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December 6, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414

Dear Mr. Denton:

On November 3 and 4, 1982, representatives from Duke Power Company, the NRC Mechanical Engineering Branch (MEB), and Pacific Northwest Labs (MEB consultant), met in Charlotte to discuss the open items which remained from previous meetings. Attached is a list of attendees and a meeting summary.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

ROS/php
Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
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Atlanta, Georgia 30303

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Catawba Nuclear Station

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B001 Aperture Card Dist
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cc: Mr. Jesse L. Riley
Carolina Environmental Study Group
854 Henley Place
Charlotte, North Carolina 28207

Mr. Henry A. Presler, Chairman
Charlotte-Mecklenburg Environmental Coalition
943 Henley Place
Charlotte, North Carolina 28207

Mr. Gordon Beeman
Pacific Northwest Laboratories
Richland, Washington 99352

November 3, 1982

Meeting: Duke Power with NRC-MEB

Attendance

M. L. Childers	Duke-SRAL/Licensing
R. O. Sharpe	Duke-Nuclear Production/Licensing
R. R. Weidler	Duke-Design Engineering/M&N
L. B. Castles	Duke-Design Engineering/M&N
J. N. Underwood	Duke-Design Engineering/M&N
W. L. Culpepper	Duke-Design Engineering/M&N
C. L. Ray, Jr.	Duke-Design Engineering/M&N
D. Terao	NRC/MEB
Grant Marr	PNL
Gordon Beeman	PNL
K. N. Jabbour	NRC/DL/LB#4
R. W. Bonsall	Duke-Design Engineering/M&N
D. L. Caldwell	Duke-Design Engineering/M&N

November 4, 1982

Site Visit - Mechanical Engineering Branch

Attendance

M. L. Childers	Duke-Design Engineering/Licensing
R. R. Weidler	Duke-Design Engineering/M&N
L. B. Castles	Duke-Design Engineering/M&N
W. L. Culpepper	Duke-Design Engineering/M&N
Gordon Beeman	PNL
Grant Marr	PNL

DUKE - NRC/MEB MEETING
November 3 and 4, 1982

The following items were discussed. Agenda item numbers from previous meetings are noted where appropriate.

1. Criteria for Jet Impingement and documentation of exceptions (Item 17).

   Resolution - Duke discussed their reasons for extending pipe whip criteria to include the effects of jet impingement. The NRC was in basic agreement and it was agreed that Duke would provide examples to demonstrate the validity of this approach. The proposed FSAR revision is attached. This item was closed.

2. Postulated Break Locations (Item 18).

   Resolution - The NRC reviewed Duke's proposed FSAR revision (attached). This item was closed.

3. Alternate Criteria for Reanalysis.

   Resolution - The NRC reviewed Duke's proposed FSAR revision (attached). This item was closed.

4. Terminal End Breaks (Item 19).

   Resolution - Duke provided a proposed FSAR revision (attached) that would assure that breaks were postulated in non-seismic piping and for interaction/effects with components. This item was closed.

5. Documentation of Limited Break Areas (Item 20).

   Resolution - Duke agreed to provide an FSAR revision (attached) that would address this concern. This item was closed.

6. Unrestrained Whipping Pipe Inside Containment (Item 22).

   Resolution - Duke agreed to provide an FSAR revision (attached) that would provide justification for isolated cases and would include system geometry, hanger location, hanger configuration, etc. This item was closed following the site visit.

7. Piping Vibration Test Program (Items 77 and 78)

   Resolution - Duke agreed to submit a response as soon as possible. This item remains open.

8. Interaction of Non-Seismic Piping Systems with Seismic Category 1 Systems.

   Resolution - See Item 4 above.

9. Damping Valves

   Resolution - This item will be resolved between Duke and the NRC's Structural Engineering Branch.

10. Functional Capability of Essential Systems (Item 86).

   Resolution - Duke agreed to provide an analysis based on the attached criteria.

11. NF Jurisdictional Boundaries and use of .78 Critical Buckling (Item 76)

   Resolution - Duke referred to the attached response. This item was closed following the site visit.

12. Response to IEB 79-02.

   Resolution - This item is under review by the NRC.

13 and 14. Inservice Testing

   Resolution - Duke agreed to submit a program description by April 1983.

15. Bellows Penetrations (Item 35)

   Resolution - The attached response was provided. This item was closed.

16. Computer Code Verification (Item 63)

   Resolution - Duke agreed to provide a response by January 1, 1983.

17. Postulated Break Locations

   Resolution - Duke agreed to provide a summary of postulated break locations in FSAR Section 3.6.2.5.

18. Classification of Systems (Items 117, 118, 119, 120, and 121).

   Resolution - Duke agreed to provide a response (attached).

<u>Summary of November 4, 1982 Site Visit</u>

<u>Unrestrained Pipe Whip</u>

NRC and Duke representatives field reviewed the two inside containment unrestrained pipe whip cases. Both postulated breaks occur on the 2 inch line from the regenerative heat exchanger to the letdown heat exchanger. NRC representatives concluded that there were no problems with these two break locations being unrestrained based upon as-erected piping configurations and targets.

Math Models

The NRC and Duke representatives reviewed two math model break postulation problems, NI-07 (loop 1D accumulator tank injection line into the cold leg) and NI-12 (loops 1A and 1D hot leg injection from the safety injection pump).

a) NI-07 - The field review included postulated break locations and pipe whip arresters, i.e., pipe whip restraints and the crane wall, for restraint function.

b) NI-12 - The field review included postulated break locations and pipe whip arresters, i.e., pipe whip restraints, for restraint function.

19. Intersystem LOCA (Item 96)

Resolution - Duke provided a draft copy of Technical Specification 3/4.4.7.2, Operational Leakage. This specification will be submitted as discussed in FSAR Chapter 16. A copy of this Technical Specification and the referenced Flow Diagrams are attached.

