VALVE INLET FLUID CONDITIONS FOR PRESSURIZER SAFETY AND RELIEF VALVES IN WESTINGHOUSE-DESIGNED PLANTS

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RESEARCH PROJECT V102-19
(PHASE C)

INTERIM REPORT, FEBRUARY 1982

Prepared by

WESTINGHOUSE ELECTRIC CORPORATION

Nuclear Energy Systems
P.O. Box 355
Pittsburgh, Pennsylvania 15230

PRINCIPAL INVESTIGATORS

A. Meliksetian
A. M. Sklencar

Prepared for

PARTICIPATING PWR UTILITIES

and

ELECTRIC POWER RESEARCH INSTITUTE
3412 Hillview Avenue
Palo Alto, California 94304

EPRI Project Manager

J. Hosler
Nuclear Power Division

Prepared by

Westinghouse Electric Corporation

Pittsburgh, Pennsylvania

ABSTRACT

The overpressure transients for Westinghouse-designed NSSSs are reviewed to determine the fluid conditions at the inlet to the PORV and safety valves.

The transients considered are:

1. Licensing (FSAR) Transients
2. Extended Operation of High Pressure Safety Injection System
3. Cold Overpressurization

The results of this review, presented in the form of tables and graphs, define the range of fluid conditions expected at the inlet to pressurizer safety and power-operated relief valves utilized in Westinghouse-designed PWR units. These results will provide input to the PWR utilities in their justification that the fluid conditions under which their valve designs were tested as part of the EPRI/PWR Safety and Relief Valve Test Program indeed envelop those expected in their units.

EPRI PERSPECTIVE

PROJECT DESCRIPTION

This report, developed under RPV102-19 in support of the EPRI/PWR Safety and Relief Valve Test Program, presents the expected range of fluid inlet conditions for pressurizer safety and relief valves utilized in PWR units designed by Westinghouse. These conditions are determined based on consideration of FSAR, Extended High Pressure Liquid Injection, and Cold Overpressurization Events.

PROJECT OBJECTIVE

The objective of this report is to assist PWR utilities with Westinghouse plants in demonstrating that the fluid conditions under which their valve designs are tested as part of the aforementioned program envelop those expected in their unit(s).

PROJECT RESULTS

FSAR events are found to result in challenges to both relief and safety valves under steam conditions with valve inlet pressures as high as 2682 psia. Liquid discharge through relief and safety valves is predicted for only one FSAR event, the feedline break accident. Liquid temperatures and surge rates for this event range from 553 to 672 degrees Fahrenheit and 224 to 2989 gallons per minute, respectively.

Extended High Pressure Liquid Injection events are found to result in no relief and safety valve challenges in two loop plants, relief valve challenges in both three and four loop plants, and safety valve challenges only in four loop plants. In cases when the valves are challenged, liquid discharge is also predicted for these events. Liquid temperatures and surge rates for these events range from 498 to 598 degrees Fahrenheit and 0 to 1104 gallons per minute, respectively.

Cold overpressurization events challenge only relief valves. Liquid discharge is predicted for these events at pressures ranging from 280 to 2350 psia with temperatures ranging from 100 to 650 degrees Fahrenheit.

John Hosler, Project Manager
Nuclear Power Division

ACKNOWLEDGEMENT

The contributions of J. M. Thompson, S. L. Ellenberger and C. Allen in the preparation of this report are gratefully acknowledged.

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SUMMARY

This report provides documentation of the expected range of pressurizer safety and relief valve fluid inlet conditions for Westinghouse designed plants. It is intended for use by PWR utilities with Westinghouse units in their justification that the fluid inlet conditions under which their valve designs are tested, as part of the EPRI/PWR Safety and Relief Valve Test Program, envelop those expected in their unit(s). These conditions are determined based on consideration of FSAR, Extended High Pressure Liquid Injection, and Cold Overpressurization events.

The methodology used to determine these conditions includes the grouping of Westinghouse PWR units by design and layout philosophy. For each group, a reference plant is then selected based on maximizing a nondimensional parameter which incorporates the critical plant parameters affecting the severity of FSAR overpressurization events resulting in steam discharge. Valve fluid inlet conditions resulting from limiting FSAR events, which result in steam discharge and an Extended High Pressure Liquid Injection event and which may result in liquid discharge, are presented for each reference plant. These conditions envelop those expected by the plants represented by each reference plant.

From the FSAR events that may result in liquid discharge, the feedline break accident is considered. For this event, plant specific valve inlet conditions are presented where applicable.

Fluid inlet conditions are presented for Cold Overpressurization events which envelop those expected in all units for which Westinghouse has provided the plant specific Cold Overpressurization Protection System design and analysis. These analyses consider the limiting mass and heat input events for each unit evaluated.

Section 1

INTRODUCTION

1.1 BACKGROUND

Following the Three Mile Island Unit 2 (TMI-2) incident, the Nuclear Regulatory Commission (NRC) published NUREG-0578, "TMI-2 Lessons Learned - Task Force Status Report and Short-Term Recommendations," which required utilities operating and in the process of constructing pressurizer water reactor (PWR) power plants to develop a program to demonstrate the operability of power operated relief valves (PORVs) and self-actuated safety valves (PSVs) used in the protection of reactor coolant systems. The requirements of NUREG-0578 were later clarified in NUREG-0737. In response to NUREG-0578 and NUREG-0737 requirements, the PWR utilities assigned EPRI the responsibility of conducting a comprehensive test program to demonstrate the operability of the various types of PORVs and safety valves used by participating utilities. The primary objective of that program is to obtain performance data applicable to each of the various types of reactor coolant system safety and relief valves in PWR plant service for the range of conditions under which they may be required to operate.

As part of their response to the NUREG requirement, each PWR licensee or applicant is required to provide evidence that the conditions under which valves representative of those installed in their unit(s) are tested, are representative of those expected in their units. Such conditions include valve inlet piping configurations, backpressure and dynamic loading as well as fluid inlet state, pressures, and temperature.

To assist in the development of test conditions to be applied to the valves selected for testing, each PWR NSSS vendor was contracted to develop a "Plant Conditions Justification Report" describing the range of fluid conditions expected at the inlets of relief and safety valves installed in plants of their design.

1.2 OBJECTIVE

The objective of this report is to document the justification for a set of limiting fluid inlet conditions to be used as input to the selection of fluid conditions for testing power-operated relief valve (PORV) and safety valve designs used in

Westinghouse plants. This report will be referenced by PWR utilities with Westinghouse plants in their justification that the fluid conditions under which their valve designs are tested, as part of the EPRI program, envelop those expected in their unit(s).

1.3 SCOPE OF WORK

The evaluation of expected fluid inlet conditions is based on consideration of FSAR events, Extended Safety Injection events, and cold overpressurization events. Standard licensing methodology is used to determine the fluid conditions at the inlet of the PORV and safety valves for FSAR events. The events evaluated in a plant's FSAR or in later licensing submittals that have the potential of challenging such valves are considered.

Because of the large number of Westinghouse plants with varying design and layout, reference Plants have been selected for evaluation. The analysis performed on the reference Plant represents the expected behavior of all plants in that group. The reference plants are selected by performing dimensional analyses on their critical parameters and considering the similarity of characteristics, design, and layout. Reference plants are used for licensing-type accident analyses (with the exception of the feedline break accident) and extended high pressure liquid injection events. Plant specific fluid conditions are presented for the feedline break accident, where applicable.

Not all Westinghouse units are covered by this report since several units no longer utilize Westinghouse fuel or have their reload analyses performed by Westinghouse. Those that are covered with respect to FSAR and Extended Safety Injection events are shown below. Of these plants, those for which fluid conditions for cold overpressurization events are presented are identified with an asterisk. Conditions resulting from cold overpressure transients are presented only for plants for which Westinghouse performed the specific design and analysis of their cold overpressurization protection system.

Two-Loop Plants	Name	Owner
RGE	R. E. Ginna	Rochester Gas & Electric Corp.
WEP	Point Beach #1	Wisconsin Electric Power Utilities
WIS	Point Beach #2	Wisconsin Electric Power Utilities

Two-Loop

Plants	Name	Owner
NSP	Prairie Island #1	Northern States Power
NRP	Prairie Island #2	Northern States Power
WPS	Kewaunee	Wisconsin Public Service

Three-Loop

Plants	Name	Owner
SCE	San Onofre #1	Southern California Edison
CPL	H. B. Robinson #2	Carolina Power & Light Co.
FPL	Turkey Point #3	Florida Power & Light Co.
FLA	Turkey Point #4	Florida Power & Light Co.
VPA	Surry #1	Virginia Electric & Power Co.
VIR	Surry #2	Virginia Electric & Power Co.
DLW	Beaver Valley #1	Duquesne Light Co.
VRA	North Anna #1	Virginia Electric & Power Co.
ALA	Joseph M. Farley #1	Alabama Power Co.
VGB	North Anna #2	Virginia Electric & Power Co.
APR	Joseph M. Farley #2	Alabama Power Co.
CGE	Virgil C. Summer #1	South Carolina Electric & Gas
DMW	Beaver Valley #2	Duquesne Light Company
CQL	Shearon Harris #1	Carolina Power & Light Co.
CRL	Shearon Harris #2	Carolina Power & Light Co.
CSL	Shearon Harris #3	Carolina Power & Light Co.
CTL	Shearon Harris #4	Carolina Power & Light Co.

