PRESENTATION ON THE NUMBER

OF OBE FATIGUE CYCLES FOR

BWR NSSS DESIGN

(EXCEPT PIPING)

SEPTEMBER 15, 1981
D.K. HENRIE
SEISMIC & DYNAMIC ANALYSIS
GENERAL ELECTRIC

DKH-1

NUMBER OF OBE FATIGUE CYCLES

NSSS EQUIPMENT

- SRP RECOMMENDATION- 5 OBE WITH 10 CYCLES
- GE RECOMMENDATION- 10 PEAK OBE CYCLES GENERICALLY

GE STUDY SHOWS 10 PEAK OBE CYCLES OVER PLANT LIFE CONSERVATIVE

DKH-2
9/15/81

- NRC NUREG/CR-1151 - RECOMMENDED REVISIONS TO NRC SEISMIC DESIGN CRITERIA

  - THE NUREG STATES THAT NRC REQUIREMENT "OF FIVE OBE CYCLES IS EXCESSIVELY CONSERVATIVE"

  - ALSO INDICATES THAT ON THE AVERAGE, THE OBE DESIGN ACCELERATION HAS A NEP OF 90% IN A 50 YEAR LIFE

  - WASH-1400, OCTOBER 1975 - PROBABILITY OF OBE IS ONE IN 100 TO 125 YEARS AND NOT FIVE IN 40 YEARS.

<u>CONCLUSION</u> - 5 OBE EXCESSIVELY CONSERVATIVE

DKH-3

9/15/81

- GE STUDY ON PROBABILITY OF OBE (1973)

  - o BASIS - A STUDY OF 26 PSAR AND FSAR PLANTS
  - o FOUR 40 YEAR PERIODS
    - 1810 - 1849
    - 1850 - 1889
    - 1890 - 1929
    - 1930 - 1969
  - o MAXIMUM SITE INTENSITY EARTHQUAKE CHOSEN FROM EACH PERIOD FOR EACH SITE
  - o RATIO OF MAXIMUM GROUND ACCELERATION TO SSE DESIGN BASIS GROUND ACCELERATION CALCULATED FOR EACH 40 YEAR PERIOD
    - MAXIMUM = 0.16
    - MINIMUM = INSIGNIFICANT
    - MEAN = 0.051
    - STANDARD DEVIATION = 0.039

DKH- 4
9/15/81

TABLE 1

GENERIC SUMMARY

	EL CENTRO	TAFT	GOLDEN GATE	CLINTON	HANFORD	PERRY	SUSQUEHANNA
DURATION (SEC.)	29.4	30.0	13.2	10.0	16.0	10.0	15.0
MAX. SITE ACCEL.(g) (RECORDED/ESTIMATED)	0.33	0.18	0.13	.015	.015	.007	.007
SSE DESIGN BASIS MAX. ACCEL. (g)	-	-	-	.25	.25	.15	.10
MAX. HORIZ. ACCEL (g)/ 90% NEP IN 50 YEARS	-	-	-	<0.04	<0.10	0.07	<0.04

DKH-5

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- NUMBER OF FATIGUE CYCLES PER EARTHQUAKE
  - OBTAINED BY TIME HISTORY ANALYSIS
  - RANDOM VS. PERIODIC EXCITATION
  - TABLE 2 - ENVELOPED AND AVERAGED CYCLES FROM THREE EARTHQUAKES AND SIX MAJOR NSSS COMPONENTS
  - TABLE 3 - % OF CYCLES < 50% OF PEAK
  - TABLE 4 - % OF CYCLES < 25% OF PEAK
  - INDEPENDENT OF EARTHQUAKE OR COMPONENT FREQUENCY
    - 99.5% OF STRESS REVERSALS OCCUR BELOW 75% OF MAXIMUM STRESS
    - 95% BELOW 50%
    - 85% BELOW 25%
  - TABLE 5 - SUMMARY OF EQUIVALENT STRESS CYCLES OF ALL MAGNITUDES

<u>CONCLUSION</u> - 10 PEAK OBE CYCLES ARE CONSERVATIVE

DKH-6
9/15/81

TABL(

AVERAGE NUMBER OF STRESS CYCLES OF ALL MAGNITUDES

LONG DURATION EARTHQUAKE	DURATION (SEC)	PEAK ACCEL. (g)	NORMALIZED PEAK ACCEL. (g)	NUMBER OF CYCLES FREQUENCY BANDS 0-10 Hz	10 - 20 Hz	20-50 Hz
EL CENTRO(1)	29.4	0.33	0.25	$\boxed{168}$	337	425
TAFT(2)	30.0	0.18	0.25	163	$\boxed{359}$	$\boxed{643}$
GOLDEN GATE(3)	13.2	0.13	0.25	94	171	316

NOTES: (1) May 18, 1940, El Centro, N/S Component, 29.4 sec

(2) July 21, 1952, Taft, S69^{O}E Component, 30.0 sec.

(3) March 22, 1952, Golden Gate, S80^{O}E Component, 13.2 sec.

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TA[illegible] 8

PERCENTAGE OF STRESS CYCLES WITH STRESS AMPLITUDES BELOW 50% OF THE MAXIMUM VALUE

Frequency Range	0 - 10 HZ			10 - 20 Hz			20 - 50 Hz		
Earthquake	El Centro	Taft	Golden Gate	El Centro	Taft	Golden Gate	El Centro	Taft	Golden Gate
Duration, sec	29.4	30.0	13.2	29.4	30.0	13.2	29.4	30.0	13.2
Component				PERCENTAGES					
A (STABILIZER)	99.2	98.0	99.9	94.7	95.6	90.8	94.7	99.2	96.8
B (STAR TRUSS)	99.2	97.1	99.9	97.9	97.9	96.8	97.6	99.[illegible]	97.7
C (REACTOR SKIRT)	99.0	95.8	99.9	95.4	96.4	92.8	99.6	99.8	99.3
D (SHROUD SUPPORT)	99.1	96.9	99.5	96.2	97.2	94.1	99.2	99.6	98.9
E (FUEL)	99.1	95.8	99.9	95.3	96.8	91.3	95.7	99.8	99.1
F (CRD HOUSING)	98.5	95.7	99.4	96.6	96.5	94.4	95.6	98.6	97.9
Average (Over-All Average 97.4)	99.0	96.6	99.8	96.0	96.7	93.4	97.1	99.4	98.3

Time History Input Cycles Below 50% of Peak

El Centro 93%

Taft 90%

Golden Gate 95%

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TABLE 4

PERCENTAGE OF STRESS CYCLES WITH STRESS AMPLITUDES BELOW 25% OF THE MAXIMUM VALUE

Frequency Range	0 - 10 Hz			10 - 20 Hz			20 - 50 Hz		
Earthquake	El Centro	Taft	Golden Gate	El Centro	Taft	Golden Gate	El Centro	Taft	Golden Gate
Duration, sec	29.4	30.0	13.2	29.4	30.0	13.2	29.4	30.0	13.2
Component	PERCENTAGE								
A	85.5	85.4	96.2	80.6	77.6	81.3	81.9	89.4	87.5
B	85.8	84.9	96.2	80.4	78.0	80.6	82.6	80.3	89.3
C	93.1	81.2	99.4	82.3	77.3	81.9	92.1	92.8	94.0
D	86.0	81.0	98.0	84.3	79.6	83.4	90.0	93.0	91.4
E	89.4	79.4	97.6	82.0	79.1	80.6	80.9	91.2	90.4
F	86.6	79.5	92.2	84.4	78.9	83.3	79.5	89.0	89.0
Average	87.7	81.9	96.6	82.3	78.4	81.9	84.5	89.3	90.3
(Over-all average	85.9)								

Time History Input Cycles Below 25% of Peak

El Centro	78%
Taft	70%
Golden Gate	90%

TABLE 5

NUMBER OF STRESS CYCLES OF ALL MAGNITUDES
DURING A LONG DURATION EARTHQUAKE

	FREQUENCY BANDS (CORRESPONDS TO COMPONENT FUNDAMENTAL FREQUENCIES)					
	0 - 10 Hz		10 - 20 Hz		20 - 50 Hz	
TOTAL NUMBER OF STRESS CYCLES	168		359		643	
NUMBER OF CYCLES BETWEEN 75% AND 100% OF PEAK VALUE (0.5% OF TOTAL)	1	(1)	2	(2)	3	(3)
NUMBER OF CYCLES BETWEEN 50% AND 75% OF PEAK VALUE (4.5% OF TOTAL)	8	(1)	16	(2)	29	(4)
NUMBER OF CYCLES BETWEEN 25% and 50% OF PEAK VALUE (10% OF TOTAL)	17	(1)	36	(1)	64	(1)
NUMBER OF CYCLES LESS THAN OR EQUAL TO 25% OF PEAK VALUE (85% OF TOTAL)	143	(1)	305	(1)	547	(1)
TOTAL NUMBER OF EQUIVALENT PEAK STRESS CYCLES		(4)		(6)		(9)

DKH-10

9/15/81

QUESTION NO. 10
(3.9.1)

No seismic transients are specified for the majority of the components and the components for which they are specified require only one OBE cycle. Justification is required.

