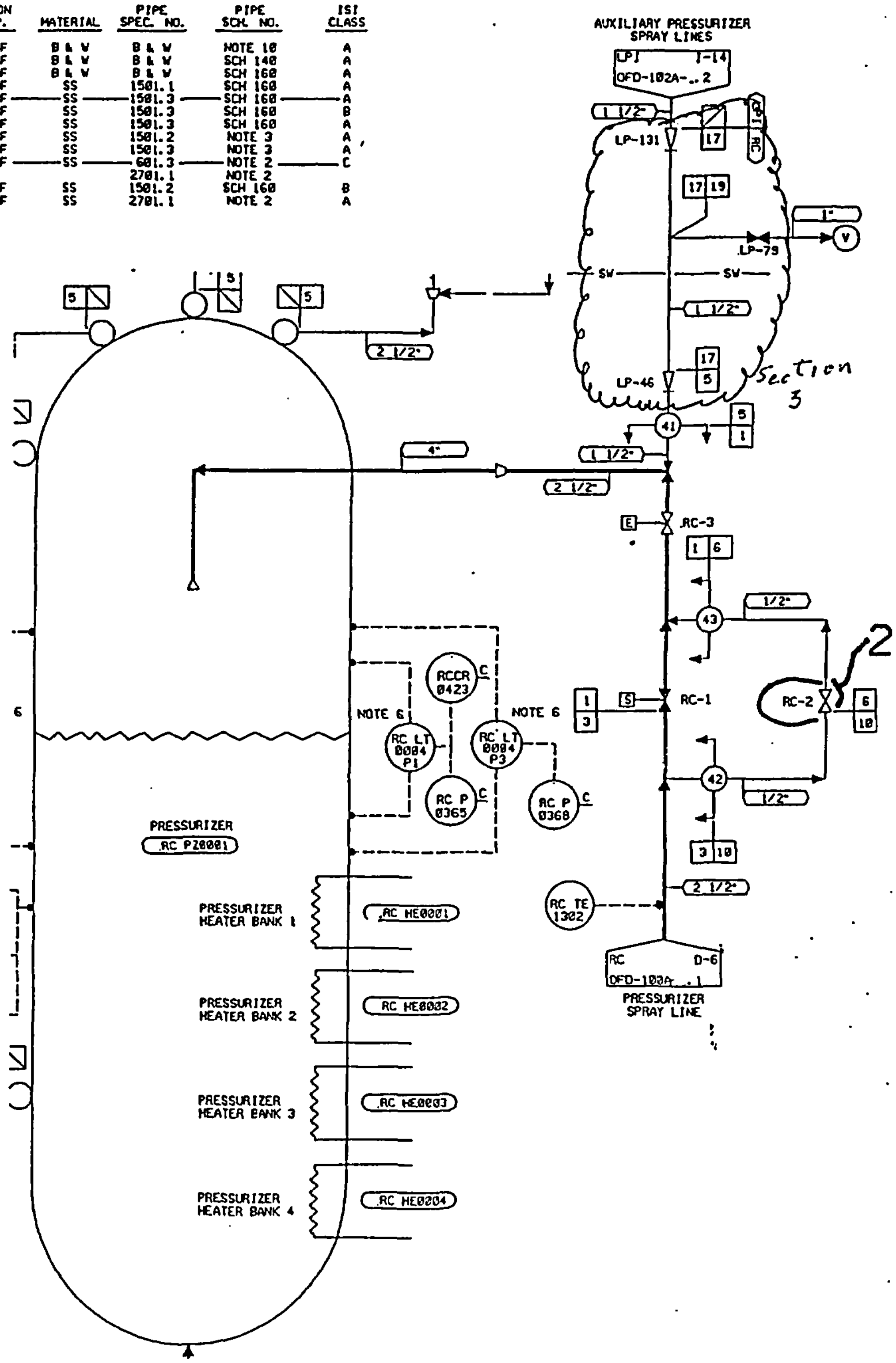
LINE NO.	DUKE CLASS	DESIGN PRESSURE	DESIGN TEMP.	MATERIAL	PIPE SPEC. NO.	PIPE SCH. NO.	ISI CLASS
70	E	470 PSIG/505 PSIG	300°F/250°F	SS	2701.2	NOTE 21	-
80	BC	200 PSIG	300°F	SS	301.3	NOTE 11	B
82	B	2500 PSIG	300°F	SS	2701.1	-	A
86	B	370 PSIG	300°F	SS	301.2	NOTE 12	B
96	B	2500 PSIG	650°F	SS	1501.2	NOTE 12	A
97	B	2500 PSIG	425°F	SS	1501.2	NOTE 12	A
98	H	400 PSIG	425°F	SS	301.4	NOTE 10	-