PART OF Item 1 ~~#17~~

CNS

3) Seismic loadings equivalent to the Operating Basis Earthquake (OBE) are used in the analysis of piping systems for the purpose of postulating break locations. Protective structures are designed to withstand the effects of the postulated piping failure in combination with loadings associated with the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) within the respective design load limits for the structures.

e) Consideration is given to the potential for a random single failure of an active component subsequent to the postulated pipe rupture. Where the postulated piping break is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system (i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping rupture), single failures of components in the other train or trains of that system only are not assumed; provided the system is designed to seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems.

f) In the event of a postulated break in the piping in one unit, safe reactor shutdown of the affected unit cannot preclude the capability for safe shutdown of the reactor of the unaffected unit(s).

g) Containment structural integrity is maintained by limiting the combination of break sizes and types to the design basis capability (i.e., temperature, pressure, and leakage rate) of the containment.

h) For those postulated breaks classified as a loss-of-reactor coolant, the design leak tightness of the containment fission product barrier shall be maintained.

i) The conditions within the control room or any other location where manual action is required to assure safe shutdown to the cold condition are such as to assure habitability and comply with the requirements of General Design Criterion 19.

j) A whipping pipe or jet is assumed not to cause failure of other pipes of equal or greater size and equal or greater thickness. Smaller and thinner pipes are assumed to encounter unacceptable damage upon impact. A whipping pipe or jet is considered capable of developing through-wall leakage cracks in equal or larger nominal pipe sizes with thinner wall thicknesses, except where experimental or analytical data for the expected range of impact energies demonstrate the capability to withstand the impact without failure.

Add below here.

If such exception is taken, the analytical technique or experimental data used will be documented in the FSAR.

k) Piping Breaks Within The LOCA Boundary (See Figure 3.6.2-1)

1) All LOCA breaks are allowed to damage any non-LOCA line except essential systems, and steam and feedwater lines.

Item 2 (Item # 18)

Terminal ends are considered as piping originating at structures or components (such as vessel and equipment nozzles and structural piping anchors) that act as rigid constraint to the piping thermal expansion. Typically, the anchors assumed for the piping code stress analysis would be terminal ends. The branch connection to the main run is one of the terminal ends of a branch run, except intersections of runs of comparable size and fixity which have a significant effect on the main run need not be considered terminal ends when the stress analysis model includes both the run and branch piping and the intersection is not rigidly constrained to the building structure.

a) Breaks in Duke Class A piping are postulated at the following locations (see Table 3.2.2-3 for class correlations):

1) The terminal ends of the pressurized portions of the run.

2) At intermediate locations selected by either one of the following methods:

i) At each weld location of potential high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges, and welded attachments, or

ii) At all intermediate locations between terminal ends where the following stress and fatigue limits are exceeded,

a) The maximum stress range shall not exceed 2.4 S_m except as noted below.

b) The maximum stress range between any two load sets (including the zero load set) shall be calculated by Eq. (10) in Paragraph NB-3653, ASME Code, Section III, for normal and upset plant conditions and an operating basis earthquake (OBE) event transient.

If the calculated maximum stress range of Eq. (10) exceeds the limit (2.4 S_m) but is not greater than 3 S_m, the limit of U < 0.1 shall be met.

If the calculated maximum stress range of Eq. (10) exceeds 3 S_m, the stress ranges calculated by both Eq. (12) and Eq. (13) in Paragraph NB-3653 shall not exceed 2.4 S_m or the limit of U < 0.1.

c) U shall not exceed 0.1

where:

S_n = primary-plus secondary stress-intensity range, as calculated from Equation (10) in Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

#2 (Item 18)

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
43/66W	2.614	16,292	42,587	AWBW
94/160W	2.942	16,292	47,931	FILW
69/99W	2.480	16,328	40,493	AWBW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
80/11/03. 15.25.40.

Duke Power Company

Catawba Unit #1

REV	BY	DATE	CHECKED	DATE	eds nuclear	JOB NO	CALC NO	PAGE
-	CLS	5/26/82	JRB	5-26-82		0093-101	CC-NI-01	1 OF 1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
52/93W	2.562	16,292	41,740	AWTT
55/97W	2.563	16,292	41,756	AWBW
71,71A/ 132R,133L	2.780	16,328	45,392	BELB
71A/133L	2.756	16,328	45,000	AWTT
73/135W	2.725	16,292	44,396	AWTT
75/139W	2.564	16,292	41,773	AWBW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
80/10/31. 17.24.28.

REV	BY	DATE	CHECKED	DATE			
					Duke Power Company		
					Catawba Unit #1		
0	OLS	5/26/82	JPH	5-2-82	eds nuclear	JOB NO 0093-101 / CALC NO CC-NI-02	PAGE 1 OF 1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
19/20W	2.770	20,000	55,400	FILW
24/26W	2.419	20,000	48,380	FILW
25/27W	2.466	20,000	49,320	FILW
25,26/ 27R, 28L	2.989	20,000	59,780	SELB
26/28W	2.474	20,000	49,480	FILW
42/55W	2.743	20,000	54,860	FILW
44/58R	2.699	20,000	53,980	SELB
44A/59W	2.539	20,000	50,780	FILW
38/93B	2.665	20,000	53,300	STEE

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
80/10/09. 18.47.03.

REV	BY	DATE	CHECKED	DATE		
0	CLE	5/26/82	[illegible]	5/26/82	Duke Power Company Catawba Unit #1 eds nuclear	JOB NO 0093-101 CALC NO [illegible] CC-NC-03 PAGE 1 OF 1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
13/34W	2.503	17,430	43,627	AWTT
A14/36W	2.510	17,430	43,749	AWBW
18/47W	2.866	17,430	49,954	AWTT
C07B/30L	2.791	17,430	48,647	BELB

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
81/04/27. 15.51.51

REV	BY	DATE	CHECKED	DATE			
					Duke Power Company		
					Catawba Unit #1		
0	GLS	5/19/82	JDJ	5-19-82	eds nuclear	JOB NO 0093-101 / CALC NO CC-NI-06	PAGE 1 OF 1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
AA2B/8W	2.504	17,430	43,645	FGBW
45/15W	2.406	20,000	48,120	AWTT
45A/16W	2.447	20,000	48,940	AWTT

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
81/04/27. 17.02.28.

