UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of	)	
	)	
SACRAMENTO MUNICIPAL UTILITY DISTRICT	)	Docket No. 50-312 (SP)
(Rancho Seco Nuclear Generating Station)	)	

AFFIDAVIT OF JOHN F. STOLZ

I John F. Stolz, being duly sworn, depose and state that:

1. I am an employee of the U. S. Nuclear Regulatory Commission (NRC). My present position is Chief, Operating Reactors Branch #4, Division of Licensing with the Office of Nuclear Reactor Regulation. I am currently responsible for managing the branch activities that include the review associated with the Rancho Seco facility. My professional qualifications are attached.

2. The purpose of this affidavit is to provide the current status of the NRC staff's review of SMUD's initial response to Item No. 6 of the ASLAB Memorandum and Order dated October 7, 1981 (ALAB-655). Item No. 6 requests the following information:

"SMUD and staff schedules for HPI analysis"

3. High Pressure Injection (HPI) Nozzle Analysis

The ASLAB's Memorandum and Order, dated October 7, 1981, required that SMUD and the staff provide the following HPI nozzle analysis:

"1) the maximum allowable number of thermal cycles on the HPI nozzles;

2) methods of detecting thermal cycle effects on the nozzles;

3) possible means of prolonging the useful life of the nozzles;

4) technical specifications or operating procedures that might reduce the use of the HPI without endangering the core."

1) Regarding 1) above, SMUD proposed to include 70 additional, manually initiated, HPI cycles in the existing design basis of 40 rapid depressurization (automatic initiation) and 40 test HPI cycles. SMUD's HPI nozzle analysis was submitted to the staff in a letter dated February 3, 1982 (enclosed). Mark L. Padovan's affidavit, transmitted to the ASLAB by Richard L. Black on January 6, 1982 indicated that our review of SMUD's initial HPI nozzle analysis submittal would be completed by February 8, 1982. Our review of SMUD's initial submittal has identified questionable assumptions in SMUD's analysis.

SMUD's HPI nozzle analysis assumed a total flow of 425 gpm through the HPI nozzles after manual HPI actuation. This flow rate is appropriate for the normal reactor coolant system (RCS) pressure of 2150 psig, but B&W plant operating experience shows that RCS pressure usually drops to 1750 psig before manual activation of HPI. At 1750 psig the HPI flow would be 760 gpm. The effect of the increased flow would be to increase the heat transfer coefficient between the nozzle metal and fluid.

Our review of SMUD's stress calculations cannot proceed beyond this point, due to a lack of information. Our February 3, 1982 letter (copy enclosed) requested SMUD to provide a revised analysis based on corrected flow rate assumptions and to provide a schedule for forwarding the information to us. We encouraged the licensee to expedite the schedule for this submittal. SMUD's February 4, 1982 letter (copy enclosed) indicated that the revised analysis would be submitted to the staff by July 1, 1982. After receipt of SMUD's revised analysis, we will need six weeks to review the submittal and provide additional testimony on our review to the ASLAB.

We expect the delays incurred by the reanalysis and additional reviews would not extend beyond the time period required for the plant to reach the existing 40 thermal cycle limit. This time margin is unchanged from that discussed in Mark L. Padovan's previous affidavit of January 6, 1982.

2) Regarding 2), SMUD's December 11, 1981 testimony to the ASLAB indicated that effects of thermal cycling the HPI nozzles may be detected by non-destructive examination techniques. SMUD also indicated that the nozzles are examined under their inservice inspection (ISI) program. The NRC staff conducted telephone conversations with the licensee regarding the effectiveness of their ISI program in detecting thermal cycling effects on the HPI nozzles. We expect to issue a request for additional ISI information to the licensee by March 15, 1982, and we will request a 30 day response from the licensee. The results of our review of SMUD's 30 day response will be provided to the ASLAB within 45 days of receipt of the requested information.

3) Regarding 3), in SMUD's December 11, 1981 testimony to the ASLAB, SMUD indicated that one possible means of prolonging the useful life of the HPI nozzles is to direct HPI flow, if manually initiated, solely to the normal make-up nozzle after a reactor trip. SMUD also indicated that the make-up nozzle had not been subjected to any additional thermal cycles since the procedure to direct manually initiated HPI to the make-up nozzle was in effect. In our February 3, 1982 letter to the licensee, we requested that SMUD provide us with their analysis that shows that no additional thermal cycling is occurring on the normal make-up nozzle during manually actuated HPI. SMUD's February 4, 1982 response indicated that the make-up nozzle is

not thermally cycled since the flow increase from the manual initiation of HPI does not involve a change in water temperature. The basis for the SMUD response is that the source of the water to the HPI was realigned the make-up tank instead of the borated water storage tank. Accordingly, in our February 8, 1982 letter (copy enclosed) we requested that SMUD submit their procedure for manually actuated HPI through the normal make-up nozzl to support SMUD's response on this issue. We will review their procedure and determine its acceptability. The results of our review will be forwarded to the ASLAB within six weeks after the receipt of the requested information.

4) Regarding 4), SMUD's testimony indicated that procedures have been changed at Rancho Seco such that the HPI nozzles are no longer thermally cycled by manual initiation of HPI following a reactor trip. As indicated in 3) above, the staff has requested SMUD's procedures for manually initiating HPI after a reactor trip, and we will foward the results of our review to the ASLAB.

An incident related to the effects of thermal cycling of the HPI make-up nozzle occurred at Crystal River (CR)-3 on February 5, 1982. This occurrence is being separately reported to the ASLAB shortly. Cracking was found in the normal make-up nozzle area and thermal cycling is thought to be a contributor to the cracking. We will also request SMUD to evaluate the HPI make-up nozzle at Rancho Seco with respect to the CR-3 nozzle cracking incident.

The above statements and opinions are true and correct to the best of my personal knowledge and belief.

John F. Stolz

John F. Stolz

Enclosures:
As Stated

Subscribed and sworn to before me
this 25th day of February 1982.

Marilyn Follenster, Notary Public
My Commission Expires: July 1, 1982

JOHN F. STOLZ

PROFESSIONAL QUALIFICATIONS

OPERATING REACTORS BRANCH NO. 4

DIVISION OF LICENSING

I am the Branch Chief of the Operating Reactors Branch No. 4 of the Division of Licensing, U. S. Nuclear Regulatory Commission. This Branch is responsible for the overall safety and environmental project management for assigned licensed operating power reactors that includes the review of technical and procedural aspects of proposed amendments to operating licenses. Operating plants having Babcock and Wilcox reactor systems have been assigned to this Branch.

I accepted an appointment with the technical staff of the NRC Regulatory organization in 1969 and was assigned as Senior Project Manager for safety review of Quad-Cities Station Units 1 and 2 and Mendocino Power Plant Units 1 and 2. From April 1972 to April 1980, I have had branch supervisory responsibility for the project management of licensing reviews of BWR 4/5 and 6 plants, PWR plants using Westinghouse, B&W, and Combustion reactors, and standard designs from General Electric, Westinghouse, Stone and Webster and Fluor Pioneer for preliminary design approvals. During 1974, I also participated in the staff review of the Reactor Safety Study that was subsequently released as WASH-1400. From April 1980 to March 1981, I was Branch Chief of the Systems Interaction Branch responsible for the development of criteria and methods that can be used to identify and evaluate common cause type of failures that can lead to adverse systems interactions.

I graduated from the City College of New York in 1942 with a Bachelor of Science Degree in Civil Engineering, obtaining a Master Degree in Civil Engineering from the University of Southern California in 1966. I have also taken additional graduate level courses in nuclear engineering, structural engineering and mechanical engineering at the University of California and New York University.

My experience following my undergraduate degree, from 1942 to 1951, included military service in the Air Force, a member of the Civil Engineering staff at the City College of New York, and structural engineering and field construction with several consulting engineering and industrial firms. From 1951 to 1953, I was employed with the consulting firm of Devenco Inc., where I worked on the structural design and analysis of the first nuclear powered submarines, Sea Wolf and Nautilus. In 1953, I joined the Atomic Energy Department of North America Aviation which subsequently became Atomics International Division of North American Rockwell Corporation. My starting position of Research Engineering involved design and analysis of reactor core and system components related to a sodium-graphite reactor development program. I subsequently became supervisor of a unit responsible for the design of supporting facilities for all nuclear power prototype plants and nuclear research facilities. In 1958, I was assigned as Project Engineer for the design of the plant, fuel handling and support systems for the Hallam Nuclear Power Facility, a 75 MWe sodium-graphite reactor plant at Hallam, Nebraska. In 1959, I was assigned as Project Engineer to modify the Organic Moderated Reactor Experiment at the National Reactor Test Station in Idaho, which involved redesign of the reactor core, pressure vessel, fuel handling, instrumentation and control and process systems. From 1962 to 1965, I held the position of Group Leader directing the work of four supervised units assigned to support the development, design and qualification of compact nuclear reactor systems (SNAP Program) specifically in the areas of testing facilities required to simulate nuclear spaceflight environment, stress analysis, and mechanical and electrical design on the SNAP systems. From 1965 to 1966, I supervised a process systems unit responsible for systems design and analysis supporting the company's development projects on sodium cooled reactors, organic-moderated heavy-water cooled reactor and desalinization systems. In 1966, I spent a year as Assistant Project Manager for the preliminary design and development of a 500 MWe sodium cooled fast-breeder reactor plant, specifically responsible for developing concepts, testing programs and budgetary plans for the overall plant and fuel handling. In 1967, I assumed a project management assignment with the Autonetics Division of North American Rockwell on system analysis studies in water and transportation systems, including management of the contract studies for the State of California on the systems analyses of operations and maintenance for the California State Water Project.

I am a member of the American Society of Civil Engineers, have been past Chairman of the Nuclear Structural and Materials Committee in the Structural Division of the Society, and am still an active member of that Committee and the Publications Committee. I am registered as a Civil Engineer in the State of California and a Professional Engineer in the State of New York.

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT ☐ 6201 S Street, Box 15830, Sacramento, California 95813; (916) 452-3211

February 3, 1982

DIRECTOR OF NUCLEAR REACTOR REGULATION
ATTENTION JOHN F STOLZ CHIEF
OPERATING REACTORS BRANCH 4
U S NUCLEAR REGULATORY COMMISSION
WASHINGTON DC 20555

DOCKET 50-312
RANCHO SECO NUCLEAR GENERATING
STATION UNIT NO 1
HIGH PRESSURE INJECTION NOZZLE
CYCLES

On December 11, 1981, the District responded to the Atomic Safety and Licensing Appeal Board's Memorandum and Order, ALAB-655. Since that time, we have had numerous discussions with your staff concerning the allowable number of transient cycles for the high pressure injection nozzles. We have provided the Babcock & Wilcox Company Field Change Package, "HPI Nozzle", Document No. 04-3370-00 FC-0174-00, for your review, and following additional questioning, B&W Calculation Nos. 32-1121811-00, "HPI Nozzle Usage Factor" and 32-1119809-01, "HPI Nozzle Usage Factor". These documents form the basis for including within the total number of allowable reactor trips (400) a specific provision for up to 70 actuations of high pressure injection following such a trip. (These 70 cycles are in addition to the 80 cycles included in the original stress analysis.) As noted on page 4 of Document 32-1121811-00, the original 80 cycles resulted in a usage factor of only .44, which demonstrates the conservation in the analysis. These documents are attached to this letter for your reference.

As discussed in our response to ALAB-655, these analyses do not explicitly address the Appeal Board's question concerning the ultimate number of cycles these nozzles are physically capable of safely withstanding. Our response provided the justification for not responding to this question. We feel, however, that these additional cycles demonstrate that safety limits are not being approached, and that no safety concern exists due to cycle usage at Rancho Seco Unit No. 1. Your staff has requested additional information for their use in reviewing these calculations. The workload within the Babcock & Wilcox Company (due to such factors as the Thermal Shock issue) precludes this information being generated and submitted for your review prior to July 1, 1982.

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This information relates primarily to techniques used in the original Rancho Seco Stress Analysis and we feel this information is irrelevant to the Appeal Board's concern over high pressure injection nozzle usage, but should satisfy your needs in reviewing this change to the Rancho Seco Administrative Procedures. We expect to receive a written request for information from you which will confirm the information to be supplied in July of this year.

John J. Mattimoe

John J. Mattimoe
Assistant General Manager
and Chief Engineer

Attachments

cc: Mr. Tom Baxter - Shaw, Pittman, Potts & Trowbridge
Mr. Dave Holt - Babcock & Wilcox Company
Mr. Mark Padovan - Nuclear Regulatory Commission

T-143

DOCUMENT SUBMITTAL FORM

SMUD
6201 S Street
Sacramento, CA 95813

Mr. R.J. Rodriguez

DATE 10/22/80
B&W CONTRACT NO. 620-0011
CUST. SMUD
CUST. ORDER NO. GRS95243
SHEET 1 OF 1
TYPE DOC Field Change Package

pkgs. ☐ UNDER SEP COVER ☒ ENCLOSED
☐ UNDER SEP COVER ☐ ENCLOSED

RENCES

☐ FOR COMMENTS & APPROVAL BY
☐ FOR INFORMATION ONLY
☐ HAS BEEN REVISED AS PER YOUR PREVIOUS COMMENTS OF
☒ FOR FINAL DISTRIBUTION
☐ FURTHER EXPLANATION REQUIRED SEE BELOW OR ATTACHMENTS

DOCUMENT DESCRIPTION

SUPPLIER/CUSTOMER DOC. NO.	B&W DOC. NO.	PART/ TASK NO.	GRP. NO.	SEQ. NO.	DOC. TITLE
	04-3370-00 FC-0174-00	50	01		HPI Nozzle

R.P. Oubre'
D.G. Raasch
R.A. Dieterich
J.V. McColligan
L.G. Sweiger
J.T. Janis

DC Holt
D.C. Holt
Engineering Product Manager

FIELD CHANGE AUTHORIZATI(FC-0174-00 (ABCOCK & WILCOX

CUSTOMER	ORGINATOR	DATE	CONT. NO.	DOC. I.D.	PCA NO.	REV. NO.
SMUD	K.C. Smith	6-11-80	620-0011	04-3370-00		

SUPPLIER	P.A. NO.	PART NO., OR TASK NO.	GROUP NO.	SEQ. NO.
Babcock & Wilcox		50	001	001

TITLE (MAX 30 CHARACTERS)
HPI Nozzle

DESCRIPTION AND JUSTIFICATION OF CHANGE:

It has been reported that the SMUD operators have been starting a second make-up pump following each reactor trip to obtain additional make-up flow. This was done to prevent loss of indicated pressurizer level during the transient. All four of the high pressure injection nozzles have been used for these occurrences and the three nozzles which are not continuously adding make-up flow receive a thermal shock from the cold BWST water.