RESPONSE

See the text revision attached.

Summation - This item is closed.

PCY:rf/45E4
8/18/81

(6)

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

This section shows the transients which are used in the design of the ASME Code Class 1, control rod drive components, reactor assembly including core supports and reactor internals, main steam and recirculation systems. The number of cycles or events for each transient are included. The design transients shown in this section are included in the design specifications for the components. Transients or combinations of transients are classified with respect to the component operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing" in the ASME Boiler and Pressure Vessel Code if applicable. The cycles due to SSE and OBE used in the fatigue analysis are shown in Table 3.7-4.

3.9.1.1.1 Control Rod Drive (CRD) Transients

The normal and test service load cycles used for design purposes for the 40 year life of the control rod drives are as follows:

	Transient	Category	Cycles
a.	Reactor startup/ shutdown	normal/upset	120
b.	Vessel pressure tests	normal/upset	130
c.	Vessel overpressure	normal/upset	10
d.	Scram test plus startup scrams	normal/upset	300
e.	Operational scrams	normal/upset	300
f.	Jog cycles	normal/upset	30,000
g.	Shim/drive cycles	normal/upset	1000

10

In addition to the above cycles, the following have been considered in the design of the CRD.

	Transient	Category	Cycles
h.	Scram with inoperative buffer	normal/upset	10
i.	Scram with stuck control blade	normal/upset	1
	faulted		
j	OBE *	normal/upset	10
k	SSE **	faulted	1

All ASME Class I components of the CRD have been analyzed according to ASME Section III Boiler and Pressure Vessel Code.

The capability of the CRD's to withstand emergency and faulted conditions is verified by test rather than analysis.

3.9.1.1.2 CRD Housing and Incore Housing Transients

The number of transients, their cycles, and classification as considered in the design and fatigue analysis of the CRD housing and incore housing are as follows:

	Transient	Category	Cycles
a.	Normal startup & shutdown	normal/upset	120
b.	Vessel pressure tests	normal/upset	130
c.	Vessel overpressure tests	normal/upset	10
d.	Interruption of feedwater flow	normal/upset	80
e.	Scram	normal/upset	200

* The frequency of occurrence of this transient would indicate emergency category. However, for conservatism, the OBE condition was analyzed as an upset condition. Ten peak OBE cycles are postulated.

** SSE is a faulted condition, however in the stress analysis it was treated as emergency with lower stress limits.

(10)

	Transient	Category	Cycles
f.	OBE Δ	normal/upset	1 10
g.	SSE**	emergency	1
CRD Housing Only			
h.	Stuck Rod Scram	normal/upset	1
i.	Scram no Buffer	normal/upset	1 10

~~** SSE is a faulted condition; however, in the stress analysis report it was treated as emergency with lower stress limits.~~

~~Δ The frequency of this cycle would indicate emergency category. However, for conservatism, this OBE condition was analyzed as upset but without fatigue considerations.~~

10

3.9.1.1.3 Hydraulic Control Unit Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40-year life and the Hydraulic Control Unit are as follows:

	Transient	Category	Cycles
a.	Normal startup & shutdown	normal/upset	120
b.	Vessel pressure tests	normal/upset	130
c.	Vessel overpressure tests	normal/upset	10
d.	Scram tests (cold)	normal/upset	300
e.	Operational scrams (hot)	normal/upset	300
f.	Jog cycles	normal/upset	30,000
g.	Drive cycles	normal/upset	1,000
h.	Scram with stuck scram discharge valve	normal/upset	1
i.	OBE	normal/upset	~~1~~ 10
k.	SSE	faulted	1

The frequency of occurrence of this event would indicate emergency category. However, for conservatism, this event was analyzed as normal and upset condition with 10 cycles considered for fatigue evaluation.

(10)

3.9.1.1.4 Core Support and Reactor Internals Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40 year life of the CS and Internals are ~~as follows:~~ shown on Table 3.9-1.

	Transients	Category	Cycles
a.	Startup	normal/upset	120
b.	Power cycles	normal/upset	12,400
c.	Loss of feedwater heater	normal/upset	80
d.	Scram	normal/upset	198
e.	Reduction to 0% power, hot standby shutdown, & vessel flooding	normal/upset	111
f.	Unbolting	normal/upset	123
g.	Scram (Auto. blow-down & reactor overpressure)	normal/upset	2
h.	Improper start of cold recirc. loop	emergency	1
i.	Sudden start of pump in cold recirc. loop	emergency	1
j.	Improper startup	emergency	1
k.	Pipe rupture & blowdown	faulted	1

(10)

3.9.1.1.5 Main Steam System Transients

The following transients are considered in the stress analysis of the main steam piping:

Main Steam Transients

	Transient	Category	Cycles
a.	Startup	normal	121
b.	Loss of F.W. pumps isolation valves closed	upset	10
c.	Scram	upset	180
d.	Shutdown	normal	111
e.	Reactor overpressure delayed scram	emergency	1
f.	Single S/RV blowdown	upset	8
g.	Automatic blowdown	emergency	1
h.	Hydrotest	test	130
i.	OBE	upset	50

3.9.1.1.6 Recirculation System Transients

The following transients are considered in the stress analysis of the recirculation piping:

Recirculation Transients

	Transient	Category	Cycles
a.	Startup	normal	121
b.	Turbine roll and increase to power	normal	120

(10)

	Transient	Category	Cycles
c.	Loss of feedwater heater	upset	10
d.	Partial feedwater heater bypass	upset	70
e.	Scrams	upset	180
f.	Shutdown	normal	111
g.	Loss of F.W. pumps isolation valves closed	upset	10
h.	Reactor overpressure with delayed scram	emergency	1
i.	Single S/RV blow-down	upset	8
j.	Automatic blowdown	emergency	1
k.	Hydrotest	test	130
l	OBE	upset	50

3.9.1.1.7 Reactor Assembly Transients

The reactor assembly includes the reactor pressure vessel, support skirt, shroud support, and shroud plate. The cycles listed in Table 3.9-1 were specified in the reactor assembly design and fatigue analysis.

3.9.1.1.8 Main Steam Isolation Valve Transients

The main steam isolation valves are designed for the following service conditions and thermal cycles:

	Transient	Category	Cycles
a.	Pre-op @100^{o}F/hr	normal/upset	150
b.	Startup (heating 100^{o}F/hr)	normal/upset	120.

(11)

TABLE 3.9-1

PLANT EVENTS

		No. of Cycles
Normal, Upset, and Testing Conditions		
a.	Bolt Up*/Unbolt	123
b.	Design Hydrostatic Test	130
c.	Startup (100°F/hr Heatup Rate)**	120
d.	Daily Reduction to 75% Power*	10,000
e.	Weekly Reduction to 50% Power*	2,000
f.	Control Rod Pattern Change*	400
g.	Loss of Feedwater Heaters (80 Cycles Total):	80
h.	Operating Base Earthquake Event at Rated Operating Conditions	~~*****~~ 10/50 ~~1****~~
i.	Scram:	
	1) Turbine Generator Trip, Feedwater on, Isolation Valves Stay Open	40
	2) Other Scrams	140
	3) Loss of Feedwater Pumps, Isolation Valves Closed	10
	4) Single Safety or Relief Valve Blowdown	8
j.	Reduction to 0% Power, Hot Standby, Shutdown (100°F/hr Cooldown Rate)**	111
k.	HPCS Operation (10), SLC Operation (10)	20

(10)

TABLE 3.9-1 (Continued)

		No. of Cycles
Emergency Conditions		
l.	Scram:	
	1) Reactor Overpressure with Delayed Scram, Feedwater Stays on, Isolation Valves Stay Open	1***
	2) Automatic Blowdown	1***
m.	Improper Start of Cold Recirculation Loop	1***
n.	Sudden Start of Pump in Cold Recirculation Loop	1***
o.	Improper Startup with Reactor Drain Shut Off Followed by Turbine Roll and Increase to Rated Power	1***
Faulted Condition		
p.	Pipe Rupture and Blowdown	1***
q.	Safe Shutdown Earthquake at Rated Operating Conditions	1***
ASME Hydrostatic Test		
r.	1.25 x Design Pressure Hydrostatic Test (per NB 6222 and NB 3114)	10

*Applies to reactor pressure vessel only.