REV	BY	DATE	CHECKED	DATE
0	CLS	5/19/82	JRD	5-19-82

Duke Power Company

Catawba Unit #1

eds nuclear

JOB NO 0093-101

CALC NO CC-NI-07

PAGE 1 OF 1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
10A/10W	2.949	16,820	49,602	AWBW
12B,13/ 14R,15L	2.480	16,820	41,714	BRED
13/15W	2.471	16,820	41,562	FILW
15A/17W	2.429	20,000	48,580	FILW
16/20W	2.482	20,000	49,640	FILW
17/21W	2.472	20,000	49,440	FILW
17AA/22W	2.427	20,000	48,540	FILW
148/135W	2.997	16,820	50,410	FILW
149D/141R	2.843	16,820	47,819	CRVP
149E/142L	2.986	16,820	50,225	CRVP

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
81/02/09. 17.29.36.

REV	BY	DATE	CHECKED	DATE			
					Duke Power Company		
					Catawba Unit #1		
						JOB NO 0093-101	PAGE 1
0	CLS	5/26/82	JPG	5-26-82	eds nuclear	CALC NO CC-NI-20	OF 1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
98/92W	2.466	16,820	41,478	FILW
98,99/ 92R,93L	2.479	16,820	41,697	SRED
181D/144W	2.784	20,000	55,680	FILW
185A/146W	2.879	20,000	57,580	FILW
187/148W	2.666	16,820	44,842	FILW
187,187A/ 148R,149L	2.663	16,820	44,792	SRED
189B/153W	2.876	16,820	48,375	AWBW
193/158W	2.616	16,820	44,001	AWBW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
81/01/30. 08.40.56.

Duke Power Company
Catawba Unit #1

eds nuclear

REV	BY	DATE	CHECKED	DATE
0	CLC	5-19-81	JRG	5-26-82

JOB NO 0093-101
CALC NO CC-NI-21

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
C01A/4W	2.435	19,800	48,213	AWBW
69A/87R	2.787	20,000	55,740	BELB
70/88L	2.810	20,000	56,200	BELB

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
82/02/04. 19.04.55.

REV	BY	DATE	CHECKED	DATE			
0	CLS	5/26/82	JPD	5-26-82	eds nuclear	JOB NO 0093-101 CALC NO CC-ND-01	PAGE 1 OF 1

Duke Power Company
Catawba Unit #1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
C31A/4W	2.706	19,800	53,579	AWBW
C71A/86R	2.531	20,000	50,620	CRVP
C71B/87L	2.581	20,000	51,620	CRVP

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
82/02/04. 19.05.13

REV	BY	DATE	CHECKED	DATE			
					Duke Power Company		
					Catawba Unit #1		
0	CLS	5/26/82	JRD	5-26-82	eds nuclear	JOB NO 0093-101 CALC NO CC-ND-02	PAGE 1 OF 1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
34A/74W	2.455	16,820	41,293	AWBW
36/76B	2.716	20,000	54,320	BTEE
50A/85W	2.824	16,862	47,618	AWTT
C15A/86W	2.527	16,862	42,610	AWTT
C15A/86R	2.432	16,862	41,008	BELB
C15B/87L	2.638	16,862	44,482	BELB
C14A/90W	2.625	16,820	44,153	AWBW
C14B/91WR	2.898	16,820	48,744	AWBW
44/94R	2.599	16,862	43,824	BELB
42/96L	2.692	16,862	45,393	BELB
C11A/98W	2.713	16,820	45,633	AWBW
C11B/99W	2.645	16,820	44,489	AWBW
C18A/107W	2.586	16,820	43,496	AWBW
C18B/108W	2.677	16,820	45,027	AWBW
66/113R	2.585	16,862	43,588	BELB

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
82/04/14. 07.02.18

Duke Power Company

Catawba Unit #1

REV	BY	DATE	CHECKED	DATE
0	CLS	5/17/82	DJT	5-17-82

eds nuclear

JOB NO 0093-101

CALC NO CC-NC-03

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
68/115L	2.573	16,862	43,386	BELB
C21A/117W	2.924	16,820	49,182	AWBW
C21B/118W	2.700	16,820	45,414	AWBW
C22A/121W	2.664	16,862	44,920	AWTT
C22A/121R	2.722	16,862	45,898	BELB
C22B/122L	2.550	16,862	42,998	BELB
C22B/122W	2.617	16,862	44,128	AWTT
73A/123W	2.992	16,862	50,451	AWTT

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
82/04/14. 07.02.18

REV	BY	DATE	CHECKED	DATE			
0	CLS	5/19/82	JPD	5-26-82	Duke Power Company Catawba Unit #1 eds nuclear	JOB NO 0093-101 CALC NO CC-NC-03	PAGE 2 OF 2

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
4/4R	2.447	16,840	41,207	BELB
7AA/9R	2.549	16,840	42,925	BELB
7BB/10L	2.489	16,840	41,915	BELB
7CC/11R	2.439	16,840	41,073	BELB
7DD/12L	2.506	16,840	42,201	BELB
8/14R	2.591	16,840	43,632	BELB
9/15L	2.614	16,840	44,020	BELB
16/28R	2.949	16,532	48,753	BELB
17/29L	2.944	16,532	48,670	BELB
20W/36W	2.653	16,520	43,828	AWTT
20C/39R	2.864	16,532	47,348	BELB
80/48K	2.854	16,292	46,497	CRVP
81/49L	2.871	16,292	46,774	CRVP
81C/50R	2.832	16,328	46,241	BELB
81E/52L	2.810	16,328	45,882	BELB
81D/51W	2.917	16,328	47,629	AWBW
83/56W	2.403	16,292	39,150	AWBW
66/114W	2.981	16,820	50,140	FILW
65/115W	2.776	20,000	55,520	FILW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
82/03/26. 17.03.29.

