Westinghouse Energy Systems

WESTINGHOUSE CLASS 3

WCAP-12938

Structural Evaluation of Indian Point Units 2 and 3 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification

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EXECUTIVE SUMMARY

Thermal stratification has been identified as a concern which can affect the structural integrity of piping systems in nuclear plants since 1979, when a leak was discovered in a PWR feedwater line. In the pressurizer surge line, stratification can result from the difference in densities between the hot leg water and generally hotter pressurizer water. Stratification with large temperature differences can produce very high stresses, and this can lead to integrity concerns. Study of the surge line behavior has concluded that the largest temperature differences occur during certain modes of plant heatup and cooldown.

This report has been prepared to support compliance with the requirements of NRC Bulletin 88-11 for Indian Point Units 2 and 3. Prior to the issuance of the bulletin, the Westinghouse Owners Group had a program in place to investigate the issue, and recommend actions by member utilities. That program provided the technical basis for the analysis reported here for Indian Point Units 2 and 3.

The transient development utilized a number of sources, including plant operating procedures, surge line monitoring data from other similar units, and historical records for each unit. This transient information was used as input to a structural and stress analysis of the surge line for the two units. A review and comparison of the piping and support configurations for the units led to the conclusion that the surge lines are nearly identical, and thus one analysis could be done to apply to both units, for the stratification transient development and structural evaluation.

The existing configurations for both Indian Point units have been analyzed in this WCAP. The analysis results are provided in Section 3 for ASME code stress, Section 4 for piping displacements at support and restraint locations and Section 5 for ASME Code fatigue cumulative usage factors.

SECTION 1.0
BACKGROUND AND INTRODUCTION

Indian Point Units 2 and 3 are four-loop pressurized water reactors, designed to be as nearly identical as practical, in both hardware and operation. This report has been developed to provide the technical basis and results of a plant-specific structural evaluation for the effects of thermal stratification of the pressurizer surge lines for both of these units.

The operation of a pressurized water reactor requires the primary coolant loops to be water solid, and this is accomplished through a pressurizer vessel, connected to one of the hot legs by the pressurizer surge line. A typical four-loop arrangement is shown in Figure 1-1, with the surge line highlighted.

The pressurizer vessel contains steam and water at saturated conditions with the steam-water interface level typically between 25 and 60% of the volume depending on the plant operating conditions. From the time the steam bubble is initially drawn during the heatup operation to hot standby conditions, the level is maintained at approximately 25% to 35%. During power ascension, the pressurizer level varies between 22% and 50% depending on reactor thermal power. The steam bubble provides a pressure cushion effect in the event of sudden changes in Reactor Coolant System (RCS) mass inventory. Spray operation reduces system pressure by condensing some of the steam. Electric heaters, at the bottom of the pressurizer, are energized to raise the liquid temperature to generate additional steam and increase RCS pressure.

As illustrated in Figure 1-1, the bottom of the pressurizer vessel is connected to the hot leg of one of the coolant loops by the surge line. The surge lines of Units 2 and 3 are both 14 inch schedule 140 stainless steel.

1.1 Background

During the period from 1982 to 1988, a number of utilities reported unexpected movement of the pressurizer surge line, as evidenced by crushed insulation, gap closures in the pipe whip restraints, and in some cases unusual snubber movement. Investigation of this problem revealed that the movement was caused by thermal stratification in the surge line.

Thermal stratification had not been considered in the original design of any pressurizer surge line, and was known to have been the cause of service-induced cracking in feedwater line piping, first discovered in 1979. Further instances of service-induced cracking from thermal stratification surfaced in 1988, with a crack in a safety injection line, and a separate occurrence with a crack in a residual heat removal line. Each of the above incidents resulted in at least one through-wall crack, which was detected through leakage, and led to a plant shutdown. Although no through-wall cracks were found in surge lines, inservice inspections of one plant in the U.S. and another in Switzerland mistakenly claimed to have found sizeable cracks in the pressurizer surge line. Although both these findings were subsequently disproved, the previous history of stratified flow in other lines led the USNRC to issue Bulletin 88-11 in December of 1988. A copy of this bulletin is included as Appendix B.

The bulletin requested utilities to establish and implement a program to confirm the integrity of the pressurizer surge line. The program required both visual inspection of the surge line and demonstration that the design requirements of the surge line are satisfied, including the consideration of stratification effects. Visual inspections were conducted in accordance with task 1a of the Bulletin at both Indian Point Units 2 and 3 [16], [17].

Prior to the issuance of NRC Bulletin 88-11, the Westinghouse Owners Group had implemented a program to address the issue of surge line stratification. A bounding evaluation was performed and presented to the NRC in April of 1989. This evaluation compared all the WOG plants to those for which a detailed plant specific analysis had been performed. Since this evaluation was unable to demonstrate the full design life for all plants, a generic justification for continued operation was developed for use by each of the WOG plants, the basis of which was documented in references [1] and [2].*

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*Numbers in brackets refer to references listed in Section 7.

The Westinghouse Owners Group implemented a program for generic detailed analysis in June 1989, and this program involved individual detailed analyses of groups of plants. This approach permitted a more realistic approach than could be obtained from a single bounding analysis for all plants, and the results were published in June of 1990 [3].

The followup to the Westinghouse Owners Group Program is a performance of evaluations which could not be performed on a generic basis. The goal of this report is to accomplish these followup actions, and to therefore support completion of the requirements of NRC Bulletin 88-11 for Indian Point Units 2 and 3.

1.2 Description of Surge Line Thermal Stratification

It will be useful to describe the phenomenon of stratification, before dealing with its effects. Thermal stratification in the pressurizer surge line is the direct result of the difference in densities between the pressurizer water and the generally cooler RCS hot leg water. The warmer, lighter pressurizer water tends to float on the cooler, heavier hot leg water. The potential for stratification is increased as the difference in temperature between the pressurizer and the hot leg increases and as the insurge or outsurge flow rates decrease.

At power, when the difference in temperature between the pressurizer and hot leg is relatively small, the extent and effects of stratification have been observed to be small. However, during certain modes of plant heatup and cooldown, this difference in system temperature could be as large as 320°F, in which case the effects of stratification are significant, and must be accounted for.

Thermal stratification in the surge line causes two effects:

- o Bending of the pipe different from that predicted in the original design.

- o Potentially reduced fatigue life of the piping due to the higher stress resulting from stratification and striping.

1.3 Scope of Work

The primary purpose of this work was to develop transients applicable to the Indian Point surge lines which include the effects of stratification, and to evaluate the structural integrity of the surge lines. This work will therefore support the demonstration of compliance with the requirements of NRC Bulletin 88-11.

The transients were developed following the same general approach originally established for the Westinghouse Owners Group. Conservatisms inherent in the original approach were refined through the use of monitoring results, plant operating procedures, operator interviews, and historical data on plant operation. This process is discussed in Section 2.

The resulting transients were used to perform an analysis of the surge line, wherein the existing support configuration was carefully modeled, and surge line displacements, stresses, support loads and nozzle loads were determined. This analysis and its results are discussed in Sections 3 and 4.

The stresses were used to perform a fatigue analysis for the surge line, and the methodology and results of this work are discussed in Section 5. The summary and conclusions of this work are summarized in Section 6.