**Bulk average vessel coolant temperature change in any 1-hour period.

***The annual encounter probability of the one cycle events is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted events.

~~****Includes 10 maximum load cycles per event.~~

~~****~~ 50 peak OBE cycles for NSSS piping, 10 peak OBE cycles for other NSSS equipment and components

50 peak OBE cycles are postulated for all BOP piping and components.

QUESTION NO. 11
(3.9.1)

Paragraph 3.9.1.1, Design Transients, referring to Table 3.7-4, "Reactor Building-Seismic Analysis Natural Frequency and Natural Period," appears to be in error. Clarification is required.

RESPONSE

The statement is deleted. See the text revision attached.

Summation - This item is closed.

PCY:rf/45E5
8/18/81

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

This section shows the transients which are used in the design of the ASME Code Class 1, control rod drive components, reactor assembly including core supports and reactor internals, main steam and recirculation systems. The number of cycles or events for each transient are included. The design transients shown in this section are included in the design specifications for the components. Transients or combinations of transients are classified with respect to the component operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing" in the ASME Boiler and Pressure Vessel Code if applicable. The cycles due to SSE and OBE used in the fatigue analysis are shown in Table 3.7-4.

3.9.1.1.1 Control Rod Drive (CRD) Transients

The normal and test service load cycles used for design purposes for the 40 year life of the control rod drives are as follows:

	Transient	Category	Cycles
a.	Reactor startup/shutdown	normal/upset	120
b.	Vessel pressure tests	normal/upset	130
c.	Vessel overpressure	normal/upset	10
d.	Scram test plus startup scrams	normal/upset	300
e.	Operational scrams	normal/upset	300
f.	Jog cycles	normal/upset	30,000
g.	Shim/drive cycles	normal/upset	1000

QUESTION NO. 12
(3.9.1)

Table 3.9-15, Applicable Seismic Cycle Loading, is indicated as "Later". Provide a schedule for its inclusion in the FSAR.

RESPONSE

Table 3.9-15 is deleted.

The statement in Section 3.9.1.1.13 that references Table 3.9-15 has also been deleted.

Summation - This item is closed.

Pressure Transient	Cycles
g. 110% design pressure at 575°F	1
h. 1300 psi at 100°F installed hydrostatic test	130
i. 1670 psi at 100°F installed hydrostatic test	3

3.9.1.1.13 Balance of Plant Transients

The transients used in design and fatigue analysis of the balance of plant components are listed in Table 3.9-1, with an exception that 50 maximum stress cycles due to OBE are used in fatigue evaluation ~~A complete list of applicable seismic cyclic loading for operating basis earthquake is shown in Table 3.9-15~~.

3.9.1.2 Computer Programs Used in Analysis

The following sections list the computer programs used in the analysis of specific components. These programs are described in 3.12.

3.9.1.2.1 Reactor Vessel

The following programs are used in the analysis of the Reactor Vessel:

a. CB&I Program 711 "GENOZZ"

b. CB&I Program 948 "NAPALM"

c. CB&I Program 1027

d. CB&I Program 846

e. CB&I Program 781 "KALNINS"

f. CB&I Program 979 "ASFAST"

g. CB&I Program 766 "TEMAPR"

h. CB&I Program 767 "PRINCESS"

TABLE 3.9-15

APPLICABLE SEISMIC CYCLIC LOADING

~~(LATER)~~

DELETED

QUESTION NO. 13
(3.9.1)

Methods of verification are required for all NSSS computer codes used in the analysis.

RESPONSE

The NSSS programs can be divided into two categories.

GE Programs

The verification of the following GE programs has been performed in accordance with the requirements of 10CFR50, Appendix B. Evidence of the verification of input, output, and methodology is documented in GE Design Record Files.

(a) MASS
(b) SNAP (MULTISHELL)
(c) GASP
(d) NOHEAT
(e) FINITE
(f) DYSEA
(g) SHELLS
(h) HEATER
(i) FAP-71 *
(j) CREEP-PLAST
(k) PISYS
(l) ANSI7
(m) SAP4G
(n) FTFLG01
(o) ANSYS
(p) BSTIF01

Vendor Programs

The verification of the following two groups of vendor programs is assured by contractual requirements between GE and the vendors. Per the requirements, the quality assurance procedure of these proprietary programs used in the design of N-stamped equipment is in full compliance with 10CFR50, Appendix B.

Byron Jackson Program

RTRMEC

CB&I Programs

(a) 711 GENOZZ
(b) 948 NAPALM *
(c) 1027
(d) 846
(e) 781 KALNINS
(f) 979 ASFAST
(g) 766 TEMAPR
(h) 767 PRINCESS
(i) 928 TGRV
(j) 962 E0962A
(k) 984
(l) 992 GASP
(m) 1037 DUNHAM'S
(n) 1335
(o) 1606 & 1657 HAP
(p) 1635
(q) 953

Accordingly, the FSAR text is revised as attached.

Summation - This item is closed pending NRC audit.

* To be audited by NRC.

PCY:ggt:rf/45L2
9/23/81

Pressure Transient	Cycles
g. 110% design pressure at 575°F	1
h. 1300 psi at 100°F installed hydrostatic test	130
i. 1670 psi at 100°F installed hydrostatic test	3

3.9.1.1.13 Balance of Plant Transients

The transients used in design and fatigue analysis of the balance of plant components are listed in Table 3.9-1.

A complete list of applicable seismic cyclic loading for operating basis earthquake is shown in Table 3.9-15.

3.9.1.2 Computer Programs Used in Analysis

The following sections list the computer programs used in the analysis of specific components. These programs are described in 3.12.

3.9.1.2.1 Reactor Vessel

The following programs are used in the analysis of the Reactor Vessel:

a. CB&I Program 711 "GENOZZ"

b. CB&I Program 948 "NAPALM"

c. CB&I Program 1027

d. CB&I Program 846

e. CB&I Program 781 "KALNINS"

f. CB&I Program 979 "ASFAST"

g. CB&I Program 766 "TEMAPR"

h. CB&I Program 767 "PRINCESS"

3.9.1.2 Computer Programs Used in Analysis

The following sections discuss computer programs used in the analysis of the major safety-related components. (Computer programs were not used in all components, hence not all components are listed.) The NSSS programs can be divided into two categories.

GE Programs

The verification of the following GE programs has been performed in accordance with the requirements of 10CFR50, Appendix B. Evidence of the verification of input, output, and methodology is documented in GE Design Record Files.

(a) MASS
(b) SNAP (MULTISHELL)
(c) GASP
(d) NOHEAT
(e) FINITE
(f) DYSEA
(g) SHELL5
(h) HEATER
(i) FAP-71
(j) CREEP-PLAST
(k) PISYS
(l) ANSI7
(m) SAP4G
(n) FTFLG01
(o) ANSYS
(p) BSTIF01

Vendor Programs

The verification of the following two groups of vendor programs is assured by contractual requirements between GE and the vendors. Per the requirements, the quality assurance procedure of these proprietary programs used in the design of N-stamped equipment is in full compliance with 10CFR50, Appendix B.

Byron Jackson Program

RTRMEC

CB&I Programs

(a) 711 GENOZZ
(b) 948 NAPALM
(c) 1027
(d) 846
(e) 781 KALNINS
(f) 979 ASFAST
(g) 766 TEMAPR
(h) 767 PRINCESS
(i) 928 TGRV
(j) 962 E0962A
(k) 984
(l) 992 GASP
(m) 1037 DUNHAM'S
(n) 1335
(o) 1606 & 1657 HAP
(p) 1635
(q) 953

3.9.1.2.1 Reactor Vessel and Internals

3.9.1.2.1.1 Reactor Vessel

CB&I Programs (a) through (q) listed above are used to analyze the reactor pressure vessel. Detailed descriptions are provided in Section 3.12.