REV	BY	DATE	CHECKED	DATE			
					Duke Power Company		
					Catawba Unit #1		
0	TW	5/25/82	CLS	5/26/82	eds nuclear	JOB NO 0093-101 CALC NO CC-NC-07	PAGE 1 OF 2

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
34B/123R	2.517	16,820	42,336	CRVP
34C/124L	2.532	16,820	42,588	CRVP
45B/144L	2.460	16,800	41,328	CRVP
46A/146R	2.644	16,800	44,419	CRVP
46B/147L	2.663	16,800	44,738	CRVP
47A/149R	2.627	16,800	44,134	CRVP
47B/150L	2.472	16,800	41,530	CRVP
50/156R	2.545	16,800	42,756	CRVP
71/183W	2.794	20,000	55,880	FILW
72/184W	2.773	20,000	55,460	FILW
117/192W	2.552	16,292	41,577	AWBW
142/226R	2.989	16,800	50,215	BRED
145/230L	2.826	16,800	47,476	BRED

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
82/03/26. 17.03.29.

REV	BY	DATE	CHECKED	DATE		
					Duke Power Company	
					Catawba Unit #1	
0	TW	5/15/82	CLS	5/26/82	eds nuclear — JOB NO 0093-101; CALC NO CC-NC-07	PAGE 2 OF 2

DCP/	S_n/S_m	S_m	S_n	Comp. type
31/32	2.975	16,328	48,576	BELB
32M	2.623	16,328	42,828	BELB
32N	2.582	16,328	42,159	BELB
32O	2.748	16,328	44,869	BELB
32P	2.834	16,328	46,274	BELB
33	2.877	16,292	46,872	CRVP
34	2.770	16,292	45,129	CRVP
35	2.555	16,328	41,718	BELB
36	2.585	16,328	42,208	BELB
62	2.560	16,328	41,800	BELB
63	2.888	16,840	48,634	BELB
69	2.980	16,840	50,183	BELB
75A	2.842	16,520	46,950	AWTT
78B	2.588	16,520	42,754	AWTT
101Z	2.823	16,800	47,426	AWBW
101Y	2.414	16,800	40,555	AWBW
101A	2.854	16,800	47,947	CRVP
101C	2.723	16,800	45,746	CRVP
101I	2.700	16,800	45,360	AWBW
117	2.885	20,000	57,700	FILW
112	2.554	16,292	41,610	AWBW
118	2.774	20,000	55,480	FILW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
82/03/23. 17.47.35.

REV	BY	DATE	CHECKED	DATE			
					Duke Power Company		
					Catawba Unit #1		
					eds nuclear	JOB NO 0093-101	PAGE 1
0	TL	5/25/82	CLS	5/26/82		CALC NO CC-NC-08	OF 2

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
126	2.766	20,000	55,320	FILW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
82/03/23. 17.47.35.

Duke Power Company

Catawba Unit #1

REV	BY	DATE	CHECKED	DATE		JOB NO	CALC NO	PAGE
0	TL	5/25/82	CLS	5/26/82	eds nuclear	0093-101	CC-NC-08	2 OF 2

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
83,83/ 50R,51L	2.985	20,000	59,700	BELB
107,108/ 81R,82L	2.624	18,040	47,337	BRED
108/82W	2.422	16,830	40,762	AWBW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
80/09/29. 16.55.38

					Duke Power Company
					Catawba Unit #1
0	CLS	5/19/82	[illegible]	5-1-82	eds nuclear — JOB NO 0093-101 — CALC NO CC-NC-12 — PAGE 1 OF 1
REV	BY	DATE	CHECKED	DATE	

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
88/123W	2.580	20,000	51,600	FILW
89/124W	2.870	20,000	57,400	FILW
94/130W	2.508	20,000	50,160	FILW
95/131B	2.750	20,000	55,000	STEE
104/137W	2.602	20,000	52,040	FILW
110/148W	2.456	20,000	49,120	FILW
110,111/ 148R,149L	2.877	20,000	57,540	SELB
111/149W	2.442	20,000	48,840	FILW
118,118A/ 157R,158L	2.751	20,000	55,020	SELB
118B/159W	2.404	20,000	48,080	FILW
118B,120/ 159R,160L	2.888	20,000	57,760	SELB
120/160W	2.534	20,000	50,680	FILW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
82/01/11. 17.13.12.

REV	BY	DATE	CHECKED	DATE			
					Duke Power Company		
					Catawba Unit #1		
0	CLC	5/17/82	JRD	5-17-82	eds nuclear	JOB NO 0093-101 / CALC NO [illegible] CC-NC-09	PAGE 1 OF 1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
S4/4W	2.511	20,000	50,220	FILW
S6/8W	2.681	20,000	53,620	FILW
S7/9W	2.734	20,000	54,680	FILW
S20/25W	2.467	20,000	49,340	FILW
S31/39W	2.531	20,000	50,620	FILW
S35/44W	2.835	20,000	56,700	FILW
S36/45W	2.594	20,000	51,880	FILW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
81/08/13. 20.06.49.

REV	BY	DATE	CHECKED	DATE			
					Duke Power Company		
					Catawba Unit #1		
0	CLS	5/19/82	JPT	5-19-82	eds nuclear	JOB NO 0093-101 / CALC NO CC-NV-10	PAGE 1 OF 1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
81C/7W	2.662	20,000	53,240	FILW
77/10W	2.778	20,000	55,560	FILW
76/11W	2.847	20,000	56,940	FILW
74/13W	2.999	20,000	59,980	FILW
73/15W	2.810	20,000	56,200	FILW
72/16W	2.757	20,000	55,140	FILW
70,71/ 17R,18L	2.590	20,000	51,800	SELB
69,68/ 19R,20L	2.668	20,000	53,360	SELB
65/24W	2.645	20,000	52,900	FILW
63L/25W	2.597	20,000	51,940	FILW
63/37B	2.402	20,000	48,040	STEE
62/38W	2.502	20,000	50,040	FILW
61,60/ 39R,40L	2.755	20,000	55,100	SELB
59/41W	2.430	20,000	48,600	FILW
58/42W	2.815	20,000	56,300	FILW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
81/08/13. 18.48.05.

REV	BY	DATE	CHECKED	DATE			
0	QLS	5/14/82	JRP	5-14-82	Duke Power Company, Catawba Unit #1, eds nuclear	JOB NO 0093-101; CALC NO CC-NV-11	PAGE 1 OF 1

DCP/SOP	S_n/S_m	S_m	S_n	Comp. type
6,7/9R,10L	2.454	20,000	49,080	SELB
12/16W	2.845	20,000	56,900	FILW
13/17W	2.617	20,000	52,340	FILW
14,15/ 18R,19L	2.523	20,000	50,460	SELB
16,17/ 20R,21L	2.968	20,000	59,360	SELB
17/21W	2.531	20,000	50,620	FILW
21/28W	2.541	20,000	50,820	FILW
31/40W	2.422	20,000	48,440	FILW
16/20W	2.412	20,000	48,240	FILW

S_n-Equation 10 stress

S_m-allowable stress

S_n and S_m stresses are given in psi.