Four-Loop

Plants	Name	Owner
IPP	Indian Point #2	Consolidated Edison Co. of New York
INT	Indian Point #3	Power Authority, State of New York
CWE	Zion #1	Commonwealth Edison
COM	Zion #2	Commonwealth Edison
AEP	Donald C. Cook #1	American Electric Power Co.
AMP	Donald C. Cook #2	American Electric Power Co.
PGE	Diablo Canyon #1	Pacific Gas & Electric Power

Four-Loop Plants	Name	Owner
PEG	Diablo Canyon #2	Pacific Gas & Electric Power
POR	Trojan	Portland General Electric
TVA	* Sequoyah #1	Tennessee Valley Authority
TEN	* Sequoyah #2	Tennessee Valley Authority
PSE	Salem #1	Public Service Electric & Gas
PNJ	Salem #2	Public Service Electric & Gas
DAP	W. B. McGuire #1	Duke Power Co.
DBP	W. B. McGuire #2	Duke Power Co.
WAT	* Watts Bar #1	Tennessee Valley Authority
WBT	* Watts Bar #2	Tennessee Valley Authority
CAE	* Byron #1	Commonwealth Edison Co.
GAE	Alvin W. Vogtle #1	Georgia Power Co.
GBE	Alvin W. Vogtle #2	Georgia Power Co.
NEU	Millstone #3	Northeast Utilities
NAH	Seabrook #1	Public Service Co. of New Hampshire
NCH	Seabrook #2	Public Service Co. of New Hampshire
DCP	Catawba #1	Duke Power Co.
DDP	Catawba #2	Duke Power Co.
TBX	* Comanche Peak #1	Texas Utilities
TCX	* Comanche Peak #2	Texas Utilities
CCE	* Braidwood #1	Commonwealth Edison Co.
CDE	* Braidwood #2	Commonwealth Edison Co.
TGX	* South Texas #1	Houston Light & Power
THX	* South Texas #2	Houston Light & Power
PBJ	Marble Hill #1	Public Service of Indiana
PCJ	Marble Hill #2	Public Service of Indiana
CBE	* Byron #2	Commonwealth Edison Co.
SAP	* Wolf Creek (SNUPPS)	Kansas Gas & Electric Co.
SCP	* Callaway #1 (SNUPPS)	Union Electric Co.
SFP	* Callaway #2 (SNUPPS)	Union Electric Co.

1.4 QUALITY ASSURANCE

The work performed in the development of this report is in accordance with 10CFR50 Appendix B, Quality Assurance requirements.

Section 2

GENERAL DESCRIPTION OF EVENTS THAT HAVE POTENTIAL FOR CHALLENGING SAFETY AND RELIEF VALVES

The events that cause overpressurization of the reactor coolant system are grouped into licensing (FSAR) transients, transients that result from automatic initiation of the high pressure injection system, and cold overpressurization transients.

2.1 LICENSING (FSAR) TRANSIENTS

The transients that result in the actuation of safety and relief valves and are normally analyzed for safety analysis reports can be grouped under Class II and Class IV events.

2.1.1 Class II Events

Class II events are incidents of moderate frequency that may occur during a calendar year for a particular plant. The transients in this class are described in the following paragraphs.

2.1.1.1 Loss of Load. In the event of the loss of external electrical load without bypass, a sudden reduction in steam flow will cause an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and reactor coolant system (RCS) pressure rise.

Both the pressurizer safety valves and main steam safety valves may open for the loss of load event. Only steam is discharged from the pressurizer safety and relief valves and no water discharge is observed.

2.1.1.2 Loss of Normal Feedwater. A loss of normal feedwater results in a reduction of the secondary system's capability to remove the heat

generated in the reactor core. If an alternative supply of feedwater were not supplied, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer occurs.

The analysis normally shows that the pressurizer steam space does not completely fill with water, and therefore only steam discharge through PORV and safety valves is observed.

2.1.1.3 Accidental Depressurization of the Secondary System. The accidental depressurization of the main steam system may result from the inadvertent opening of a single steam dump, relief, or safety valve. This event results in a small increase in nominal steam flow. A much larger increase in steam flow can be caused from steam line rupture.

Initial increase in steam flow increases the energy removal rate from the primary system and causes a reduction of reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a positive reactivity insertion. If the most reactive rod cluster control assembly is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down due to boric acid injection by the Safety Injection System. At hot shutdown, actuation of the safety injection system occurs early enough to prevent criticality.

The above discussion also covers Minor Steam Line Rupture, which is a class III event.[1] For both events the extended operation of the Safety Injection System can result in pressurization of the primary system which will result first in steam discharge and later in water discharge, if safety valves and PORVs are actuated.

[1]Class III events are events which may occur very infrequently during the life of the plant.

2.1.1.4 Loss of Off-Site Power. In the event of complete loss of offsite power and turbine trip, there will be a loss of power to the station auxiliaries, (reactor coolant pumps, condensate pump, and so forth). Reactor coolant flow coasts down to natural circulation flow rates. Main feedwater flow is lost and the auxiliary feed pumps automatically start. As the system pressure rises (due to decay heat input) following the trip, the system's PORVs are automatically opened. If the steam flow rate through the PORVs is not adequate, safety valves may lift to dissipate the excess energy by passing steam.

2.1.1.5 Uncontrolled Rod Withdrawal at Power. A continuous uncontrolled rod cluster control assembly (RCCA) withdrawal at power due to faulty operator actions or malfunction of the reactor instruments will result in an increase in the core heat flux. Following the event, the steam generator heat removal rate will lag behind the core power generation rate until the steam generator pressure reaches the main steam safety/relief valve setpoint. This unbalanced heat removal rate will cause the reactor coolant temperature and pressure to rise and eventually may actuate the PORVs. If the steam flow through the PORVs is not adequate, safety valves may also be actuated to dissipate the excess energy by passing steam.

2.1.2 Class IV Events

Class IV events are limiting faults that are not expected to take place, but are postulated because their consequences include the potential for the release of significant amounts of radioactive material.

2.1.2.1 Main Feedwater Pipe Rupture. A major feedwater line rupture is defined as a break in the feedwater line large enough to prevent the addition of sufficient feedwater to the steam generator to maintain shell side fluid inventory in the steam generators. If the break is postulated to occur in the feedline between the check valve and the steam generators, fluid from steam generators will be discharged through the break. Feedwater flow to the steam generators may be reduced. This can cause the reactor coolant temperature to increase prior to reactor trip. For certain locations and sizes of the

postulated breaks the PORVs and safety valves are challenged and will pass steam followed by slightly subcooled water.

2.1.2.2 Steam Line Rupture. The scenario for this accident is the same as discussed for accidental depressurization of the secondary system due to inadvertent opening of a single steam dump, relief, or safety valve. The effects of minor secondary system pipe breaks are bounded by the analysis presented for this event. The safeties and PORVs, when actuated, will discharge steam and in the long term (beyond scope of FSAR analysis) they may also discharge water. As far as liquid discharge through the safety and relief valves is concerned, depressurization of the secondary system due to steam line rupture or inadvertent opening of main steam safety/relief valves is bounded by the analysis performed for feedline break and spurious actuation of the safety injection system at power.

2.1.2.3 Locked Rotor. This accident is postulated to result from a sudden locking of one rotor on one of the primary reactor coolant pumps. This causes a rapid reduction in core flow rate, reducing the heat transfer rate in one steam generator and increasing the temperature of the coolant, causing severe pressure increases. Departure from nucleate boiling may occur due to flow reduction and the resultant power-coolant mismatch. Both the PORVs and safety valves are challenged and are required to flow steam.

2.1.2.4 Rod Ejection. This accident is the result of the assumed mechanical failure of a control rod mechanism pressure housing, such that the reactor coolant system pressure ejects the control rod and drive shaft to the fully withdrawn position. This mechanical failure at most leads to a rapid reactivity insertion together with a higher core power distribution peak and high reactor coolant pressure.

As the system pressure increases the PORVs may be actuated to discharge steam. However, if the steam flow through the PORVs is not adequate, safety valves may open on steam to prevent excessive pressurization.

2.2 EXTENDED HIGH PRESSURE INJECTION EVENTS

The safety injection system (SIS) is designed to provide emergency core cooling in the case of a LOCA or steam break accident. The system is designed to maintain its protective capability in case of single failure. The system layout varies somewhat for different groups of plants. However, they retain the same basic functional design criteria, the main difference being the use of certain pumps for different purposes. The system operation is initiated by the Safety Injection Signal, which can be actuated by any of the following:

- Low pressurizer pressure
- High containment pressure
- High steam line differential pressure
- Low steamline pressure
- Manual actuation from control board

In the following sections, incidents that actuate the SIS signal are considered.

2.2.1 Transients Expected to Result in Initiation of High Pressure Safety Injection and Challenge the Pressurizer Relief or Safety Valves

For events described in this section steam is discharged through PORV and safety valves when the valve is first lifted, but later, when the pressurizer is filled with water, water discharge is also predicted.