3.9.1.2.1.2 Reactor Internals

The following computer programs are used in the analysis of the core support structures and other safety-related reactor internals: MASS, SNAP (MULTISHELL) GASP, NOHEAT, FINITE, DYSEA, SHELL5, HEATER, FAP-71, and CREEP-PLAST. Detailed descriptions of these programs are provided in Section 4.1.

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i. CB&I Program 928 "TGRV"

j. CB&I Program 962 "E0962A"

k. CB&I Program 984

l. CB&I Program 992 "GASP"

m. CB&I Program 1037 "DUNHAM'S"

n. CB&I Program 1335

o. CB&I Programs 1606 and 1657 "HAP"

p. CB&I Program 1635

q. CB&I Program 953

3.9.1.2.2 Piping

The following programs are used in the analysis of piping:

a. ADLPIPE

b. DYNAMIC ANALYSIS OF PIPING SYSTEMS

c. PLATE PANEL, SPACE STRUCTURAL ANALYSIS (MASS)

d. SHELL ANALYSIS PROGRAM (MULTISHELL)

e. TIME DEPENDENT PIPE FORCE

f. PIPE DYNAMIC ANALYSIS PROGRAM (PDA)

3.9.1.2.3 Recirculation Pump

No computer programs were used in the design of the recirculation pumps.

3.9.1.2.4 ECCS Pumps and Motors

An equivalent static computer analysis was performed on the ECCS pump motor rotor shafts. The model consisted of lumped masses simulating the distribution of mass in the system,

3.9.1.2.2 Piping

3.9.1.2.2.1 Piping Analysis Program/PISYS

PISYS is a computer code specialized for piping load calculations. It utilizes selected stiffness matrices representing standard piping components, which are assembled to form a finite element model of a piping system. The technique relies on dividing the pipe model into several discrete substructures, called pipe elements, which are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment, and loading of the structure becomes possible. PISYS is based on the linear classical elasticity in which the resultant deformation and stresses are proportional to the loading, and the superposition of loading is valid.

PISYS has a full range of static and dynamic analysis options which include distributed weight, thermal expansion, differential support motion modal extraction, response spectra, and time history analysis by modal or direct integration. The PISYS program has been benchmarked against five Nuclear Regulatory Commission piping models for the option-of-response-spectrum analysis and the results are documented in a report to the Commission, "PISYS Analysis of NRC Problem," NEDO-24210, August, 1979.

3.9.1.2.2.2 Component Analysis/ANSI7

The ANSI 7 computer program determines stress and accumulative usage factors in accordance with NB-3600 of the ASME Code, Section III. The program was written to perform stress analysis in accordance with the ASME Code sample problem, and has been verified by reproducing the results of the sample problem analysis.

3.9.1.2.3 ECCS Pumps and Motors

3.9.1.2.3.1 Rotor Assembly Analysis Program/RTRMEC

RTRMEC is a computer program which calculates and displays results of mechanical analysis of motor rotor assembly when acted upon by external forces at any point along shaft (rotating parts only). The shaft deflection due to magnetic and centrifugal forces was analyzed. The calculation for the seismic condition assumes that the motor is operating and that the seismic, magnetic, and centrifugal forces all act simultaneously and in phase on the rotor-shaft assembly. Note that the distributed rotor assembly weight is lumped at the various stations, with the shaft weight at a station being the sum of one-half the weight of the incremental shaft length just before the station, plus one-half the weight of the adjacent incremental shaft length just after the station. Bending and shear effects are accounted for in the calculations.

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3.9.1.2.3.2 Structural Analysis Program/SAP4G

SAP4G is used to analyze the structural and functional integrity of the ECCS pump/motor systems. This is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve the displacements and stresses of each element of the structure. The structure can be composed of unlimited number of three-dimensional truss, beam, plate, shell, solid, plate strain-plane stress and spring elements that are axisymmetric. The program can treat thermal and various forms of mechanical loading. The dynamic analysis includes mode superposition, time history, and response spectrum analysis. Seismic loading and time-dependent pressure can be treated. The program is versatile and efficient in analyzing large and complex structural systems. The output contains displacements of each nodal point as well as stresses at the surface of each element.

3.9.1.2.3.3 Effects of Flange Joint Connections/FTFLG01

The flange joints connecting the pump bowl castings are analyzed using FTFLG01. This program uses the local forces and moments determined by SAP4G to perform flat flange calculations in accordance with the rules set forth in Appendix II and Section III of the ASME Boiler and Pressure Vessel Code.

3.9.1.2.3.4 Structural Analysis of Discharge Head/ANSYS

ANSYS is used to analyze the pump discharge head flange and bolting taking into account of the prying action developed by the flat face contact surface. The program is described in detail in 3.12.

3.9.1.2.4 RHR Heat Exchangers

3.9.1.2.4.1 Structural Analysis Program/SAP4G

SAP4G is used to analyze the structural and functional integrity of the RHR heat exchangers. The description of this program is provided in Subsection 3.9.1.2.3.2.

3.9.1.2.4.2 Local Stiffness Calculations/BSTIF01

BSTIF01 is used to estimate the local stiffness of the heat exchanger shell at the attachment point of the supports. The method used in this program is based on the shell stiffness calculations by P. P. Bijlaard as groundwork for Welding Research Council Bulletin 107. The results of BSTIF01 are used to determine equivalent beam properties of the lower and upper heat exchanger support bracket to shell attachments included in the finite element model of the heat exchanger.

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connected by massless elastic members, simulating the distribution of shaft stiffness. The analysis was performed iteratively to obtain compatibility between the rotor displacements and the magnetic and centrifugal forces acting on the rotor.

All other analysis of specific motor components and pump components consisted of hand calculations.

3.9.1.2.5 RHR Heat Exchangers

Following are the computer programs used in dynamic and static analysis to determine structural and functional integrity of the RHR heat exchangers:

Support Load Seismic Analysis (ED-6)

Stress Analysis of Supports (ED-8)

3.9.1.2.6 Other Computer Programs Used in Analyses

Other computer programs used in the dynamic and static analyses of structural and functional integrity of Seismic Category I systems, components, equipment and supports are listed in 4.1.4 and 3.12

3.9.1.3 Experimental Stress Analysis

When experimental stress analysis is used in lieu of analytical methods for Seismic Category I ASME Code items, the requirements for experimental testing enumerated in the ASME Code which are applicable for the specific components under test shall be applied. When testing is required for Seismic Category I non-ASME Code parts account shall be taken of size effects and dimensional tolerances which exist between the actual part and the test part or parts as well as differences which may exist in the ultimate strength or other governing material properties of the actual part and the tested parts, to assure that the loads obtained from the test are a realistic or conservative representation of the load carrying capability of the actual structure under the postulated loading.

Results of both ISOFINITE and NASTRAN are given in Table 3.12-2. As can be seen, there is close correlation between the deflections, with NASTRAN giving larger values throughout the flued head than ISOFINITE. This is due to the lack of rotational freedom at nodes with NASTRAN over the more flexible shear elements in ISOFINITE. This leads to prediction of higher stresses using ISOFINITE(as can be seen by comparing pages 2 and 3 of Table 3.12-2). The computer program ISOFINITE is therefore a conservative method for determining stresses in flued head fittings.

This program is referred to in 3.8.6.4.4.

3.12.10 ADLPIPE

ADLPIPE is a digital computer program developed by the Arthur D. Little Co. and used for static and dynamic analyses of complex piping systems. Input data preparation uses piping language and output information is presented for easy interpretation. The input data may be pre-processed and plots made for input and model evaluation. To aid in rapid data preparation there are many input error diagnostics. The output automatically includes a stress analysis as required by ANSI B31.1 (1967); B31.1 (1973); ASME Code Section III, Class 1, Class 2 and Class 3 (1971 and 1974). The ASME Code, Section III, Class 1 analysis includes calculation of fatigue usage factor and simplified elastic-plastic analysis. All forces, moments, deflections, rotations and a summary stress report are included in the output. Additionally, the program has orthographic, isometric, and stereoscopic plotting capability to aid checking input and interpreting computed results.

The piping system is modeled as a series of sections that lie between network points. A section is composed of straight and curved members, and each member may have common or different loads and physical properties. The network points may be free, partially or fully restrained and have specified displacements that represent thermal anchor displacements or seismic anchor motion. Intermediate springs to ground or joining other members may be placed within the section to represent spring hangers, pipe bellows, skew and guided restraints, support and equipment stiffness. Transfer matrix techniques are used to reduce the size of the stiffness matrix.