REFERENCE:
Computer Seq. #
81/08/14. 02.07.58.

REV	BY	DATE	CHECKED	DATE	
0	CLS	5/26/82	JPB	5-26-82	Duke Power Company; Catawba Unit #1; eds nuclear; JOB NO 0093-101; CALC NO CC-NV-12; PAGE 1 OF 1

ITEM 3

S_m = allowable design stress-intensity value, as defined in Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

U = the cumulative usage factor, as calculated in accordance with Subarticle NB-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

3) If there are no intermediate locations where S exceeds 2.4 S_m or U exceeds 0.1, a minimum of two locations are chosen based upon highest stress. Intermediate breaks are not postulated in sections of straight pipe where there are no pipe fittings, flanges, valves or welded attachments.

4) As a result of piping reanalysis which follows the completion of the original problem interaction analysis, the highest stress locations may be shifted; however, the initially determined intermediate break locations will not be changed unless one of the following conditions exists.

i) Maximum stress ranges or cumulative usage factors exceed the threshold levels in 2)ii) a), b) or c) above.

ii) A change is required in pipe parameters, such as major differences in pipe size or wall thickness.

iii) Breaks at the new highest stress locations are determined to have significantly higher stress values from the original locations, and result in consequences to safety related systems requiring additional safety protection.

b) Breaks in Duke Class B and C piping are postulated at the following locations (See Table 3.2.2-3 for class correlations):

1) The terminal ends of the pressurized portions of the run.

2) At intermediate locations selected by either one of the following methods:

i) at each weld location of potential high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges and welded attachments, or

ii) at all locations where the stress, S, exceeds 0.8 $(1.2S_h + S_A)$,

where:

S = stresses under the combination of loadings associated with the normal and upset plant condition loadings and an OBE event, as calculated from the sum of equations

(9) and (10) in Subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

S_h = basic material allowable stress at maximum (hot) temperature from the allowable stress tables in Appendix I of the ASME Boiler and Pressure Vessel Code, Section III.

S_A = allowable stress range for expansion stresses, as defined in Subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

3) If there are not at least two intermediate locations where S exceeds 0.8 (1.2 S_h + S_A), a minimum of two separated locations are chosen based upon highest stress. Intermediate breaks are not postulated in sections of straight pipe where there are no pipe fittings, flanges, valves, or welded attachments. The pattern of postulated intermediate break locations is determined separately for the normal plant condition load combination and for that upset plant condition which has the highest stress.

4) As a result of piping reanalysis which follows the completion of the original problem interaction analysis, the highest stress locations may be shifted; however, the initially determined intermediate break locations will not be changed unless one of the following conditions exists.

i) Maximum stress ranges exceed the threshold level of 0.8 (1.2S_h + S_A).

ii) A change is required in pipe parameters, such as major differences in pipe size or wall thickness.

iii) Breaks at the new highest stress locations are determined to have significantly higher stress values from the original locations, and result in consequences to safety related systems requiring additional safety protection.

c) Breaks in Duke Class E, F, G and H piping are postulated at the following locations (See Table 3.2.2-3 for class correlations)

1) The terminal ends of the pressurized portions of the run.

2) At intermediate locations selected by one of the following methods:

i) For Class E, F, G, and H Piping:

At each weld location of potential high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges, and welded attachments; or

ii) For Class F Piping:

At all locations where the stress, S, Exceeds 0.8 (1.2 S_h + S_A), where:

S = stresses under the combination of loadings associated with the normal and upset plant condition loadings and an OBE event, as calculated from the sum of equations (9) and (10) in subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

S_h = basic material allowable stress at maximum (hot) temperature, per ANSI B31.1.0.

S_A = allowable stress range for expansion stresses, per ANSI B31.1.0.

3) For Class F Piping:

If there are not at least two intermediate locations where S exceeds 0.8 (1.2 S_h + S_A), a minimum of two separate locations are chosen based upon highest stress. Intermediate breaks are not postulated in sections of straight pipe where there are no pipe fittings, flanges, valves or welded attachments.

4) As a result of piping reanalysis which follows the completion of the original problem interaction analysis, the highest stress locations may be shifted; however, the initially determined intermediate break locations will not be changed unless one of the following conditions exists.

i) Maximum stress ranges exceed the threshold level of 0.8 (1.2S_h + S_A).

ii) A change is required in pipe parameters, such as major differences in pipe size or wall thickness.

iii) Breaks at the new highest stress locations are determined to have significantly higher stress values from the original locations, and result in consequences to safety related systems requiring additional safety protection.

3.6.2.1.2.2 Postulated Piping Break Locations For Moderate-Energy Piping Systems

Systems identified as containing moderate-energy piping are examined by detailed drawing review for postulated through-wall cracks as defined below. Systems analyzed for consequences of postulated piping cracks are listed in Table 3.6.1-2.

To assure protection of safety-related structures, systems or components,

break locations is determined separately for the normal plant condition load combination and for that upset plant condition which has the highest stress.

c) breaks in Duke Class E, F, G and H piping are postulated at the following locations (See Table 3.2.2-3 for class correlations)

1) The terminal ends of the pressurized portions of the run.

2) At intermediate locations selected by one of the following methods:

i) For Class E, F, G, and H Piping:

At each intermediate weld location of potential high stress or fatigue. ~~such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges, and welded attachments; or~~

ii) For Class F Piping:

At all locations where the stress, S, Exceeds 0.8 (1.2 S_h + S_A), where:

S = stresses under the combination of loadings associated with the normal and upset plant condition loadings and an OBE event, as calculated from the sum of equations (9) and (10) in subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

S_h = basic material allowable stress at maximum (hot) temperature, per ANSI B31.1.0.