The contract Functional Specification, CS(F)3-92/NSS-11/0372, defines forty cycles of high pressure injection actuation and forty cycles of high pressure injection testing. At SMUD's present usage rate, 31 cycles, the nozzles would exceed their allowable cycles before the end of their forty-year life.

B&W has performed a HPI nozzle stress analysis evaluation of the actual experienced transients (actual transients are less severe than functional specification transients) to determine their actual fatigue life expectancy. In reviewing the SMUD HPI Design Analysis, it was discovered that the original analysis was performed as a generic analysis and the fatigue analysis was performed to a later, more restrictive design code. In analyzing the HPI nozzle to the contractual design code, 1968 draft USAS, B31.7 with June 1968 errata, the number of allowable HPI actuations can be approximately doubled. The re-analysis must conform to the requirements of ASME Section XI which now applies to the site and allows the use of earlier codes.

This Field Change Authorization is being written to document the increase in allowable HPI system actuation cycles and therefore no hardware changes are required.

REASON FOR CHANGE:

- [x] SITE PROBLEM
- [ ] ENGINEERING REQUIREMENT
- [x] CUSTOMER REQUEST
- [ ] IMPROVEMENT
- [ ] OTHER (SPECIFY) ____

SPR. DOC. ~~xxxxxxxxxx~~ 13-11-361-00

TASKS AFFECTED

TASK NO.	TASK TITLE	NO. DOCS	COMPLETED (T.E. SIGN)	DATE
17	RCS Comp.	0	[signature]	10-[illegible]
50A	RC Pipe-Sys.	Des. 1	K.C. Smith	[illegible]
50	RC Piping	1	[signature]	[illegible]
10	Proj. Engr.	0	[signature]	[illegible]
09	Power Systems & Controls	1	[signature]	[illegible]

CONCUR TO PROCEED	APPROVALS NAME	DATE
KCS	[signature]	10/10/80
CBR	[signature]	10/10/80
CJH	[signature]	10/13/80
CH	[signature]	10/14/80

CUSTOMER/CUSTOMER AGENT DISPOSITION OF FIELD CHANGE:

- [ ] IMPLEMENTED
- [ ] NOT IMPLEMENTED

REF. ____

PROJECT MANAGER ____ DATE ____

AUTH. CHANGE NO. | PCA NO.

FIELD CHANGE AUTHORIZATION - AFFECTED DOCUMENTATION

CUSTOMER: SMUD CONTRACT NO. 620-0011 FCA NO. 04-3370 REV. NO. 00

DOCUMENT TITLE	*DOCUMENT NUMBER	DOCUMENT TITLE	*DOCUMENT NUMBER
SPECIFICATION FOR REACTOR COOLANT SYSTEM PIPING	CS(F)3-37/NSS-11/0372 DTD. 3-22-72		
SPECIFICATION FOR REACTOR COOLANT SYSTEM	CS(F)-3-92/NSS-11/0372 DTD. 3-22-72		
STRESS REPORT FOR PRIMARY PIPING (620-0011-50)			

*REVISION LEVEL SAME AS DOCUMENT DELIVERED TO CUSTOMER

PDS-21048A-4(12-76)

FIELD CHANGE AUTHORIZATION-CONTINUATION

☐ EXPEDITE
☒ NORMAL

REFERENCE DOCUMENT TITLE	REF DOCUMENT NO.
Specification for Reactor Coolant System Piping	CS(F)3-37/NSS-11/0372
	SHEET 1 OF 1

DESCRIPTION OF CHANGE:

1. Revise paragraph 1.1 of Appendix 1 of the reference document to read as follows:

   1.1 General Functional Specification for Reactor Coolant System Components, Specification No. CS(F)-3-92/NSS-11/0672 as modified by Field Change Authorization 04-3370-00.

2. Add paragraph 2.2 in Appendix 1 of the reference document to read as follows:

   2.2 ASME Boiler and Pressure Vessel Code

   a. Section XI, Inservice Inspection — 1974 Edition with Summer 1975 Addenda

   The reanalysis of the HPI nozzle is to be in accordance with the requirements of the code listed above.

The changes to the reference document above are certified to be correct and complete and in compliance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

Thomas C. Helms PROFESSIONAL ENGINEER

Virginia
STATE

06223
REGISTRATION NO.

10/10/80
DATE

COMMONWEALTH OF VIRGINIA
Thomas C. Helms
06223
PROFESSIONAL ENGINEER

REVISED REQUIREMENTS (CONTINUATION SHEET)		SUPPLIER REVISIONS		TASK NO.	TASK TITLE
PREPARED BY	DATE	REVIEWED BY	DATE	50A	R.C. Pipe - System Designer
				FCA NO.	
REVIEWED BY	DATE				
A. Lewis	10/10/80			04-3370-00	
APPROVED BY	DATE				

PDS-21048A-4(12-76)

FIELD CHANGE AUTHORIZATION-CONTINUATION

☐ EXPEDITE
☒ NORMAL

REFERENCE DOCUMENT TITLE	REF DOCUMENT NO.
GENERAL FUNCTIONAL SPECIFICATION FOR REACTOR COOLANT SYSTEM	CS(F) 3-92/NSS-11/0672 SHEET 1 OF 6

DESCRIPTION OF CHANGE:

Change the RCS Functional Specification as per FCA continuation sheets 2 through 6.

The changes to be incorporated in this revision (03) of the Spec include:

1. A revision to Section 6.9, Transient 8 - Reactor Trip. Specifically, paragraphs will be added to Sections 6.9.2 and 6.9.3.1 of the text to provide for manual actuation of the high pressure injection system after a reactor trip.

2. Table 2 will be revised to include 70 design cycles for the above noted change.

3. Figure 8-14 will be added to the spec.

REVISED REQUIREMENTS (CONTINUATION SHEET)	SUPPLIER REVISIONS	TASK NO.	TASK TITLE
PREPARED BY R. B. Park DATE 8/15/80	REVIEWED BY DATE	09	POWER SYSTEMS + CONTROLS
REVIEWED BY DATE		FCA NO. 04-3370-00	
APPROVED BY DATE			

Figure No.	Title
8-8	Surge Line Temperature and Flow.
8-9	Makeup and Spray Flow Rate and Temperature.
8-10	Reactor Trip, Cooldown and Normal Power Recovery - Primary System.
8-11	Reactor Trip, Cooldown and Normal Recovery - Secondary System.
8-12	Reactor Trip from Full Power for Trans. 8A&B - Steam Gen. Conditions.
2 8-12A	Feedwater Conditions Following Reactor Trip (8A&B).
8-13	Loss of Feedwater Reactor Trip-Steam Feedwater System Parameters.
BP /5/80 8-14	MAKEUP FLOW AND TEMPERATURE - MANUAL ACTUATION OF HIGH PRESSURE INJECTION SYSTEM
	Transient No. 9 - Rapid Depressurization
9-1	Reactor Coolant Conditions.
9-2	Surge Flow and Temperature.
9-3	Makeup and Spray Flow and Temperature.
9-4	Feedwater and Reactor Coolant Flow Range.
9-5	Steam Temperature and Test.
9-6	Steam Generator Pressure.
9-7	Feedwater Temperature Range.
	Transient No. 10 - Change of Flow
10-1	Reactor Coolant Temperature.
10-2	Surge Line Water Temperature and Flow Rate.
10-3	Makeup and Spray Line Flow and Water Temperature.
10-4	FW-Steam System Parameters for Steam Generator with Two Reactor Coolant Pumps.
10-5	FW-Steam System Parameters for Steam Generator with One Pump.
	Transient No. 11 - Rod Withdrawal Accident
11-1	Reactor Coolant Temperature and Pressurizer Liquid Level and Pressure.
11-2	Spray and Makeup Temperature and Flow Rate.

2 in Transient 3 (Plant loading 8% to 100%). These return to power events are included in the 18000 cycles specified for Transient 3.

2) 56 Reactor trip events, followed by cooldown as described in Transient 1B (Cooldown), followed by heatup as described in Transient 1A (heatup) and Transient 3 (Plant Loading). These 56 heatup and cooldown cycles following reactor trip are included in the 240 cycles specified for Transient 1.

Insert indicated addition shown on following page. RBP 8/15/80

6.9.3 Transient Data

6.9.3.1 Reactor Coolant System

Reactor Coolant System conditions for the reactor trip transients are presented in the following figures:

Type A Reactor Trip - Figures 8-1, 8-1A, 8-2, 8-3
Type B Reactor Trip - Figures 8-4, 8-4A, 8-5, 8-6, 8-10
Type C Reactor Trip - Figures 8-7, 8-7A, 8-8, 8-9

Insert indicated addition shown on following page. RBP 8/15/80

6.9.3.2 Steam Generators

Steam Generator conditions for the reactor trip transients are presented in Figures 8-11, 8-12, 8-12A and 8-13.

2 6.10 Transient 9 - Rapid Depressurization (Upset Condition)

6.10.1 General Description

Rapid depressurization is a short term, rapid cooling of the Reactor Coolant System by the steam generators in order to reduce the reactor coolant system pressure to a value less than the design pressure (1065 psia) of the steam generators within 15 minutes. The objective of the rapid depressurization is to isolate a tube leak.

The initial conditions at the start of the transient are assumed to be hot standby with decay heat removal by the steam generators dumping steam to the condenser. The turbine bypass control pressure is assumed to be 1050 psia. This gives an average reactor
2 coolant system temperature of about 558.7°F.

When average reactor coolant temperature has been reduced to 500°F, and reactor coolant pressure is equal to or less than 1065 psia (15 minutes), normal cooldown as described in Transient 1B is initiated.

6.10.2 Cycles

1 The number of rapid depressurization events for design shall be 40. A complete cycle consists of a) power reduction from 100% power as described in Transient 4 (Plant Unloading); b) Rapid de-

Insert in §6.9.2
RBP
8/15/80

Manual Actuation of High Pressure Injection System After Reactor Trip

Occasionally after a reactor trip, it may be necessary to manually actuate the high pressure injection system in order to obtain additional makeup flow. This should only be done to prevent the loss of indicated pressurizer level during the transient. For the total of 400 reactor trip events, an allowance of 70 cycles of high pressure injection system actuations shall be considered for design purposes.

Insert in §6.9.3.1
RBP
8/15/80

Manual Actuation of High Pressure Injection System After Reactor Trip - Figure 8-14

Table 2

Operating Transient Cycles

	Transient Number	Transient Description (ASME Category)		Design Cycles	Figure Nos.
2	1A	Heatup from 70°F to 8% Full Power (Normal)		240	1A-1 - 1A-[illegible]
2	1B	Cooldown from 8% Full Power (Normal)		240	1B-1 - 1B-[illegible]
	2	Power change 0 to 15% and 15 to 0% (Normal)		1440	2A-1 - 2A-[illegible] 2B-1 - 2B-5
2	3	Power Loading 8% to 100% power (Normal)		18,000	3-1 - 3-5
2	4	Power Unloading 100% to 8% power (Normal)		18,000	4-1 - 4-5
	5	10% Step Load Increase (Normal)		8,000	5-1 - 5-4
	6	10% Step Load Decrease (Normal)		8,000	6-1 - 6-5
	7	Step Load Reduction (100% to 8% Power)(Upset)			
		Resulting from turbine trip	160		7-1 - 7-5
		Resulting from electrical load rejection	150		
		Total		310	
	8	Reactor Trip (Upset)		--	
		Type A	40		8-1 - 8-13
		Type B	160		
		Type C	88		
		Trips included in transient numbers 11, 15 16, 17 & 21	112		
		Total *		400	
		* MANUAL ACTUATION OF HIGH PRESSURE INJECTION SYSTEM AFTER REACTOR TRIP		70	8-14
1	9	Rapid Depressurization (Upset)		[illegible]	[illegible]
	10	Change of Flow (Upset)		20	10-1 -10-5
	11	Rod Withdrawal Accident (Upset)		40	11-1 -11-4
2	12	Hydrotests (Test)		35	
	13	Steady-State Power Variations (Normal)		∞	13-1
	14	Control Rod Drop (Upset)		40	14-1 -14-4
	15	Loss of Station Power (Upset)		40	15-1 -15-7
	16	Steam Line Failure (Faulted)		1	16-1 -16-3
	17A	Loss of Feedwater to One Steam Generator (Upset)		20	17A-1 - 17A-[illegible]
	17B	Stuck Open Turbine Bypass Valve (Emergency)		10	17B-1 - 17B-[illegible]