3.12.10 ANSYS

ANSYS is a general-purpose finite element computer program designed to solve a variety of problems in engineering analysis.

The ANSYS program features the following capabilities:

1. Structural analysis including static elastic, plastic and creep, dynamic, seismic and dynamic plastic, and large deflection and stability analysis.

2. One-dimensional fluid flow analyses.

3. Transient heat transfer analysis including conduction, convection, and radiation with direct input to thermal-stress analyses.

4. An extensive finite element library, including gaps, friction interfaces, springs, cables (tension only), direct interfaces (compression only), curved elbows, etc. Many of the elements contain complete plastic, creep, and swelling capabilities.

5. Plotting - Geometry plotting is available for all elements in the ANSYS library, including isometric and perspective views of three-dimensional structures.

6. Restart Capability - The ANSYS program has restart capability for several analyses types. An option is also available for saving the stiffness matrix once it is calculated for the structure, and using it for other loading conditions.

The program is maintained current by Swanson Analysis Systems, Inc. of Pittsburgh, Pennsylvania and is supplied to General Electric for use on the Honeywell 6000.

The ANSYS program has been used for productive analyses since early 1970. Users now include the nuclear, pressure vessels and piping, mining, structures, bridge, chemical, and automotive industries, as well as many consulting firms.

The static loads on the piping system may be thermal, deadweight, static "g" seismic loads, externally applied forces and moments, and wind loads. The dynamic loads are computed using normal mode theory and seismic response spectra or time history forcing functions in one or more directions.

The approach used in ADLPIPE to compute the response of piping system to ground shock inputs is based upon a normal mode or modal superposition method. The formulation in terms of normal modes, which is particularly advantageous for transient reponse problems, follows generally the form discussed by Young (Reference 3.12-15).

The first step in the application of this method is the determination of the natural frequencies and mode shapes of the free vibrations of the system. For a conservative linearly-elastic lumped mass system, the governing matrix equation is:

$$M \ddot{u} + K_R u = 0$$

where

M is the (diagonal) inertia matrix,

K_R is the stiffness matrix, and

u is the column matrix of the displacement coordinates.

The stiffness matrix K_R utilized in the dynamics formulation differs from the stiffness matrix K_R developed by ADLPIPE for the network points. The latter matrix, developed by transfer matrix techniques, includes mass points and interior network points. The stiffness matrix for the dynamics formulation requires stiffness values at mass points only. Thus, K_R is a reduced form of K_R, and can be shown to be equal to:

$$K_R = A - B \cdot E^{-1} \cdot D$$

where:

$$K = \begin{matrix} A & B \\ D & E \end{matrix}$$

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and A = Mass points sub-matrix

B,D = Coupling sub-matrices

E = Branch points submatrix.

Determination of the natural modes can proceed by any one of several methods. The two eigenvalue routines used in ADLPIPE are the Jacobi (Reference 3.12-16) rotation scheme and the Givens-Householder (Reference 3.12-16) scheme; the latter has been modified to incorporate a suggestion made by Wilkinson (Reference 3.12-17). In the Jacobi routine, the operations are carried out in core memory and the number of degrees of freedom is limited by available core. The Givens-Householder routine is unlimited by core utilization of secondary storage and produces the lowest eigenvalues and associated eigenvectors.

For a system having N degrees of freedom, the eigenvalue routines will produce up to N eigenvalues (natural frequencies) ωi ($i=1 \ldots N$) and up to N sets of eigenvectors, ϕij ($i=1, \ldots N$, $j=1, 2, \ldots N$). The j^{th} column of the ($N \times N$) array ϕij is called the eigenvector for the j^{th} mode while the array itself is called the modal matrix.

The normal mode formulation of the response of a lumped system to a shock displacement can be carried out by considering the kinetic and potential energies of the loaded system. Assuming that the elastic displacement of the i^{th} coordinate is u_i, and the total displacement equal to $u_i + s_i$, where s_i is the shock displacement of the i^{th} coordinate "normal" coordinates q_n (t) and $p^{(t)}$ (t) are defined by the linear transformation:

$$u_i(t) = \sum_{n=1}^{N} \phi_{in} q_n (t)$$

$$s_i(t) = \sum_{n=1}^{N} \phi_{in} p_n (t)$$

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where ϕ_{in} is the i^{th} element of the n^{th} eigenvector. These transformations are useful because the resulting equations of motion in terms of the normal coordinates are completely decoupled from one another.

The kinetic energy T and the potential energy V of the system, in terms of the normal coordinates, are given by (reference 3.12-15):

$$T = \frac{1}{2}\sum_{i=1}^{N} m_i\dot{q}_i^2 + \dot{s}\sum_{i=1}^{N} m_i \sum_{n=1}^{N}\phi_{in}\dot{q}_n + \frac{1}{2}\dot{s}^2\sum_{i=1}^{N} m$$

where: m_i are the individual mass elements, and

M_i = generalized mass for the i^{th} mode, defined by the relation:

$$M_i = \sum_{j=i}^{N} m_j\phi_{ji}^2$$

Substitution of these energy expressions into Lagrange's equation leads to the equations of motion:

$$\ddot{q}n + \omega_n^2 qn = -\ddot{p}_n$$

The solution of these equations for the modal amplitudes q_n is:

$$q_n(t) = \frac{1}{\omega_n}\int_0^t \bar{p}_n(T)\sin\omega_n(t-T)dT$$

where T is a dummy variable of integration.

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From the transformation equation, we have

$$\bar{P}_n = \sum_{\ell}\left[\phi^{-1}_{n\ell}\bar{s}_{\ell}\right]$$

where $\Sigma\phi^{-1}_{n\ell}\phi_{\ell m} = I_{nm}$ = the identity matrix.

Hence,

$$q_n(t) = \frac{1}{\omega_n}\left[\sum_{\ell}\left(\phi^{-1}_{n\ell}\right)\int_0^t \bar{s}_{\ell}(T)\sin\omega_n(t-T)\,dT\right]$$

Defining $R_n(t) = \dfrac{1}{\omega_n}\displaystyle\int_0^t \bar{s}_{\ell}(T)\sin\omega_n(t-T)\,dt$

the modal response can be written simply as:

$$q_n(t) = \left(\sum_{\ell}\phi^{-1}_{n\ell}\right)\left(R_n(t)\right)$$

The expression $\sum_{\ell}\phi^{-1}_{n\ell}$ represents the portion of the maximum modal response developed by each normal coordinate, and may be thought of as a measure of the extent of which the n^{th} normal mode participates in the synthesis of the total response of the structural system. As such, the square array $\phi^{-1}_{n\ell}$, which is the inverse of the modal matrix, is termed the modal participation matrix, with each element of the matrix corresponding to the "participation" in the overall response synthesis of mode n, and mass point ℓ.

The term $R_n(t)$, expressed by the convolution integral, represents the response of mode n as a function of time, assuming that mode n is uncoupled from the other modes of the system; i.e., $R_n(t)$ is the response of a single degree of freedom system to the transient loading given by s(t).

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The three dimensional shock input displacement, $D_{\ell n}$, is given in terms of a maximum-valued spectrum (such as the Housner earthquake input spectra) for each principal axis. For instance, the vertical response may be different from the horizontal response. Thus, the prescribed input for each mode is the maximum value of the response $R_n(t)$ developed during the overall duration of the response. (These values are obtained, for example, by measurement with displacement meters such as cantilever reed gages which record the peak value of the displacement during the response period.)

Thus,

$$(D_\ell)_n = \left| R_n(t) \right| \text{ maximum}$$

Therefore, the modal amplitudes become

$$q_n(t) \leq \sum_\ell \left[\phi^{-1}_{n\ell}\right] (D_\ell)_n$$

For each normal mode, the amplitudes at each coordinate i are given by

$$x_i = \phi_{in} q_n$$

where:

$$u_i = \sum_{n=i}^{N} x_i$$ = elastic amplitude at coordinate i summed over all modes.

This then provides a set of displacements, x_i, for each of the n modes. These individual sets of displacements can then be applied to the system as equivalent static deflections on a mode-by-mode basis. The corresponding network forces are obtained by ADLPIPE.