S_A = allowable stress range for expansion stresses, per ANSI B31.1.0.

3) For Class F Piping:

If there are not at least two intermediate locations where S exceeds 0.8 (1.2 S_h + S_A), a minimum of two separate locations are chosen based upon highest stress. Intermediate breaks are not postulated in sections of straight pipe where there are no pipe fittings, flanges, valves or welded attachments.

3.6.2.1.2.2 Postulated Piping Break Locations For Moderate-Energy Piping Systems

Systems identified as containing moderate-energy piping are examined by detailed drawing review for postulated through-wall cracks as defined below. Systems analyzed for consequences of postulated piping cracks are listed in Table 3.6.1-2.

Item # 3 5 #20

CNS

a) Cracks in Duke Class B, C and F piping are postulated at the following locations:

1) The terminal ends of the pressurized portions of the run.

2) At intermediate individual locations of potential high stress or fatigue (e.g. pipe fittings, valves, flanges and welded attachments) that result in the maximum effects from fluid spraying, flooding or environmental conditions except in portions of piping where the maximum stress range is less than 0.4 ($1.2 S_b + S_A$) as defined in items b)2)ii) and c)2)ii) of Section 3.6.2.1.2.1.

b) Cracks in Duke Class E, G and H piping are postulated at the following locations:

1) The terminal ends of the pressurized portions of the run.

2) At intermediate individual locations of potential high stress or fatigue (e.g. pipe fittings, valves, flanges and welded attachements) that result in the maximum effects from fluid spraying, flooding or environmental conditions.

3.6.2.1.2.3 Postulated Break Type, Size, and Orientation

a) Circumferential Pipe Breaks

The following circumferential breaks are postulated in high-energy fluid system piping at the locations specified in Section 3.6.2.1.2.1.

1) Circumferential breaks are postulated in fluid system piping and branch runs exceeding a nominal pipe size of 1 inch, except where the maximum stress range exceeds the limits of Section 3.6.2.1.2.1, items b) and c)2)ii) but the circumferential stress range is at least 1.5 times the axial stress range.

2) Where break locations are selected in fittings in accordance with Section 3.6.2.1.2.1 without the benefit of detailed stress calculations, breaks are postulated at each weld, in piping greather than one inch NPS, to the fitting, valve, or welded attachment. Alternately, a single break location at the section of maximum stress range may be selected as determined by detailed stress analyses or tests on a pipe fitting.

3) Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints. No limited break areas will be used for compartment pressurization calculations. If limited break areas are used for jet impingement reviews, the basis will be the installation of rigid rupture restraints; and any such limited break areas along with their locations will be documented in the FSAR.

Added

Item 6 #22

4) The dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

5) ~~Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction. For the purpose of analysis, breaks are assumed to reach full opening size within one (1) millisecond after break initiation.~~

Replace with words on attached page.

b) Longitudinal Pipe Breaks

The following longitudinal breaks are postulated in high-energy fluid system piping at the locations specified in Section 3.6.2.1.2.1.

1) Longitudinal breaks in fluid system piping and branch runs are postulated in nominal pipe sizes 4 inches and larger, except where the maximum stress range exceeds the limits of Section 3.6.2.1.2.1, items b) and c)2)ii) but the axial stress range is at least 1.5 times the circumferential stress range.

2) Longitudinal breaks are not postulated at:

a) Terminal ends provided the piping at the terminal ends contains no longitudinal pipe welds.

b) At intermediate locations where the criterion for a minimum number of break locations must be satisfied.

3) Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are oriented (but not concurrently) at two diametrically-opposed points on the piping circumference such that the jet reaction causes out-of-plane bending of the piping configuration. Alternately, a single split may be assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

4) The dynamic force of the fluid jet discharge is based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

Item 6

Unrestrained Whipping Pipe

Postulated pipe whip for target review will be defined by engineering judgement based on piping geometry, jet thrust direction, break location analysis type, and hanger location and type. When further confirmation is required, postulated piping breaks and targets are field reviewed after the drawing based analysis has been completed. For the purposes of analysis, breaks are assumed to reach full opening size in one millisecond after break initiation.

#10

NRC MECHANICAL ENGINEERING BRANCH QUESTION 86

3.9.3.1, pages 3.9-40 to 43

This section does not address the criteria used to assure the functional capability of essential systems when they are subjected to loads in excess of those for which Service Limit B Limits are specified. By essential systems are meant those ASME Class 1, 2 and 3 and any other piping systems which are necessary to shut down the plant following, or to mitigate the consequences of an accident. Provide such criteria.

#10

RESPONSE TO QUESTION 86

On Catawba Nuclear Station Duke will apply the following criteria to essential piping in order to assure that functional capability is maintained when subjected to loads in excess to those for which Upset Limits are specified.

1. Reference 1 will be applied to determine the acceptance of functional capability for ASME Class 1 piping and fittings.
2. ASME Class 2 & 3 piping and fittings with the exception of stainless steel elbows will be accepted as meeting functional capability when the following equation is met:

$$\frac{P_{max} D_o}{4t_n} + \frac{0.75i\,(M_A + 1.875M_B)}{Z} \leq 1.8\ S_h \qquad (1)$$

3. ASME Class 2 & 3 piping and fittings that do not meet equation 1 will be accepted as meeting functional capability when $D_o/t_n \leq 50$ and the following equation is met.

$$\frac{B_1 P_{max} D_o}{2t_n} + \frac{B_2\,(M_A + 1.875M_B)}{Z} \leq 2.25\ S_h \text{ But not greater than } 1.8\ S_y \qquad (2)$$

Eq. 9 ASME Code NC-3600 W'81 addenda

4. ASME Class 2 & 3 stainless steel elbows will be accepted as meeting functional capability when $D_o/t_n \leq 50$ and the following equation is met.

$$\frac{B_1 P_{max} D_o}{2t_n} + \frac{B_2\,(M_A + 1.875M_B)}{Z} \leq 1.8\ S_y \qquad (3)$$

Eq. as developed by Westinghouse for Comanche Peak Steam Electric Station.

5. Reference 1 will be applied to determine the acceptance of functional capability for ASME Class 2 & 3 stainless steel elbows and those that do not meet equation 1 when $50 < D_o/t_n < 100$.