2.2.1.1 Accidental Depressurization of the Secondary System/Steam Line Rupture. An accidental depressurization of the main steam system may result from the rupture of steam line, inadvertent opening of main steam safety/relief valves, or a single steam dump. The steam release results in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The temporary increase in energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. Due to the negative moderator coefficient, the cooldown results in a positive reactivity insertion. If the most reactive rod cluster control assembly is assumed stuck in its fully withdrawn

position after reactor trip, there is the possibility of the core returning to power. The primary cooldown and pressure drop actuates the SIS, which eventually results in a system repressuration as the pressurizer fills with liquid. Extended operation of the SIS would result in cycling of the PORVs (or safety valves if PORVs were assumed unavailable) on steam followed by subcooled water.

2.2.1.2 Main Feedline Rupture Accident. This accident is discussed in paragraph 2.1.2.1 and later in paragraph 4.1.2.1.

2.2.2 Spurious Initiation of High Pressure Safety Injection at Power

Inadvertent or spurious actuation of the safety injection system at power can be caused by operator error or a false electrical actuating signal.

Following the spurious actuation, the coolant charging pumps force highly concentrated boric acid solution through injection lines into the cold legs of each loop. Depending on the type of the plant, the residual heat removal pumps, safety injection pumps, and the passive injection system are also actuated but provide no flow when the RCS is at normal pressure.

An SIS signal results in reactor trip followed by a turbine trip; however, for extra conservativism another case was reviewed where it was assumed that the trip was delayed.

Both PORV and safety valves may be challenged depending on the pressure-head characteristics of the safety injection system. Valves, if actuated, lift on steam, and for extended operation of the safety injection system, subcooled water discharge may be observed. This event is more limiting from the viewpoint of surge flow and range of liquid temperatures at the valve inlet, and is selected for subsequent analysis.

2.3 COLD OVERPRESSURE TRANSIENTS

2.3.1 Mass Input Events

Based on probability of occurrence and in-plant operating experience, the most credible mass input events producing a net injection of mass into the reactor coolant system (RCS) involve failure in the air supply system, which causes the charging flow control valve to open, and/or isolation of letdown. Mass injection based on single charging pump operation is the most likely mass input mechanism, producing typical charging rates up to 120 gpm following isolation of letdown, and higher rates for air supply system failure.

Although precluded at low temperature by administrative procedure, two-charging-pump operation was considered in all plants to develop maximum input capability and thus provide additional flexibility in the operation of the cold overpressure mitigation system. Maximum input capability associated with this mechanism as applied to all plants analyzed to date is shown in Figure 2-1. The PORV inlet conditions presented in Figure 5-1 also include this mechanism.

Operation of the PORV at a predetermined setpoint pressure is employed by Westinghouse in the Cold Overpressure Mitigation System (OMS) to arrest the pressure transient caused by the above mechanisms. Mitigation of the transient on valve opening results in the RCS pressure turning over. This produces a transient peak overpressure. The PORV continues to open until valve capacity matches the net mass injection rate, after which the reset pressure is reached and the valve begins to close. PORV closure arrests the decreasing RCS pressure and reinitiates the pressure increase to complete the pressure transient cycle. This mimimum pressure is termed the transient pressure undershoot and is determined by the blowdown setting of the PORVs (nominally 20 psi). Pressure cycling continues until action is taken to remove the mass input mechanism.

Selection of PORV setpoints for pressure control of mass input-induced transients are based on a water-solid reactor coolant system, which produces pressure excursions significantly higher than for a RCS with

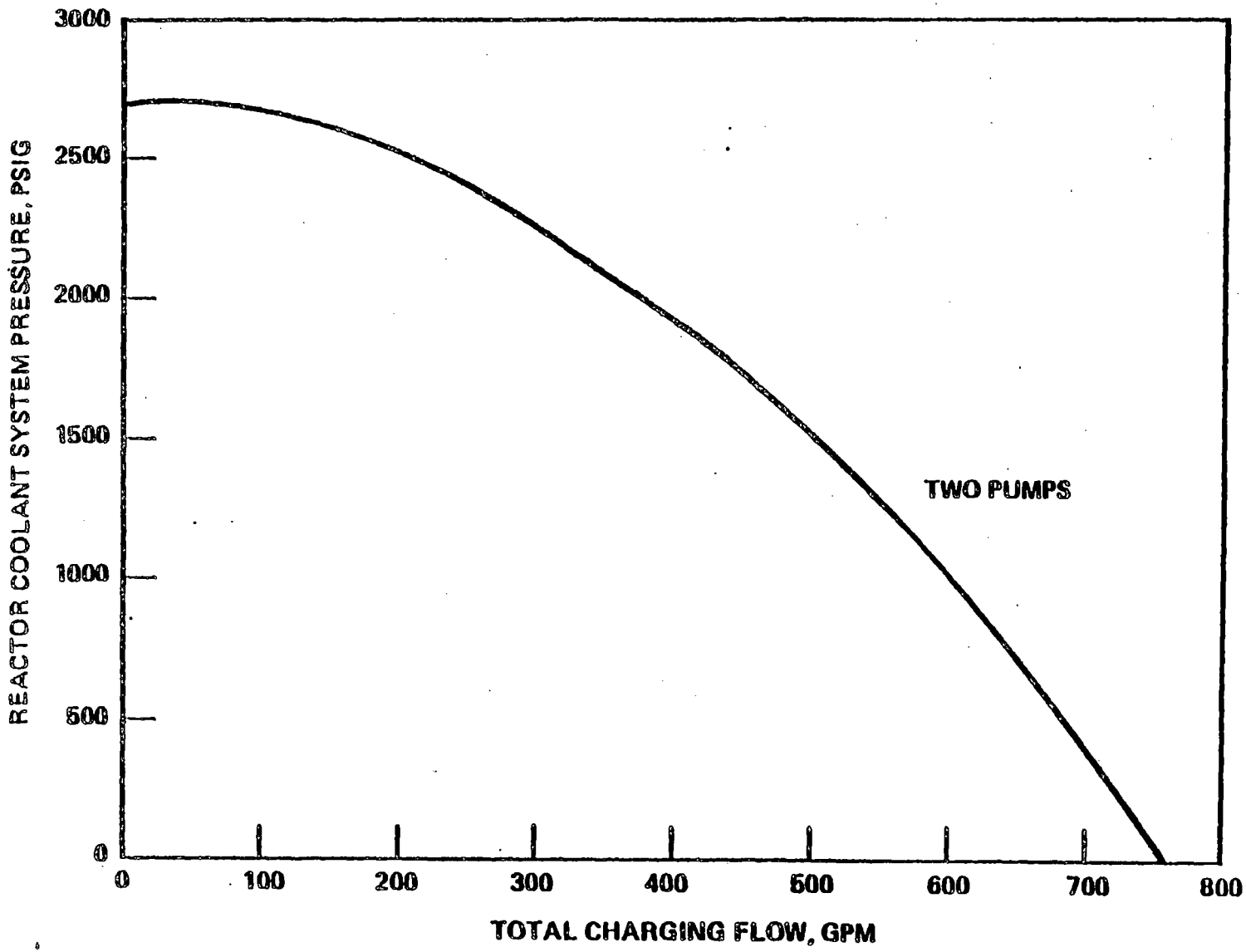

Figure 2-1. Maximum Charging Capacity versus Reactor Coolant System Pressure

a pressurizer steam cushion. Setpoint selection is also based on an algorithm which considers the reactor vessel NDT pressure limit and the integrity of the reactor coolant pump No. 1 Seal.

Valve opening and closure times of 2 seconds are assumed, and valve setpoints are staggered such that operation of the first valve will mitigate the event so that the other valve will not be challenged.

2.3.2 Heat Input Events

The heat input case which has the potential for the most severe pressure transient is that in which the steam generators exhibit a higher temperature than the remainder of the reactor coolant system. The magnitude of the difference in temperature is dependent on the means by which the temperature asymmetry was achieved, but a typical difference is considered to be about 50°F.

For the heat input transient with the initial reactor coolant temperature 50°F less than the temperature in the steam generators and with all reactor coolant pumps off, one of the two reactor coolant pumps is started to circulate the reactor coolant through the warmer steam generators. As the coolant flow begins, the warm water in the tubes of the steam generator in the active loop is forced out and into the reactor coolant pump where it is pumped into and mixed with the colder reactor coolant. In the inactive loops, the warmer water from the tubes of the steam generator is forced out in a reverse direction due to the backflow in the inactive loops, and also mixed with the cooler reactor coolant. This initial mixing of the warm water with the larger volume of cooler water causes an initial shrinkage effect which tends to decrease the initial coolant pressure.

Simultaneously, the cooler reactor coolant that enters the steam generator begins to be heated as it moves through the tube bundle. As heat is added to the coolant due to heat transfer from the secondary water in the steam generator, the coolant attempts to expand and cause a resultant pressure increase. The net effect of the expansion due to the heat transferred to the coolant and the shrinkage effect due to the mixing of the warm water with the cooler coolant is a relatively

constant coolant pressure in the initial few seconds of the transient. Then, as the flow rate increases and the heat transfer mechanism becomes predominant, the coolant pressure increases rapidly.