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~~ADLPIRE computes the non-mass network force-moments sets for each mode. As seen previously, the network stiffness matrix formed is generated by the transfer matrix of a series of many individual members. This same accumulated transfer matrix is used to compute the force-moment sets at interior points of the piping system (including the mass points).~~

~~The cumulative effect of all the modes is estimated by taking the square root of the sum of squares of the force-moment sets at each position in the piping system. For closely spaced frequencies, an option exists which enables the addition of the absolute value of those modal moments and then forming the square of that sum in the square root of square summation.~~

~~This program is referred to in 3.9.1.2.2.~~

3.12.11 RELAP3

This program describes the behavior of water-cooled nuclear react ors during postulated accidents such as loss-of-coolant, pump failure, or power transients. The behavior of the primary cooling system and the reactor is emphasized. The program calculates flaws, mass inventories, energy inventories, pressures, temperatures, and qualities along with variables associated with reactor power, reactor heat transfer, or control systems.

RELAP3 is an NRC accepted computer program and is in the public domain. For a complete discussion of this program see Reference 3.12-18.

This program is referred to in 3.6.2.2.1b and 3.6.2.3.1.

3.12.11.1 RELAP4/MOD5

RELAP4 is a computer program written in FORTRAN IV for the digital computer analysis of nuclear reactors and related systems. It is primarily applied in the study of system transient response to postulated perturbations such as coolant loop rupture, circulation pump failure, power excursions, etc. The program was written to be used for water-cooled (PWR and BWR) reactors and can be used for scale models such as LOFT and SEMISCALE. Additional versatility extends its usefulness to related applications, such as ice condenser and containment subcompartment analysis. Specific options are available for reflood (FLOOD) analysis and for the NRC Evaluation Model.

QUESTION NO. 14

All computer programs used in the design and analysis of systems and components within the BOP scope must be listed. Methods of verification are required for all BOP programs.

RESPONSE

See revised FSAR pages (attached).

Summation - This item is closed.

3.9.1.2.7 BOP Computer Programs

A list of the principal computer programs used in dynamic and static analyses in the BOP scope is given in Table 3.9-18. With the exception of the Burns and Roe developed program, these programs are recognized and widely used in the industry with a history of successful applications. The Burns and Roe developed program listed in Table 3.9-18 is documented, verified and maintained by Burns and Roe as described in SRP 3.9.1 II2.b.

3.9.1.2.7.1 SRVDAM

SRVDAMA (Safety Relief Valve Discharge Analyses Model 4) is a computer model which simulates the transient flow of steam, air and water in a safety relief valve discharge line (S/RVDL) for a time period of approximately 0.5 seconds after S/RV opening. The model calculates transient fluid properties, forces and thermal distributions in the S/RVDL.

The piping system is initially filled with air and a water slug at the exit submerged in the suppression pool. Upon S/RV actuation, steam enters the line and compresses the air which expels the water slug. The piping system is represented by two models: (1) a gas (steam and air) and (2) a water slug, which are coupled by common pressure and velocity at the air-water interface. The gas flow equations are expressed in finite difference form solved with the method of characteristics. Provision for axial variation in flow area is included. Motion of the water slug is solved with a one-dimensional ordinary differential form of the momentum equation which is integrated axially to determine flowrate and displacement.

SRVDAM4 is based on the analytical model described in the G.E. Report NEDE-23749-P (ref. 1) and G.E. computer code RVFORO4 described in NEDE-24695 (ref. 2).

<u>Program Version and Computer</u>

Currently SRVDAM version 4 is being used by Burns and Roe, Inc. in conjunction with a CDC Computer.

<u>Extent of Application</u>

SRVDAM4 is a transient piping fluid analysis program which began development in 1975 and is supported by Burns and Roe. It has been used on several in-house projects.

Test Problems

SRVDAM4 has been benchmarked against problems provided in references 1 and 2 which have been compared with in-plant test data from Quad Cities, Monticello and CAORSO BWR plants.

References

1) "Analytical Model for Computing Transient Pressures and Forces in the S/RVDL", NEDE-23749-P, General Electric Co., February 1978.

2) "RVFORO4 User's Manual, SRVDL Clearing Transient For X-Quencher Devices", NEDE-24695, General Electric Co., December 1979.

TABLE 3.9-18

Computer Programs Used For Dynamic and Static Analyses in the BOP Scope

Computer Program	Document Traceability	System Used
ADLPIPE	Arthur D. Little, Inc. Acorn Park Cambridge, Ma. 02140	(1) CDC 7600 (2) CDC 172 (3) National Advanced Systems AS/5000
ANSYS	Swanson Analysis Systems, Inc. Elizabeth, Pa. 15037	(1) CDC 176 (2) National Advanced Systems AS/5000
RELAP	Argonne National Laboratory	(1) CDC 176 (2) CDC 7600
SRVDAM 4	Burns and Roe	CDC 175

QUESTION NO. 15
(3.9.1)

The computer code utilized in the analysis of the ECCS Pump Motor Rotor Shafts addressed in Paragraph 3.9.1.2.4, ECCS Pumps and Motors, is not identified. This code should be identified and data presented for the validity and applicability for use of the code.

RESPONSE

Referring to the response to Question No. 13, RTRMEC was used by the motor vendor to estimate the ECCS motor shaft displacement. The results of the calculation have been verified by (1) comparison with motor rotor bend test data and (2) comparison with the SAP4G results obtained by GE. The comparison demonstrated the conservatism of RTRMEC.

Summation - This item is closed.

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QUESTION NO. 16
(3.9.1)

Provide additional details concerning the test program performed on the orificed fuel support to establish the validity of the program. In addition, provide justification for using the allowable limits by applying a 0.65 quality factor to the ASME Code allowables of 1.5 Sm for upset condition.

RESPONSE

1. Test Program

The following is a detailed description of the test program. (Note: WNP-2's orificed fuel support is not required to conform with the ASME Code; however, the test program is designed to conform to the code in order to verify the design adequacy.)

Two separate tests were conducted, and each test was designed to be in conformance with Appendix II of th ASME code Section III. The first test series verified the structural capability of the fuel support casting to sustain vertical design loads. A production fuel support was stresscoated and subjected to an extremely high vertical load to identify the location and principal stress directions of the highest stressed regions. A second fuel support was instrumented with strain gauges: 12 uniaxial gauges were used where the principal stress directions were known from the previous stresscoat test. Six rosettes were used where the principal stress axes could not readily be determined. (All the gauges used in the experimental stress analysis were put in the regions of highest stress as determined by the previous stresscoat test.) The fuel support was mounted in a fixture simulating the geometric characteristics of both the load and support in the reactor. Vertical loads only were applied, simulating the weight load of the fuel assemblies.

It was found that the fuel support could sustain a vertical load of 104,000 pounds before the onset of yielding in the highest stressed region. This 104,000 pound load represents a safety factor in excess of 35 based on yielding over the normal applied vertical load.

A second series of tests were conducted to investigate the resulting stresses induced in the fuel support by a horizontal (or lateral) load applied by the fuel assemblies during a seismic event. A fuel support was instrumented with 15 three-element rosette strain gauges. The location of these gauges were determined from an initial computer analysis, and represented the areas of highest stress plus a few key locations of minimal material thickness.

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The test fixtures used were designed to apply equal loads on all four pods. This was achieved by using two hydraulic cylinders to load two spreader bars. The load was transmitted into each spreader bar through balls which prevented moment build-up. Each spreader bar then loaded two arms, which in turn loaded dummy fuel lower tie plates. At the interface of the tie plates in the fuel support, the dimensions of these dummy tie plates were identical to those used in the production components. During loading, weight was placed at the top of the load arms approximately in the center of the fuel support. This loading simulated a vertical load which would be present due to the fuel assembly weight.

During the initial phases of the testing, it was discovered that the stresses induced by a horizontal load were a maximum when the applied vertical load was a minimum. Because the fuel support is not attached to the guide tube and sits on a chamfered seat on the guide tube provided for that purpose, it was found that an increased downward vertical load actually enhanced the fuel support's ability to sustain a horizontal load. (With increased vertical load, additional rigidity was provided to the fuel support casting by the guide tube.)

A load cell was calibrated and installed on the lower hydraulic cylinder. Load data was recorded on a continuous recorder, and strain gauge data was recorded on a multi-channel recorder. The total applied load was twice the load cell readings.

The first horizontal loading applied simulated the ASME code upset condition. For this condition the total vertical load was calculated to be just under 1,000 pounds with a horizontal load of 2,600 pounds being applied. The calculated vertical load applied to the fuel casting included its weight, the upward component of a 1/2g seismic load, and the differential pressure across the fuel and the fuel support. The 2,600 pound load was taken from the fuel support design specification for the upset event. A horizontal test load of 3,000 pounds was applied to compensate for possible increased hydraulic piston friction, changes in friction due to a small amount of misalignment and/or cocking of the load arm in relation to the piston travel direction.