RBP
8/15/80

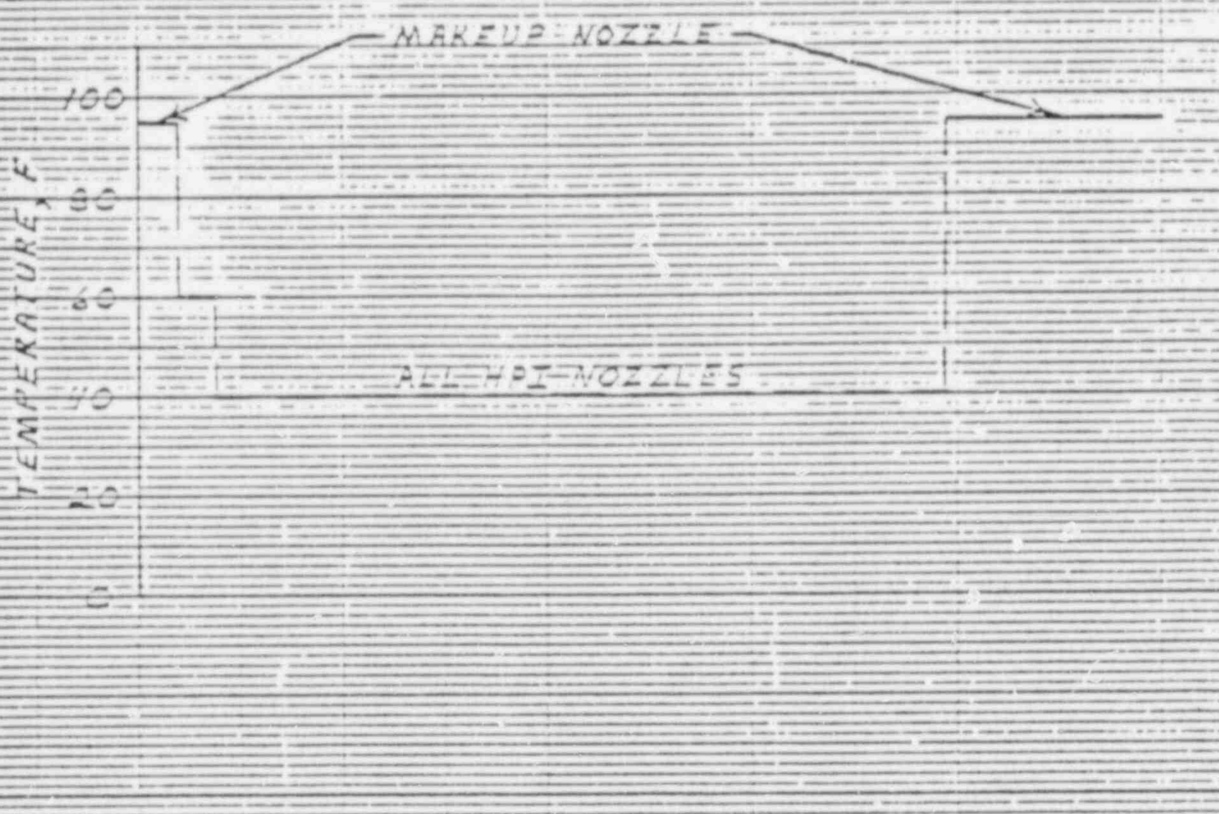

FIGURE 8-14

TRANSIENT NO. 8 (REACTOR TRIP)

MAKEUP FLOW AND TEMPERATURE - MANUAL

ACTUATION OF HIGH PRESSURE INJECTION SYSTEM

SPECIFICATION NO

PDS-21048A-4(12-76)

FIELD CHANGE AUTHORIZATION-CONTINUATION

☐ EXPEDITE
☒ NORMAL

REFERENCE DOCUMENT TITLE	REF DOCUMENT NO.
STRESS REPORT FOR PRIMARY PIPING (620-0011-50)	NONE
	SHEET 1 OF 1

DESCRIPTION OF CHANGE:

The following documents are added as addenda to the original stress report:

32-1119809-01 HPI Nozzle Usage Factor

32-1121811-00 HPI Nozzle Usage Factor

I certify that the two (2) above mentioned analysis when taken as a whole justify the modifications to be made under the authority of FCA No. 04-3370-00 and that these two (2) analysis when considered as a whole meet the requirements imposed by the applicable Equipment Specification, as modified by FCA No. 04-3370-00. Attested to this date: October 10, 1980

COMMONWEALTH OF VIRGINIA
GEORGE J. VAMES
CERTIFICATE NO.
7507
CERTIFIED PROFESSIONAL ENGINEER

G J Vames

George J. Vames
Babcock & Wilcox
Nuclear Power Generation Division
Lynchburg, Virginia

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Virginia State Board
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Engineers

REVISED REQUIREMENTS (CONTINUATION SHEET)		SUPPLIER REVISIONS		TASK NO.	TASK TITLE
PREPARED BY	DATE	REVIEWED BY	DATE	50	Reactor Coolant Piping
R R Schaefer	10/9/80	R B Reynolds	10/10/80		
REVIEWED BY	DATE			FCA NO.	
G. J. Vames	10-10-80	R Chevins	10/10/80	04-3370-00	
APPROVED BY	DATE				

DOCUMENT RELEASE NOTICE (DRN)

CONTRACT/STANDARD NO.	RELEASE DATE	PAGE
620-0011	OCT 13 1980	1 of 1

PART-MARK/TASK-GROUP-SEQ.	B&W DOCUMENT NO.	DOCUMENT TITLE	PUL STAT.	REL. NO.
50/001/001	32-1121811-00	HPI NOZZLE USAGE FACTOR	~	~

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CALCULATION DATA/TRANSMITTAL SHEET

DOCUMENT IDENTIFIER CALC. 32 - 1121811 - 00

TRANS. 86 - -

TYPE: ___ RESEARCH & DEVELOPMENT ___ SAFETY ANALYSIS REPORT ___ NUC. SERV. INPUT ___ DESIGN RQMT. X DESIGN VERIF. ___ OTHER

TITLE HPI Nozzle Usage Factor

PREPARED BY R R Schaefer REVIEWED BY Alvin D. McKim

TITLE Senior Engineer DATE 10/9/80 TITLE Supervisory Engineer DATE 10/9/80

PURPOSE:

To determine if the HPI nozzle can withstand 70 HPI actuations following a reactor trip condition. These will be in addition to the 40 rapid depressurization and 40 test transient cycles specified in the contract functional specification CS(F)3-92/NSS-11/0372.

SUMMARY OF RESULTS (INCLUDE DOC. ID'S OF PREVIOUS TRANSMITTALS & SOURCE CALCULATIONAL PACKAGES FOR THIS TRANSMITTAL)

The HPI Nozzle can withstand 40 test transient cycles, 40 rapid depressurization transients and 70 reactor trip transients as described in FCA 04-3370-00.

DISTRIBUTION

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GENERAL CALCULATIONS

PURPOSE

The purpose of this calculation is to amend CDS 32-1119809-00, Ref (1). Ref (1) justified an additional 70 cycles of rapid depressurization transient. This calculation will justify 70 cycles of HPI actuation following a reactor trip instead of 70 additional rapid depressurization transient cycles was done in Ref (1).

DISCUSSION

CDS 32-1119809-00, Ref. (1), determined the maximum allowable number of rapid depressurization transient cycles. This calculation was based upon the assumption that FCA 04-3370-00 would characterize the 70 additional HPI transients as required by SPR #13-11-361-00, Ref. (5) as rapid depressurization transients.

When FCA 04-3370-00 was issued the additional 70 HPI transients were assumed to follow a reactor trip transient. Thus, this calculation will show that the HPI can withstand 70 cycles of HPI Actuation following a reactor trip transient instead of 70 additional rapid depressurization cycles as shown in Ref. (1).

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GENERAL CALCULATIONS

32 DOC I.D. | 1121811 SERIAL NUMBER | 00 REV.

RESULTS

MAXIMUM PRIMARY + SECONDARY STRESS INTENSITY RANGE

$S = 128.9$ KSI $> 3S_m$ (PG. 5, REF [1])

THEREFORE AN ELASTIC-PLASTIC FATIGUE ANALYSIS WAS PERFORMED.

USAGE FACTOR = 0.441 < 1.0 FOR 40 TEST TRANSIENTS AND 40 RAPID DEPRESSURIZATION TRANSIENTS. (PG. 5, REF. [1])

USAGE FACTOR = 0.26 FOR 70 HPI ACTUATIONS FOLLOWING A REACTOR TRIP TYPE TRANSIENT.

TOTAL CUMMULATIVE USAGE FACTOR IS;

$U_T = 0.44 + 0.26 = 0.70 < 1.0$
(PG. 16)

CONCLUSION

THE HPI NOZZLE CAN WITHSTAND 40 TEST TRANSIENT AND 40 RAPID DEPRESSURIZATION TRANSIENT CYCLES. IN ADDITION THE HPI NOZZLE CAN WITHSTAND 70 ADDITIONAL CYCLES OF HPI ACTUATION FOLLOWING A REACTOR TRIP TRANSIENT AS DESCRIBED IN REF. [7].

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CONT NO 620.00
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THERMAL PARAMETERS

PAGE 4-2 OF REF [2] GIVES THE FILM COEFFICIENT VALUES USED IN THE ANALYSIS FOR THE RAPID DEPRESSURIZATION TRANSIENT. THIS FILM COEFFICIENT CALC. ASSUMED A FLOW RATE OF 425 GPM THROUGH EACH NOZZLE. THE ACTUAL FLOW IS 425 GPM TOTAL FLOW SPLIT EQUALLY BETWEEN ALL 4 HPI NOZZLES. THUS, THE FILM COEFFICIENT SHOULD HAVE BEEN;

IN BRANCH (PER PG. 4-2, REF. [2]

FOR 60°F WATER

$C_p = 1.$ $\rho = 62.34\ LB/FT^3$ $\mu = 2.71$
$K = 0.344$ $D = 0.177\ FT$ $A = 0.0246\ FT^2$

$$hD^{0.2} = 0.023K\left(\frac{G}{\mu}\right)^{0.8}\left(\frac{C_p\mu}{K}\right)^{0.4}$$

$$G = \frac{(425\ GPM/4)\left(60\ \frac{MIN}{HR}\right)\left(0.1337\ \frac{FT^3}{GAL}\right)\left(62.34\ \frac{LB}{FT^3}\right)}{0.0246\ FT^2}$$

$G = 2159948\ LB/FT^2\text{-}HR$

$$hD^{0.2} = 0.023\left[\frac{2159948}{2.71}\right]^{0.8}\left[\frac{(1.0)(2.71)}{0.344}\right]^{0.4}(0.344)$$

$hD^{0.2} = 950.8$

$h = 1345\ BTU/HR\text{-}FT^2\text{-}°F$

$h = 4100\ BTU/HR\text{-}FT^2\text{-}°F$ (USED IN REF'S. [1 & 2])

THE FILM COEFFICIENT USED IN REFERENCES [1 & 2] FOR THE TEST TRANSIENT IS 1300 BTU/HR-FT²-°F. BOTH OF THESE TRANSIENTS INITIATE AT 550°F.

FROM PAGE 4-11, REF. [2], ITERS 3, 6 & 8 EACH OCCUR IN THE FIRST TEN SECONDS. FROM PAGE 6 OF REF. [1] THE MAXIMUM STRESS INTENSITY OCCURS AT ITERATION 8 WHICH HAS A TIME POINT EQUAL TO 5.7 SECONDS. THUS, THE MAXIMUM STRESS INTENSITY AT THE NOZZLE END FOR THE TEST TRANSIENT IS FULLY DEVELOPED BEFORE THE TRANSIENT ENDS.

THUS, THE STRESS INTENSITY CALCULATED FOR THE TEST TRANSIENT INITIATION IS ESSENTIALLY WHAT THE ACTUAL RAPID DEPRESSURIZATION TRANSIENT SHOULD HAVE BEEN. THE ONLY SIGNIFICANT DIFFERENCE BETWEEN THE TWO TRANSIENTS IS THE RETURN TO POWER AND TEMPERATURE. THIS PORTION OF THE ANALYZED RAPID DEPRESSURIZATION TRANSIENT IS CORRECT.

HOWEVER, FOR THE PURPOSES OF THIS CALCULATION, THE TEST TRANSIENT CYCLES (40 CYCLES) AND THE RAPID DEPRESSURIZATION TRANSIENT CYCLES (40) WILL BE JUSTIFIED BY CDS-32-1119809-00. THEY HAVE A CUMMULATIVE USAGE FACTOR OF U=0.441, AS SHOWN ON PAGE 23 OF REF. [1].

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BABCOCK & WILCOX
GENERAL CALCULATIONS

THE ADDITIONAL 70 CYCLES OF HPI ACTUATION FOLLOWING A REACTOR TRIP WILL BE JUSTIFIED BY ADJUSTING THE STRESS INTENSITY RANGES DISCUSSED FOR THE TEST AND RAPID DEPRESSURIZATION TRANSIENTS PREVIOUSLY. CALCULATIONS AND DATA ARE PRESENTED ON THE FOLLOWING PAGES FOR EACH TRANSIENT JUSTIFICATION.

CUSTOMER	PROP NO / FILE NO
SUB	DWG NO / COMP NO
CONT NO 620-00	GROUP NO
CALC BY RRS DATE 1/0/80	SHEET NO 7 of 15

TABLE OF REACTOR TRIP CONDITIONS

(PER REF. 3)

TRANSIENT	HIGHEST TEMP	INITIAL TEMP. AT 45 SEC	RETURN TEMP	PRESSURE AT 45 SEC	RETURN PRESSURE
8A	570	550	550	2200	2150
8B	575	550	550	1850	2200
8C	590	575	565	2200	2200
11	592	585	540	2300	1800
15	575	550	550	2200	2150
16	575	560	560		
17A	577	565	550	2100	1850
17B	565	565	560	2450	2200
21	575	575	300	1600	0

FROM THE ABOVE TABLE, TRANSIENT 11 EXHIBITS THE HIGHEST INITIAL TEMPERATURE AND PRESSURE. THE HIGHEST RETURN TEMPERATURE IS 565. THE JUSTIFICATION WILL CONSERVATIVELY USE THE MAXIMUM VALUES.

TO JUSTIFY THE ANALYSIS OF THE 70 CYCLES FOR THE HIGHER STARTING TEMPERATURE, THREE ANALOGOUS THERMAL STRESS RUNS WERE MADE. B&W COMPUTER PROGRAM P91232 WAS USED. USING THIS SLAB TEMPERATURE PROGRAM, THE HPI RAPID DEPRESSURIZATION AND TEST TRANSIENTS WERE ANALYZED AS TO THEIR EFFECT ON THE NOZZLE END THERMAL STRESS WITHOUT DISCONTINUITY EFFECTS. THE RESULTS OF THIS ANALYSIS ARE CONTAINED ON REF FICHE SHAFØGZ AND ARE TABULATED BELOW.