As in the mass input scenario described in Section 2.3.1, the reactor coolant pressure increases until the pressure reaches the PORV setpoint. The valve opens and pressure is mitigated. When the valve opens sufficiently to provide a capacity in excess of the expansion rate of the coolant and the coolant pressure decreases rapidly after reaching an overshoot, PORV closure at the reset pressure setpoint is then required to arrest the pressure decay and start the pressure transient cycle over again.

The heat input transient due to temperature asymmetry in the reactor coolant system is unique in that it is self-limiting; i.e., when the temperatures are brought to equilibrium by the reactor coolant flow, the transient is ended. The use of a relief valve to mitigate the pressure transient results in a valve cycling of the coolant as it is heated, but the valve is only required to cycle a few times until the temperatures in the system are brought to equilibrium and coolant expansion ceases. The first cycle results in the largest setpoint overshoot. Subsequent valve cycles result in diminishing overshoots as the coolant expansion rate diminishes until eventually the valve closes and remains closed.

The heat input event is considered in the algorithm utilized for cold overpressure system setpoint selection for the relief valves as described above.

Section 3

GROUPING OF WESTINGHOUSE-DESIGNED NSSSs

A complete analysis of all overpressure transients in Westinghouse-designed plants is prohibitive due to the large number of these plants. Therefore, a method was established to group the Westinghouse NSSSs under selected reference plants. Various methods could have been utilized to group the plants and select reference plants. The method utilized in this report has two important objectives:

a. to maintain the similarity of plant characteristics and design philosophy within each group, and

b. to represent the performance of the plants in each group with the results of transient analyses performed on the respective reference plant.

The following method was used to generate the groups and to select reference plants for each group.

3.1 DEFINITION OF CRITICAL PARAMETERS

The parameters that may have an effect on peak reactor coolant pressure and the rate of pressurization for overpressure transients are listed in Table 3-1. The list was developed from sensitivity analyses and engineering judgment.

The grouping of Westinghouse plants was done in two steps.

In the first step, the Westinghouse-designed NSSSs are collected into groups according to differences in design and physical layout. This initial grouping was made by number of loops. The number of loops is important, from the overpressure transient analysis view, since the limiting overpressure transients are not the same for plants with different numbers of loops.

The number and capacities of safety and relief valves in the primary system are also different for plants with different numbers of coolant loops.

Table 3-1

LISTING OF CRITICAL PARAMETERS

Parameters	Units
1. Number of Loops	---
2. Steam Generator Type	---
3. NSSS Power	MWt
4. Vessel Average Temperature	°F
5. Coolant Flow Rate	lb/hr
6. Number of Pressurizer Safety Valves	---
7. Capacity of Pressurizer Safety Valves	lb/hr
8. Number of Pressurizer Relief Valves	---
9. Capacity of Pressurizer Relief Valves	lb/hr
10. Safety Valve Opening Setpoint	psig
11. Relief Valve Opening Setpoint	psig
12. Safety Injection Charging Rate Versus Pressure	gpm

There are significant differences in the design of the SIS for the various two-, three-, and four-loop plants while they retain the same basic functional design criteria. The main difference is the use of certain pumps for single duty in one type and multiple duty in others. Other differences include the number, shut-off

pressure, and size of passive injection system and the number, shut-off pressure, and run-out flow rate of low and high head safety injection pumps.

There are other differences between plants with different numbers of loops that have less effect on the overpressure transients, such as the auxiliary and feedwater system, the containment pressure transient, and the protection logic.

Therefore, it is logical and necessary to group the Westinghouse designed NSSSs under two-, three-, and four-loop plants.

In the second step, it is necessary to select reference plants for each group considering the remaining critical parameters, which are:

- Steam Generator Type
- NSSS Power Generation
- Coolant Flow Rate
- Vessel Average Temperature
- Safety and Relief Valve Opening Setpoint
- Safety and Relief Valve Number and Capacity Within Each Group

The above critical parameters are utilized in the next section to select reference plants for each group.

3.2 BASIS FOR SELECTION OF REFERENCE PLANTS

The parameter that utilizes the effect of the above critical parameters for selecting the reference plants is the ratio of asymptotic surge rate to safety valve capacity.

The asymptotic surge rate is defined as follows:

$$W = \frac{\text{(Reactor power) x (Total volumetric expansion per unit temp. change)}}{\text{Combined heat capacity of primary and secondary system}}$$

Selection of the plant within a group having the higher "W" should result in the most severe valve inlet conditions for the plants in that group. The volumetric expansion is considered in three parts: (1) cold volume, which includes the

volume of the steam generator outlet plenum, the volume of cold legs, the volume of the downcomer, and the volume of the reactor vessel lower plenum; (2) the medium temperature volume, which includes the volume of steam generator tubes, the active fuel region, and the volume of bypass flow region; and (3) hot volume, which is made up of the volume from the active fuel region discharge to steam generators.

A linear variation of volumetric expansion versus temperature in a small neighborhood about an average temperature in each region is assumed, from which the total volumetric expansion of the primary coolant due to temperature change is calculated. The heat capacity of the primary system is evaluated from the basics of thermodynamics and that of the secondary system is obtained using empirical correlations:

- Steam Generator Type -- The effect of the steam generator type enters through the empirical correlation used to calculate the heat capacity of the secondary system. The sensitivity analysis performed on the effect of the existence of preheaters in the steam generators indicates that peak pressure reached during the course of overpressure transients is insensitive to the existence of the preheaters. Therefore, steam generator type is not considered a key parameter in this evaluation and was not used in plant grouping.

- NSSS Power Generation -- This parameter appears in the definition of the asymptotic surge rate explicitly.

- Vessel Average Temperatures and Nominal Pressure -- The effects of these parameters are included in the calculation of volumetric expansion rates calculated at average temperature and pressure in each region.

- Coolant Flow Rate -- The effect of coolant flow rate is incorporated through pressure drop calculations that take into account the losses across the core inlet, outlet, and upper and lower tie plates, and frictional, fuel spacers, hydrostatic and other acceleration losses throughout the primary loop. Except for the hydrostatic losses, all other pressure drops are obtained from experimental results and correlations are based on coolant flow rate. The coolant flow rate and energy transfer are intimately related to the heat transfer film coefficient, which is

strongly dependent on the coolant velocity. Therefore, the use of temperature and pressures in calculating the volumetric expansion rate implicitly incorporates the effect of coolant flow rate.

- Safety and Relief Valve Capacity and Opening Setpoint -- The effects of these parameters are considered not in the asymptotic surge rate, but in the ratio of asymptotic surge rate to the valve capacity. The volumetric discharge rate of the valve is calculated at the valve opening setpoint and depends on the capacity (area of the valve).

Therefore, by maximizing one nondimensional parameter (asymptotic surge rate to safety valve capacity) reference plants for each group can be selected which would be expected to have fluid conditions enveloping those for the other plants in that group.

The results of this analysis are presented in Table 3-2, where the values of critical parameters and the ratio of asymptotic surge rate to PSV capacity for each plant are listed. The reference plants for two-, three-, and four-loop plants whose parameters are shown in Tables 3-3, 3-4, and 3-5 have the ratio of asymptotic surge rate to safety valve capacities of 1.563, 1.424, and 1.619 respectively. For 2 and 4-loop plants, reference plants were selected with the highest surge rate to valve capacity ratio for that group. For 3-loop plants the reference plant selected represents the majority of the plants in that group, rather than being the plant with the largest asymptotic surge rate to valve capacity for the following reasons:

1. More than 87 percent of the 3-loop plants are represented by a plant with an asymptotic surge rate of 1.424.

2. The results of transient analysis performed for 4- and 2-loop plants (with surge rate to valve capacity of 1.619 and 1.563 respectively) envelop the results of transient analyses for 3-loop plants. (See table 5-1.

The reference plants from this point forward will be considered generic plants representing two-, three-, and four-loop plants.