The test results simulating the upset horizontal loading conditions produced a maximum stress of 10,833 psi. The differential pressure stresses across the castings were computed. The 1,580 psi value obtained from the computation was then added to the test results. (Differential pressures across the fuel support were not simulated in the test program.) The total resultant stress was 12,413 psi for the upset condition. The total stress resultant was less than the ASME code allowable of 15,580 psi for the upset condition.

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A second series of test loadings were applied to the support casting and were designed to simulate the faulted conditions. No vertical load was applied during this phase of the testing because of the net result of 1g downward force due to gravity and the 1g upward component of force due to the safe shutdown seismic faulted event. The horizontal test load was applied to simulate 5,200 pounds of force for the faulted event.

Testing simulating the faulted horizontal loading produced a maximum stress intensity of 21,225 psi. A computed stress value of 1,580 psi for the internal pressure was added to the test result similar to that of the upset event described above. The addition of these two stresses resulted in a maximum stress intensity of 23,505 psi, which is significantly less than the 35,400 psi allowed by ASME code for the faulted conditions.

2. Quality Factor

The 0.65 quality factor accounts for the fact that not all castings are fully volumetrically examined. It is specified in the ASME Code, 1976 Edition, Summer 1976 Addenda, Paragraph NG-2571.2(a).

Summation - This item is closed.

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Question 17

Expand Paragraph 3.9.1.4.1.2 (page 3.9-18) to describe the actual mounting of the hydraulic control units and to justify the validity of the assumption utilized in the FSAR.

Response

Please refer to revised 3.9.1.4.1.2 (page 3.9-18) for the information requested.

Summation - This item is closed.

No analysis has been made for the non-code components of the CRD for the abnormal condition.

The design adequacy of non-code components of the CRD has been verified by extensive testing programs on components parts, specially instrumented prototype drives and production drives. The testing included postulated abnormal events as well as the service life cycle listed in 3.9.1.1.1.

3.9.1.4.1.2 Hydraulic Control Unit

The Hydraulic Control Unit (HCU) was analyzed for the SSE faulted conditions, through the implementation of the computer code SAMIS (See 3.12). Using the method of "Sum of Absolute Values of the Modal Loads," the maximum stress on the HCU frame was calculated to be 54,310 psi. The maximum allowable for SSE is 60,000 psi for the HCU. These stresses were obtained by ~~assuming~~ ~~that~~ two HCUs ~~are braced~~ together back to back ~~on the "H" beam~~ at the top, mid-depth and bottom of the HCU. ; add attached INSERT (A)

The fundamental frequency of the HCU is close to the frequency at which peak seismic shock will occur. This results in overstressed conditions in the piping connected to the HCU during the safe shutdown earthquake (SSE). By the application of bolt-on stiffening struts to the HCU frames and diagonal braces along the rows of installed HCU's, the fundamental frequency is raised sufficiently to avoid peak seismic response. The stresses in the connected piping are thus reduced to acceptable values.

The analysis of the HCU under faulted condition loads establishes the structural integrity of the system.

3.9.1.4.1.3 CRD Housing

The SSE is classified as a faulted condition; however, in the CRD housing analysis the SSE event has been treated as an emergency condition. The maximum membrane stress intensity occurs at the tube to tube weld near the center of the housing. The stresses are within elastic limits and are shown in Table 3.9-2(v).

Response to MEB SER
Question #17

Insert A to FSAR Page 3.9-18

each pair of tied HCUs is supported in each of three mutually perpendicular directions by means of struts and diagonal bracing connected from the HCUs to a three dimensional seismic support frame enclosing rows of HCUs and anchored to the concrete foundation. See attached Figure Q 17-1.

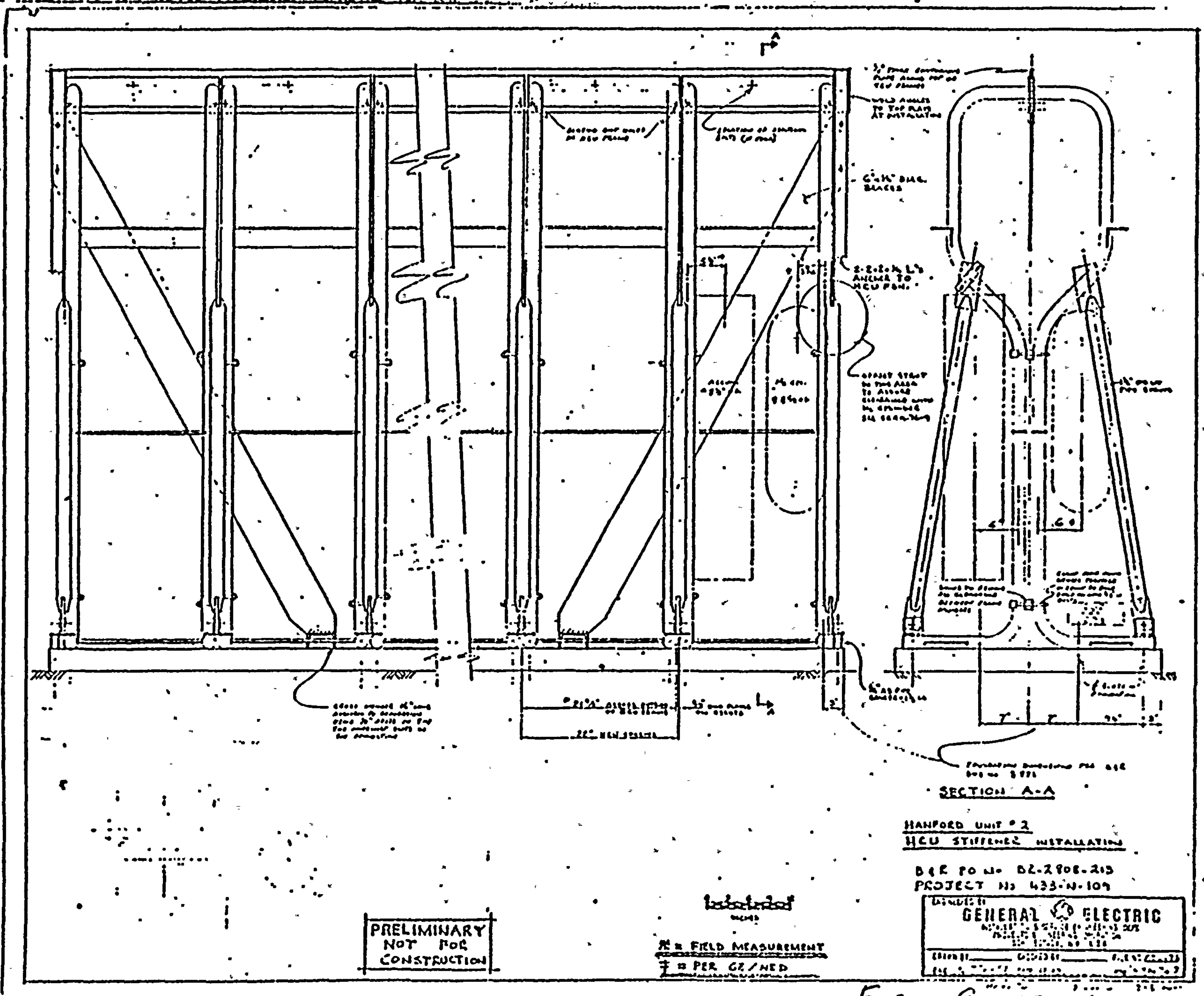
SECTION A-A
HANFORD UNIT # 2
HCU STIFFENER INSTALLATION
B&R PO No. B2-2908-213
PROJECT No 433-N-109
GENERAL ELECTRIC
PRELIMINARY
NOT FOR
CONSTRUCTION
= FIELD MEASUREMENT
‡ = PER GE/NED

FIG. A. 17-1

Question

18. Provide a commitment in the FSAR stating that all required piping restraints, components and component supports have been installed in the piping systems prior to testing.

Response: Paragraph 14.2.4.1.2 indicates that certain test prerequisites must be satisfied prior to the initiation of any preoperational test. System lineup tests (SLTs) which require, as part b of paragraph 14.2.4.1.2 states, that pipe support inspections and adjustments be completed are examples of these preoperational test prerequisites. In addition a separate, distinct SLT governing verification of proper installation and adjustment of component supports has been generated. Execution of applicable portions of this SLT on each piping system provides formal documentation of required support operability.