CASE	INITIAL TEMP	ΔT (T-60)	FILM COEFF.	ΔT RATIO (1)	LINEAR THERMAL STRESS	STRESS RATIO (2)
1	550	490	4100	1.00	73106	1.00
2	570	510	4100	1.04	76023	1.04
3	585	525	4100	1.07	78204	1.07
4	550	490	1300	1.00	57131	1.00
5	570	510	1300	1.04	59350	1.04
6	585	525	1300	1.07	61005	1.07

NOTES (1) ΔT RATIO $= \Delta T / \Delta T_1 = \Delta T / 490$

(2) STRESS RATIO $= S_{2 \text{ OR } 3} / S_1$

OR $= S_{5 \text{ OR } 6} / S_4$

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GENERAL CALCULATIONS

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TO JUSTIFY THE ANALYSIS OF THE 70 CYCLES FOR A HIGHER RETURN TEMPERATURE, THE FOLLOWING RESULTS FROM P91232 ARE TABULATED. THESE RESULTS ARE FROM REF. FICHE SHRFØGZ.

CASE	RETURN TEMP	ΔT (T-40)	FILM COEFF.	ΔT RATIO (1)	LINEAR THERMAL STRESS	STRESS RATIO (2)
1	550	510	35	1.00	-5658	1.00
2	570	530	35	1.04	-5876	1.04
3	585	545	35	1.07	-6039	1.07

FROM THE TABULATIONS ON PAGE 6 AND ABOVE, THERE IS SUFFICIENT JUSTIFICATION FOR USING A ΔT RATIO METHOD TO JUSTIFY THE DIFFERING TEMPERATURE FROM SPR #13-11-361-00 VERSUS THE RAPID DEPRESSURIZATION TRANSIENT ANALYZED. IN ORDER TO CALCULATE A STRESS INTENSITY RANGE FOR THE REACTOR TRIP FOLLOWED BY HPI ACTUATION, THE FOLLOWING METHOD WILL BE USED.

THERMAL GRADIENT STRESSES FROM PAGE 6 OF REF. [1] WILL BE MULTIPLIED BY THE RATIO, 1.07. PRESSURE STRESS WILL APPROPRIATELY ADJUSTED.

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THERMAL DISCONTINUITY

THE THERMAL DISCONTINUITY EFFECTS WILL ALSO BE LOWER WITH THE LOWER FILM COEFFICIENTS BEING USED. THE AXIAL ΔT WOULD BE SMALLER. SINCE, THE LONGITUDINAL AND HOOP STRESSES ARE ADDITIVE, AS LONG AS THE LINEAR RADIAL GRADIENT STRESSES ARE LESS, THE DISCONTINUITY STRESSES WILL BE LESS.

PRIMARY PLUS SECONDARY STRESS (INSIDE)

USING TEMPERATURES AND PRESSURE FROM PAGE 6 AND STRESS FOR ITER 8.

PRESSURE STRESS = (4.84 KSI)(2300 PSI/2200 PSI)
= 5.06 KSI (INSIDE)

THERMAL EXPANSION = 5.1 KSI (INSIDE)

THERMAL STRESS = 78.8 (1.07) = 84.32 KSI

TOTAL STRESS = 5.06 + 5.1 + 84.32 = 94.48 KSI

ITERATION 6335 (MAXIMUM THERMAL STRESS)

PRESSURE STRESS = (1.54 KSI)(1800 PSI / 700 PSI)
= 3.96 KSI (INSIDE)

THERMAL EXPANSION = 5.1 KSI (INSIDE)

THERMAL STRESS = (-7.7 KSI)(565/500)
= -8.7 KSI (INSIDE)

TOTAL STRESS = 3.96 + 5.1 - 8.7 = 0.36 KSI

MAXIMUM PRIMARY + SECONDARY STRESS INTENSITY RANGE

σ_{P+S} = 94.48 + 20.3E = 94.48 + 20.0 = 114.48 KSI

114.48 KSI > $3S_m$ = 51.2 KSI

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GENERAL CALCULATIONS

PRIMARY PLUS SECONDARY STRESS (OUTSIDE)

ITER 8

PRESSURE STRESS = 1.32 (2300/2200) = 1.35 KSI

THERMAL EXPANSION = 7.3 KSI

THERMAL GRADIENT STRESS = -71.9 (1.07) = -76.93 KSI

TOTAL STRESS = -68.3 KSI

ITER 6338

PRESSURE STRESS = 0.42 (1800/700) = 1.08 KSI

THERMAL EXPANSION = 7.3 KSI

THERMAL GRADIENT STRESS = 7.0 (565/500) = 7.91 KSI

TOTAL STRESS = 1.08 + 7.30 + 7.91 = 16.3 KSI

MAXIMUM P + S STRESS INTENSITY RANGE

σ_{P+S} = -68.3 + 16.3 + 28.1 = 112.7 KSI

PEAK STRESS INTENSITY (INSIDE)

ITER 8 (STRESS FROM PG. 12 OF REV 00)

$\sigma = 5.5(2300/2200) + 6.5 + 113.3(1.07)$

$\sigma = 133.48$ KSI (INSIDE)

ITER 6338

$\sigma = 1.8(1800/700) + 6.5 - 9.6(565/500)$

$\sigma = -0.3$ KSI (INSIDE)

TOTAL STRESS INTENSITY RANGE

$S_{L-R} = 133.48 + 0.3 = 133.8$ KSI

$\bar{Y}^{\text{RANGE}}_{\text{PEAK}} = 133.8$ KSI $+ 2OBE = 133.8 + 25.5$

$= 159.3$ KSI

NOTE: OBE STRESS INTENSITY RANGES WERE ALSO ADDED TO THE ANALYSIS OF 70 ADDITIONAL CYCLES. THIS PROVIDES ADDITIONAL CONSERVATISM TO THE ANALYSIS.

ELASTIC-PLASTIC ANALYSIS FOR THE 70 ADDITIONAL HPI TRANSIENTS

FOLLOWING THE PROCEDURE OF REF [5], PARAGRAPH F-105.2.7, THE FINAL PEAK ALTERNATING STRESS INTENSITY, σ_{ALT}, IS:

$$\sigma_{ALT} = \frac{1}{2} K_f K_\epsilon S_{Pij}^{(n)}$$

$$K_f = K_t + A(K_t - 1)$$

$$K_t = \sigma_{PEAK} / \sigma_{P+S} = 159.3/114.5 = 1.39$$

$$A = 0.7 \quad (\text{PG } 17, \text{ REF. } [1])$$

$$K_f = 1.39 + 0.7(1.39 - 1.0) = 1.65$$

K_ϵ IS FROM FIG. F-105(a), REF. [5]

$$\frac{S_n}{3S_m} = \frac{114.5}{51.3} = 2.23$$

$$|Q_m| = \left| \frac{(94.48+10.0)+(-68.3+10.0)}{2} - \frac{(0.36-10.0)+(16.3-1\ldots}{2} \right.$$

$$|Q_m| = 23.09 - (-1.67) = 24.76 \text{ KSI}$$

$$Q_b = (104.48 - 23.09) - [-9.64 - (-1.67)]$$

$$Q_b = 81.39 + 7.97 = 89.36 \text{ KSI}$$

$$\frac{Q_m}{Q_m + Q_b} = \frac{24.76}{24.76 + 89.36} \doteq 0.22$$

$\therefore K_\epsilon = 1.75$

$\sigma_{ALT} = \frac{1}{2}(1.65)(1.75)(114.48) = 165.3$ KSI

ALLOWABLE CYCLES = 270 {PER REF. [5] FIG. F-106(b)}

$U_A = 70/270 = 0.26$

CUMMULATIVE USAGE FACTOR (U_{TOTAL})

USAGE FACTOR FOR ALL HPI TRANSIENTS EXCEPT THE 70 ADDITIONAL CYCLES IS 0.441 PER PAGE 23, REF. [1]. THEREFORE, THE TOTAL USAGE FACTOR FOR HPI NOZZLE CONSIDERING THE 70 ADDITIONAL HPI CYCLES DESCRIBED IN REF. [7] IS;

$U_{TOTAL} = 0.441 + 0.26 = \underline{\underline{0.70}}$

$= 0.70 < 1.0$

∴ ACCEPTABLE

DS.21036 (2.73)

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GENERAL CALCULATIONS

32 DOC I.D. | 1121811 SERIAL NUMBER

REFERENCES

1. CDS 32-1119809-00, "HPI NOZZLE USAGE FACTOR."
2. REACTOR COOLANT PRIMARY PIPING STRESS REPORT, B&W CONTRACT NO. 620-0011-50, DESIGN REPORT NO. 5, "HPI NOZZLE ANALYSIS."
3. REACTOR COOLANT SYSTEM FUNCTIONAL SPECIFICATION CS(F)3-92/NSS-11/0672.
4. SPR #13-11-361-00, B&W CONTRACT NO. 620-0011.
5. USAS B31.7 DRAFT EDITION, DATED FEBRUARY 1968 WITH JUNE 1968 ERRATA.
6. B&W COMPUTER PROGRAM P91232, VERSION 1.0, "ONE DIMENSIONAL IMPLICI TRANSIENT TEMPERATURE DISTRIBUTION.
7. FCA NO. 04-3370-00, B&W CONTR 620-0011-50.
8. MICROFICHE, SHAFOGZ, SCHAEFER PR, 8/27/80. (ATTACHED)

CUSTOMER	PROP NO	FILE NO
SU	DWG NO	COMP NO
CONT NO 620.00	GROUP NO	

0 (2-79)

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GENERAL CALCULATIONS

32 DOC I.D | 1121811 SERIAL NUMBER | 00 REV

REFERENCES (ATTACHED)

FICHE SNAFOGZ, DATED 8/27/80.

	PROP NO / FILE NO
CUSTOMER	DWG NO / COMP NO
SUBJECT	GROUP NO
CONT NO 620.00	
ORC DATE 8/27/80	SHEET NO 12

BWNP-20032A-4 (10-78)

CONTRACT/STANDARD NO. 620-0011

RELEASE DATE: OCT 13 1980

PAGE 1 OF 1

DOCUMENT RELEASE NOTICE (DRN)

PART-MARK/TASK-GROUP-SEQ.	B&W DOCUMENT NO.	DOCUMENT TITLE	PUL STAT.	RRL.NO.
50/001/001	32-1119509-01	HPI NOZZLE USAGE FACTOR	~	~

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REVIEWED BY: A.D. McKim (NAME) 10/9/80 (DATE)

CALCULATION DATA/TRANSMITTAL SHEET

DOCUMENT IDENTIFIER — CALC. 32 - 1119809 - 01
TRANS. 86 - -

TYPE: ___ RESEARCH & DEVELOPMENT ___ SAFETY ANALYSIS REPORT ___ NUC. SERV. INPUT ___ DESIGN RQMT. X DESIGN VERIF. ___ OTHER

TITLE HPI Nozzle Usage Factor

PREPARED BY RR Schaefer REVIEWED BY Alvin D. McKim
TITLE Senior Engineer DATE 10/9/80 TITLE Supervisory Engineer DATE 10/9/80

PURPOSE:

This calculation is being revised to show that it is not the controlling stress document for SPR #13-11361-00 anymore.

This document is now amended by CDS #32-1121811 -00

SUMMARY OF RESULTS (INCLUDE DOC. ID'S OF PREVIOUS TRANSMITTALS & SOURCE CALCULATIONAL PACKAGES FOR THIS TRANSMITTAL)

See CDS-32-1121811-00 and CDS-32-1119809-00.

DISTRIBUTION

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DETERMINATION OF ALLOWABLE RAPID DEPRESSURIZATION CYCLES ASSUMMING 40 CYCLES OF TEST TRANSIENT	24
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PURPOSE & INTRODUCTION

B&W Site Problem Report #13-11-361-00 described a problem at SMUD where the operators have been starting a second makeup pump following each reactor trip to obtain additional make up flow to prevent loss of indicated pressure level during the transient. All four of the HPI nozzles have been used for these occurrences, and the three nozzle which do not have continuous makeup flow to the system receive a thermal shock from the cold BWST water. The B&W Reactor System Functional Specification defines the operational events for the HPI nozzles which include 40 cycles of HPI actuations and 40 cycles of HPI testing. At the current rate of usage, SMUD will exceed their allowable lifetime cycles before the end of the 40 year life.

The purpose of this document is to calculate the fatigue usage factor for the SMUD HPI nozzles utilizing the contractual elastic-plastic fatigue analysis in reference #1. This analysis method is not as restrictive as the method utilized in the Stress Report, reference # 3. With the present usage rate at SMUD, it is important to eliminate the Stress Report conservatisms and justify the maximum number of allowable cycles the HPI nozzles can withstand in an attempt to justify the nozzle for their lifetime cycles.

CONCLUSION & RESULTS

MAXIMUM PRIMARY + SECONDARY STRESS RANGE = 128.9 ksi > 3 S_m ; therefore an elastic - plastic fatigue analysis was performed.

Usage Factor = 0.441 < 1.0 for 40 test transients and 40 rapid depressurization transient

Maximum number of rapid depressurization cycles is 110 in conjunction with 40 test transient cycles.

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GENERAL CALCULATIONS

32-1119809 01

PRIMARY + SECONDARY STRESS INTENSITIES
SEGMENT # 8 (L-R INTENSITY)

Iteration	Pressure (Psi)	Pressure Stress (ksi)		Thermal Expansion Stress (Ksi)		Stress Report Thermal Stress (ksi)		Total Stress Intensity (Ksi)	
		Ins.	Out.	Inside	Outside	Inside	Outside	Inside	Outside
1	2200	4.84	1.32	5.1	7.3	63.7	-58.2	73.64	-49.58
6	2200	4.84	1.32	5.1	7.3	77.5	-70.8	87.44	-62.18
8	2200	4.84	1.32	5.1	7.3	78.8	-71.9	88.74	-63.28
15	2200	4.84	1.32	5.1	7.3	69.3	-63.0	79.24	-54.38
30	2200	4.84	1.32	5.1	7.3	7.7	-6.9	17.64	1.72
1211	2200	4.84	1.32	5.1	7.3	-0.4	0.3	9.54	8.92
1491	2200	4.84	1.32	5.1	7.3	0.0	0.0	9.94	8.62
42021	2200	4.84	1.32	5.1	7.3	0.0	0.0	9.94	8.62
27957	2200	4.84	1.32	5.1	7.3	-0.7	0.6	9.24	9.22
5072	2200	4.84	1.32	5.1	7.3	86.0	-78.3	95.94	-69.68
5060	2200	4.84	1.32	5.1	7.3	73.1	-66.8	83.04	-58.18
5069	2200	4.84	1.32	5.1	7.3	97.9	-89.3	107.84	-80.68
5124	2100	4.62	1.26	5.1	7.3	33.0	-29.7	42.72	-21.14
5237	1800	3.96	1.08	5.1	7.3	11.5	-10.4	20.56	-2.02
6059	1000	2.2	0.6	5.1	7.3	0.4	-0.4	7.7	7.50
6367	800	1.76	0.48	5.1	7.3	0.4	0.4	7.26	8.18
6338	700	1.54	0.42	5.1	7.3	-7.7	7.0	-1.06	14.72
6850	600	1.32	0.36	5.1	7.3	-3.5	3.1	2.92	10.76
1×OBE	0.0	0.0	0.0	0.0	0.0	± 0.0	± 0.0	± 19.9	± 28.1
OBE	0.0	0.0	0.0	0.0	0.0	0.0	0.0	± 10.0	± 14.0

See the following sheet for references and table explaination.