TABLE 3-2
CRITICAL AND NONDIMENSIONAL PARAMETERS FOR
WESTINGHOUSE-DESIGNED NSSSs

Plant	Number of Loops	Steam Generator Type	NSSS Power (MWt)	Vessel Average Temp. (°F)	Coolant Flow (lb/hr x 10^8)	PORVs			Safety Valves			Ratio of Asymptotic Surge Rate to Safety Valve Capacity
						Number	Setpoint (psia)	Capacity (lb/hr)	Number	Setpoint (psia	Capacity (lb/hr)	
RGE	2	44	1520	573.5	.680	2	2350	178900	2	2500	288000	1.563
WEP	2	44	1518.5	573.9	.671	2	2350	179000	2	2500	288000	1.563
WIS	2	44	1518.5	573.9	.671	2	2350	179000	2	2500	288000	1.563
NSP	2	51	1650	567.3	.682	2	2350	179000	2	2500	345000	1.259
NRP	2	51	1650	567.3	.682	2	2350	179000	2	2500	345000	1.259
WPS	2	51	1650	567.3	.682	2	2350	210000	2	2500	350000	1.244
SCE	3	27	1351	575	7.38x	2	2350	108000	3	2500	240000	NA
CPL	3	44	2200	574.2	1.015x1	2	2350	210000	3	2500	288000	1.425
FPL	3	44	2208	574.2	.965	2	2350	210000	3	2500	288000	1.425
FLA	3	44	2208	574.2	.965	2	2350	210000	3	2500	288000	1.425
VPA	3	51	2441	574.3	1.007	2	2350	210000	3	2500	293330	1.517
VIR	3	51	2441	574.3	1.007	2	2350	210000	3	2500	293000	1.517
DLW	3	51	2660	576.2	1.008	3	2350	210000	3	2500	345000	1.415
VRA	3	51	2785	580.3	1.05	2	2350	210000	3	2500	380000	1.425
ALA	3	51	2660	577.2	1.007	2	2350	210000	3	2500	345000	1.415
VGB	3	51	2785	580.3	1.05	2	2350	210000	3	2500	380000	1.425
APR	3	51	2660	577.3	1.007	2	2350	210000	3	2500	345000	1.415
CGE	3	D3-1	2785	587.4	1.096	3	2350	210000	3	2500	420000	1.425
DMW	3	51	2660	576.2	1.008	3	2350	210000	3	2500	345000	1.415
CQL	3	D4-2	2785	587.5	1.092	3	2350	210000	3	2500	380000	1.425
CRL	3	SD4-1	2785	587.5	1.092	3	2350	210000	3	2500	380000	1.425
CSL	3	SD5	2785	587.5	1.092	3	2350	210000	3	2500	380000	1.425
CTL	3	D5	2785	587.5	1.092	3	2350	210000	3	2500	380000	1.425

TABLE 3-2 (cont)
CRITICAL AND NONDIMENSIONAL PARAMETERS FOR
WESTINGHOUSE-DESIGNED NSSSs

Plant	Number of Loops	Steam Generator Type	NSSS Power (MWt)	Vessel Average Temp. (°F)	Coolant Flow (lb/hrx10^8)	PORVs			Safety Valves			Ratio of Asymptotic Surge Rate to Safety Valve Capacity
						Number	Setpoint (psia)	Capacity (lb/hr)	Number	Setpoint (psia	Capacity (lb/hr)	
IPP	4	44	2758	569.5	1.361	2	2350	179000	3	2500	408000	1.222
INT	4	44	3025	571.5	1.363	2	2350	179000	3	2500	420000	1.334
CWE	4	51	3250	562.2	1.350	2	2350	210000	3	2500	420000	1.352
COM	4	51	3250	562.2	1.350	2	2350	210000	3	2500	420000	1.352
AEP	4	51	3250	567.8	1.350	3	2350	179000	3	2500	420000	1.352
AMP	4	51	3403	573.8	1.346	3	2350	210000	3	2500	420000	1.483
PGE	4	51	3350	576.6	1.329	3	2350	210000	3	2500	420000	
PEG	4	51	3423	577.6	1.339	3	2350	210000	3	2500	420000	1.619
PDR	4	51A	3423	584.7	1.326	2	2350	210000	3	2500	420000	1.619
TVA	4	51	3423	578.2	1.38	2	2350	179000	3	2500	420000	1.619
TEN	4	51	3423	578.2	1.38	2	2350	179000	3	2500	420000	1.619
PSE	4	51	3350	576.8	1.323	2	2350	210000	3	2500	420000	1.619
PNJ	4	51	3423	578	1.322	2	2350	210000	3	2500	420000	1.619
DAP	4	D2	3425	588.2	1.448	2	2350	210000	3	2500	420000	1.619
DBP	4	D3	3425	588.2	1.448	2	2350	210000	3	2500	420000	1.619
WAT	4	D3-2	3425	588.2	1.448	2	2350	210000	3	2500	420000	1.619
WBT	4	D3-2	3425	588.2	1.448	2	2350	210000	3	2500	420000	1.619
CAE	4	D4	3425	587.7	1.405	2	2350	210000	3	2500	420000	1.619
GAE	4	F	3425	588.5	1.421	2	2350	210000	3	2500	420000	1.619
GBE	4	F	3425	588.8	1.421	2	2350	210000	3	2500	420000	1.619
NEU	4	F	3425	587.1	1.408	2	2350	210000	3	2500	420000	1.619
NAH	4	F	3425	588	1.421x	2	2350	210000	3	2500	420000	1.619
NCH	4	F	3425	588.5	1.421	2	2350	210000	3	2500	420000	1.619
DCP	4	D3-2	3427	590.8	1.434	3	2350	210000	3	2500	420000	1.619

TABLE 3-2 (cont)
CRITICAL AND NONDIMENSIONAL PARAMETERS FOR
WESTINGHOUSE-DESIGNED NSSSs

Plant	Number of Loops	Steam Generator Type	NSSS Power (Mwt)	Vessel Average Temp. (°F)	Coolant Flow (lb/hrx10^8)	PORVs			Safety Valves			Ratio of Asymptotic Surge Rate to Safety Valve Capacity,
						Number	Setpoint (psia)	Capacity (lb/hr)	Number	Setpoint (psia	Capacity (lb/hr)	
DDP	4	D5	3427	590.8	1.434	3	2350	210000	3	2500	420000	1.619
TDX	4	D4-2	3425	588	1.403	2	2350	210000	3	2500	420000	1.619
TCX	4	D5	3425	588.5	1.421	2	2350	210000	3	2500	420000	1.619
CCE	4	D5	3425	587.7	1.405	2	2350	210000	3	2500	420000	1.619
CDE	4	D5	3425	587.7	1.405	2	2350	210000	3	2500	420000	1.619
OPS	4	F	3425	588.5	1.403	2	2350	210000	3	2500	420000	1.619
OQS	4	F	3425	588.5	1.403	2	2350	210000	3	2500	420000	1.619
TGX	4	E2	3817	593	1.396	2	2350	210000	3	2500	420000	1.619
THX	4	E2	3817	593	1.396	2	2350	210000	3	2500	420000	1.619
PBJ	4	D4	3425	587.7	1.405	2	2350	179000	3	2500	420000	1.619
PCJ	4	D5	3425	587.7	1.405	2	2350	210000	3	2500	420000	1.619
CDE	4	D5	3425	587.7	1.405	2	2350	210000	3	2500	420000	1.619
SAP	4	F	3425	588.5	1.421	2	2350	210000	3	2500	420000	1.619
SCP	4	F	3425	588.5	1.421	2	2350	210000	3	2500	420000	1.619
SFP	4	F	3425	588.5	1.421	2	2350	210000	3	2500	420000	1.619

Table 3-3

REFERENCE PLANT FOR WESTINGHOUSE TWO-LOOP PLANTS

NSSS Power (MWt)	1520
Thermal Design Flow (gpm)	83700
Reactor Coolant Pressure (psia)	2250
Reactor Coolant Temperature (^{o}F)	
Core Outlet	612.2
Vessel Outlet	609.8
Core Average	583.7
Vessel Average	581.2
Vessel/Core Inlet	552.5
Steam Generator Outlet	552.5
Steam Generator	
Type	44
Steam Temperature (^{o}F)	521.2
Steam Pressure (psia)	821
Steam Flow (10^6 lb/hr total)	6.62
Feed Temperature (^{o}F)	435.7
Zero Load Temperature (^{o}F)	547
Pressurizer Safety Valves	
Number	2
Set Points (opening/closing)	2500/2500
Capacity (lb/hr)	288000
Pressurizer Relief Valves	
Number	2
Set Points (opening/closing)	2350/2350
Capacity (lb/hr)	175000
Ratio of Asymptotic Surge Rate to SV Capacity	1.563

Table 3-4

REFERENCE PLANT FOR WESTINGHOUSE THREE-LOOP PLANTS

NSSS Power (MWt)	2787
Thermal Design Flow (gpm)	95000
Reactor Coolant Pressure (psia)	2250
Reactor Coolant Temperature (^{o}F)	
Core Outlet	622.8
Vessel Outlet	620.1
Core Average	591.1
Vessel Average	587.8
Vessel/Core Inlet	555.5
Steam Generator Outlet	555.5
Steam Generator	
Type	51
Steam Temperature (^{o}F)	532.0
Steam Pressure (psia)	900
Steam Flow (10^6 lb/hr total)	12.2
Feed Temperature (^{o}F)	437
Zero Load Temperature (^{o}F)	547
Pressurizer Safety Valves	
Number	3
Set Points (opening/closing)	2500/2500
Capacity (lb/hr)	380000
Pressurizer Relief Valves	
Number	3
Set Points (opening/closing)	2350/2350
Capacity (lb/hr)	210000
Ratio of Asymptotic Surge Rate to SV Capacity	1.425

Table 3-5

REFERENCE PLANT FOR WESTINGHOUSE FOUR-LOOP PLANTS

NSSS Power (MWt)	3425
Thermal Design Flow (gpm)	94400
Reactor Coolant Pressure (psia)	2250
Reactor Coolant Temperature (^{o}F)	
Core Outlet	621.1
Vessel Outlet	617.8
Core Average	591.1
Vessel Average	587.7
Vessel/Core Inlet	557.6
Steam Generator Outlet	557.3
Steam Generator	
Type	D4
Steam Temperature (^{o}F)	543.3
Steam Pressure (psia)	990
Steam Flow (10^6 lb/hr total)	15.13
Feed Temperature (^{o}F)	440
Zero Load Temperature (^{o}F)	557
Pressurizer Safety Valves	
Number	3
Set Points (opening/closing)	2500/2500
Capacity (lb/hr)	420000
Pressurizer Relief Valves	
Number	3
Set Points (opening/closing)	2350/2350
Capacity (lb/hr)	210000
Ratio of Asymptotic Surge Rate to SV Capacity	1.619

Table 4-1

TRANSIENT RESPONSE PARAMETERS OF OVERPRESSURE EVENTS
COMPARED FOR A TYPICAL PLANT (2-Loop)

Transient	Discharge Fluid Condition	Peak Valve Inlet Pressure (psia)	Pressure Rate at Safety Valve Opening (psi/sec)	Maximum Enthalpy of Fluid Discharge (Btu/lbm)
Locked Rotor	Steam	2675	236	1123
Loss of Load	Steam	2553	73	1126.7
Loss of Normal Feedwater	Steam	2529	7	1126.1
Station Blackout	Steam	2529	7	1126.1
Rod Ejection *	Steam	2341	N/A	N/A
Rod Withdrawal at Power	Steam	2504	9	1125.6

*Accident is not normally analyzed for overpressurization.