The administrative frame work imposed upon the preoperational test program as described in FSAR Chapter 14 provides a commitment which requires that all required piping restraints, components and component supports have been installed prior to testing. In summary, sufficient discussion presently exists in the FSAR to address concerns in this area.

Summation - This item is closed.

Question

19. The applicant's preoperational test program covers the vibration and dynamic effects. However, the thermal expansion effects required in SRO 3.9.2.II-1.d, e and f are not adequately addressed. The thermal motion monitoring program should deal specifically with verification of snubber movement, adequate clearances and gaps to allow free movement of the pipe during heatup and cooldown and should include acceptance criteria and test procedures. Additional information on this program is required.

Response: The WNP-2 Thermal Expansion Program is conducted during the Startup Test Program which is described in FSAR section 14.2. The specific thermal expansion program is described in section 14.2.12.3.17. This section prescribes test purposes, prerequisites, a test description and acceptance criteria. This program will be applied to systems which experience an operating temperature greater than 250°F and are classified in one of the following categories:

- ASME Code Class 1, 2 or 3 piping system
- High energy piping system inside Seismic Category 1 structures
- High energy system whose failure could reduce the functioning of a Seismic Category 1 feature to an unacceptable safety level
- Seismic Category 1 portions of moderate energy piping system located outside containment
- Condensate/feedwater piping per Reg. Guide 1.68.1 c.2.f&g

A combination of visual inspection and remote monitoring of certain inaccessible locations on critical piping will provide data to make evaluations which address the defined test purposes. Specifically on selected systems, a pre-heatup visual inspection to establish baseline test conditions is performed which confirms that no potential obstruction thermal movement exists, pipe hangers are at their "cold positions", snubbers are at the mid-range of travel and adequate pipe whip restraint clearance exists. At an intermediate point and again at rate temperature, a visual examination of the selected piping systems is performed to confirm proper thermal movement relative to the baseline conditions. At corresponding reactor system temperatures, data is also recorded from the remote monitoring devices and compared against test acceptance criteria. Following several heatup and cooldown cycles, the thermal movement measurements are recorded a second time to determine that proper "shakedown" of the systems has occurred. Appropriate action based upon the test results is taken which includes a review of the system performance by the responsible piping design engineering organization and issuance of their findings.

During the visual inspections, special attention is directed to the following areas of piping/reactor system support components:

- Pipe whip restraint to pipe clearance at rated temperature

- Snubber expected movement and swing clearances at various temperatures including rated

- Control rod drive support structure to CRD housing gap at rated temperature

- Main steam piping penetration guide movement at rated temperature

- Reactor vessel seismic supports operability

  vessel to sacrificial shield stabilizers
  sacrificial shield to biological shield stabilizers

- Safety related process instrument piping movement such as:

  Reactor Vessel Level instrument piping
  Main Steam Flow instrument piping
  RCIC Steam Flow instrument piping

- Hot pipe containment penetration temperature profiles

The remote monitoring locations have not been finalized at present. Piping systems to be monitored have been tentatively identified that include the main steam, recirculation, feedwater, reactor core isolation cooling and, safety relief valve discharge line piping. The actual locations and selected piping systems will be established after an iterative selection process which consists of an assessment of the most advantageous measurement locations coupled with a review of possible monitoring locations. Both the responsible piping design organization and the Startup Test organization will thus cooperate to achieve an effective piping thermal movement monitoring program.

The finalized, detailed test procedure which delineates selected piping systems, applicable test acceptance criteria, visual inspection techniques, remote monitoring locations and required test conditions will be available on site for NRC inspection 60 days prior to commencement of the Startup Test Program on a schedule consistent with the preparation of other Startup Test procedures.

Summation - The Supply System will provide a reference in 3.9.2 to Chapter 14 of the FSAR. This item is closed.

NOTE: Chapter 14 will be revised per the above response.

QUESTION NO. 20
(3.9.2.1)

The applicant has not given a clear description of the acceptance criteria for steady-state piping vibrat.ons. The staff's position is that acceptance limi^s for vibration should be based on half the endurance limit as defined by the ASME Code at 10^6 cycles.

RESPONSE

For steady-state vibration, the piping peak stress due to vibration only (neglecting pressure) will not exceed 10,000 psi for Level 1 criterion and 5,000 psi for Level 2 criterion. These limits are below the piping material fatigue endurance limits as defined in Design Fatigue Curves in Appendix I of ASME Code for 10^6 cycles. The definitions of Level 1 and Level 2 criteria are clarified in the text revision attached.

Summation - This item is closed.

The FSAR will be changed to quantity Level 1 and Level 2 Criteria as indicated above.

NOTE: References in Chapters 3 and 14 will be verified.

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amplitude of displacements and number of cycles per transient of the main steam and recirculation piping are measured and the displacements compared with acceptance criteria. The deflections are correlated with stresses to verify the pipe stresses remain within Code limits. Remote vibration and deflection measurements are taken during the following transients:

a. Recirculation pump starts;

b. Recirculation pump at 100% of rated flow;

c. Turbine stop valve closure at 100% power;

d. Manual discharge of each S/R valve at 1,000 psig and at planned transient tests that result in S/R valve discharge.

3.9.2.1.5 Test Evaluation and Acceptance Criteria

The piping response to test conditions are considered acceptable if the organization responsible for the stress report reviews the test results and determines that the tests verify that the piping responded in a manner consistent with the predictions of the stress report and/or that the tests verify that piping stresses are within Code limits. To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. These criteria, designated Level 1 and 2, are described in the following paragraphs.

3.9.2.1.5.1 Level 1 Criteria

If in the course of the tests, measurements indicate that the piping is responding in a manner that would make test termination prudent, the test is terminated. Level 1 criteria establishes bounds on movement that, if exceeded, make a test hold or termination mandatory. The limits on movement are based on maximum allowable Code stress limits.

3.9.2.1.5.2 Level 2 Criteria

Conformance with Level 2 criteria demonstrates that the piping is responding in a manner consistent with the stress report

3.9.2.1.5.1 Level 1 Criterion

Level 1 establishes the maximum limits for the level of pipe motion which, if exceeded, makes a test hold or termination mandatory.

If the Level 1 limit is exceeded, the plant will be placed in a satisfactory hold condition, and the responsible piping design engineer will be advised. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

3.9.2.1.5.2 Level 2 Criterion

Level 2 specifies the level of pipe motion which, if exceeded, requires that the responsible piping design engineer be advised. If the Level 2 limit is not satisfied, plant operating and startup testing plans would not necessarily be altered. Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer. Depending upon the nature of such resolution, the applicable tests may or may not have to be repeated.

~~predictions. Failure to meet Level 2 criteria does not mean that the piping response is unsatisfactory; it means that the system is not responding in accordance with theoretical predictions and further analyses based on test results is necessary. Level 2 criteria is intended to screen out test results that are consistent with predictions and need no analytical review from those that must be evaluated.~~

3.9.2.1.6 Corrective Actions

During the course of the tests, the remote measurements are regularly checked to determine compliance with Level 1 criteria. If trends indicate that Level 1 criteria may be violated, the measurements are monitored at more frequent intervals. The test is held or terminated as soon as Level 1 criteria is violated. As soon as possible after the test hold or termination, the following corrective actions will be taken:

a. Installation Inspection. A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. If vibration exceeds criteria, the source of the excitation must be identified to determine if it is related to equipment failure. Action is taken to correct any discrepancies before repeating the test.

b. Instrumentation Inspection. The instrumentation installation and calibration are checked and any discrepancies corrected. Additional instrumentation is added, if necessary.

c. Repeat Test. If actions (a) and (b) identify discrepancies that could account for failure to meet Level 1 criteria, the test is repeated.

d. Resolution of Findings. If the Level 1 criteria is violated on the repeat test or no relevant discrepancies are identified in (a) and (b), the organization responsible for the stress report shall review the test results and criteria to determine if the test can be safely continued.

Item # 21 Snubber Testing

The Supply System's response to the letter from R. Tedesco to R. Ferguson "Preservice Inspection and Testing of Snubbers" dated March 6, 1981 is contained in the letter from J. Shannon to R. Tedesco, G02-81-313, "Preservice Inspection and Testing of Snubbers," dated September 24, 1981. This letter states that the Supply System will comply with all of the requirements contained in NRC letter of March 6, 1981.

Summation - This item is closed.