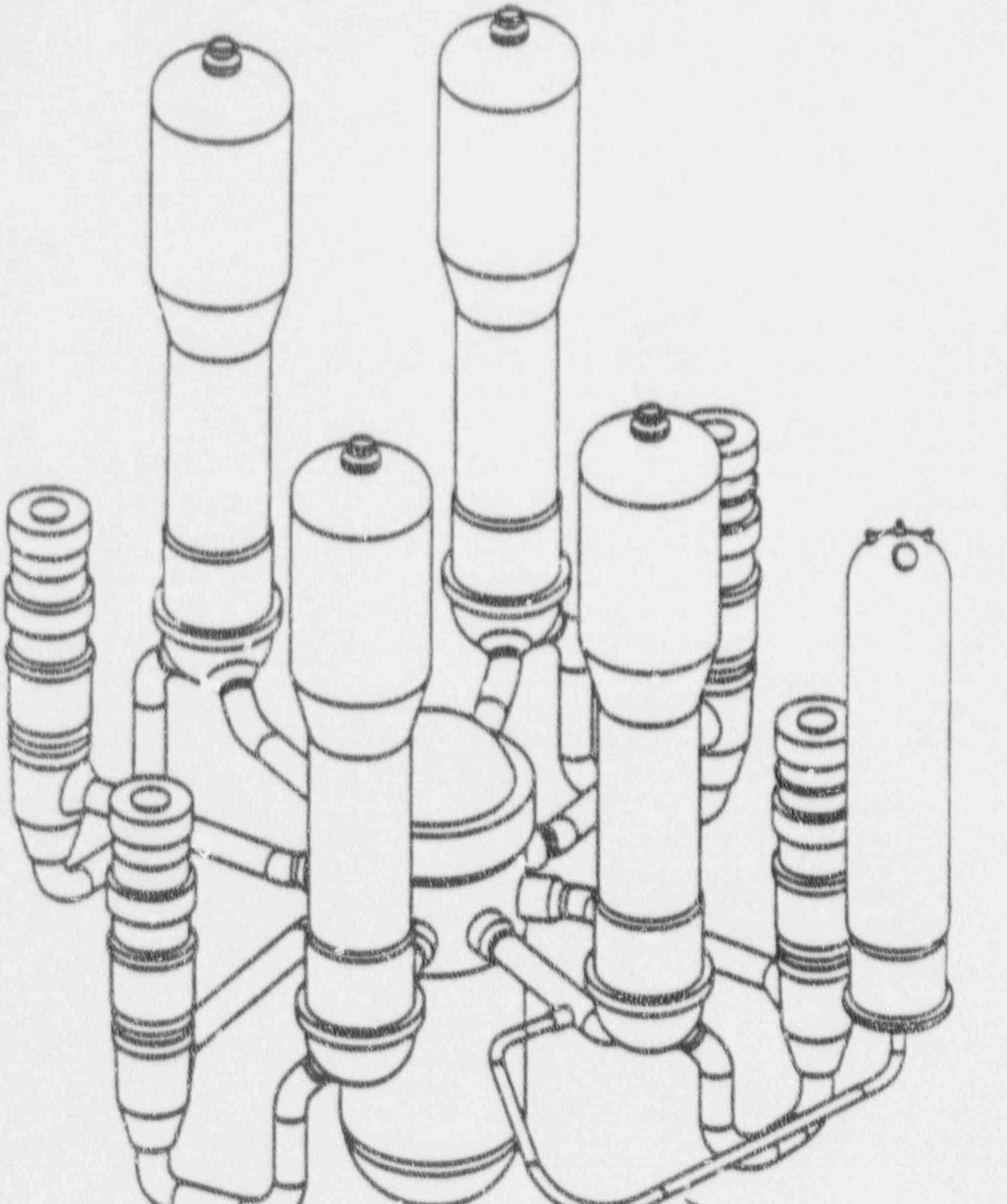

Figure 1-1. Typical 4-Loop Plant Loop Layout

SECTION 2.0
SURGE LINE TRANSIENT AND TEMPERATURE PROFILE DEVELOPMENT

2.1 General Approach

The transients for the pressurizer surge line were developed from a number of sources, including the most recent Westinghouse Systems Standard Design transients. The heatup and cooldown transients, which involve the majority of the severe stratification occurrences, were developed from review of the design transients, plant operating procedures, operator interviews, monitoring data and historical records for each unit. The total number of heatup and cooldown events specified remains unchanged at 200 each, but a number of transient events within each heatup and cooldown cycle have been defined to reflect stratification effects, as described in more detail later.

The normal and upset transients, except for heatup and cooldown, used for the Indian Point Units 2 and 3 surge lines are provided in Table 2-1. For each of the transients the surge line fluid temperature was modified from the original design assumption of uniform temperature to a stratified distribution, according to the predicted temperature differentials between the pressurizer and hot leg, as listed in the table. The transients have been characterized as either insurge/outsurges (I/O in the table) or fluctuations (F). Insurge/outsurge transients are generally more severe, because they result in the greatest temperature change in the top or bottom of the pipe. Typical temperature profiles for insurges and outsurges are shown in Figure 2-1.

Transients identified as fluctuations (F) typically involve low surge flow rates and smaller temperature differences between the pressurizer and hot leg, so the resulting stratification stresses are much lower. This type of cycle is important to include in the analysis, but is generally not the major contributor to fatigue usage.

In addition to the plant specific operating history discussed above, the development of transients which are applicable to Indian Point Units 2 and 3 was based on the work already accomplished under programs completed for the Westinghouse Owners Group [1,2,3]. In this work all the Westinghouse plants were grouped based on the similarity of their response to stratification. The three most important factors influencing the effects of stratification were found to be the structural layout, support configuration, and plant operation.

The transient development for the Indian Point units took advantage of the similarity in the surge line layout for the two units, as well as general similarities in the operating procedures. A detailed comparison of the piping and support configurations for the units appears in Section 3.1.

The transients developed here, and used in the structural analysis, have taken advantage of the monitoring data collected during the WOG program, as well as historical operation data for the Indian Point units. Each of these will be discussed in the sections which follow.

2.2 System Design Information

The thermal design transients for a typical Reactor Coolant System, including the pressurizer surge line, are defined in Westinghouse Systems Standard Design Criteria.

The design transients for the surge line consist of two major categories:

(a) Heatup and Cooldown transients

(b) Normal and Upset operation transients (by definition, the emergency and faulted transients are not considered in the ASME Section III fatigue life assessment of components).

In the evaluation of surge line stratification, the transient events considered encompass the normal and upset design events defined in the FSAR chapter 4.1.

2.3 Development of Normal and Upset Transients

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2.4 Monitoring Results and Operational Practices

2.4.1 Monitoring

Monitoring information collected as part of the Westinghouse Owners Group generic detailed analysis [3] was utilized in this analysis. The pressurizer surge line monitoring programs utilized externally mounted temperature sensors (resistance temperature detectors or thermocouples). The temperature sensors were attached to the outside surface of the pipe at various circumferential and axial locations. In all cases these temperature sensors were securely clamped to the piping outer wall, taking care to properly insulate the area against heat loss due to thermal convection or radiation.

The typical temperature sensor configuration at a given pipe location consists of two to five sensors mounted as shown in Figure 2-2. Temperature sensor configurations were mounted at various axial locations. The multiple axial locations give a good picture of how the top to bottom temperature distribution may vary along the longitudinal axis of the pipe. In addition, many pressurizer surge line monitoring programs utilized displacement sensors mounted at various axial locations to detect horizontal and vertical movements, as shown in Figure 2-2. Typically, data was collected at [ ]^{a,c,e} intervals or less, during periods of high system delta T.

Existing plant instrumentation was used to record various system parameters. These system parameters were useful in correlating plant actions with stratification in the surge line. A list of typical plant parameters monitored is given below.

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Data from the temporary sensors was stored on magnetic floppy disks and converted to hard copy time history plots with the use of common spreadsheet software. Data from existing plant instrumentation was obtained from the utility plant computer.

2.4.2 Operational Practices

Based on a review of the Indian Point Units 2 and 3 heatup and cooldown operating procedures and operational interviews conducted at a number of WOG utilities (including Indian Point Unit 2 in September, 1989), it was determined that both units heat up and cool down in manners somewhat similar to other plants that heat up with a steam bubble in the pressurizer. Heatups and cooldowns are used here to characterize plant operation because they represent the periods during which the temperature difference between the pressurizer and the hot leg is potentially the greatest. Brief descriptions of the Indian Point units' heatup and cooldown procedures follow.

Unit 2

The heatup and cooldown procedures at Indian Point Unit 2 are generally similar to those used at other steam bubble plants, but utilize a nitrogen bubble to maintain RCS pressure during periods early in the heatup or late in the cooldown. The heatup procedure begins with the RCS full and the pressurizer level at 75-85%. At this point, a nitrogen bubble is established in the pressurizer at a pressure of 400-450 psig. The pressurizer level and RCS pressure are maintained in these ranges during nitrogen bubble operation using charging flow, letdown flow and nitrogen supply. After the nitrogen bubble is established, the RCS is vented, and at least one reactor coolant pump (RCP) is started before the RCS temperature exceeds 180°F. Oxygen levels are also checked and hydrazine added, if necessary, with the RCS held between 170°F and 190°F.

After this, a steam bubble is established in the pressurizer by energizing the pressurizer heaters, reaching saturation temperature at RCS pressure of 400-450 psig. After the steam bubble is formed, pressurizer spray is begun. The nitrogen is returned into solution and transferred from the pressurizer to the volume control tank. Normal pressurizer level is established, and RHR is isolated, once the RCS temperature is approximately 350°F. The remaining RCP's are started to complete the heatup within specified limits. Plant procedures allow a maximum RCS heatup rate of 50°F/hr in the range of 70-350°F, and a maximum allowable pressurizer heatup rate of 100°F/hr. During the entire process, a limit of 320°F is imposed on the difference between pressurizer and spray fluid temperatures. This inherently imposes the same limit on the difference between pressurizer and hot leg temperature during this period of operation.

The cooldown process is essentially the reverse of the heatup. Once RCS pressure is below 450 psig and temperature is below 350°F, RHR is put in service, and the steam bubble is exchanged for a nitrogen bubble in the pressurizer. Pressurizer spray and letdown are used to control the process. The cooldown is completed using RHR, with at least one RCP operating until the RCS temperature is below 180°F.

Unit 3

The heatup and cooldown procedures for Indian Point Unit 3 are typical of most plants that use the steam bubble method. At the beginning of the heatup, the RCS is filled and vented, and pressurized to 400-450 psig, using charging and letdown flow to maintain the pressure. (In the past, nitrogen was also an option for controlling RCS pressure at this stage, but according to plant operations, has not been used extensively and is not used presently.) At least one RCP is started, and various checks (including chemistry and oxygen) are performed during the ensuing process prior to reaching limits of 200°F and 250°F.

After the RCP is started, with the RCS solid and pressure greater than 400 psig, a steam bubble is established in the pressurizer. This is accomplished by closing the spray valves and energizing the pressurizer heaters until the steam bubble is drawn. RCS pressure is maintained between 400-450 psig using letdown flow, charging and/or pressurizer heaters until the bubble is formed, at which point the pressurizer level is decreased to 32% using charging and letdown. The RCS pressure is then controlled using pressurizer spray and/or heaters. Once the RCS pressure is stabilized (between 400 and 450 psig) and pressurizer normal water level is established, the RHR is shut down and isolated. From this point, the plant heatup is commenced by starting the remaining RCP's and using pressurizer heaters, if necessary, to supplement the heatup rate. The cooldown procedure is basically the reverse of the heatup.

During these processes, administrative limits are imposed as follows: RCS allowable heatup rate of 50°F/hr; pressurizer allowable heatup rate of 100°F/hr; maximum allowable delta T between the pressurizer and reactor coolant loops of 320°F; spray should not be used if delta T between the pressurizer and the spray fluid exceeds 320°F.

Summary

From the operating procedures, the possible ranges of the system delta T could vary, but are bounded by the administrative limit of 320°F. The actual impact of these plant operating procedures on the analysis was determined in conjunction with review of the plants' past operating histories, and is discussed in the following section.