Section 4

METHODOLOGY TO DETERMINE RANGE OF EXPECTED PORV AND SAFETY VALVE INLET CONDITIONS

In this section the methodology used in determining the range of inlet fluid conditions at the PORV and safety valves is discussed. Two different methods are used; one for the extended operation of the high pressure injection system and transients that result in steam discharge through safety valves, and another for a main feedline rupture accident.

4.1 LICENSING-TYPE TRANSIENTS

The transients that are analyzed for reload and licensing can be divided into two groups; those that result in steam discharge and those that result in liquid discharge through PORV and safety valves.

4.1.1 Transients Resulting in Steam Discharge

The results of analyses of overpressure transients that result in steam discharge through PORV and safety valves are summarized in Table 4-1 for a typical two-loop plant. The standard Safety Analysis Report (SAR) type analysis was utilized. Listed are the peak pressure and enthalpy of steam being discharged and the pressurization rate at valve opening.

Comparison of the peak pressure for each event indicates that the limiting transients for steam discharge are loss of load and locked rotor.

4.1.1.1 Loss of Load. This transient is analyzed to make certain that the reactor coolant and steam generators are not overpressurized and the increase of reactor coolant system temperature does not result in Departure from Nucleate Boiling

(DNB) in the core. In addition, fuel temperature and fuel clad strain limit should not be exceeded.

The initial power level is assumed to be at maximum allowable value plus 2 percent uncertainty. However, the sensitivity analysis performed shows that the RCS peak pressure for loss of load is relatively insensitive to the initial conditions of temperature and pressure.

For standard DNB design procedures, the reactor coolant average temperature corresponds to the initial power level and the pressure is at nominal, including allowance for calibration and instrument errors.

Control systems are assumed to function only if their operation results in more severe accident results.

Two cases are analyzed, both with and without automatic pressure control, to assure that the reactor is protected for both modes of plant operation. Cases are also analyzed for maximum and minimum reactivity feedback.

To maximize the peak pressure and pressurization rate for loss of load transients for SAR-type analyses, no steam dump is assumed, and minimum values for overall heat transfer coefficients are used. For the control system, it is assumed that the pressurizer spray system and heaters are off, and no credit is taken for the rod control system. The safety injection system and auxiliary feed water system are not assumed to operate. The same conservative assumptions are made for a locked rotor incident.

4.1.1.2 Locked Rotor/Loss of Flow. For this accident, two separate procedures are used. One procedure is used for calculating reactor coolant pressure and clad temperature and the other is used to calculate the number of rods in the DNB. For this accident, the reactor coolant pressure is expected to remain below 110 percent of design pressure and the clad temperature

must remain below 2700°F. The initial power level is assumed to be at maximum allowable value plus 2 percent uncertainty. The initial system pressure for RCS pressure calculation is assumed to be a nominal pressure plus allowance for calibration and instrument error. Control system is assumed to function only if their operations increases the severity of the event.

4.1.2 Transients Resulting in Liquid Discharge

Past analysis indicates that the most limiting transient resulting in liquid discharge through the PORV and safety valves is the feedline break accident. Water discharge through safety and relief valves is predicted during standard SAR analysis of feedline break.

4.1.2.1 Main Feedline Rupture. The purpose of analyzing a feedwater line rupture incident is to ensure that the plant is maintained in a safe condition for a range of feedline breaks up to and including a break equivalent in area to double-ended rupture of the largest water line. The calculated radioactive material release of these events should not exceed the guideline value of 10CFR100. To achieve this, the pressure in the reactor coolant system is conservatively maintained below 110 percent of the design pressure, and the fuel damage that may occur during the course of the transient must be limited so that the core will remain geometrically intact with no loss of core cooling capability. Based on sensitivity analysis, two cases which represent the worst conditions are presented in the SAR. Both cases assume a double-ended rupture occurring downstream of the main feedline check valve with the core at 102 percent power. The RCS flow is assumed at thermal design flow with temperature and pressure at nominal condition with allowance for calibration and instrument error.

Following a main feedline rupture, steam line pressure and steam generator water level begin to drop. The low steam line pressure signal initiates a steam line isolation signal and a

safety injection signal, which initiates flow of borated water into the RCS.

No reactor control system is assumed to function during the accident unless its function results in a more severe transient. Core nuclear parameters are chosen to maximize the energy input to the coolant.

4.1.2.2 Small Steam Break. This accident is similar to feedline break and considers the event of potential reactor cooldown resulting from a secondary pipe rupture.

This accident is bounded, as far as the rate of liquid discharge through PORVs and safety valves is concerned, by the feedline break analysis.

4.2 EXTENDED HIGH PRESSURE INJECTION EVENTS

As discussed earlier, the limiting Extended High Pressure Injection event is spurious SIS actuation at power. This event is analyzed to make sure that the critical heat flux is not exceeded. The pressures calculated in the reactor coolant and main steam system are below 110 percent of the design pressure, and fuel temperature and fuel clad strain limits are not exceeded. The peak linear heat generation rates are below a value that could cause the fuel centerline to melt. The range of safety and relief valve inlet conditions for this event depends on the maximum safety and relief valve opening setpoint, initial core boron concentration, boron worth, and maximum safety injection flow rate versus reactor coolant pressure. As briefly described before, the current design of the safety injection system (SIS) for two-, three-, and four-loop Westinghouse-designed NSSSs vary in layout and philosophy. The initial power is assumed to be at maximum allowable NSSS power plus 2 percent uncertainty. The reactor coolant pressure and average temperature are assumed to be at nominal temperature and pressure including allowance for calibration and instrument errors. Pressurizer water volume corresponds to the programmed reactor coolant average temperature, and feedwater temperature corresponds to the initial power level.

The accident is simulated by initiating injection of borated water into each of reactor coolant cold legs with no direct reactor trip resulting from the SI signal.

The only protection system assumed available is the low pressurizer pressure reactor trip. The control systems are assumed to function only if their operation results in more severe accident results. For this accident, no control system are assumed operable.

Two cases with maximum and minimum boron worth for a given boron concentration, which yields a high boron worth per ppm increase in boron concentration, were analyzed. Minimum values of overall fuel heat transfer coefficients are used that delay the heat addition to reactor coolant system, which in turn increases the primary coolant expansion rate and thus the discharge rate through safety and relief valves.

Maximum flow rate is assumed for the safety injection system versus reactor coolant pressure with all pumps running and all lines injecting.

Surge flowrates presented for this event include pump flow as well as any system expansion which might occur during this event.

4.3 COLD OVERPRESSURIZATION EVENTS

The range of relief valve inlet conditions for cold overpressure operation is dictated by several factors: Cold Overpressure Mitigation System (OMS) relief valve setpoint, the existence of a steam bubble in the pressurizer, minimum allowable system pressure, and maximum relief valve setpoint.

OMS setpoints are based on consideration of the most severe, credible mass input and heat input events. Analytical assumptions include pressurization of a water-solid, cold reactor coolant system; staggered valve operation, which takes credit for only single valve operation; and reactor vessel and reactor coolant pump no. 1 seal pressure limit constraints. Setting OMS setpoints on the above bases provides the system the capability to accommodate all expected transients and fluid conditions at the valve inlet. The setpoints also depend on acceptable performance of at least one of the PORVs.

The maximum pressure expected to occur during a cold overpressure incident at any RCS temperature is that associated with actuation of PORV #2 only, the relief valve at the higher setpoint. This valve setpoint has an upper limit of 2335 psig,

which corresponds to the normal (high temperature, high pressure) setpoint. PORV #2 operation would normally occur during cold overpressurization in the event of failure of PORV #1, the relief valve at the lower setpoint.

The OMS setpoint is variable with RCS temperature, from a low value at low temperature to the upper 2335 psig limit. The highest possible setpoints were determined to provide the plant operator with maximum pressure margin for plant operation during shutdown.

The minimum pressure expected during any cold overpressure incident at any applicable RCS temperature is the reactor coolant pump no. 1 seal pressure limit used in setpoint determination. Theoretically, operating pressure may be established as low as this limit and the minimum temperature in the pressurizer operating with a steam cushion is based on this limit.