CUSTOMER SMUD	PROP NO	CONT NO 620-0	
SUBJECT HPI NOZZLE USAGE FACTOR	DWG NO	FILE NO	
	COMP NO 50	GROUP NO	
CALC BY ACM DATE 6/2/80 CHK'D BY RB DATE 6/24/80		SHEET NO 6	

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GENERAL CALCULATIONS 32-1119809 01

TABLE REFERENCES AND EXPLAINATIONS

- Pressure stresses are obtained by multiplying the pressure stresses from reference # 3, page 5-17 (Iteration #1) by the ratio of actual pressure / 1000 psi (pressure used in reference #3).

- Thermal expansion stresses are from reference #3, page 1-6 for section 2.

- Stress Report thermal stresses are from reference #3, page 5-17.

- The cross section position that experiences positive thermal expansion stresses is used to maximize the inside intensity which is the critical intensity.

- 2X OBE stresses are from reference #3, page 1-6 for section 2. This is the stress state range for a cold earthquake

- Hot earthquake stresses and cycles will be applied and analyzed for in the primary + secondary and peak stress intensity ranges sheets in this document. OBE stresses are from reference # 3, page 1-6. OBE stresses = OBE + Thermal - Thermal.

- A graph of the L-R inside intensities is shown on sheet 8 to aid in determining ranges for maximum and minimum intensity values.

CUSTOMER SMUD		PROP NO	CONT NO 620-001
SUBJECT HPI NOZZLE USAGE FACTOR		DWG NO	FILE NO
		COMP NO 50	GROUP NO
CALC BY AHM DATE 6/2/80	CHK'D BY RB DATE 6/24/80		SHEET NO 7

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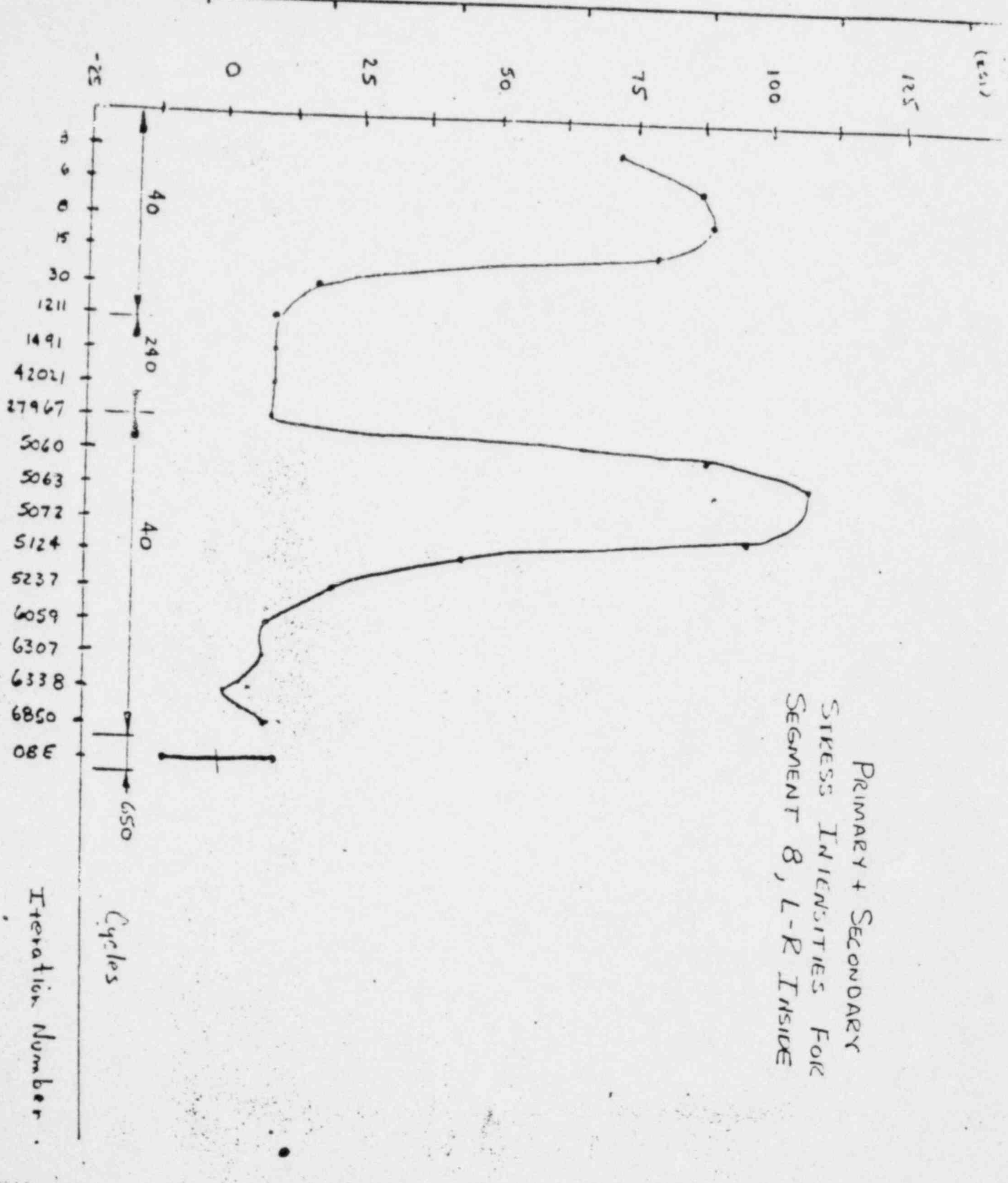
GENERAL CALCULATIONS

32-1119809 0



CUSTOMER SMUD			
SUBJECT		PROP NO	CONT NO
		DWG. NO	FILE NO
		COMP. NO	GROUP NO
CALC. BY AJM DATE 6/2/80	CHK'D BY KS DATE 6/24/80		SHEET NO 8

Determination Of The Number Of Stress Cycles The Primary + Secondary Stress Intensity Range Exceeds $3S_m$

The primary + secondary stress intensity range limit per reference #1, paragraph F-104.4, is $3S_m$. The $3S_m$ value at the critical location, segment #8, is 51.3 ksi which is exceeded for a number of stress intensity cycles. Since the $3S_m$ limit is exceeded, an elastic-plastic fatigue analysis must be performed in accordance with reference #1, paragraph F-105.2.7. This fatigue method is valid only if the number of cycles that exceed $3S_m$ are less than 250. The number of cycles that exceed $3S_m$ are determined on sheets 10 and 11.

MAXIMUM PRIMARY + SECONDARY STRESS INTENSITY RANGE

Range is comprised of $|(\text{Iteration } 6338 + \overset{-\text{Sign}}{OBE})| +$ $(\text{Iteration } 5063 + \overset{+\text{Sign}}{OBE})$ and can occur for 40 cycles.

$$\sigma_{Range}^{Pri+Sec} = 1.06 + 10.0 + 107.84 + 10.0 = 128.9\,ksi > 3S_m$$

2nd MAXIMUM PRIMARY + SECONDARY STRESS INTENSITY RANGE

Range is comprised of $|0 \text{ stress state} + \overset{-\text{Sign}}{OBE}| +$ $(\text{Iteration } 8 + \overset{+\text{Sign}}{OBE})$ and can occur for 40 cycles

$$\sigma_{Range}^{Pri+Sec} = 0 + 10.0 + 88.74 + 10.0 = 108.74\,ksi > 3S_m$$

3rd MAXIMUM PRIMARY + SECONDARY STRESS INTENSITY RANGE

Range is comprised of $|0 \text{ stress state} + \overset{-\text{Sign}}{OBE}| +$ $(\text{Iteration } 4021 + \overset{+\text{Sign}}{OBE})$ and can occur for 40 cycles

$$\sigma_{Range}^{Pri+Sec} = 0 + 10.0 + 9.94 + 10.0 = 39.94\,ksi > 3S_m$$

CUSTOMER SMUD	PROP NO	CONT NO 620-0C
SUBJECT HPI NOZZLE USAGE FACTOR	DWG NO	FILE NO
	COMP NO 5C	GROUP NO
CALC. BY /!! M DATE 6/2/80 CHK'D BY RB DATE 6/24/80		SHEET NO 10 of

Remaining stress intensity ranges are less than $3S_m$.

Total number of cycles in which the primary + secondary stress intensity range exceeds $3S_m$ is 80; therefore, the elastic-plastic fatigue analysis is valid.

PEAK STRESS INTENSITIES SEGMENT #8 L-R INSIDE

Iteration	Pressure Psi	Pressure Stress Ksi	Thermal Exp. σ (ksi)	Stress Rpt. Therm σ (ksi)	Total Stress (ksi)
3	2200	5.5	6.5	109.0	121.0
6	2200	5.5	6.5	115.0	127.0
8	2200	5.5	6.5	113.3	125.3
15	2200	5.5	6.5	95.0	107.0
30	2200	5.5	6.5	8.7	20.7
1211	2200	5.5	6.5	-0.4	11.6
1491	2200	5.5	6.5	0.0	12.0
42021	2200	5.5	6.5	0.0	12.0
47967	2200	5.5	6.5	-0.8	11.2
5072	2200	5.5	6.5	117.3	129.3
5060	2200	5.5	6.5	147.4	159.4
5063	2200	5.5	6.5	143.3	155.3
5124	2100	5.3	6.5	39.9	51.7
5237	1800	4.5	6.5	14.2	25.2
6059	1000	2.5	6.5	0.5	9.5
6307	800	2.0	6.5	0.5	9.0
6338	700	1.8	6.5	-9.6	-1.3
6550	600	1.5	6.5	-1.3	6.7
2xOBE	0.0	0.0	0.0	0.0	25.5
SSE	0.0	0.0	0.0	0.0	±12.7

See the following sheet for references and table explainations.

CUSTOMER SMUD		PROP NO	CONT NO 620-0011
SUBJECT HPI NOZZLE USAGE FACTOR		DWG NO	FILE NO
		COMP. NO 50	GROUP NO
CALC BY [illegible] DATE [illegible]	CHK'D BY KB DATE 6/24/80		SHEET NO 12 of 25

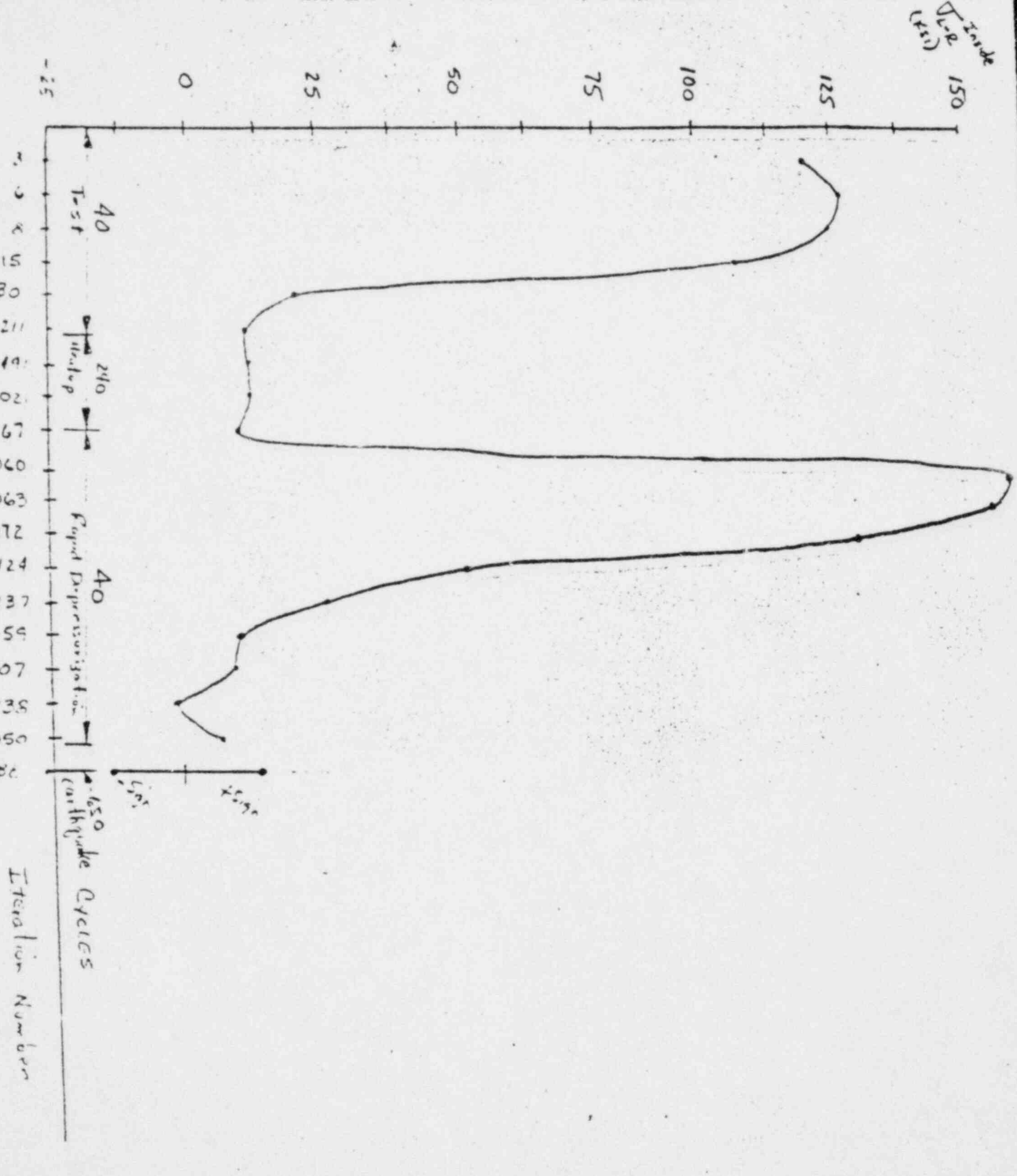
TABLE REFERENCES AND EXPLAINATIONS

- Pressure stresses are obtained by multiplying the peak pressure from reference # 3, page 5-17 (Iteration #1) by the ratio of actual pressure divided by 1000 psi (pressure used in reference #3)
- Thermal expansion stresses are from reference #3, page 1-6 for section 2 multiplied time the bending stress concentration factor from reference #3, page 1-10.
- Stress Report thermal stresses are from ref. #3, page 5-17.
- The cross sectional position that experiences positive thermal expansion stresses is used to maximize the critical inside intensity.
- 2 × OBE stresses are from reference #3, page 1-6 for section 2 multiplied times the bending stress concentration factor from reference #3, page 1-10. This is the range for a cold earthquake.
- A graph of the L-R inside peak stress intensities is shown on the following sheet to aid in determining ranges for maximum and minimum intensity values.