2.5 Historical Operation

Historical records from the plants (operator logs, surveillance test reports, etc.) were reviewed in December, 1990, supplemented with confirmatory investigations of the data by the utilities [14, 15]. The purpose of the review was to obtain a distribution of maximum system delta T, and to identify heatup or cooldown events where the maximum system delta T exceeded the 320°F limit. To date (as of April, 1991), Unit 2 has experienced 85 heatups and 85 cooldowns, and Unit 3 has experienced 37 heatups and 36 cooldowns. The data available represents only a portion of these events. Therefore, the delta T distribution is expressed in terms of the events in a predetermined range as a percentage of the total number of events for which data was available. A summary of the results for available data is presented below.

	Unit 2		Unit 3	
System ΔT Range (°F)	Number of Heatups or Cooldowns	% of Total	Number of Heatups or Cooldowns	% of Total
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				]a,c,e
Total events	7		28	

For Unit 2, pressurizer temperature was not specified in the data initially available for review, which represented 58 events, either heatups or cooldowns. Since a nitrogen bubble is used to maintain RCS pressure during periods where system delta T is typically maximum in steam-bubble plants, the corresponding saturation temperature at the indicated RCS pressure could not be used. For a subset of seven of these events, available plant computer data that did indicate pressurizer temperature was investigated, and it was found that actual system delta T's were less than 320°F for the events investigated [14]. Since plant procedures inherently specify a limit on system delta T of 320°F (see Section 2.4.2), it was assumed that all past events for Unit 2 had system delta T values below 320°F. This assumption is conservative with respect to plant operations until May, 1977, when water-solid heatup and cooldown operations were used. Typical maximum system delta T for water-solid plant operation is 210°F [3].

For Unit 3, there was some recorded data that could lead to the conclusion that the 320°F system delta T limit had been exceeded. An investigation was made by the utility [15], which concluded that the 320°F administrative limit had not been exceeded for past operation. Thus, the events in question were assumed to occur with a maximum system delta T of 320°F. Data for Unit 3 was available for 28 events, either heatups or cooldowns, that occurred in 17 of the past heatup/cooldown cycles.

The comparison of these past system delta T distributions to that used in the analysis is illustrated in Figure 2-3. (Development of the analytical system delta T distribution is discussed further in Section 2.6.) [

]^{a,c,e} From this comparison, it is evident that if future operation for Unit 2 continues as assumed for the past, the analysis is conservative.

2.6 Development of Heatup and Cooldown Transients

The heatup and cooldown transients used in the analysis were developed from a number of sources, as discussed in the overall approach. The transients were built upon the extensive work done for the Westinghouse Owners Group [1,2,3], coupled with plant specific considerations for Indian Point Units 2 and 3.

The transients were developed based on monitoring data, historical operation and operator interviews conducted at a large number of plants. For each monitoring location, the top-to-bottom differential temperature (pipe delta T) vs. time was recorded, along with the temperatures of the pressurizer and hot leg during the same time period. The difference between the pressurizer and hot leg temperature was termed the system delta T.

From the pipe and system delta T information collected in the WOG[1,2,3] effort, individual plants' monitoring data was reduced to categorize stratification cycles (changes in relatively steady-state stratified conditions) using the rainflow cycle counting method. This method considers delta T range as opposed to absolute values.

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The resulting distributions (for I/O transients) were cycles in each RSS range above 0.3, for each mode. (In the surge line analyses, RCS temperature ranges of ≤200°F, 200-350°F, >350°F at hot standby, and >350°F during startup were labelled as modes 5,4,3 and 2, respectively). A separate distribution was determined for the reactor coolant loop nozzle and for a chosen critical pipe location. Next, a representative RSS distribution was determined by multiplying the average number of occurrences in each RSS range by two. Therefore, there is margin of 100% on the average number of cycles per heatup in each mode of operation.

Transients, which are represented by delta T pipe with a corresponding number of cycles, were developed by combining the delta T system and cycle distributions. For mode 5, delta T system is represented by a historical distribution developed from plant operating records from a number of plants, and is represented in Figure 2-3 as "used in analysis". As discussed in Section 2.5, this historical delta T system distribution was assumed to encompass the prior operating history of the Indian Point units, and to account for future operation. For modes 4, 3 and 2, the delta T system was defined by maximum values. The values were based on the maximum system delta T obtained from the monitored plants for each mode of operation.

An analysis was conducted to determine the average number of stratification cycles per cooldown relative to the average number of stratification cycles per heatup. [

]^{a,c,e} The transients for all modes were then enveloped in ranges of ΔT_{pipe}, i.e., all cycles from transients within each ΔT_{pipe} range were added and assigned to the pre-defined ranges. These cycles were then applied in the fatigue analysis with the maximum ΔT_{pipe} for each range. The values used are as follows:

For Cycles Within Pipe Delta T Range	Pipe Delta
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This grouping was done to simplify the fatigue analysis.

The final result of this complex process is a table of transients corresponding to the subevents of the heatup and cooldown process. The actual number of transient cycles used in the analysis to represent 200 heatup/cooldown event cycles is shown in Table 2-2. A mathematical description of the methodology used is given in Appendix C. [

]^{a,c,e} The critical location is the location with the highest combination of pipe delta T and number of stratification cycles.

Because of main coolant pipe flow effects, the stratification transient loadings at the RCS hot leg nozzle are different. These transients have been applied to the main body of the nozzle as well as the pipe to nozzle girth butt weld.

Plant monitoring included sensors located near the RCS hot leg nozzle to surge line pipe weld. Based on the monitoring, a set of transients was developed for the nozzle region to reflect conditions when stratification could occur in the nozzle. The primary factor affecting these transients was the flow in the main coolant pipe. Significant stratification was noted only when the reactor coolant pump in the loop with the surge line was not operating. Transients were then developed using a conservative number of "pump trips."

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]^{a,c,e} Therefore, the fatigue analysis of the RCS hot leg nozzle was performed using the "nozzle transients" and the "pipe transients." The analysis included both the stratification loadings from the nozzle transients, and the pressure and bending loads from the piping transients.

The total transients for heatup and cooldown are identified as HC1 thru HC9 for the pipe, and HC1 thru HC9 for the RCS hot leg nozzle as shown in Tables 2-2(a) and 2-2(b), respectively. Transients HC8 and HC9 for the pipe and HC9 for the nozzle represent transients which occur during later stages of the heatup.

2.7 Axial Stratification Profile Development

In addition to transients, a profile of the [

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Two types of profile envelope the stratified temperature distributions observed and predicted to occur in the line. These two profiles are a [

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Review and study of the monitoring data for all the plants revealed a consistent pattern of development of delta T as a function of distance from the hot leg intersection. This pattern was consistent throughout the heatup/cooldown process, for a given plant geometry. This pattern was used along with plant operating practices to provide a realistic yet somewhat conservative portrayal of the pipe delta T along the surge line.

The combination of the hot/cold interface and pipe delta T as functions of distance along the surge line forms a profile for each individual plant analyzed. Since Unit 2 and Unit 3 have similar surge line configurations, the profile applies to both units. [

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2.8 Striping Transients

The transients developed for the evaluation of thermal striping are shown in Table 2-3.

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Striping transients use the labels HST and CST denoting striping transients (ST). Table 2-3 contains a summary of the HST1 to HST8 and CST1 to CST7 thermal striping transients which are similar in their definition of events to the heatup and cooldown transient definition.

These striping transients were developed during plant specific surge line evaluations and are considered to be a conservative representation of striping in the surge line[3]. Section 5 contains more information on specifically how the striping loading was considered in the fatigue evaluation.

TABLE 2-1

SURGE LINE TRANSIENTS WITH STRATIFICATION

NORMAL AND UPSET TRANSIENT LIST - INDIAN POINT UNIT 2 OR UNIT 3

			TEMPERATURES (°F)		
			MAX	NOMINAL	
LABEL	TYPE	CYCLES	ΔT_{Strat}	PRZ T	RCS T

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]a,c,e

See notes on next page

TABLE 2-1 (Cont'd.)

SURGE LINE TRANSIENTS WITH STRATIFICATION

NORMAL AND UPSET TRANSIENT LIST - INDIAN POINT UNIT 2 OR UNIT 3

LABEL	TYPE	CYCLES	MAX ΔT_{Strat}	TEMPERATURES (°F) NOMINAL PRZ T	RCS T
[					

]a,c,e

TABLE 2-2a

SURGE LINE PIPE TRANSIENTS WITH STRATIFICATION - INDIAN POINT UNIT 2 OR 3

HEATUP/COOLDOWN (HC) - 200 CYCLES TOTAL

			TEMPERATURES (°F)		
			MAX	NOMINAL	
LABEL	TYPE	CYCLES	ΔT_{Strat}	PRZ T	RCS T

[

]a,c,e

TABLE 2-2b

SURGE LINE NOZZLE TRANSIENTS WITH STRATIFICATION - INDIAN POINT UNIT 2 OR 3

HEATUP/COOLDOWN (HC) - 200 CYCLES TOTAL

			TEMPERATURES (°F)		
			MAX	NOMINAL	
LABEL	TYPE	CYCLES	ΔT_{Strat}	PRZ T	RCS T

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]a,c,e

TABLE 2-3

SURGE LINE TRANSIENTS - STRIPING

FOR HEATUP (H) and COOLDOWN (C) - INDIAN POINT UNIT 2 OR 3

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Figure 2-1. Typical Insurge-Outsurge (I/O) Temperature Profiles

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Figure 2-2. Typical Monitoring Locations

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Figure 2-3. Summary of Historical Data Distribution from Indian Point Units 2 and 3, Compared to the Distribution Used in the Analysis

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Figure 2-4. Example Axial Stratification Profile for Low Flow Conditions

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Figure 2-5. Geometry Considerations

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Figure 2-6. Temperature Profile Analyzed for Indian Point Units 2 and 3

SECTION 3.0
STRESS ANALYSIS

The flow diagram (Figure 3-1) describes the procedure to determine the effects of thermal stratification on the pressurizer surge line based on transients developed in section 2.0. [

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3.1 Surge Line Layouts

The Indian Point Units 2 & 3 surge line layouts are documented in references 6 and 7, and the layout is shown schematically in Figure 3-2. The layout dimensions for the two Indian Point units are identical. The support configurations of the two Indian Point surge lines are similar. Below is a table summarizing the existing Indian Point surge line support configuration.