While setpoints are determined for the conservative situation whereby the RCS is assumed to be in a water-solid configuration, in reality, a steam cushion could exist in the pressurizer via compliance with administrative procedures to ameliorate the effects of cold overpressurization events.

Depending on the progress made by the operator in implementing these procedures at the time of a cold overpressure event, and the duration of the event, the state of the pressurizer could be saturated steam, saturated water, or subcooled water. The condition of the fluid at the PORV inlet during a cold overpressure event can therefore vary from saturated steam to subcooled water, as discussed in Section 5.4.

Section 5

EXPECTED FLUID CONDITIONS AT SAFETY & RELIEF VALVE INLETS

The expected fluid conditions at the inlet of PORVs and safety valves for both steam and liquid discharge are discussed. Normally for SAR-type analyses of overpressure incident, no credit is taken for the operation of PORVs. Therefore, SAR-type analysis is used to obtain fluid condition at the inlet to the safety valves. To obtain the inlet condition to the PORVs, the above analysis is repeated with PORVs assumed operational. In no case is credit taken for the pressurizer sprays.

Two limiting transients, loss of load and locked rotor, are considered for steam discharge. Both accidents are analyzed for each reference plant. The results are presented in tables. For water discharge through the valves, the extended operation of the high pressure injection system and feedline break events are analyzed. The Extended Operation of High Pressure Injection Event is analyzed for each reference plant, whereas for feedline break, only results of analyses for plants that have feedline break analyses are presented. This is further discussed in appropriate sections.

5.1 FSAR TRANSIENTS RESULTING IN STEAM DISCHARGE

The FSAR transients that result in steam discharge, locked rotor, and loss of load are selected as the limiting events in determining the inlet fluid conditions for PORV and safety valves. The reasons for this selection, as discussed before, are the peak pressurizer pressures reached during the course of the transients and the rate of pressurization reached.

5.1.1 Reference Plant -- Two-Loop Group

The inlet fluid conditions for PORV and safety valves are presented for locked rotor and loss of load in Table 5-1. The inlet fluid conditions are also summarized where credit is taken for PORV operation. The limiting transient for steam discharge in two-loop plants is the locked rotor, during which peak reactor coolant pressure

Table 5-1

VALVE INLET CONDITIONS FOR FSAR EVENTS
RESULTING IN STEAM DISCHARGE

Reference Plant	Valve Opening Pressures (psia)	Maximum Pressurizer Pressure (psia)/ Limiting Event	Maximum Pressure Rate (psia/sec)/ Limiting Event
SAFETY VALVES ONLY			
2 - Loop	2500	2682/Locked Rotor	240/Locked Rotor
3 - Loop	2500	2592/Locked Rotor	216/Locked Rotor
4 - Loop	2500	2555/Loss of Load	144/Locked Rotor
SAFETY AND RELIEF VALVES			
2 - Loop	2350	2573/Locked Rotor	202/Locked Rotor
3 - Loop	2350	2555/Locked Rotor	200/Locked Rotor
4 - Loop	2350	2532/Loss of Load	130/Locked Rotor

and maximum pressurizations rates are observed. A peak pressure of 2682 psia and a pressurization rate of 240 psia/sec is observed when no credit is taken for the operation of PORVs. Assuming the PORVs operable reduces the peak pressure by 4.1 percent and the rate of pressurization by 16.5 percent.

5.1.2. Reference Plant -- 3-Loop Group

Table 5-1 also presents the results of loss of load and locked rotor analysis for plants with three loops. As for two-loop plants, the fluid inlet conditions expected at the inlets of the safety valves and PORVs are also included in Table 5-1. Locked rotor can also be considered as the limiting overpressurization transient for three-loop plants.

The peak pressure reached during the locked rotor accident is 2592 psia and the maximum rate of pressurization is about 216 psia/sec. When PORVs are assumed operational, a 1.4 percent and 7.4 percent reduction is peak pressure and maximum rate of pressurization are observed, respectively. It should be noted that the result of analysis for the two-loop reference plant envelops the predicted valve inlet conditions of three-loop reference plant for transients resulting in steam discharge.

5.1.3 Reference Plant -- 4-Loop Group

In terms of peak pressurizer pressure, loss of load is the limiting transient for the four-loop plants. However, the rate of pressurization is higher for the locked rotor compared to that for loss of load. The fluid conditions expected at the inlets of the safety valves and PORVs are presented in Table 5-1.

The peak pressure reached during the loss of load accident for four-loop plants is 2555 psia. The maximum rate of pressurization of the reactor coolant system for loss of load was below that for the locked rotor transient. Hence, the maximum pressurization rate of 144 psia/sec from locked rotor analysis for four-loop plants is recorded in Table 5-1. If PORVs are assumed operational, a 0.9 percent reduction in peak pressure and 9.7 percent reduction in the rate of reactor coolatn pressurization is observed.

5.2 PLANT-SPECIFIC VALVE INLET CONDITIONS FOR MAIN FEEDLINE BREAK

The fluid conditions at the inlet to the safety valves for feedline rupture accidents are summarized in Table 5-2 for plants that have feedline break accident analyses. The information presented in Table 5-2 must be considered in light of the following discussion.

The feedline break analyses for Beaver Valley Unit 1 and North Anna Units 1 and 2 were done in a somewhat more conservative fashion due to the Code limitations. For these plants, it was assumed that the safety valves did not open until 2575 psia instead of their actual setpoint of 2500 psia. These analyses also overspecified the liquid enthalpy in the pressurizer. More recent analysis have used better models to predict valve opening characteristics and fluid enthalpy.

For all cases presented, maximum pressurization rates are taken when valves open on water (the valves initially open on steam; however, the pressurization rate is enveloped by those presented for the locked rotor and loss of load events). When the pressurizer is filled and begins discharging liquid, the pressurization rate is small.

For the feedline break, as mentioned above, the results of standard FSAR analysis are reported. The range of pressures for liquid discharge is from 2500 to 2575 psia and enthalpies from 570 to 742 Btu/lb. This defines the range of fluid temperatures possible from 570°F, which corresponds to subcooled water at 2500 psia and 570 Btu/lb enthalpy, and 673°F, which corresponds to saturated water at 2580 psia and 741.9 Btu/lb enthalpy. The range of the pressurization rates is from 1.6 to 12 psia/sec. The range of surge rates through the pressurizer when valves are discharging liquid is from 0.6129 to 6.66 ft^3/sec (224 GPM to 2989 GPM).

5.3 EXTENDED HIGH PRESSURE INJECTION EVENTS

The limiting Extended High Pressure Injection Event was the spurious activation of the safety injection system at power. This transient is a Condition II event which, at worst, will result in a reactor shutdown with the plant capable of returning to operation. Condition II events should not cause more serious events, that is, Condition III or IV events. Other criteria stated in the USNRC Standard Review Plan are stated in Section 4. The results of the analyses are presented in Table 5-3.

Table 5-2

SAFETY VALVE INLET CONDITIONS FOR FSAR EVENT RESULTING IN LIQUID DISCHARGE (MAIN FEEDLINE BREAK)

Plant	Safety Valve Opening Setpoint (psia)	Maximum Pressurizer Pressure (psia)	Maximum Pressurization Rate (psia/sec)	Maximum Liquid Surge Surge Rate Into Pressurizer When Valve Is Passing Liquid (GPM)	Range of Liquid Temperatures at Valve Inlet (1st or Subsequent Openings) (°F)
CAE/CBE/ CCE/CDE PBJ/PCJ	2500	2507.7	3.5	569.1	615.0 - 635.1
DAP/DBP	2500	2501.8	5.0	659.3	611.9 - 622.5
DCP/DDP	2500	2507.7	5.0	543.1	613.7 - 631.3
NAH/NCH	2500	2504.9	3.0	275.1	568.7 - 584.1
TBX/TCX	2500	2503.2	5.0	1109.5	608.2 - 614.9
TGX/THX	2500	2505.4	6.0	408.2	607.1 - 609.6
CQL/CRL/ CSL/CTL	2500	2504.0	4.0	313.7	620.1 - 623.4
SNUPPS	2500	2535.0	12.2	2512.5	613.4 - 632.7
DMW	2500	2503.7	8.0	224.4	553.8 - 572.0
VRA/VGB	2575	2575.0	4.0	507.2	634.5 - 636.6
CGE	2500	2510.9	6.0	535.9	623.6 - 644.3
DLW	2575	2575.0	1.7	2010.8	644.6 - 672.0
ALA/APR	2575	2575.0	5.2	2989.2	646.0 - 672.0

Table 5-2 (Continued)

SAFETY VALVE INLET CONDITIONS FOR FSAR EVENT RESULTING IN LIQUID DISCHARGE (MAIN FEEDLINE BREAK)

Plant	Safety Valve Opening Setpoint (psia)	Maximum Pressurizer Pressure (psia)	Maximum Pressurization Rate (psia/sec)	Maximum Liquid Surge Surge Rate Into Pressurizer When Valve Is Passing Liquid (GPM)	Range of Liquid Temperatures at Valve Inlet (1st or Subsequent Openings) (°F)
POR	2575	2575.0	3.6	1575.4	646.0 - 672.0
PGE/PEG/ PSE/PNJ/ TVA	2575	2575.0	3.4	646.3	654.2 - 658.0
WAT	2575	2575.0	1.6	430.4	630.8 - 637.0
AMP	No Water Discharge Observed				