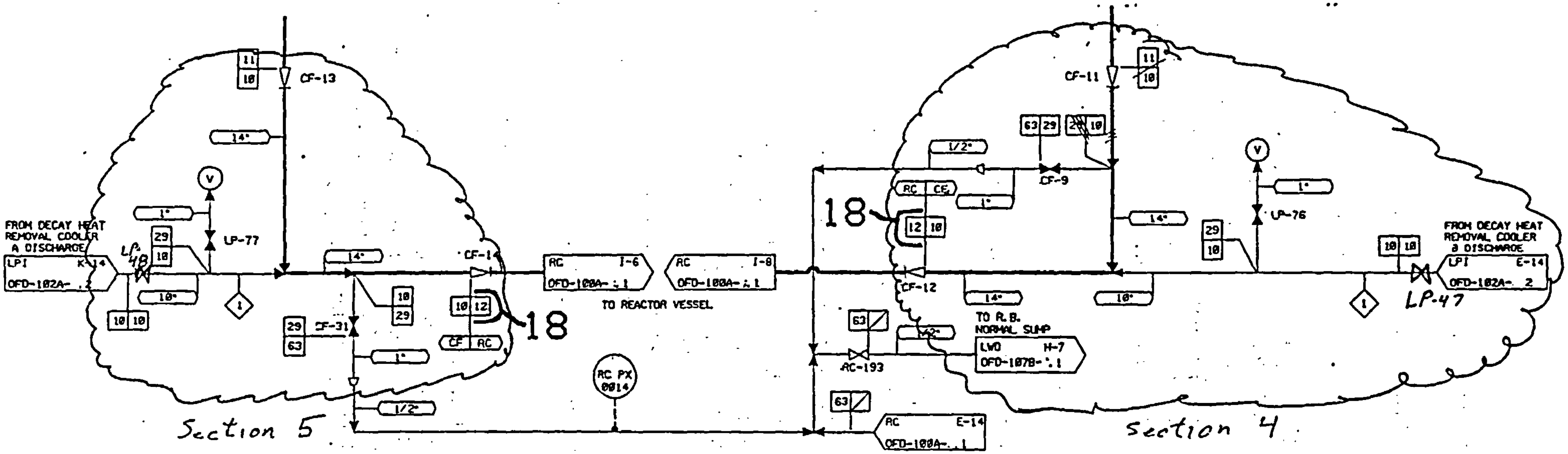
Attachment # 2
Request For Relief 04-ON-012

OFD 100A-x.2

DESIGN PARAMETERS

LINE NO.	DUKE CLASS	DESIGN PRESSURE	DESIGN TEMP.	MATERIAL	PIPE SPEC. NO.	PIPE SCH. NO.	ISI CLASS
1	A	2500 PSIG	670°F	B & W	B & W	NOTE 10	A
2	A	2500 PSIG	670°F	B & W	B & W	SCH 140	A
3	A	2500 PSIG	650°F	B & W	B & W	SCH 160	A
5	A	2500 PSIG	670°F	SS	1501.1	SCH 160	A
6	AC	2500 PSIG	670°F	SS	1501.3	SCH 160	A
7	BC	2500 PSIG	670°F	SS	1501.3	SCH 160	B
10	AC	2500 PSIG	650°F	SS	1501.3	SCH 160	A
17	B	2500 PSIG	300°F	SS	1501.2	NOTE 3	A
19	BC	2500 PSIG	300°F	SS	1501.3	NOTE 3	A
21	C	700 PSIG	500°F	SS	601.3 2701.1	NOTE 2 NOTE 2	C
27	B	2500 PSIG	300°F	SS	1501.2	SCH 160	B
40	B	2500 PSIG	300°F	SS	2701.1	NOTE 2	A