Indian Point Units 2 and 3

Support			
Unit 2	Unit 3	Node	Type
PWR-122	PWR-122	1500	Pipe Whip Restraint
PWR-123	PWR-123	2100	Pipe Whip Restraint
-	PWR-124	2700	Pipe Whip Restraint
PWR-120	PWR-120	3400	Pipe Whip Restraint
PWR-121	PWR-121	3500	Pipe Whip Restraint
PWR-125	PWR-125	4300	Pipe Whip Restraint
RCH-76	PW-H-63-1	3100	Variable Spring Hanger
RCH-78	-	3450	Swing Brace

It can be seen from the table above that both of the Indian Point surge lines contain one variable spring hanger with Unit 2 having five pipe whip restraints and Unit 3 having six pipe whip restraints. In addition, there is a horizontal sway brace in the Unit 2 surge line. In some cases these supports can cause higher thermal loads if displacement from thermal stratification exceed available gap limits. The piping sizes are 14 inch schedule 140, and the pipe material is stainless steel, SA 376-Type 316, for both units.

3.2 Piping System Global Structural Analysis

The Indian Point Units 2 and 3 piping systems were modeled by pipe, elbow, and non-linear spring elements using the ANSYS computer code described in Appendix A. The geometric and material parameters are included. [

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Each thermal profile loading defined in section 2 was broken into [

]^{a,c,e} Table 3-1 shows the loading cases considered in the analysis. To encompass all plant operations, [

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[ ]^{a,c,e} Consequently, all the thermal transient loadings defined in section 2 could be evaluated.

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In order to meet the ASME Section III Code stress limits, global structural models of the surge lines were developed using the information provided by references 6 and 7 and the ANSYS general purpose finite element computer code. Each model was constructed using [ ]^{a,c,e} to reflect the layout of straight pipe, bends and field welds as shown in Figure 3-2.

For the stratified condition, [ ]^{a,c,e} These temperature distributions were established from the transients, as discussed in section 2.0. The maximum system delta T was taken as 320°F for the future condition. This corresponds to [

]^{a,c,e}

The global piping stress analysis was based on two structural models for the Indian Point Units. The first model represents the existing support configuration of Unit 2 and the second model represents the existing support configuration of Unit 3. The existing configuration has the actual gaps at al. whip restraints. In the analysis, no spring can bottom-out condition was assumed. This assumption will be assured and verified for Indian Point Units 2 and 3 by Con Ed and NYPA respectively. In addition, the beneficial effect of insulation crushability was taken into account for the existing configuration. The results of the ANSYS global structural analyses provide the thermal expansion moments. The ASME Section III equation (12) stress intensity range was evaluated for both units. For the Indian Point units, the maximum ASME equation (12) stress intensity range in the surge line for a system delta T of 320°F was found to be under the code allowable [4] of 3Sm for the existing configuration, without the spring hanger bottomed out. Maximum equation (12) and equation (13) stress intensity ranges are shown in Table 3-2.

The pressurizer nozzle loads from thermal stratification in the surge line based on no spring bottomed out configuration, were also evaluated according to the requirements of the ASME Code [18]. The evaluation using transients detailed in Reference [13] plus the moment loading from this analysis calculated primary plus secondary stress intensities and the fatigue usage factors. For the Unit 2 and 3 pressurizer nozzles, the maximum intensity range is 44.9 ksi compared to the code allowable value of 57.9 ksi for a material of SA 216 Grade WCC. The maximum fatigue usage factor will be reported in Section 5. It was found that the Indian Point pressurizer surge nozzles met the code stress requirements.

3.3 Local Stresses-Methodology and Results

3.3.1 Explanation of Local Stress

Figure 3-3 depicts the local axial stress components in a beam with a sharply nonlinear metal temperature gradient. Local axial stresses develop due to the restraint of axial expansion or contraction. This restraint is provided by the material in the adjacent beam cross section. For a linear top-to-bottom temperature gradient, the local axial stress would not exist. [

]a,c,e

3.3.2 Finite Element Model of Pipe for Local Stress

A short description of the pipe finite element model is shown in Figure 3-4. The model with thermal boundary conditions is shown in Figure 3-5. Due to symmetry of the geometry and thermal loading, only half of the cross section was required for modeling and analysis. [

]a,c,e

3.3.3 Pipe Local Stress Results

Figure 3-6 shows the temperature distributions through the pipe wall [

]a,c,e

3.3.4 RCL Hot Leg Nozzle Analysis

Detailed surge line nozzle finite element models were developed to evaluate the effects of thermal stratification. The 14 inch schedule 140 model is shown in Figure 3-10. Loading cases included [

]a,c,e A summary of stresses in the RCL nozzle (location 1) due to thermal stratification is given in Tables 3-3A and 3-3B. A summary of representative stresses for unit loading is shown in Table 3-4.

3.4 Total Stress from Global and Local Analyses

[

]a,c,e

[

]a,c,e

3.5 Thermal Striping

3.5.1 Background

At the time when the feedwater line cracking problems in PWR's were first discovered, it was postulated that thermal oscillations (striping) may significantly contribute to the fatigue cracking problems. These oscillations were thought to be due to either mixing of hot and cold fluid, or turbulence in the hot-to-cold stratification layer from strong buoyancy forces during low flow rate conditions. (See Figure 3-11 which shows the thermal striping fluctuation in a pipe). Thermal striping was verified to occur during subsequent flow model tests. Results of the flow model tests were used to establish boundary conditions for the stratification analysis and to provide striping oscillation data for evaluating high cycle fatigue.

Thermal striping was also examined during water model flow tests performed for the Liquid Metal Fast Breeder Reactor (LMFBR) primary pipe loop. The stratified flow was observed to have a dynamic interface region which oscillated in a wave pattern. These dynamic oscillations were shown to produce significant fatigue damage (primary crack initiation). The same interface oscillations were observed in experimental studies of thermal striping which were performed in Japan by Mitsubishi Heavy Industries. The thermal striping evaluation process was discussed in detail in references 3, 8, 9, and 10.

3.5.2 Thermal Striping Stresses

Thermal striping stresses are a result of differences between the pipe inside surface wall and the average through wall temperatures which occur with time, due to the oscillation of the hot and cold stratified boundary. (See Figure 3-12, which shows a typical temperature distribution through the pipe wall). [

]^{a,c,e}

The peak stress range and stress intensity was calculated from a 3-D finite element analysis. [

]^{a,c,e} The methods used to determine alternating stress intensity are defined in the ASME Code [4]. Several locations were evaluated in order to determine the location where stress intensity was a maximum.

Stresses were intensified by K_3 to account for the worst stress concentration for all piping elements in the surge line. The worst piping element was the butt weld.

[

]a,c,e

3.5.3 Factors Which Affect Striping Stress

The factors which affect striping are discussed briefly below:

[

]a,c,e

[

]a,c,e

TABLE 3-1

TEMPERATURE DATA USED IN THE ANALYSIS

Type of Operation	Max System ΔT(°F)	Analysis Cases	Pressurizer Temp (°F)	RCL Temp (°F)	T_{Top} (°F)	T_{Bot} (°F)	Pipe ΔT (°F)
[							]^{a,c,e}

TABLE 3-2

Summary of Indian Point Units 2 & 3 Surge Lines
Thermal Stratification Stress Results

ASME Code Equation	Stress Unit 2	Stress Unit 3	Code Allowable (ksi)
12	41.4*	52.6**	52.9
13	46.6	46.6	50.1

*at 5D bend underneath the pressurizer nozzle
**at pipe side of pressurizer nozzle safe end

TABLE 3-3A

INDIAN POINT UNIT 2 SURGE LINE
MAXIMUM LOCAL AXIAL STRESS AT ANALYZED LOCATIONS

Profile Location*	Surface	Local Axial Stress (psi) Maximum Tensile	Maximum Compressive

[ ]a,c,e

* See Figure 3-5

** RCL nozzle transition

*** RCL nozzle safe end and weld

[ ]a,c,e

TABLE 3-3B

INDIAN POINT UNIT 3 SURGE LINE

MAXIMUM LOCAL AXIAL STRESS AT ANALYZED LOCATIONS

Profile Location*	Surface	Local Axial Stress (psi) Maximum Tensile	Maximum Compressive
[			]a,c,e