'05-21037'

GENERAL CALCULATIONS

32-11-9809-001



CUSTOMER SMUD		PROP NO	CONT NO
SUBJECT		DWG NO	FILE NO
		COMP NO	GROUP NO
CALC BY [illegible] DATE 6/2/80	CHK'D BY RB DATE 6/24/80		SHEET NO 14 of 25

FATIGUE ANALYSIS

The fatigue analysis is performed in accordance with paragraph F-105.2.7 of reference # 1, "Simplified Elastic-Plastic Discontinuity Analysis". The primary plus secondary stress intensity ranges determined on sheets 10 and 11 shows that the primary plus secondary ranges do not exceed $3S_m$ for more than 250 cycles; therefore, this method of fatigue evaluation is validated.

The peak stress intensities and cycles are presented in graphic form on sheet 14. This is used to determine maximum peak stress intensity ranges and cycles used in the usage factor calculations. Actual peak stress stress intensity value can be obtained from the tabulation on sheet 12. Primary plus secondary intensities are also used in the fatigue analysis:

These values are obtained from the tabulation on sheet 6.

MAXIMUM PEAK STRESS INTENSITY RANGE

Range is comprised of (Iteration 5060 $\overset{+\text{sign}}{+\ \text{OBE}}$) + (Iteration 6338 $\overset{-\text{sign}}{+\ \text{OBE}}$) and can occur for 40 cycles.

$$\sigma_{peak}^{range} = 159.4 + 1.3 + 25.5 = 186.2 \text{ ksi}$$

The primary + secondary range associated with these transients = 128.9 ksi > $3S_m$ = 51.3 ksi (see sht. 10, 2×OBE + iterations 6338 and 5063); therefore, an elastic plastic analysis must be performed for these 40 cycles. Following the procedure of reference #1, paragraph F-105.2.7, the final Peak Alternating Stress Intensity, σ_{alt}, is:

$$\sigma_{alt} = \tfrac{1}{2} K_f K_\epsilon S_{rij}^{(n)}$$

where: $K_f = K_t + A(K_t - 1)$

$$K_t = \frac{S_{rij}^{(p)}}{S_{rij}^{(n)}} = \frac{\text{Peak Stress Intensity Range}}{\text{Primary + Secondary Intensity Range}}$$

$$= 186.2/128.9 = 1.44$$

A = 0.7 for Stainless Steel, from Figure D-201, reference #1

$\therefore K_f = 1.44 + 0.7(1.44 - 1.0) = 1.7?8$

K_E is determined from Figure F-?5(a). Below is the calculation of the parameters necessary to obtain K_E:

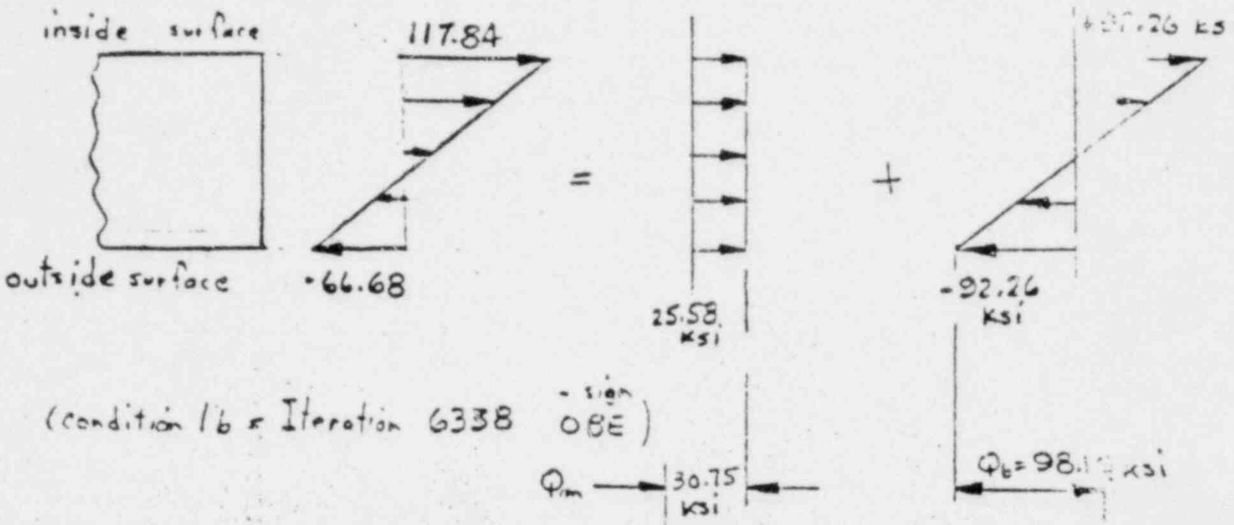
$$\frac{S_n}{3S_m} = \frac{\text{Primary + Secondary Intensity Range}}{3S_m}$$

$$\frac{S_n}{3S_m} = 128.9/51.3 = 2.51$$

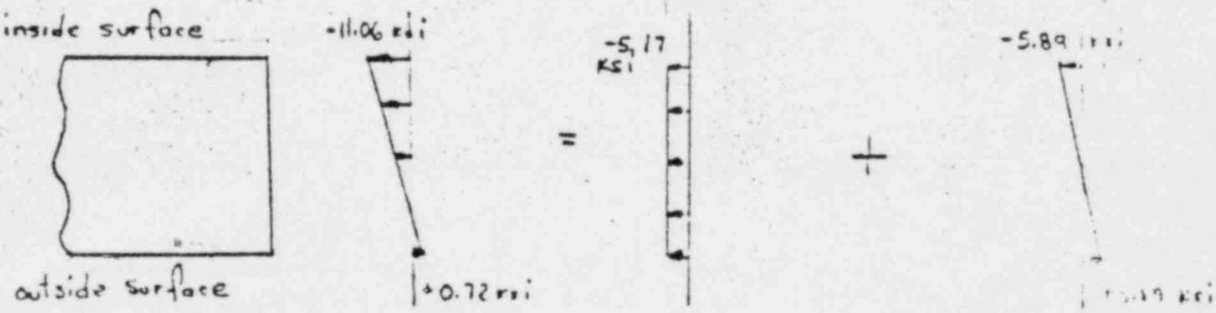
$|Q_m| / [|Q_m| + |Q_b|]$ is determined on the next sheet.

CUSTOMER SMUD		PROP NO	CONT NO 620-00
SUBJECT HPI NOZZLE USAGE FACTOR		DWG. NO	FILE NO
		COMP NO 50	GROUP NO
CALC. BY ABM DATE 6/2/80	CHK'D BY RB DATE 6/24/80		SHEET NO. 17 of

(condition 1a - Iteration 5063 + $\overset{+\text{sign}}{\text{OBE}}$)



(condition 1b = Iteration 6338 $\overset{-\text{sign}}{\text{OBE}}$)



$$\frac{Q_m}{Q_m + Q_b} = \frac{30.75}{30.75 + 98.15} = 0.239$$

$\therefore K_E = 1.95$ from Figure E-105 reference #1

CUSTOMER SMUD		PROP NO	CONT NO 620-00
SUBJECT HPI NOZZLE USAGE FACTOR		DWG NO	FILE NO
		COMP NO 50	GROUP NO
CALC BY ADM DATE 6/2/80	CHK'D BY RS DATE 6/24/80		SHEET NO 18 of

$\sigma_{alt} = \frac{1}{2} K_f K_\epsilon S_{rij}^{(n)}$

$\sigma_{alt} = (\frac{1}{2})(1.748)(1.95)(128.9) = 220.0$ ksi

Allowable cycles = 130 from reference #1, Figure F-106(b)

Usage Factor = U_1 = $40/130 = 0.308$

2nd MAXIMUM PEAK STRESS INTENSITY RANGE

Range is comprised of (Iteration 6 + $\overset{+\text{sign}}{\text{OBE}}$) + |(0 stress + $\overset{-\text{sign}}{\text{OBE}}$)| and can occur for 40 cycles.

$\sigma_{peak}^{range} = (127 + 12.7) + |(0 - 12.7)| = 152.4$ ksi

The primary + secondary range associated with these transients = $(88.74 + 10) + |(0 - 10.0)| = 108.74^{*}$ ksi

$> 3S_m = 51.3$ ksi ; therefore, an elastic-plastic analysis must be performed for these 40 cycles. Following

* Iteration 8 + OBE + 0 stress state + OBE (see sheet 6)

CUSTOMER SMUD	PROP NO	CONT NO 620-001
SUBJECT HPI NOZZLE USAGE FACTOR	DWG. NO	FILE NO
	COMP NO 50	GROUP NO
CALC. BY DDM DATE 6/2/80 CHK'D BY RD DATE 6/24/80		SHEET NO 19 of

the procedure of reference #1, paragraph F-105.2.7, the final Peak Alternating Stress intensity, σ_{alt}, is :

$$\sigma_{alt} = \tfrac{1}{2} K_f K_\epsilon S_{rij}^{(n)}$$

where: $K_f = K_t + A(K_t - 1.0)$

$$K_t = \frac{S_{rij}^{(p)}}{S_{rij}^{(n)}} = \frac{\text{Peak Stress Intensity Range}}{\text{Primary + Secondary Intensity Range}}$$

$$= 152.4/108.74 = 1.40$$

A = 0.7 for Stainless Steel, from Figure D-201, reference #1

$$\therefore K_f = 1.40 + 0.7(1.40 - 1.0) = 1.68$$

K_ϵ is determined from Figure F-105(a). Below is the calculation of the parameter necessary to obtain K_ϵ :

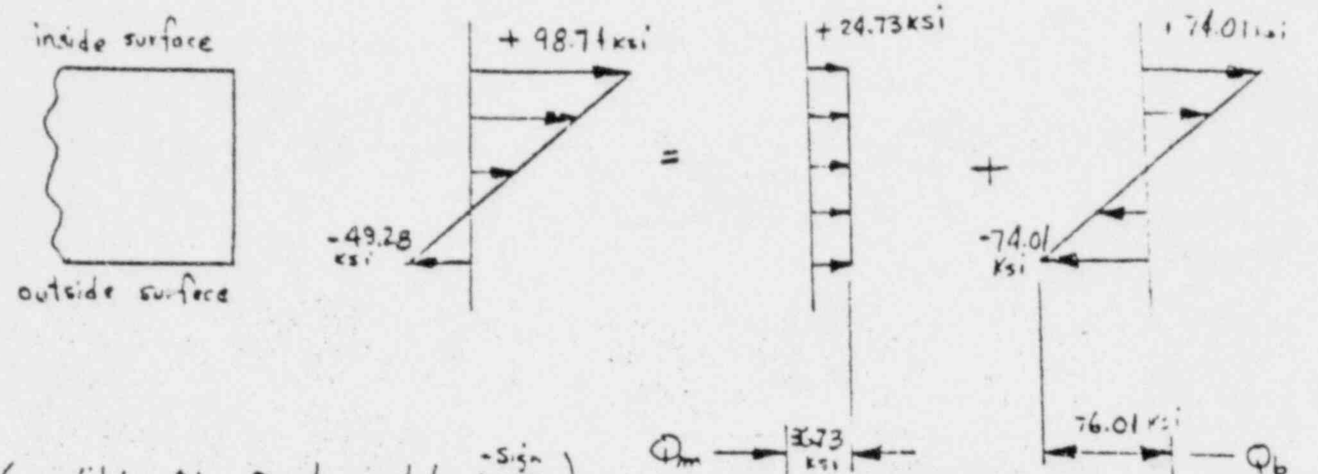
$$S_n/3S_m = \frac{\text{Primary + Secondary Intensity Range}}{3S_m}$$

$$S_n/3S_m = 108.74/51.3 = 2.12$$

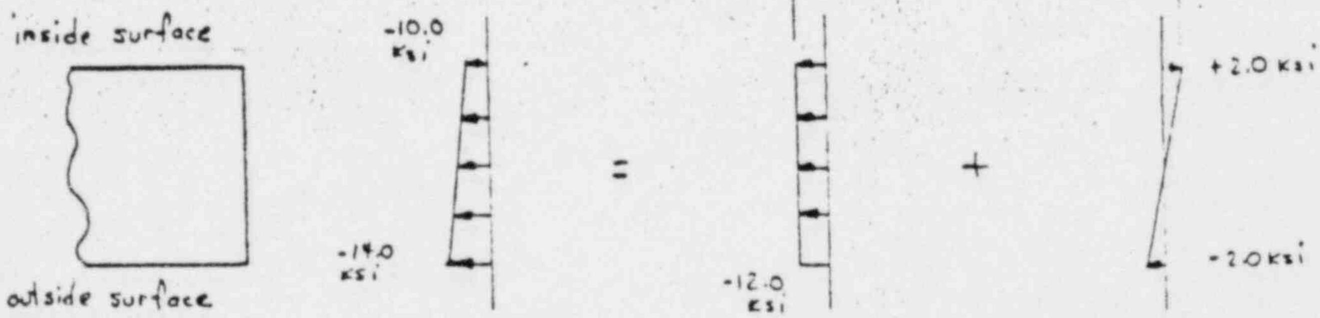
$|Q_m| / [|Q_m| + |Q_b|]$ is determined on the next sheet.