The criteria as shown in 1 through 5 will be included in the FSAR.

Reference

1. NEDO-21985 "Functional Capability Criteria for Essential Mark II Piping," General Electric Company San Jose, CA 95125.

#11

Item 76 - 3.9.1.4.7, page 3.9-24

Provide your interpretation of jurisdictional boundaries as they pertain to NF supports. Justify your position.

Response:

Subsection NF of the ASME Boiler and Pressure Vessel Code defines requirements for structural elements for both component supports and piping supports. In reviewing application of this subsection to structural elements of pipe supports Duke Power Company has defined jurisdictional boundaries to be within the scope of subsection NF when the ASME code provisions are clearly applicable and result in rational structural requirements for design and construction. Many aspects of structural steel design have a long and established history of adequate and reasonable application by structural engineers. For this reason, Duke Power Company uses applicable codes and standards other than ASME standards for the majority of structrual steel design associated with pipe supports. Boundaries between items designed and fabricated to Subsection NF and those designed and fabricated to codes and standards applicable to building structural items are clearly designated on all drawings released for construction. Guidelines for defining these standard boundaries are provided in Duke specification CNS-1206.00-04-0001 Design Specification For Nuclear Safety Related Component Supports (QA Condition 1). A copy of this document has been previously provided to NRC representatives.

As justification for our position we have been asked by the staff to show approximate equivalence across jurisdictional boundaries for several specific items.

1. Presence of clearly defined structural requirements on both sides of the jurisdictional boundary.

   The Design Specification clearly defines design criteria to be followed on both sides of the NF boundary. For both normal and upset conditions, structural steel is designed to normal allowable limits per AISC requirements, regardless of boundary. For the faulted condition, additional conservatism over and above that required by Appendix F is provided by limiting steel stress to 1.33 x AISC allowables within the NF boundary and to 1.5 x AISC allowables outside the NF boundary. This is consistent with allowable values used for building structural steel. Limiting stresses under faulted conditions to the elastic stress range as provided by these measures provides that the structure performs as predicted by elastic analysis procedures that are well documented and provides a degree of conservatism due to additional capacity in the inelastic range of materials which is not utilized.

2. Consideration of the relevance of buckling as a factor in support design.

   For normal and upset conditions, use of AISC allowable stresses results in appropriate safety factors against buckling for these conditions.

   Use of the 1.5 factor on AISC allowable stresses for the faulted condition subsequently reduces the safety factor against buckling but maintains adequate margins for this condition. In the region of "short column" buckling (for fy=36, $\frac{kl}{r} < 126$)formulas presented by AISC and use of the 1.5 factor allows values of allowable stress to approach 0.9 x yield strength. Critical buckling as presented by the Euler approach is not the predominant failure mode in this region. For KL/r values above the critical buckling coefficient, AISC limits stresses to 12/23 x critical buckling stress. Use of the 1.5 for faulted loading increases this limit to 18/23 x critical buckling stress. This value is slightly higher than the 2/3 value proposed by the staff.

In actuality, buckling strength is seldom the controling factor in structural steel for pipe support design. Bending and shear forces are the predominant limiting stress considerations. Coupling these limits with deflection limitations imposed on the supports, required to make the supports relatively stiff compared to the piping for stress analysis validation, removes most consideration of buckling from the analytical approach to pipe support design. Predomanance of tube steel in support design, primarily due to superior properties for torsional resistance, provides relatively low L/r ratios for design, further obviating the need for a stringent review of buckling as a critical factor. However, changing the design specification to account for a 2/3 factor on buckling rather than 18/23 would require individual review of each of more than 20,000 calculations already completed for Nuclear Safety Related pipe supports at the Catawba Station. In short, we feel the slight difference in buckling limits has little, if any, relevance in support design and the methods and allowable stresses used for Catawba present a safe and adequate situation.

3. Use of initial overdesign as an added conservatism.

Beginning in December 1979, new support designs initiated for Catawba Nuclear Station were conservatively designed by incorporation of an additional 25% of piping loads supplied into design loads for the support. Rational for this conservatism included reduction of changes due to revised loadings by reanalysis and increased margin available when field conditions would not allow all elements of the design drawing to be fabricated per drawing requirements. Although such changes have resulted in encrouchment into the margin in many cases, for an even greater number of cases, the margin is now increased as piping loads have decreased. Coupled with the fact that the full capacity of a member is seldom used when structural engineers provide a design which is limited only by a maximum criteria, pipe support designs for the Catawba Station are certainly conservative and provide high confidence that adequate structures for pipe supports are produced.

4. Materials control and mill test certification reports.

Materials purchased for Catawba Nuclear Station are classified on site as either safety related or non-safety related. Materials used within the NF jurisdictional boundary which are not "bulk stock" items, such as component standard supports are purchased from an authorized supplier per Subsection NA-3700 and NCA-3800. Certified mill test reports in accordance with NF-2130(a) are obtained showing compliance with Section III Subsection NF requirements for Class I materials. Traceability of component standard supports are maintained for Class A supports. Bulk stock items, such as miscellaneous steel are received on site with certified mill test reports. Tube steel inside the NF boundary is purchased to requirements of Code Case N-71-8.

Design drawings define materials specifications inside the NF jurisdictional boundary as SA/A showing that the appropriate ASTM specification and ASME specification are equivalent. Tube steel furnished per Design drawings reference Code Case N-71-8. Material furnished on Design drawings outside the NF boundary reference the appropriate ASTM specification. Both material inside and outside the NF boundary for structural items other than component standard supports are furnished from Nuclear Safety Related field bulk stock and are equivalent as certified mill test reports are furnished as previously mentioned. Receipt of CMTR's are part of initial receiving inspection procedures.

5. Weld inspection and NDE requirements.

   Weld inspection inside the NF boundary is performed per Duke Power Company QA requirements for ASME Code work and meets requirements of Subsection NF, Article NF-5000. An Authorized Nuclear Inspector, independent of Duke Power Company, reviews each support package prior to fabrication. The ANI has opportunity to add inspection hold points and additional NDE requirements as appropriate.