* See Figure 3-5

** RCL nozzle safe end

*** RCL nozzle safe end weld

[ ]a,c,e

TABLE 3-4
SUMMARY OF PRESSURE AND BENDING INDUCED STRESSES
IN THE SURGE LINE RCL NOZZLE FOR UNIT LOAD CASES

			All Stress in psi			
			Linearized Stress Intensity Range		Peak Stress Intensity Range	
Location	Diametral Location	Unit Loading Condition	Inside	Outside	Inside	Outside

a,c,e

TABLE 3-5

STRIPING FREQUENCY AT 2 MAXIMUM LOCATIONS FROM 15 TEST RUNS

	Frequency (HZ)						Total Duration # Cycles Lgth. in Seconds
	Min	% (Duration)	Max	% (Duration)	Avg	% (Duration)	

a,c,e

a,c,e

Figure 3-1. Schematic of Stress Analysis Procedure

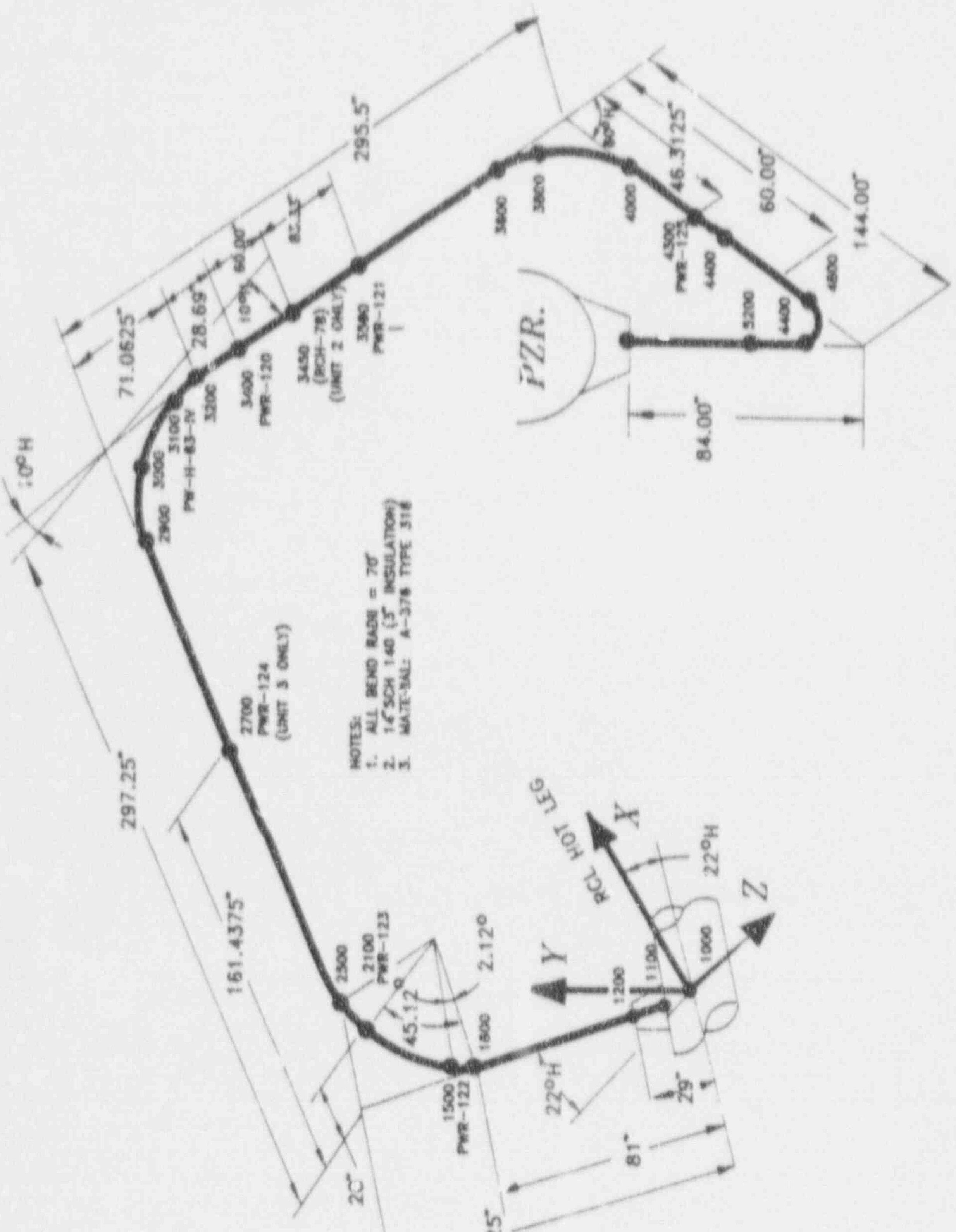

Figure 3-2. Pressurizer Surge Line Layout: Indian Point Units 2 & 3

a,c,e

Figure 3-3. Local Axial Stress in Piping Due to Thermal Stratification

a,c,e

Figure 3-4. Local Stress - Finite Element Models/Loading

a,c,e

Figure 3-5. Piping Local Stress Model and Thermal Boundary Conditions

a,c,e

Figure 3-6. Surge Line Temperature Distribution at [ ]^{a,c,e} Axial Locations

a,c,e

Figure 3-7. Surge Line Local Axial Stress Distribution at [ ]a,c,e Axial Locations

a,c,e

Figure 3-8. Surge Line Local Axial Stress on Inside Surface at [ ]^{a,c,e} Axial Locations

a,c,e

Figure 3-9. Surge Line Local Axial Stress on Outside Surface at [ ]^{a,c,e} Axial Locations

a,c,e

Figure 3-10. Surge Line RCL Nozzle 3-D WECAN Model: 14 Inch Schedule 140

a,c,e

Figure 3-11. Thermal Striping Fluctuation

a,c,e

Figure 3-12. Thermal Striping Temperature Distribution

SECTION 4.0
DISPLACEMENTS AT SUPPORT LOCATIONS

The Indian Point Units 2 and 3 plant specific piping displacements at the pipe whip restraints and hanger locations along the surge line were calculated under the thermal stratification and normal thermal loads.

Table 4-1 shows the maximum surge line piping displacements (from all stratified conditions) at whip restraint and spring hanger locations for Units 2 and 3. Table 4-2 shows the maximum surg' line piping displacements from the normal operation thermal condition at whip restraint and spring hanger locations for both units. The support configuration used in analysis is based on the existing whip restraint gaps and no bottomed-out spring hanger. The pipe whip restraint gaps used in the analysis are listed in Table 4-3.

In the analysis of Unit 2, PWR-120 contacts the pipe when maximum system ΔT of 320°F is considered. The resulting load on PWR-120 is 8.3 kips, which is judged to be small compared to its design capacity.

In the analysis of Unit 3, PWR-124 contacts the pipe when maximum system ΔT of 320°F is considered. The resulting load on PWR-124 is 20.7 kips which is judged to be small compared to its design capacity.

TABLE 4-1

Maximum Piping Displacement Under Stratified Conditions*

	Unit 3			Unit 2		
Node No.	Ux (in)	Uy (in)	Uz (in)	Ux (in)	Uy (in)	Uz (in)

[

]a,c,e

* All stratified cases are defined in Table 3-1

**Variable spring hanger location (RCH-76 for Unit 2 and PW-H-63-1 for Unit 3)

TABLE 4-2

Maximum Piping Displacement Under Normal Thermal Conditions

	Units 2 & 3		
Node No.	Ux (in)	Uy (in)	Uz (in)

[

]a,c,e

* Variable spring hanger location (RCH-76 for Unit 2 and PW-H-63-1 for Unit 3)

TABLE 4-3

PIPE WHIP RESTRAINT GAPS

PWR-	Side*	Gap (inches) Unit 2	Unit 3
[			
			]a,c,e

*Looking toward pressurizer from RCL branch nozzle

**This restraint limits pipe movement in other (vertical) directions

SECTION 5.0
ASME SECTION III FATIGUE USAGE FACTOR EVALUATION

5.1 Methodology

Surge line fatigue evaluations have typically been performed using the methods of ASME Section III, NB-3600 for all piping components [

]^{a,c,e} Because of the nature of the stratification loading, as well as the magnitudes of the stresses produced, the more detailed and accurate methods of NB-3200 were employed using finite element analysis for all loading conditions. Application of these methods, as well as specific interpretation of Code stress values to evaluate fatigue results, is described in this section.

Inputs to the fatigue evaluation included the transients developed in section 2.0, and the global loadings and resulting stresses obtained using the methods described in section 3.0. In general, the stresses due to stratification were categorized according to the ASME Code methods and used to evaluate Code stresses and fatigue cumulative usage factors. It should be noted that, [

]^{a,c,e}

5.1.1 Basis

The ASME Code, Section III, 1986 Edition [4] was used to evaluate fatigue on surge lines with stratification loading. This was based on the recommendation of NRC Bulletin 88-11 (Appendix B of this report) to use the "latest ASME Section III requirements incorporating high cycle fatigue". Specific requirements for class 1 fatigue evaluation of piping components are given in NB-3653. These requirements must be met for Level A and Level B type loadings according to NB-3653 and NB-3654.

According to NB-3611 and NB-3630, the methods of NB-3200 may be used in lieu of the NB-3600 methods. This approach was used to evaluate the surge line components under stratification loading. Since the NB-3650 requirements and equations correlate to those in NB-3200, the results of the fatigue evaluation are reported in terms of the NB-3650 piping stress equations. These equations and requirements are summarized in Tables 5-1 and 5-2.

The methods used to evaluate Code limits for the surge line components are described in the following sections.

5.1.2 Fatigue Stress Equations

Stress Classification

The stresses in a component are classified in the ASME Code based on the nature of the stress, the loading that causes the stress, and the geometric characteristics that influence the stress. This classification determines the acceptable limits on the stress values and, in terms of NB-3653, the respective equation where the stress should be included. Table NB-3217-2 provides guidance for stress classification in piping components, which is reflected in terms of the NB-3653 equations.

The terms in Equations 10, 11, 12 and 13 include stress indices which adjust nominal stresses to account for secondary and peak effects for a given component. Equations 10, 12 and 13 calculate secondary stresses, which are obtained from nominal values using stress indices C1, C2, C3 and C3' for pressure, moment and thermal transient stresses. Equation 11 includes the K1, K2 and K3 indices in the pressure, moment and thermal transient stress terms in order to represent peak stresses caused by local concentration, such as notches and weld effects. The NB-3653 equations use simplified formulas to determine nominal stress based on straight pipe dimensions. [

]a,c,e

For the RCL nozzles, three dimensional (3-D) finite element analysis was used as described in Section 3.0. [

]a,c,e

Classification of local stress due to thermal stratification was addressed with respect to the thermal transient stress terms in the NB-3653 equations. Equation 10 includes a Ta-Tb term, classified as "Q" stress in NB-3200, which represents stress due to differential thermal expansion at gross structural discontinuities. [

]a,c,e The impact of this on the selection of components for evaluation is discussed in Section 5.1.3.