DESIGN PARAMETERS

18

LINE NO.	DUKE CLASS	DESIGN PRESSURE	DESIGN TEMP.	MATERIAL	PIPE SPEC. NO.	PIPE SCH. NO.	ISI CLASS
9	B	700 PSIG	300°F	SS	601.2	NOTE 3	B
10	A	2500 PSIG	300°F	SS	1501.1	NOTE 2	A
11	B	2500 PSIG	300°F	SS	1501.2	NOTE 2	B
12	A	2500 PSIG	650°F	SS	1501.1	NOTE 2	A
18	E	100 PSIG	200°F	CS	150.4	NOTE 7	-
20	E	150 PSIG	300°F	SS	151.4	NOTE 4	-
24	F	100 PSIG	200°F	CS	150.4	SCH. 160	B
29	AC	2500 PSIG	300°F	SS	1501.3	NOTE 2	A
30	BC	2500 PSIG	300°F	SS	1501.3	NOTE 2	B
31	BC	700 PSIG	300°F	SS	601.3	NOTE 3	B
45	E	700 PSIG	300°F	SS	601.4	NOTE 3	-
55	BC	700 PSIG	300°F	SS	601.3	SCH. 80S	B
57	BC	700 PSIG	300°F	SS	601.3	SCH. 40	B
61	E	700 PSIG	300°F	SS	601.4	SCH. 80S	-
63	E	2500 PSIG	650°F	SS	2701.2*	-	-
76	E	700 PSIG	300°F	SS	2701.2	NOTE 12	-
87	BC	700 PSIG	300°F	SS	601.3	SCH. 160	B
		NOTE 5					
88	BC	700 PSIG	300°F	SS	601.3	SCH. 80S	B
		NOTE 5					
108	E	2500 PSIG	300°F	SS	1501.4	NOTE 2	-

Attachment # 3
Request For Relief 04-ON-012
OFD 102A-x.3

The following information was copied from the latest revision of the Oconee UFSAR.

3.1.1 Criterion 1 - Quality Standards (Category A)

3.1.1.1 Oconee QA-1 Program

To meet the requirements of 10CFR50 Appendix B, Oconee has defined its QA-1 program. The QA-1 program shall be applied to the "essential systems and components" listed above. The scope of these systems and components is provided in greater detail below. The QA-1 program shall also be applied to the Reactor Protective System, and shall be applied to any systems and components committed to the NRC as being classified as QA-1 per any correspondence subsequent to the orginal QA-1 licensing basis.

Therefore, the general criteria used to determine if a SSC is QA-1 is divided into two categories:

> First category - provides general QA-1 criteria based on the original licensing basis of ONS, and
>
> Second Category - provides general criteria for SSCs that were added to the QA-1 licensing basis after issuance of the original operating licenses for ONS.

First Category, Original Oconee QA-1 Licensing Basis

This first category includes the integrity of SSCs essential to prevention and mitigation of the Large Break LOCA coincident with loss of offsite power for the following five SSCs: 1) Reactor Coolant System, 2) Reactor Vessel Internals, 3) Reactor Building, 4) Engineered Safeguards System, and 5) Emergency Electric Power Sources. In addition, 6) Reactor Protective System, another system not addressed in FSAR Section 3.1.1, was interpreted to be included in the QA-1 scope, even though not listed.

Clarification regarding the six SSCs identified above is provided below.

1. Reactor Coolant System

   From a quality assurance perspective, the Reactor Coolant System consists of all connecting piping, valve bodies, pump casings, heat exchangers, or vessels out to and including the first isolation valve. The integrity of the pressure boundary of the connecting piping, valve bodies, pump casings, heat exchangers, or vessels is the function which determines applicability of the quality assurance program.

3.2.2.1 System Classifications

Class I

This class is limited to the Reactor Coolant System (RCS) and Reactor Coolant Branch lines, as described herein. The Reactor Coolant Branch lines include connecting piping out to and including the first isolation valve. This section of piping is Class I in material, fabrication, erection, and supports and restraints. A Class I analysis of the piping to the first isolation valve has been completed for the following systems:

1. High Pressure Injection (Emergency Injection)
2. High Pressure Injection (Normal Injection)
3. High Pressure Injection (Letdown)
4. Low Pressure Injection (Decay Heat Removal Drop-line)
5. Low Pressure Injection (Core Flood)
6. Reactor Coolant Drain Lines
7. Pressurizer Spray
8. Pressurizer Relief Valve Nozzles

Modifications that affect the Reactor Coolant System and the Class I portion of the branch lines must demonstrate that the impact on the Class I piping is acceptable. The impact may be assessed by performing a Class I analysis or by other conservative techniques to assure Class I allowable limits are not exceeded. Isolation valves can be either stop, relief, or check valves. Piping 1 inch and less is excluded from Class I.

Attachment # 4
Request For Relief 04-ON-012

Attachment # 1
Request For Relief 04-ON-012

OFD 102A-x.1

Attachment # 2
Request For Relief 04-ON-012

OFD 100A-x.2

Attachment # 3
Request For Relief 04-ON-012

OFD 102A-x.3