Table 5-3

SAFETY & RELIEF VALVE INLET CONDITIONS RESULTING FROM SPURIOUS INITIATION OF HIGH PRESSURE INJECTION AT POWER WHEN VALVES ARE DISCHARGING LIQUID

Reference Plant	Valve Opening Setpoint (psia)	Fluid State on Valve Opening (a)	Maximum Pressurizer Pressure (psia)	Range of Pressurization Rates (psi/sec)	Range of Surge Rates When Valve Is Passing Liquid (GPM)	Range of Liquid Temperature at Valve Inlet (°F)
SAFETY VALVES						
2-Loop	No Discharge					
3-Loop	No Discharge					
4-Loop	2500	Steam/Liquid	2507	0-4	0.0-628.3	567-572
RELIEF VALVES						
2-Loop	No Discharge					
3-Loop	2350	Steam/Liquid	2352	0-12	0.0-781	498-502
4-Loop	2350	Steam/Liquid	2353	0-4	113.1-1104.1	565-569

a. First/subsequent openings

The fluid conditions at the inlet to safety valves range from 567° to 572°F at 2507 psia with a maximum discharge rate of 628.3 gpm. No liquid discharge from the safety valves of the 2- and 3-loop reference plants was observed during the analysis. The fluid conditions at the inlet to PORVs range from 498°F to 569°F at 2353 psia with a maximum discharge rate of 1104.1 gpm. In this case no liquid discharge from the PORVs of the 2-loop reference plant is observed during the interval that the transient was analyzed.

In general valves open on steam and no liquid discharge is observed until the pressurizer becomes water solid. This is plant dependent and can vary anywhere from 20 minutes to more than six hours.

5.4 PLANT-SPECIFIC VALVE INLET CONDITIONS RESULTING FROM COLD OVERPRESSURIZATION EVENTS

Setpoints for the cold overpressurization mitigation system are conservatively determined to accommodate the rapid pressurization rates (up to 100 psi/sec) produced by cold overpressure transients (Section 2.3) during water-solid, low temperature operation of the reactor coolant system. In practice, however, fluid conditions at the relief valve inlet are not restricted to low temperature, subcooled water. A variable fluid condition (steam or water) and temperature (saturated to subcooled) at the valve inlet is possible due to administrative requirements for maintaining a pressurizer steam bubble during low temperature operations when pressure excursions due to cold overpressurization events are a possibility (Section 4.3).

The maximum range of potential cold overpressure fluid conditions at the relief valve inlet, covering all Westinghouse plants analyzed to date, may be inferred from Figure 5-1. These plants include: Comanche Peak Units 1 and 2, SNUPPS, Sequoyah Units 1 and 2, Watts Bar Units 1 and 2, South Texas Units 1 and 2, and Byron/Braidwood Units 1 and 2. A description of the indexed curves used to define the range of potential fluid conditions is presented below.

Legend Applicable To Figure 5-1

Index	Description
1	Locus of maximum primary system pressures developed following PORV #2 operation (limiting condition/ water-solid RCS)

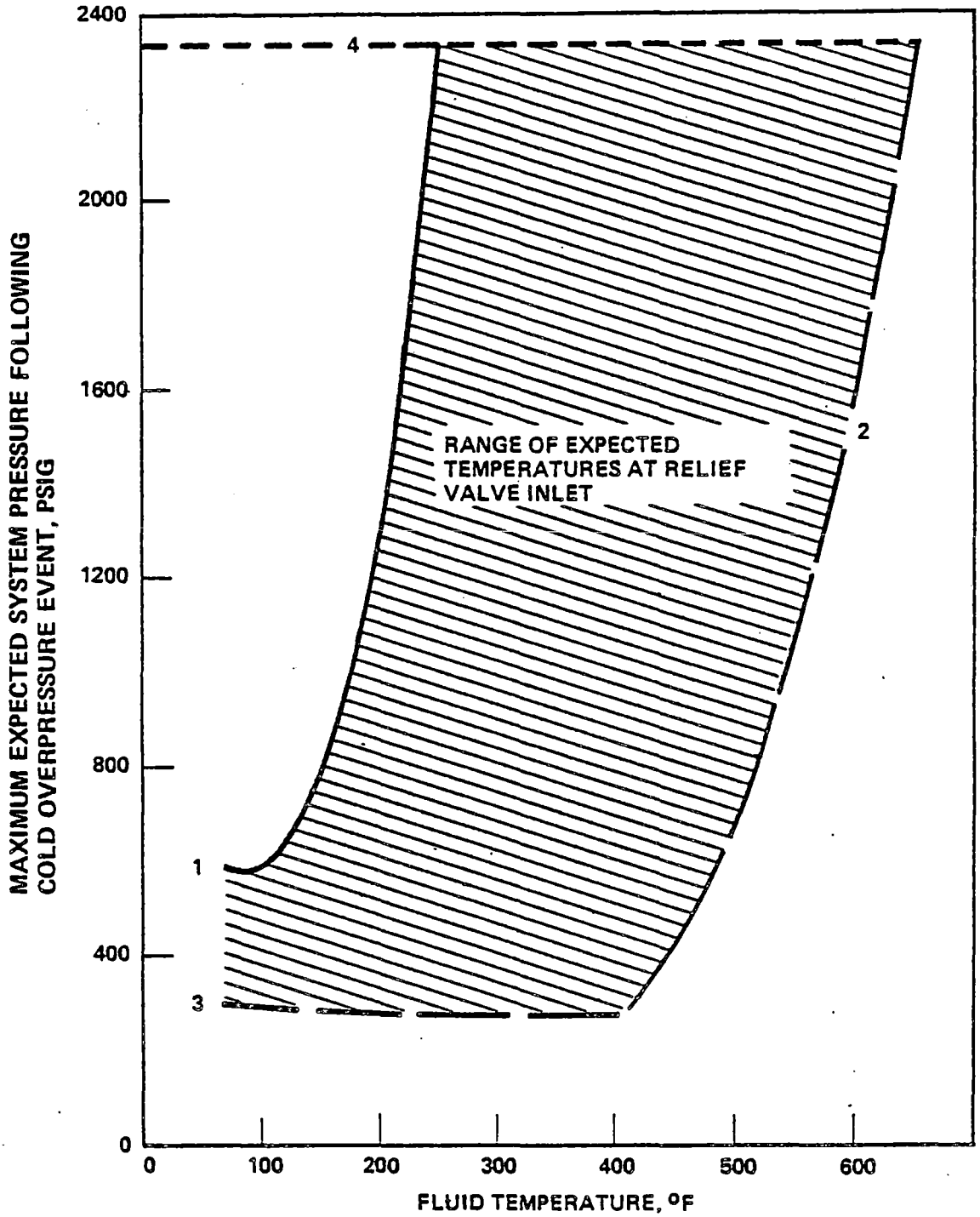

Figure 5-1. Potential Cold Overpressure Fluid Conditions at the Relief Valve Inlet

Index	Description
2	Potential steam/saturated liquid conditions in pressurizer per recommended administrative procedure
3	Minimum operating pressure limit to ensure reactor coolant pump No. 1 Seal integrity
4	Maximum relief valve setpoint based on high temperature operation

It should be noted that although possible, liquid discharge at temperatures lower than 100°F are extremely unlikely because of the limited time the plant is cold and in a condition capable of being pressurized (i.e., the RV head is off or the RCS is open for maintenance).

SECTION 6

REFERENCES

Tables 6-1 and 6-2 list the sources of data and the date that the analysis was performed. For FSAR events resulting in steam discharge (except feedline break), and spurious actuation of high pressure safety injection system, reanalysis was performed for this report. The feedline break analysis was performed as part of the FSAR analysis as required by Regulatory Guide 1.70.

TABLE 6-1

DATA SOURCE FOR FSAR EVENTS
RESULTING IN STEAM DISCHARGE
AND
SPURIOUS ACTUATION OF THE HIGH
PRESSURE INJECTION
SYSTEM

Reference Plant	Reanalysis Based On	Date Analysis Done
4-Loop	FSAR	1981
3-Loop	Cycle 3 Reload	1981
2-Loop	FSAR	1981

TABLE 6-2

DATA SOURCES FOR
FEEDLINE BREAK

PLANT	ANALYSIS DONE FOR	DATE ANALYSIS DONE
CAE/CBE/ CCE/CDE PBJ/PCJ	FSAR	1980
DAP/DBP	FSAR	1979
DCP/DDP	FSAR	1979
NAH/NCH	FSAR	1981
TBX/TCX	FSAR	1976
TGX/THX	FSAR	1977
CQL/CRL/ CSL/CTL	FSAR	1979
SNUPPS	FSAR	1980
DMW	FSAR	1978
VRA/VGB	FSAR	1976
CGE	FSAR	1978
DLW	FSAR	1974
ALA/APR	FSAR	1974
POR	FSAR	1974
PGE/PEG PSE/PNJ TVA	FSAR	1974
WAT	FSAR	1977
AMP	FSAR	1978