Stress Combinations

The stresses in a given component due to pressure, moment and local thermal stratification loadings were calculated using the finite element models described in Section 3.0. [

]^{a,c,e} This was done for specific components as follows:

[

]^{a,c,e}

[

]a,c,e

From the stress profiles created the stresses for Equations 10 and 11 could be determined for any point i: :.e section. Experience with the geometries and loading showed that certain points in the finite element models consistently produced the worst case fatigue stresses and resulting usage factors, in each stratified axial location. [

]a,c,e

Equation 12 Stress

Code Equation 12 stress represents the maximum range of stress due to thermal expansion moments as described in Section 3.2. This used an enveloping approach, identifying the highest stressed location in the model. By evaluating the worst locations in this manner, the remaining locations were inherently addressed.

Equation 13 Stress

Equation 13 stress, presented in Section 3.2, is due to pressure, design mechanical loads and differential thermal expansion at structural discontinuities. Based on the transient set defined for stratification, the design pressures were not significantly different from previous design transients. Design mechanical loads are defined as deadweight plus seismic OBE loads.

The "Ta-Tb" term of Equation 13 is only applicable at structural discontinuities. [

]a,c,e

Thermal Stress Ratchet

The requirements of NB-3222.5 are a function of the thermal transient stress and pressure stress in a component, and are independent of the global moment loading. As such, these requirements were evaluated for controlling components using applicable stresses due to pressure and stratification transients.

Allowable Stresses

Allowable stress, Sm, was determined based on note 3 of Figure NB-3222-1. For secondary stress due to a temperature transient or thermal expansion loads ("restraint of free end deflection"), the value of Sm was taken as the average of the Sm values at the highest and lowest temperatures of the metal during the transient. The metal temperatures were determined from the transient definition. When part of the secondary stress was due to mechanical load, the value of Sm was taken at the highest metal temperature during the transient.

5.1.3 Selection of Components for Evaluation

Based on the results of the global analyses and the considerations for controlling stresses in Section 5.1.2, [

]a,c,e The method to evaluate usage factors using stresses determined according to Section 3.0 is described below.

5.2 Fatigue Usage Factors

Cumulative usage factors were calculated for the controlling components using the methods described in NB-3222.4(e), based on NB-3653.5. Application of these methods is summarized below.

Transient Loadcases and Combinations

From the transients described in Section 2.0, specific loadcases were developed for the usage evaluation. [

]a,c,e

Each loadcase was assigned the number of cycles of the associated transient as defined in Section 2.0. These were input to the usage factor evaluation, along with the stress data as described above.

Usage factors were calculated at controlling locations in the component as follows:

1) Equation 10, Ke, Equation 11 and resulting Equation 14 (alternating stress - Salt) are calculated as described above for every possible combination of the loadsets.

2) For each value of Salt, the design fatigue curve was used to determine the maximum number of cycles which would be allowed if this type of cycle were the only one acting. These values, N_1, $N_2 \ldots N_n$, were determined from Code Figures I-9.2.1 and I-9.2.2, curve C, for austenitic stainless steels.

3) Using the actual cycles of each transient loadset, n_1, $n_2, \ldots n_n$, calculate the usage factors U_1, $U_2 \ldots U_n$ from $U_i = n_i/N_i$. This is done for all possible combinations. Cycles are used up for each combination in the order of decreasing Salt. When N_i is greater than 10^{11} cycles, the value of U_i is taken as zero.

[

]a,c,e

4) The cumulative usage factor, Ucum, was calculated as Ucum = U_1 + U_2 + ... + U_n. To this was added the usage factor due to thermal striping, as described below, to obtain total Ucum. The Code allowable value is 1.0.

5.3 Fatigue Due to Thermal Striping

The usage factors calculated using the methods of Section 5.2 do not include the effects of thermal striping. [

]^{a,c,e}

Thermal striping stresses are a result of differences between the pipe inside surface wall and the average through wall temperatures which occur with time, due to the oscillation of the hot and cold stratified boundary. This type of stress is defined as a thermal discontinuity peak stress for ASME fatigue analysis. The peak stress is then used in the calculation of the ASME fatigue usage factor.

[

]^{a,c,e} The methods used to determine alternating stress intensity are defined in the ASME code. Several locations were evaluated in order to determine the location where stress intensity was a maximum.

Thermal striping transients are shown as a ΔT level and number of cycles. The striping ΔT for each cycle of every transient is assumed to attenuate and follow the slope of the curve shown on Figure 5-2. Figure 5-2 is conservatively represented by a series of 5 degree temperature steps. Each step lasts [ ]^{a,c,e} seconds. [ ]^{a,c,e} is used in all of the usage factor calculations, the total fluctuations per step is constant and becomes:

[ ]^{a,c,e}

Each striping transient is a group of steps with [ ]^{a,c,e} fluctuations per step. For each transient, the steps begin at the maximum ΔT and decreases by [ ]^{a,c,e} steps down to the endurance limit of ΔT equal to [ ]^{a,c,e} The cycles for all transients which have a temperature step at the same level were added together. This became the total cycles at a step. The total cycles were multiplied by [ ]^{a,c,e} to obtain total fluctuations. This results in total fluctuations at each step. This calculation is performed for each step plateau from [ ]^{a,c,e} to obtain total fluctuations. Allowable fluctuations and ultimately a usage factor at each plateau is calculated from the stress which exists at the ΔT for each step. The total striping usage factor is the sum of all usage factors from each plateau.

The usage factor due to striping, alone, was calculated to be a maximum of [ ]^{a,c,e} This is reflected in the results to be discussed below.

5.4 Fatigue Usage Results

NRC Bulletin 88-11 [5] requests that fatigue analysis should be performed in accordance with the latest ASME III requirements incorporating high cycle fatigue and thermal stratification transients. ASME fatigue usage factors have been calculated considering the phenomenon of thermal stratification and thermal striping at various locations in the surge line. Total stresses included the [

]^{a,c,e} The total stresses for all transients in the bounding set were used to form combinations to calculate alternating stresses and resulting fatigue damage in the manner defined by the Code. Of this total stress, the stresses in the 14 inch pipe due to [

]^{a,c,e}

The maximum usage factor on Indian Point surge lines occurred at [

]^{a,c,e} In this thermal fatigue evaluation, welded attachments at PWR-120 and PWR-121 were also included, and it was found that the usage factors were smaller than the maximum value listed above due to lower total loadings and stresses at the lug locations.

It is also concluded that the Indian Point pressurizer surge nozzles meet the Code stress allowables under the thermal stratification loading from the surge line, with no spring hanger bottomed out configuration and the transients detailed in reference [13]. They also meet the fatigue usage requirements of ASME Section III, with a maximum cumulative usage factor equal to 0.26 [18].

TABLE 5-1

CODE/CRITERIA

- ASME B&PV Code Sec. III, 1986 Edition
  - NB3600
  - NB3200

- Level A/B Service Limits
  - Primary Plus Secondary Stress Intensity $\leq$ 3Sm (Eq. 10)

  - Simplified Elastic-Plastic Analysis

    - Expansion Stress, $S_e \leq$ 3Sm (Eq. 12) - Global Analysis
    - Primary Plus Secondary Excluding Thermal Bending < 3Sm (Eq. 13)
    - Elastic-Plastic Penalty Factor $1.0 \leq K_e \leq 3.333$

  - Peak Stress (Eq. 11)/Cumulative Usage Factor (U_{cum})

    - $S_{alt} = K_e S_p/2$ (Eq. 14)
    - Design Fatigue Curve
    - $U_{cum} \leq 1.0$

TABLE 5-2

SUMMARY OF ASME FATIGUE REQUIREMENTS

Parameter	Description	Allowable (if applicable)
Equation 10	Primary plus secondary stress intensity; if exceeded, simplified elastic-plastic analysis may be performed	< 3Sm
K_e	Elastic-plastic penalty factor; required for simplified elastic-plastic analysis when Eq. 10 is exceeded; applied to alternating stress intensity	
Equation 12	Expansion stress; required for simplified elastic-plastic analysis when Eq. 10 is exceeded	< 3Sm
Equation 13	Primary plus secondary stress intensity excluding thermal bending stress; required for simplified elastic-plastic analysis when Eq. 10 is exceeded	< 3Sm
Thermal Stress Ratchet	Limit on radial thermal gradient stress to prevent cyclic distortion; required for use of Eq. 13	
Equation 11	Peak stress intensity - Input to Eq. 14	
Equation 14	Alternating stress intensity - Input to Ucum	
Ucum	Cumulative usage factor (fatigue damage)	< 1.0

a,c,e

Figure 5-1. Striping Finite Element Model

a,c,e

Figure 5-2. Attenuation of Thermal Striping Potential by Molecular Conduction (Interface Wave Height of One Inch)

SECTION 6.0
ANALYSIS SUMMARY

The subject of pressurizer surge line integrity has been under intense investigation since 1988. The NRC issued Bulletin 88-11 in December of 1988, but the Westinghouse Owners Group had put a program in place earlier that year, and this allowed all members to make a timely response to the bulletin.

The Owners Group programs were completed in June of 1990, and have been followed by a series of plant specific evaluations. This report has documented the results of the plant specific evaluation for Indian Point Units 2 and 3.

Following the general approach used in developing the surge line stratification transients for the WOG, a set of transients and stratification profile were developed specifically for Indian Point Units 2 and 3. A study was made of the historical operating experience at the Indian Point Units 2 and 3, and this information, as well as plant operating procedures and monitoring results (from similar plants), was used in development of the transients and profiles.