CUSTOMER SMUD		PROP NO	CONT NO 620-0
SUBJECT HPI NOZZLE USAGE FACTOR		DWG NO	FILE NO
		COMP NO 50	GROUP NO
CALC. BY ADM DATE 6/2/80	CHK'D BY RS DATE 6/24/80		SHEET NO 20 of

(condition 2a – Iteration 8 + $\overset{+\text{Sign}}{\text{OBE}}$)



(condition 2b - O stress state $\overset{-\text{Sign}}{+\text{OBE}}$)



$$\frac{Q_m}{Q_m + Q_b} = \frac{36.73}{36.73 + 74.01} = 0.33$$

$\therefore K_E = 1.7$ from figure F-105(a) reference #1

CUSTOMER SMUD		PROP NO	CONT NO 620-00
SUBJECT HPI NOZZLE USAGE FACTOR		DWG NO	FILE NO
		COMP NO 50	GROUP NO
CALC. BY ADM DATE 6/2/80	CHK'D BY RJ DATE 6/24/80		SHEET NO 21 of 2

$$\sigma_{alt} = \tfrac{1}{2} K_f K_\epsilon S_{rij}^{(n)}$$

$$\sigma_{alt} = \tfrac{1}{2}(1.68)(1.7)(108.74) = 155.3 \text{ ksi}$$

Allowable cycles = 300 from reference #1, Figure F-136 (b)

Usage Factor = U_2 = $40/300$ = 0.133

<u>3rd</u> <u>MAXIMUM</u> <u>PEAK</u> <u>STRESS</u> <u>INTENSITY</u> <u>RANGE</u>

Range is comprised of (Iteration 27967 $\overset{+\text{sign}}{+\text{OBE}}$) + |(0 stress state + $\overset{-\text{sign}}{\text{OBE}}$)| and can occur for 240 cycles.

$$\sigma_{peak}^{range} = (11.2 + 12.7) + |(0 - 12.7)| = 36.6 \text{ ksi}$$

The primary + secondary range associated with these iterations = (9.24 + 10.0) + |(0 - 10.0)| = 29.24 ksi < $3S_m$ = 51.4 ksi ; therefore, the final Peak Alternating Stress Intensity Range, σ_{alt}, is :

$$\therefore \sigma_{alt} = \sigma_{peak}^{range} / 2.0 = 36.6/2 = 18.3 \text{ ksi}$$

CUSTOMER SMUD		PROP NO	CONT NO 620-0011
SUBJECT HPI NOZZLE USAGE FACTOR		DWG NO	FILE NO
		COMP NO 50	GROUP NO
CALC BY AJM DATE 6/2/80	CHK'D BY RS DATE 6/24/80		SHEET NO 22 of

Allowable cycles = ∞

Usage Factor = U_3 = $240/\infty$ = 0.0

4th MAXIMUM PEAK STRESS RANGE

This range is comprised of $\pm$ OBE and can occur for the remainder of the earthquake cycles = 650 - 40 - 40 - 240 = 330. The peak stress range is less than case 3 range which resulted in an usage factor of 0.0; therefore, $U_4 = 0.0$

Cummulative Usage Factor = U_{TOTAL}

$U_{TOTAL} = U_1 + U_2 + U_3 + U_4 = 0.308 + 0.133 + 0.0 + \cdot$

$U_{TOTAL} = 0.441 < 1.0$, Acceptable

DETERMINE OF ALLOWABLE RAPID DEPRESSURIZATION CYCLES ASSUMMING 40 CYCLES OF TEST TRANSIENT

The usage factor for 40 cycles of test transient = 0.133 (see sheet 22, U_2)

The remaining usage factor available for rapid depressurization transient cycles = 1.0 - 0.133 = 0.867

Each rapid depressurization transient cycle results in an usage factor contribution of 1/130 = 0.0077 (see sheet 19)

Therefore, the allowable rapid depressurization transient cycles, X, is:

$$0.867 = X\,(0.0077)$$

$$X = 112 \;,\; \text{say } 110$$

BABCOCK & WILCOX
GENERAL CALCULATIONS

32-1119809 0/1
DOC I D — SERIAL NUMBER — REV.

REFERENCES

1) USAS B31.7 Draft Edition dated February 1968 with June 1968 errata.

2) 1971 ASME Section III

3) Design Report #5 - Thermal Mechanical Analysis of 2 ½" Sch 160 Make-up and High Pressure Injection Nozzle for Sacramento Util. District - 620-0011-50, Design Report revision #1 dated 1-15-73.

CUSTOMER SMUD	PROP NO
[illegible] HPI NOZZLE FATIGUE ANALYSIS	FILE NO
CONT NO 620 00 11	GROUP NO
BY [illegible] DATE 6/7/10	SHEET NO [illegible] of 25

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 3, 1982

Docket No. 50-312

Mr. J. J. Mattimoe
Assistant General Manager and
Chief Engineer
Sacramento Municipal Utility
District
6201 S Street
P. O. Box 15830
Sacramento, California 95813

Dear Mr. Mattimoe:

SUBJECT: HPI NOZZLE ANALYSIS - REQUEST FOR INFORMATION

We have performed a preliminary review of your HPI nozzle analysis, and have identified inadequacies in your computations.

Your analysis of HPI nozzle thermal stress did not use the appropriate HPI flow rate following manual initiation of HPI. A total flow rate of 425 gpm was used, corresponding to a normal reactor coolant system (RCS) pressure of 2150 psig. Our review of B&W operating experience shows that RCS pressure usually drops to 1750 psig following reactor trip before manual actuation of HPI would occur.

At 1750 psig, the HPI flow would be 760 gpm. The effect of the increased flow would be to increase the heat transfer coefficient between the nozzle metal and the fluid.

In order for us to complete our review, the information requested in Enclosure 1 is needed. Within 7 days of receipt of this letter, please give us your schedule for supplying this information to us.

We also understand that SMUD has established procedures for manual HPI actuation following a reactor trip if pressurizer level begins to fall. Utilizing this procedure, HPI is manually initiated and directed only to the normal make-up nozzle. The purpose of this procedural change is to assure that the other three HPI nozzles are not subjected to any additional thermal cycling.

In order for us to verify that no additional cycling is occuring on the normal make-up nozzle, please submit to us your analysis in support of this conclusion. Also, please give us a copy of your procedures for keeping track of the number of cycles experienced by the four HPI nozzles. Please submit a schedule for supplying this additional information in your 7 day response.

The reporting requirements contained in this letter affect fewer than ten respondents; therefore OMB clearance is not required under P.L. 96-511.

Sincerely,

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosure:
Request for Additional
Information

cc w/enclosure:
See next page

3.0 PROCEDURE (continued)

.6 Definition and description of each transient cycle.

Following are the Transients, by number, and a description of each cycle:

.1A Heat-up from ambient temperature to 8% Full Power. (Normal Condition)

One cycle involves only the heatup and escalation to 8% of Full Power. Design Cycles = 240.

.1B Cooldown from 8% Full Power to 140 degrees F. (Normal Condition)

One cycle involves only the cooldown from 8% Full Power. Design Cycles = 240.

.2 Power change, 0 to 15% or 15 to 0%. (Normal Condition)

One cycle includes either a heat-up from 532°F to 582°F or a cooldown from 582°F to 532°F.
Design Cycles = 1440.

.3 Power Loading, 8% to 100% Power. (Normal Condition).

This is one design power loading cycle.
Design Cycles = 18,000.

.4 Power Unloading, 100% to 8% Power. (Normal Condition)

This is one design power unloading cycle.
Design Cycles = 18,000.

.5 10% Step Load Increase. (Normal Condition)

One such step is one cycle.
Design Cycles = 8,000.

.6 10% Step Load Decrease. (Normal Condition)

One such step is one cycle.
Design Cycles = 8,000.

.7 Step Load Reduction, 100% to 8% Power. (Upset Condition)

A complete cycle is defined as a turbine trip or electrical load rejection to 8% power with a return to full power.
Design Cycles = 310.

AP-17

LOGGING OF OPERATIONAL TRANSIENTS

1.0 PURPOSE

To establish the procedures for logging operational transients involving the Nuclear Steam Supply System of Rancho Seco Unit 1.

2.0 REFERENCES

.1 B & W Specifications CS(F)-3-92/NSS-11 "Reactor Coolant System Components".

.2 Rancho Seco Unit 1 Technical Specifications.

.3 Rancho Seco Unit 1 FSAR.

3.0 PROCEDURE

.1 In the design of Rancho Seco Unit 1 twenty-four types of NSSS transients have been considered. To insure that design margins and parameters are observed in operation, each and every transient cycle is to be logged. This log will be compiled from the Control Room Log, Computer Logs, Recorder Traces, and other pertinent records of plant operation.

.2 A cycle is defined as a transient deviation from some condition and a return to that initial condition. Half cycles are recorded as "cycles" in some cases, e.g. heatup from cold shutdown, or cooldown to cold shutdown.

.3 Descriptions of transients for equipment design purposes are given in Reference 2.1. | 1

.4 A record sheet will be kept for each type of transient. In addition to recording that a particular transient did occur, data will be recorded for upset and faulted condition transients that will assist in an analysis of the severity of the transient. | 1

.5 At least semi-annually, the Assistant Superintendent for Technical Support will review the logs to insure that the number of design cycles is not being approached or exceeded, and to determine what, if any, corrective action is required.

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P PDR

3.0 PROCEDURE (Continued)

.6

.8 Reactor Trip (Upset Condition)

There are two types of cycles to be considered for each trip:

.1 Reactor trip from Power, hold at hot shutdown with return to power.
Design Cycles = 30 cycles of transient 8A, 130 cycles of transient 8B, and 72 cycles of transient 8C.

.2 Reactor trip from Power, followed by cooldown.
Design Cycles = 10 cycles of transient 8A, 30 cycles of transient 8B, and 16 cycles of transient 8C.

NOTE:

There are no specific design cycles listed in Reference 2.1 for transient 8D. This transient is logged for information purposes to assist in future evaluations if required. 1

The trips are listed by their separate causes as:

Transient 8A: Reactor Trip Due to Loss of RCS Flow.

Transient 8B: Reactor Trip Due to High Temperature, pressure or power caused by Turbine Trip.

Transient 8C: Reactor Trip Due to High Temperature, pressure or power caused by loss of feedwater.

Transient 8D: Reactor trip due to Power/Imbalance/Flow, Low Pressure, Manual or other undesignated causes. 1

.9 Rapid RCS Depressurization (Upset Condition)

A complete cycle consists of power reduction, rapid depressurization from 2155 PSIG to 1050 PSIG in ≤ 15 minutes, followed by normal cooldown. Design Cycles = 40.

3.0 PROCEDURE (Continued)

.6

.10 Change of Reactor Coolant Flow without Reactor Trip (Upset Condition)

The cycle involves a change of flow, due to RCP trip, power reduction to 15%, pump restart and return to full power. Design Cycles = 20.

.11 Rod Withdrawal Accident (Upset Condition)

A cycle is defined as the withdrawal of a CRA Group at maximum speed at a power level ≤ 15%.

There are two types of cycles to be considered:

a. Rod withdrawal, reactor trip, and return to 15% power.
Design Cycles = 30.

b. Rod withdrawal, reactor trip, and cooldown.
Design Cycles = 10.

.12 Hydrotests (Test)

a. A full hydrotest cycle is used when the pressure exceeds 2750 PSIG (110% of design) in the RCS, or when the Secondary System pressure exceeds 1312.5 PSIG (125% of design). Pressures exceeding the design of 2500 PSIG for the RCS or 1050 PSIG for the Secondary System shall be logged; however, pressures less than the hydro pressures defined above do not constitute a full hydro design cycle.

b. ASME Section XI "hydros" do not require exceeding 2500 PSIG or 1050 PSIG. These hydros shall be logged, but are not limited.

Transient 12A: Primary Hydrotests - RCS.
Design Cycles = 35.

Transient 12B: Secondary Hydrotests - Steam Side.
Design Cycles - 35.

.13 Steady-State Power Variations (Normal Condition)

An infinite number of these cycles are allowed, as such they need not be recorded.

3.0 PROCEDURE (Continued)

.6

.14 Reactor Power Runback (Upset Condition)

a. Control Rod Drop with runback.

One cycle involves a CRA drop, runback to 60% power, and return to full power.
Design Cycles - 40.

b. Feedwater Pump trip with runback.

One cycle involves a single feed pump tripping, Reactor runback to 60% power, and return to full power.
Design Cycles - None listed; logged for information only. 1

.15 Loss of Station Power (Upset Condition)

Cycle involves loss of station power, restoration of power, and return to power.
Design Cycles = 40.

.16 Main Steam Line Failure (Faulted Condition)

This is an emergency condition and after OTSG isolation involves a cooldown.

Design Cycles = 1.

.17A Loss of Feedwater to One OTSG (Upset Condition)

Dry OTSG because of loss of Feedwater.
Design Cycles = 20.

.17B Stuck Open Turbine Bypass Valve (Upset Condition)

Dry OTSG because of stuck open turbine bypass or atmospheric dump valve.
Design Cycles = 10.

3.0 PROCEDURE (Continued)

.6

.18 Sudden Loss of Feedwater Heater at 100% Power (Upset Condition)

A cycle involves a sudden (< 30 Seconds) decrease in feedwater temperature of at least 50°F.
Design Cycles = 40.

.19 Feed and Bleed Operations (Normal Condition)

No logging required since no equipment cycle limits are involved.

.20 Pressurizer Spray Flow (Normal Condition)

A cycle consists of opening and closing of the main spray valve during normal plant operation.
Design Cycles = 20,000.

.21 RCS Loss of Coolant (Faulted Condition)

This is based on a double ended rupture of a 36 inch hot leg, one half cycle of blowdown from full pressure constitutes the entire transient.
Design cycles = 1.