   All welders utilized for supports on ASME Code piping are qualified for both requirements of AWS D1.1 and ASME Section IX.

   Predominant weld types on welds within and outside the NF boundary are fillet and partial penetration welds. Visual inspection is performed by qualified QC inspectors for all welds with other NDE requirements specified where appropriate.

6. Duke Power Company control of design and construction and resultant "reasonableness" of job.

   Duke Power Company designs, fabricates and erects the supports for Catawba Nuclear Station. This provides unique advantages in the enforcement of requirements both within the NF jurisdictional boundary and outside that boundary. The intent of specification requirements is maintained throughout the design and construction process through direct communication between designer, fabricator and erector. This close interface is not always possible when design, fabrication and erection are contracted to different companies.

Item 35 - Table 3.6.1-3, page 3, SAR Section 3.6.2.4

Provide an example of the analysis conducted to assure that Duke penetration designs are acceptable.

Response

Duke presented an example analysis to determine torsional rotations. This analysis was acceptable to the NRC. The manufacturer of the bellows joint of the penetration assemblies has determined that the maximum resulting torsional rotations for the Main Steam and Feedwater penetrations are acceptable as follows:

Type I Main Steam System

Condition	Torsional Rotation
Normal	.00342 radians
Upset	.00369 radians
Faulted	.00374 radians

Type II Feedwater System

Condition	Torsional Rotation
Normal	.00028 radians
Upset	.00120 radians
Faulted	.00222 radians

117. Table 3.2.1-2
Waste Evaporator Feed Tank, RCP Motor Oil Drain Tank, Vent. Unit Cond. Drain Tank
Identify the OBE and SSE seismic requirement for these components.

Response: FSAR Table 3.2.1-2 (Page 1) has been revised to indicate the seismic requirements for these components.

#117

Table 3.2.1-2 (Page 1)

Summary of Criteria - Equipment

Equipment	Scope	Quality Assurance Required	(Note 3) Category	Code	Location	(Note 2) Rad. Source	Seismic OBE	Seismic SSE	Tornado Wind	Tornado Missile	Remarks - Including Any Environmental Requirements
Cranes											
Containment Polar Crane	D	X	I	As Applicable	C	-	X	X	-	-	Containment Accident Pressure, Dead and Equipment, Live Loads, Hold Down Device, Note 1
Cask Crane	D	X	I	As Applicable	AB	-	X	X	-	-	Dead and Equipment, Live Loads, Hold Down Device, Note 1
Cranes (Excluding Reactor Building and Fuel Handling)	D	-	III	As Applicable	-	-	-	-	-	-	Dead and Equipment, Live Loads
Refueling Machine	W	X	I	As Applicable	C	-	X	X	-	-	
Fuel Handling Machine	W	X	I	As Applicable	AB	-	X	X	-	-	
Tanks											
Recycle Monitor	D	-	III	ASME VIII	AB		-	-	-	-	Class E
Laundry and Hot Shower	D	-	III	ASME VIII	AB		-	-	-	-	Class E
Waste Monitor	D	-	III	ASME VIII	AB		-	-	-	-	Class E
Mixing and Settling	D	-	III	ASME VIII	AB		-	-	-	-	Class E
Mixing and Settling Reagent	D	-	III	ASME VIII	AB		-	-	-	-	Class E
Floor Drain	D	-	III	ASME VIII	AB		-	-	-	-	Class E
Chemical Drain	D	-	III	ASME VIII	AB		-	-	-	-	Class E
[redacted]											
Waste Gas Decay	D	X	II I	ASME III	AB		X	X	-	-	
Waste Drain	D	X	I	ASME III	AB		X	X	-	-	
Waste Evaporator Feed	D	X	I	ASME VIII	AB		-	-	-	-	
Vent. Unit Cond. Drain	D	X	I	ASME VIII	AB		-	-	-	-	Not Code Stamped
Spent Resin Storage	D	X	I	ASME III	AB		X	X	-	-	
Refueling Water Storage	D	X	I	ASME III	YD		X	X	X	-	
Reactor Makeup Water Storage	D	-	III	ASME VIII	YD		-	-	X	-	Not Code Stamped
Boron Recycle Holdup	D	X	I	ASME III	AB		X	X	See Note (4)	See Note (4)	
Boric Acid	D	X	I	ASME III	AB		X	X	See Note (4)	See Note (4)	
Fuel Oil Storage	D	X	I	ASME III	YD		X	X	-	-	Buried - Not Code Stamped
Component Cooling Surge	D	X	I	ASME III	AB		X	X	-	-	
Steam Gen. Blowdown	D	-	III	ASME VIII	TB		-	-	-	-	
Backwash	D	-	III	ASME VIII	TB		-	-	-	-	Not Code Stamped
Upper Surge	D	-	III	ASME VIII	TB		-	-	-	-	
Condensate Storage	D	-	III	ASME VIII	TB		-	-	-	-	Not Code Stamped
Upper Surge Dome	D	-	III	ASME VIII	TB		-	-	-	-	
Evap. Concentrate Holdup	D	-	III	ASME VIII	AB		-	-	-	-	Class E - Not Code Stamped
Demin. Water Storage	D	-	III	ASME VIII	SB		-	-	X	-	
Heater Blowoff	D	-	III	ASME VIII	TB		-	-	-	-	
"C" Heater Drain	D	-	III	ASME VIII	TB		-	-	-	-	
RCW Storage	D	-	III	ASME VIII	SB		-	-	-	-	Not Code Stamped
Turbine Oil Transfer	D	-	III	ASME VIII	TB		-	-	-	-	Not Code Stamped
Fire Protection Pressurizer	D	-	III	ASME VIII	SB		-	-	-	-	
Filtered Water	D	-	III	ASME VIII	SB		-	-	X	-	Not Code Stamped
Y T Sulfuric Acid	D	-	III	ASME VIII	YD		-	-	X	-	Not Code Stamped
Evap. Concentrates Batch	D	-	III	ASME VIII	AB		-	-	-	-	Class E - Not Code Stamped

118. Provide a P&I Diagram showing the Steam Generator Blowdown System.

Response: See attached flow diagrams CN-1580-1.0 and CN-2580-1.0.