The analysis results are shown in Section 3.0 for ASME code stress, Section 4.0 for displacements at supports and whip restraint locations and Section 5.0 for ASME code fatigue cumulative usage factors. The results were conservatively calculated using the maximum design temperature differential and worst case assumptions for inducing thermal stratification to the system.

SECTION 7.0
REFERENCES

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16. Con Edison Memorandum IPQA#9-0354, "Pressurizer Surge Line Examination," May 23, 1989.

17. New York Power Authority Memorandum IP-89-52, "Visual Inspection of IP3's Pressurizer's Surge Line," February 15, 1989.

18. Westinghouse Letter AEA-91-100, "IPP/INT Pressurizer Surge Nozzle Analysis for Pipe Loads due to Thermal Stratification," May 2, 1991 (Westinghouse central file IPP/INT-160-2A3).

APPENDIX A

LIST OF COMPUTER PROGRAMS

This appendix lists and summarizes the computer codes used in the pressurizer surge line thermal stratification. The codes are:

1. WECAN
2. STRFAT2
3. ANSYS
4. FATRK/CMS

A.1 WECAN

A.1.1 Description

WECAN is a Westinghouse-developed, general purpose finite element program. It contains universally accepted two-dimensional and three-dimensional isoparametric elements that can be used in many different types of finite element analyses. Quadrilateral and triangular structural elements are used for plane strain, plane stress, and axisymmetric analyses. Brick and wedge structural elements are used for three-dimensional analyses. Companion heat conduction elements are used for steady state heat conduction analyses and transient heat conduction analyses.

A.1.2 Feature Used

The temperatures obtained from a static heat conduction analysis, or at a specific time in a transient heat conduction analysis, can be automatically input to a static structural analysis where the heat conduction elements are replaced by corresponding structural elements. Pressure and external loads can also be include in the WECAN structural analysis. Such coupled thermal-stress analyses are a standard application used extensively on an industry wide basis.

A.1.3 Program Verification

Both the WECAN program and input for the WECAN verification problems, currently numbering over four hundred, are maintained under configuration control. Verification problems include coupled thermal-stress analyses for the quadrilateral, triangular, brick, and wedge isoparametric elements. These problems are an integral part of the WECAN quality assurance procedures. When a change is made to WECAN, as part of the reverification process, the configured inputs for the coupled thermal-stress verification problems are used to reverify WECAN for coupled thermal-stress analyses.

A.2 STRFAT2

A.2.1 Description

STRFAT2 is a program which computes the alternating peak stress on the inside surface of a flat plate and the usage factor due to striping on the surface. The program is applicable to be used for striping on the inside surface of a pipe if the program assumptions are considered to apply for the particular pipe being evaluated.

For striping the fluid temperature is a sinusoidal variation with numerous cycles.

The frequency, convection film coefficient, and pipe material properties are input.

The program computes maximum alternating stress based on the maximum difference between inside surface skin temperature and the average through wall temperature.

A.2.2 Feature Used

The program is used to calculate striping usage factor based on a ratio of actual cycles of stress for a specified length of time divided by allowable cycles of stress at maximum the alternating stress level. Design fatigue curves for several materials are contained into the program. However, the user has the option to input any other fatigue design curve, by designating that the fatigue curve is to be user defined.

A.2.3 Program Verification

STRFAT2 is verified to Westinghouse procedures by independent review of the stress equations and calculations.

A.3 ANSYS

A.3.1 Description

ANSYS is a public domain, general purpose finite element code.

A.3.2 Feature Used

The ANSYS elements used for the analysis of stratification effects in the surge line are STIF 20 (straight pipe), STIF 60 (elbow and bends) and STIF14 (spring-damper for supports).

A.3.3 Program Verification

As described in section 3.2, the application of ANSYS for stratification has been independently verified by comparison to WESTDYN (Westinghouse piping analysis code) and WECAN (finite element code). The results from ANSYS are also verified against closed form solutions for simple beam configurations.

A.4 FATRK/CMS

A.4.1 Description

FATRK/CMS is a Westinghouse developed computer code for fatigue tracking (FATRK) as used in the Cycle Monitoring System (CMS) for structural components of nuclear power plants. The transfer function method is used for transient thermal stress calculations. The bending stresses (due to global stratification effects, ordinary thermal expansion and seismic) and the pressure stresses are also included. The fatigue usage factors are evaluated in accordance with the guidelines given in the ASME Boiler and Pressure Vessel Code, Section III, Subsections NB-3200 and NB-3600.

The code can be used both as a regular analysis program or an on-line monitoring device.

A.4.2 Feature Used

FATRK/CMS is used as an analysis program for the present application. The input data which include the weight functions for thermal stresses, the unit bending stress, the unit pressure stress, the bending moment vs. stratification temperatures, etc. are prepared for all locations and geometric conditions. These data, as stored in the independent files, can be appropriately retrieved for required analyses. The transient data files contain the time history of temperature, pressure, number of occurrence, and additional condition necessary for data flowing. The program prints out the total usage factors, and the transients pairing information which determine the stress range magnitudes and number of cycles. The detailed stress data may also be printed.

A.4.3 Program Verification

FATRK/CMS is verified according to Westinghouse procedures with several levels of independent calculations.

APPENDIX B

USNRC BULLETIN 88-11

In December of 1988 the NRC issued this bulletin, and it has led to an extensive investigation of surge line integrity, culminating in this and other plant specific reports. The bulletin is reproduced in its entirety in the pages which follow.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

December 20, 1988

NRC BULLETIN NO. 88-11: PRESSURIZER SURGE LINE THERMAL STRATIFICATION

Addressees:

All holders of operating licenses or construction permits for pressurized water reactors (PWRs).

Purpose:

The purpose of this bulletin is to (1) request that addressees establish and implement a program to confirm pressurizer surge line integrity in view of the occurrence of thermal stratification and (2) require addressees to inform the staff of the actions taken to resolve this issue.

Description of Circumstances.

The licensee for the Trojan plant has observed unexpected movement of the pressurizer surge line during inspections performed at each refueling outage since 1982, when monitoring of the line movements began. During the last refueling outage, the licensee found that in addition to unexpected gap closures in the pipe whip restraints, the piping actually contacted two restraints. Although the licensee had repeatedly adjusted shims and gap sizes based on analysis of various postulated conditions, the problem had not been resolved. The most recent investigation by the licensee confirmed that the movement of piping was caused by thermal stratification in the line. This phenomenon was not considered in the original piping design. On October 7, 1988, the staff issued Information Notice 88-80, "Unexpected Piping Movement Attributed to Thermal Stratification," regarding the Trojan experience and indicated that further generic communication may be forthcoming. The licensee for Beaver Valley 2 has also noticed unusual snubber movement and significantly larger-than-expected surge line displacement during power ascension.

The concerns raised by the above observations are similar to those described in NRC Bulletins 79-13 (Revision 2, dated October 16, 1979), "Cracking in Feedwater System Piping" and 88-08 (dated June 22, 1988), "Thermal Stresses in Piping Connected to Reactor Coolant Systems."

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Discussion:

Unexpected piping movements are highly undesirable because of potential high piping stress that may exceed design limits for fatigue and stresses. The problem can be more acute when the piping expansion is restricted, such as through contact with pipe whip restraints. Plastic deformation can result, which can lead to high local stresses, low cycle fatigue and functional impairment of the line. Analysis performed by the Trojan licensee indicated that thermal stratification occurs in the pressurizer surge line during heatup, cooldown, and steady-state operations of the plant.

During a typical plant heatup, water in the pressurizer is heated to about 440°F; a steam bubble is then formed in the pressurizer. Although the exact phenomenon is not thoroughly understood, as the hot water flows (at a very low flowrate) from the pressurizer through the surge line to the hot-leg piping, the hot water rides on a layer of cooler water, causing the upper part of the pipe to be heated to a higher temperature than the lower part (see Figure 1). The differential temperature could be as high as 300°F, based on expected conditions during typical plant operations. Under this condition, differential thermal expansion of the pipe metal can cause the pipe to deflect significantly.

For the specific configuration of the pressurizer surge line in the Trojan plant, the line deflected downward and when the surge line contacted two pipe whip restraints, it underwent plastic deformation, resulting in permanent deformation of the pipe.

The Trojan event demonstrates that thermal stratification in the pressurizer surge line causes unexpected piping movement and potential plastic deformation. The licensing basis according to 10 CFR 50.55a for all PWRs requires that the licensee meet the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Sections III and XI and to reconcile the pipe stresses and fatigue evaluation when any significant differences are observed between measured data and the analytical results for the hypothesized conditions. Staff evaluation indicates that the thermal stratification phenomenon could occur in all PWR surge lines and may invalidate the analyses supporting the integrity of the surge line. The staff's concerns include unexpected bending and thermal striping (rapid oscillation of the thermal boundary interface along the piping inside surface) as they affect the overall integrity of the surge line for its design life (e.g., the increase of fatigue).

Actions Requested:

Addressees are requested to take the following actions:

1. For all licensees of operating PWRs:

   a. Licensees are requested to conduct a visual inspection (ASME, Section XI, VT-3) of the pressurizer surge line at the first available cold shutdown after receipt of this bulletin which exceeds seven days.