.22A High Pressure Injection into RCS (Test or Upset Condition)

A cycle occurs anytime H.P. Injection pumps are used to supply water to the RCS through the H.P.I. nozzles (except during cold shutdown). The nozzle at the "A" RCP discharge is the normal makeup nozzle and would not see a cycle unless makeup flow were stopped entirely prior to start of the H.P Injection.
Design Cycles = 110 cycles per nozzle. 2

.22B Core Flooding Check Valve Tests (Test)

Each time water from the core flood tanks enters the RV with the RCS at 600 PSIG (415°F) is one cycle.
Design Cycles = 240.

.23A OTSG Secondary Side Filling (Normal Condition)

No logging necessary as this transient is assumed to occur concurrent with transient 1A.

3.0 PROCEDURE (continued)

.23B OTSG Primary Side Filling. (Normal Condition)

No logging necessary as this transient is assumed to occur concurrent with transient 1A.

.23C OTSG Chemical Cleaning. (Normal Condition)

One cycle involves repeated fill, soak, and drain operations with chemicals added to the water.
Design Cycles = 20.

.24 Hot Functional Testing. (Test)

This is a test condition expected to occur once and involve two heatup and cooldown transients.

4.0 ACCEPTANCE CRITERIA

Transients defined in this procedure have been logged and reviewed at least semi-annually to ensure the number of design cycles for each transient are not exceeded.

5.0 ENCLOSURES

.1 Transients Log Sheets

.2 Data Sources for Transient Logging 1

TRANSIENT 1A

DESIGN CYCLES 240

Heatup From Ambient Temperature to 8% Full Power

(Normal Condition)

Cycle Number	Date Heatup Started	Date 8% Power Reached	Recorded By/ Date	Remarks

TRANSIENT 1B

DESIGN CYCLES 240

Cooldown From 8% Full Power to 140°F

(Normal Condition)

Cycle Number	Date Cool-down Started	Date RCS @ 140°F or Less	Recorded By/ Date	Remarks

TRANSIENT 2

DESIGN CYCLES 1440

Power Change, 0 to 15% or 15% to 0%

(Normal Condition)

Cycle Number	Date Transient Complete	Power Change Increase or Decrease	Recorded By/ Date	Remarks

TRANSIENT 3

DESIGN CYCLES 18,000

Power Loading, 8% to 100% Full Power

(Normal Condition)

Cycle Number	Date of Transient	Power Loading Change		Recorded By/ Date	Remarks
		Initial % FP	Final % FP		

TRANSIENT 4

DESIGN CYCLES 18,000

Power Unloading, 100% to 8% Full Power

(Normal Condition)

Cycle Number	Date of Transient	Power Unloading Change		Recorded By/ Date	Remarks
		Initial % FP	Final % FP		

TRANSIENT 5

DESIGN CYCLES 8,000

10% Step Load Increase

(Normal Condition)

Cycle Number	Date of Transient	Power Level Init. % FP	Power Level Final % FP	Maximum Loading Rate % FP/min	Recorded By/ Date	Remarks

TRANSIENT 6

DESIGN CYCLES 8,000

10% Step Load Decrease

(Normal Condition)

Cycle Number	Date of Transient	Power Level Init. % FP	Power Level Final % FP	Maximum Unloading Rate % FP/min	Recorded By/ Date	Remarks

DESIGN CYCLES 310

TRANSIENT 7

Step Load Reduction, 100% to 8% Full Power

(Upset Condition)

Cycle Number	Date of Transient Runback	Reactor Power Init. % FP	Reactor Power Final % FP	Type of Trip Load Rejection, Generator Trip or Turbine Trip	Returned to Power __ hrs. Later	RCS Press. psig max.	RCS Press. psig min.	Pressurizer Level, inches max.	Pressurizer Level, inches min.	Recorded By/ Date	Remarks

TRANSIENT 8A

Reactor Trip Due to Loss of RCS Flow

(Upset Condition)

DESIGN CYCLES

A. Trip & hold @ hot shutdown 30

B. Trip & cooldown 10

Cycle Number	Date of Transient	Reactor Power @ Trip, % FP	Number of RCP's on prior to trip	Natural Circulation Exists for, hours	Maximum Cooldown Rate, °F/hr.	Stable RCS Temp. After Trip, °F	Type of Cycle		Recorded By/ Date	Remarks
							Trip & Hold Hot Standby	Trip & Cooldown		

TRANSIENT 8B

DESIGN CYCLES

A. Trip & Hold at Hot Shutdown 130

B. Trip & Cooldown 30

Reactor Trip Due to High Temperature, Pressure or Power - Caused by turbine trip

(Upset Condition)

Cycle Number	Date of Transient	Reactor Power at Trip, % FP	Maximum Cooldown Rate, ^oF/hr.	Reactor Trip Cause, Temp., Pressure or Power	Stable RCS Temp. After Trip, ^oF	Type of Cycle		Recorded By/ Date	Remarks
						Trip & Hold at Hot Shutdwn.	Trip & Cooldown		

TRANSIENT 8C.

DESIGN CYCLES
A. Trip & Hold at Hot Standby 72
B. Trip & Cooldown 16

Reactor Trip Due to High Temperature, Pressure or Power-Caused by loss of feedwater

(Upset Condition)

Cycle Number	Date of Transient	Reactor Power at Trip % FP	Which OTSG Lost Fdwtr.	Maximum Cooldown Rate, °F/hr.	Stable RCS Temp. After Trip, °F	Type of Cycle: Trip & Hold Hot Standby	Type of Cycle: Trip & Cooldown	Recorded By/ Date	Remarks

TRANSIENT 8D

INFORMATION ONLY

Reactor Trip Due to Power/Imbalance/Flow, Low Pressure, Manual and Any Other Undesignated Cause.

Cycle Number	Date of Transient	Reactor Power at Trip % FP	Maximum Cooldown Rate, °F/hr.	Stable RCS Temp. After Trip, °F	Type of Cycle		Recorded By/ Date	Remarks (Specify cause of trip)
					Trip & Hold Hot Standby	Trip & Cooldown		

TRANSIENT 9

DESIGN CYCLES 40

Rapid RCS Depressurization-
2155 to 1050psig in ≤15 minutes

(Upset Condition)

Cycle Number	Date of Transient	Reactor Power Prior to Shutdown % FP	Maximum Cooldown Rate, °F/hr.	Time to RCS 1050 psig, minutes	Recorded By/ Date	Remarks

DESIGN CYCLES 20

TRANSIENT 10

Change of RCS Flow Without Reactor Trip

(Upset Condition)

Cycle Number	Date of Transient	Initial Reactor Power, % FP	Runback to, % FP	RCP's Running Initial (check)				RCP's Running Final (check)				Recorded By/ Date	Remarks
				A	B	C	D	A	B	C	D		

TRANSIENT 11

DESIGN CYCLES

A. Reactor Trip & Return to 15% Power 30

B. Reactor Trip & Cooldown 10

Rod Withdrawl Accident

(Upset Condition)

Cycle Number	Date of Transient	Initial Reactor Power, % FP	Reactor Power at Trip, % FP	Type of Cycle		Actual Trip Parameter	Recorded By/ Date	Remarks
				Trip & Return to Power	Trip & Cooldown			

TRANSIENT 12A

DESIGN CYCLES 35

Primary Hydrotests - RCS
(Pressure ≥2500psig)

Cycle Number	Date of Hydrotest	Peak Test Pressure, psig	Minimum Metal Temp., °F	Test per Which Code	Reason For Test	Recorded By/ Date	Remarks

TRANSIENT 12B

DESIGN CYCLES 35

Secondary Hydrotests - Steam Side
Pressure ≥1050psig)

Cycle Number	Date of Hydrotest	Peak Test Pressure, psig	Minimum Metal Temp., °F	Test per Which Code	Reason For Test	Recorded By/ Date	Remarks

TRANSIENT 13

DESIGN CYCLES INFINITE

Steady-state Power Variations

(Normal Conditions)

AN INFINITE NUMBER OF THESE CYCLES ARE ALLOWED,
AS SUCH THEY NEED NOT BE RECORDED.

TRANSIENT 14A

DESIGN CYCLES 40

Control Rod Drop With Runback

(Upset Condition)

Cycle Number	Date of Transient	Reactor Power		Recorded By/ Date	Remarks
		Prior to Drop, % FP	After Runback % FP		

TRANSIENT 14B

INFORMATION ONLY

Feedwater Pump Trip With Runback

(Upset Condition)

Cycle Number	Date of Transient	Reactor Power		Recorded By/ Date	Remarks
		Prior to Runback % FP	After Runback % FP		

DESIGN CYCLES 40

TRANSIENT 15

Loss of Station Power

(Upset Condition)

Cycle Number	Date & Time of Transient	Length of Power Outage	Power Level at Trip, % FP	Maximum/Minimum RCS			Recorded By/ Date	Remarks
				Temp., ^{o}F	Press., psig	Cooldown Rate, $^{o}F/hr.$		

TRANSIENT 16

DESIGN CYCLES 1

Main Steam Line Failure

(Faulted Condition)

IF THIS TRANSIENT OCCURS A DETAILED ANALYSIS WILL BE REQUIRED PRIOR TO RECOVERY.

ATTACH THAT ANALYSIS FOR LOGGING PURPOSES.

TRANSIENT 17A

DESIGN CYCLES 20 per OTSG

Dry OTSG Because of Loss of Feedwater

(Upset Condition)

Cycle Number		Date of Transient	OTSG Which Went Dry	Recorded By/ Date	Remarks
A	B				

TRANSIENT 17B

DESIGN CYCLES 10 per OTSG

Dry OTSG Because of Stuck Open Turbine Bypass or Atmospheric Dump Valve

(Upset Condition)

Cycle Number A	Cycle Number B	Date of Transient.	OTSG Which Went Dry	Recorded By/ Date	Remarks

DESIGN CYCLES 40

TRANSIENT 18

Sudden Loss of Feedwater Heater at 100% Power

(Upset Condition)

Cycle Number	Date of Transient	Reactor Power When Htr. Lost, % FP	Which Heater Lost	Feedwater Temperature		Time to Make Temp. Change	Recorded By/ Date	Remarks
				Init. °F	Final °F			

TRANSIENT 19

Feed & Bleed Operations

(Normal Conditions)

THIS CYCLE IS NOT LOGGED SINCE
NO EQUIPMENT CYCLE LIMITS ARE INVOLVED.

TRANSIENT 20

DESIGN CYCLES 20,000

Pressurizer Spray Flow

(Normal Condition)

Recording Period, Give Dates	Cycles This Period	Total Cycles	Recorded By/ Date	Remarks

TRANSIENT 21

DESIGN CYCLES 1

RCS Loss of Coolant

(Faulted Condition)

IF THIS TRANSIENT OCCURS A DETAILED ANALYSIS [illegible] WILL BE REQUIRED PRIOR TO RECOVERY.

ATTACH THAT ANALYSIS FOR LOGGING PURPOSES.

TRANSIENT 22A

DESIGN CYCLES 80

High Pressure Injection Into RCS

(Test or Upset Condition)

Cycle Number	Date of Transient	RCS Cold Leg Temp., ^{o}F (if available)	H.P.I. Water Temp., ^{o}F (if available)	H.P.I. Nozzles Used			Recorded By/ Date	Remarks
				B	C	D		

TRANSIENT 22B

DESIGN CYCLES 240

Core Flooding Check Valve Tests

(Test Condition)

Cycle Number	Date of Test	RCS $T_{ave.}$, °F	CFT-A vol., gallons (if available)	CFT-B vol., gallons (if available)	Recorded By/ Date	Remarks

DESIGN CYCLES 20

TRANSIENT 23C

OTSG Chemical Cleaning

(Normal Condition)

Cycle Number	Date of Cleaning	OTSG Cleaned	Fill*			Soak*		Recorded By/ Date	Remarks
			Water Temp., °F	Flow Rate, gpm	Duration, hours	Water Temp., °F	Duration, hours		

*Record this information each time the fill/soak/drain evolution is repeated during one cycle.

TRANSIENT 24

DESIGN CYCLES 1

Hot Functional Testing

INCLUDE WITH THIS RECORD A CHART SHOWING THE CONTINOUS RCS AND SECONDARY PRESSURE/TEMPERATURE HISTORY VERSUS TIME FROM THE START OF HFT UNTIL RV HEAD REMOVAL FOR INITIAL FUEL LOADING.

ENCLOSURE 2

DATA SOURCES FOR TRANSIENT LOGGING

Control Room Log
Shift Supervisors' Log
Trip Reports

Recorder Charts:

- TR 21025, Unit T_{ave}
- XR 00403, Reactor Auctioneered Power
- PR 21037, RCS Narrow Range Pressure
- PR 21038, RCS Narrow Range Pressure
- PR 21092, RCS Wide Range Pressure
- FR 21027, RC Flow
- LR 21503, Pressurizer Level
- TR 21031, Unit T_{hot}
- LR 20503, A OTSG Operate Level
- LR 20504, B OTSG Operate Level
- PR 20543, Steam Pressure
- PR 30119, Steam Pressure
- FR 20535, A OTSG FW Flow
- FR 20536, B OTSG FW Flow
- LR 23502, Makeup Tank Level

Computer Points:

- G009, Reactor Auctioneered Power
- G139, 140, Unit T_{ave}
- T010, 011, 108, 109, Unit T_{hot}
- T012, 013, 014, 015, 016, 017, Unit T_{cold}
- P014, 015, 016, 017, 034, 035, 036, RCS Pressure
- F016, 017, 018, 019, 020, 021, 022, 023, RCS Flow
- L008, 009, Pressurizer Level
- T018, Pressurizer Surge Line Temperature
- T009, Pressurizer Temperature
- L035, 036, OTSG A, B Full Range Level
- L037, 038, OTSG A, B Operate Range Level
- L039, 040, OTSG A, B Startup Range Level
- T800, 801, 802, Mainsteam Temperature
- P800, 801, OTSG A, B Mainsteam Pressure
- T067, 208, Feedwater Temperature
- F013, 014, 801, 802, 807, 808, Feedwater Flow
- F015, Letdown Flow
- T217, Makeup Temperature