This inspection should determine any gross discernable distress or structural damage in the entire pressurizer surge line, including piping, pipe supports, pipe whip restraints, and anchor bolts.

b. Within four months of receipt of this Bulletin, licensees of plants in operation over 10 years (i.e., low power license prior to January 1, 1979) are requested to demonstrate that the pressurizer surge line meets the applicable design codes* and other FSAR and regulatory commitments for the licensed life of the plant, considering the phenomenon of thermal stratification and thermal striping in the fatigue and stress evaluations. This may be accomplished by performing a plant specific or generic bounding analysis. If the latter option is selected, licensees should demonstrate applicability of the referenced generic bounding analysis. Licensees of plants in operation less than ten years (i.e., low power license after January 1, 1979), should complete the foregoing analysis within one year of receipt of this bulletin. Since any piping distress observed by addressees in performing action 1.a may affect the analysis, the licensee should verify that the bounding analysis remains valid. If the opportunity to perform the visual inspection in 1.a does not occur within the periods specified in this requested item, incorporation of the results of the visual inspection in the analysis should be performed in a supplemental analysis as appropriate.

Where the analysis shows that the surge line does not meet the requirements and licensing commitments stated above for the duration of the license, the licensee should submit a justification for continued operation or bring the plant to cold shutdown, as appropriate, and implement Items 1.c and 1.d below to develop a detailed analysis of the surge line.

c. If the analysis in 1.b does not show compliance with the requirements and licensing commitments stated therein for the duration of the operating license, the licensee is requested to obtain plant specific data on thermal stratification, thermal striping, and line deflections. The licensee may choose, for example, either to install instruments on the surge line to detect temperature distribution and thermal movements or to obtain data through collective efforts, such as from other plants with a similar surge line design. If the latter option is selected, the licensee should demonstrate similarity in geometry and operation.

d. Based on the applicable plant specific or referenced data, licensees are requested to update their stress and fatigue analyses to ensure compliance with applicable Code requirements, incorporating any observations from 1.a above. The analysis should be completed no later than two years after receipt of this bulletin. If a licensee

*Fatigue analysis should be performed in accordance with the latest ASME Section III requirements incorporating high cycle fatigue.

is unable to show compliance with the applicable design codes and other FSAR and regulatory commitments, the licensee is requested to submit a justification for continued operation and a description of the proposed corrective actions for effecting long term resolution.

2. For all applicants for PWR Operating Licenses:

   a. Before issuance of the low power license, applicants are requested to demonstrate that the pressurizer surge line meets the applicable design codes and other FSAR and regulatory commitments for the licensed life of the plant. This may be accomplished by performing a plant-specific or generic bounding analysis. The analysis should include consideration of thermal stratification and thermal striping to ensure that fatigue and stresses are in compliance with applicable code limits. The analysis and hot functional testing should verify that piping thermal deflections result in no adverse consequences, such as contacting the pipe whip restraints. If analysis or test results show Code noncompliance, conduct of all actions specified below is requested.

   b. Applicants are requested to evaluate operational alternatives or piping modifications needed to reduce fatigue and stresses to acceptable levels.

   c. Applicants are requested to either monitor the surge line for the effects of thermal stratification, beginning with hot functional testing, or obtain data through collective efforts to assess the extent of thermal stratification, thermal striping and piping deflections.

   d. Applicants are requested to update stress and fatigue analyses, as necessary, to ensure Code compliance.* The analyses should be completed no later than one year after issuance of the low power license.

3. Addressees are requested to generate records to document the development and implementation of the program requested by Items 1 or 2, as well as any subsequent corrective actions, and maintain these records in accordance with 10 CFR Part 50, Appendix B and plant procedures.

Reporting Requirements:

1. Addressees shall report to the NRC any discernable distress and damage observed in Action 1.a along with corrective actions taken or plans and schedules for repair before restart of the unit.

*If compliance with the applicable codes is not demonstrated for the full duration of an operating license, the staff may impose a license condition such that normal operation is restricted to the duration that compliance is actually demonstrated.

2. Addressees who cannot meet the schedule described in Items 1 or 2 of Actions Requested are required to submit to the NRC within 60 days of receipt of this bulletin an alternative schedule with justification for the requested schedule.

3. Addressees shall submit a letter within 30 days after the completion of these actions which notifies the NRC that the actions requested in Items 1b, 1d or 2 of Actions Requested have been performed and that the results are available for inspection. The letter shall include the justification for continued operation, if appropriate, a description of the analytical approaches used, and a summary of the results.

Although not requested by this bulletin, addressees are encouraged to work collectively to address the technical concerns associated with this issue, as well as to share pressurizer surge line data and operational experience. In addition, addressees are encouraged to review piping in other systems which may experience thermal stratification and thermal striping, especially in light of the previously mentioned Bulletins 79-13 and 88-08. The NRC staff intends to review operational experience giving appropriate recognition to this phenomenon, so as to determine if further generic communications are in order.

The letters required above shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended. In addition, a copy shall be submitted to the appropriate Regional Administrator.

This request is covered by Office of Management and Budget Clearance Number 3150-0011 which expires December 31, 1989. The estimated average burden hours is approximately 3000 person-hours per licensee response, including assessment of the new requirements, searching data sources, gathering and analyzing the data, and preparing the required reports. These estimated average burden hours pertain only to these identified response-related matters and do not include the time for actual implementation of physical changes, such as test equipment installation or component modification. The estimated average radiation exposure is approximately 3.5 person-rems per licensee response.

Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to the Office of Management and Budget, Room 3208, New Executive Office Building, Washington, D.C. 20503, and to the U.S. Nuclear Regulatory Commission, Records and Reports Management Branch, Office of Administration and Resource Management, Washington, D.C. 20555.

If you have any questions about this matter, please contact one of the technical contacts listed below or the Regional Administrator of the appropriate regional office.

Charles E. Rossi
Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

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(301) 492-0904

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(301) 492-0943

N. P. Kadambi, NRR
(301) 492-1153

Attachments:
1. Figure 1
2. List of Recently Issued NRC Bulletins

Surge Line Stratification

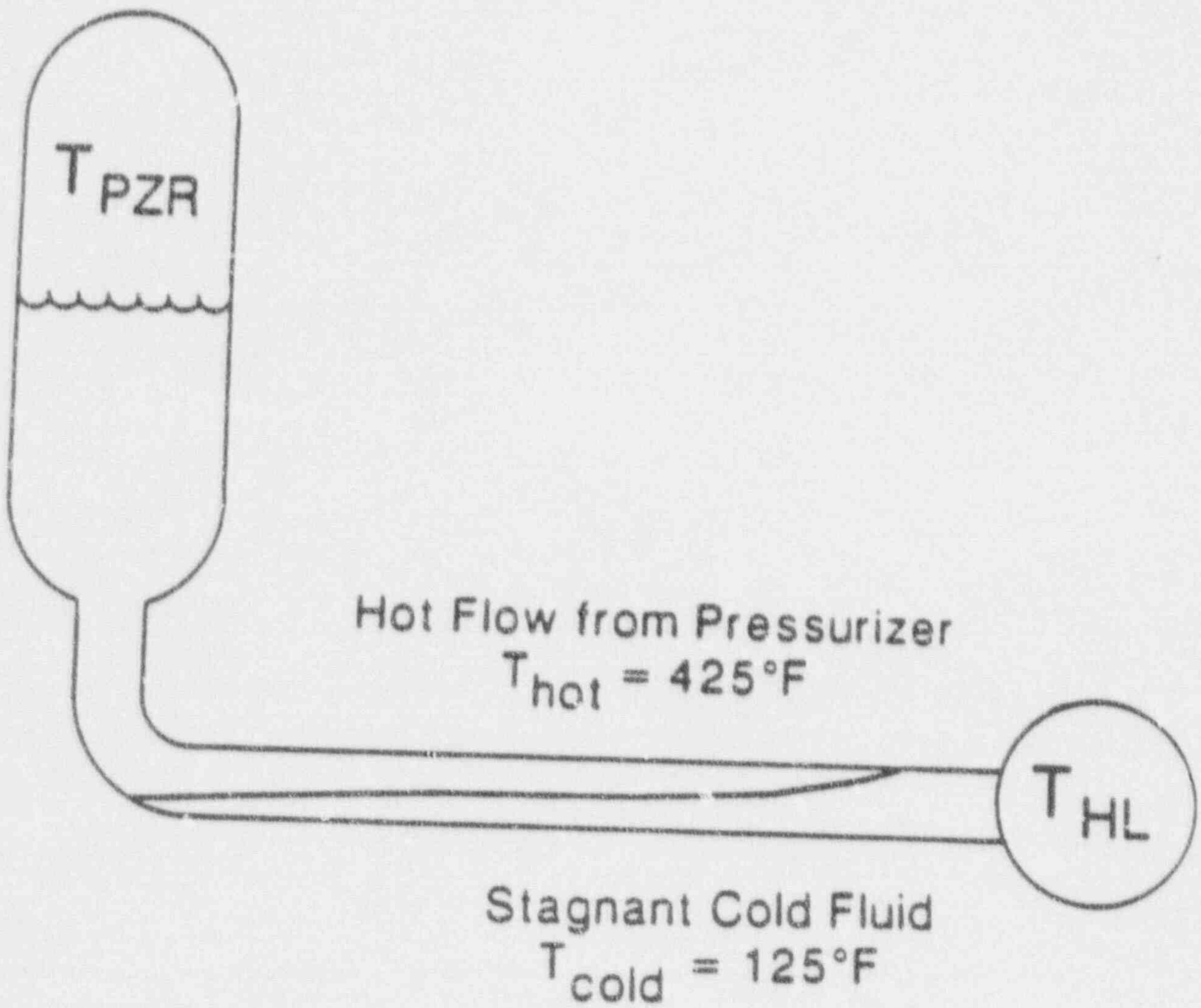

Figure 1

APPENDIX C

TRANSIENT DEVELOPMENT DETAILS

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ENCLOSURE C TO IPN-91-036

Westinghouse: (1) Letter and affidavit regarding "Application for Withholding Proprietary Information from Public Disclosure."; (2) Proprietary Information Notice; (3) Copywrite Notice.

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64