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December 1, 1976

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Office of Nuclear Reactor Regulation

Reference: (1) License No. DPR-3 (Docket No. 50-29)
(2) USNRC letter to YAEC dated 8/11/76
(3) YAEC letter to USNRC dated 9/3/76
(4) YAEC letter to USNRC dated 10/26/76

Dear Sir:

Reference (2) requested that Yankee evaluate the potential for reactor vessel overpressurization as a result of a single equipment failure or single operator error and implement design or procedural modifications as required to preclude such occurrences. Reference (3), Yankee's initial response to this request, indicated that review and necessary modifications to administrative controls and operation procedures would be accomplished within 60 days and that the balance of the information requested would be provided by February 16, 1977. Reference (4) provided additional information on procedural and administrative controls.

As a result of discussions between Yankee Atomic Electric Company and the NRC on October 29, 1976 and revised requests communicated verbally to Yankee and others during the November 3rd meeting, we are providing the following information as Appendices to this letter.

Appendix A: Details of Operating Procedures, including changes, which minimize the potential for overpressurization.

Appendix B: Plant Design Features, proposed design changes, which minimize the potential for and/or mitigate the consequences of overpressurization events. Included in this appendix will be a discussion of the design criteria to be met by the equipment comprising the plant's low temperature overpressure protection.

Appendix C: Evaluation of postulated events which could cause reactor vessel overpressurization. A transient analysis of the steam generator heat transfer event has been conducted utilizing a model developed by Yankee. The results of this analysis and a description of the model used is provided.

Appendix D: A schedule for implementation of design changes and completion of a topical report is provided.

We trust that you will find this response satisfactory; however, should you require additional information, please contact Mr. R. P. Shone at (617) 366-9011, Extension 2830.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

D.E. Vandenburgh

D. E. Vandenburgh
Senior Vice President

RPS/kg

APPENDIX A

Summary of Operating Procedures and Administrative Controls

As stated in Reference (3), many precautions and restrictions have been in existence in various operating procedures and administrative controls to minimize the potential for a reactor vessel overpressurization event. However, following our recent review, modifications were made to the following documents to provide additional precautions and safeguards against the possibility of this event:

OP-2105	Plant Cooldown from Hot Shutdown
OP-2653	Removal of the Safety Injection System from Service
OP-2140	Periodic Jogging of the Main Coolant Pumps during an Extended Shutdown
OP-2652	Preparation of the Safety Injection System for Normal Operation
AP-2002	Operations Department Personnel Shift Relief
OP-2100	Plant Startup from Cold Shutdown
OP-2125	Filling and Venting of Main Coolant System
OP-2654	Safety Injection Tank Makeup
OP-4204	Monthly Test or Special Operation of the Safety Injection Pumps

In order to illustrate the current impact of procedural and administrative controls on the potential for an overpressurization event, the following description of a plant cooldown and heatup is provided. Included in this discussion are all existing and recently added procedural steps and precautions related to vessel overpressurization.

Plant Cooldown from Hot Conditions to Cold Shutdown

Technical Specifications indicate that, on cooldown, the need for low temperature overpressure protection begins at 360°F RCS temperature. At this temperature and below, maximum allowable reactor vessel pressure is below full RCS operating pressure. Therefore, although other operational considerations are more limiting on reactor coolant pressure, the reactor vessel is protected adequately against overpressurization by the pressurizer safety valves at all temperatures above 360°F.

On cooldown, at 1800 psig, the following action is taken:

a. Automatic initiation of safety injection is blocked.

b. Safety Injection Panel switches for all high and low pressure safety injection pumps are placed in the "trip pull out" position.

RCS temperature is > 360°F at 1800 psig. This action precludes the possibility of a single equipment malfunction initiating safety injection flow; but a single operator error could start a safety injection pump. However, since RCS temperature exceeds 360°F, overpressurization of the vessel would not occur in this event.

Continuing the cooldown, at 1000 psig the following action is taken:

a. Isolate the safety injection accumulator.

b. Tag two of the three trains of ECCS out of service.

c. Close one motor operated isolation valve in the safety injection header to each RCS loop.

RCS temperature is > 360°F at 1000 psig; therefore, at this point, low temperature overpressure protection is not required. However, the action taken at 1000 psig provides protection required later in the cooldown. Item (c) above blocks all safety injection flow to the vessel. This together with item (b) performed at 1800 psig effectively precludes the possibility of a single operator error initiating safety injection flow to the vessel. Item (a) together with the fact that accumulator cover gas pressure is always ≤ 347 psig eliminates the possibility of overpressurizing the vessel by accumulator injection.

When RCS temperature is between 300°F and 330°F, the shutdown cooling system is opened to the RCS. This makes available two passive safety valves each with a capacity of 101 gpm at 400°F with 10% accumulation over a set pressure of [illegible] psig. As explained in detail in Appendix C, the availability of the shutdown cooling system safety valves at reactor coolant temperatures below 300°F provides protection against all overpressurization events postulated to occur at or below this point.

During the cooldown, system temperature is being reduced by steam dump to either the condenser or the atmosphere. The water level in all steam generators is allowed to drop to a specific low level by the time steam pressure is reduced to the point where steam dump is ineffective. Subsequent to this, the steam generators are vented to the atmosphere, and water level is increased to a high point with cool make-up water. Although no means is available to measure directly the bulk water temperature in the steam generator s[illegible]ry when no steam is being produced, calculations show that it is always [illegible]00°F. This is an important parameter since it validates the maximum tempe[illegible]re difference assumed to exist across the steam generator tubes in the reverse heat transfer transient analysis.

When RCS temperature and pressure are below 200°F and 300 psig respectively, the following action is taken:

a. The third train of ECCS is tagged out of service.

b. Isolation valves redundant to those closed in item (c) performed at 1000 psig are closed.

This action further reduces the possibility of inadvertent initiation of safety injection although previous action has adequately established protection against this event.

At a RCS temperature of 180°F, all main coolant pumps are stopped, control switches are placed in the trip pullout position, and their breakers withdrawn. The MCP's will not be operated further during the period of cold shutdown except for periodic jogging during an extended shutdown. This action is taken, among other reasons, as a redundant measure to protect the vessel against overpressurization due to reverse heat transfer that might occur with a solid plant when RCS flow is initiated.

When RCS temperature is ≤ 180°F, steps are taken to collapse the pressurizer steam bubble. Prior to attaining solid conditions in the RCS, an operator is stationed at the Main Control Board. His sole responsibility is to monitor RCS pressure while solid and take action necessary to terminate any pressure excursion. This operator, having no other duties, will serve reliably to open the pressurizer solenoid relief valve should system pressure approach Technical Specification limits. The water relief capacity of the existing pressurizer relief valve opened at 425 psig is conservatively calculated to exceed a minimum of 120 gpm at an accumulation of 10%. The capacity increases to 160 gpm at 25% accumulation (550 psia).

Once solid, RCS pressure is maintained between 50 - 150 psig by charging and bleed as necessary while the pressurizer is cooled down from approximately 400° F to 170°F. This process may require several hours. When cooled to 170°F, the pressurizer is vented to the low pressure vent header via a line capable of relieving in excess of 100 gpm at 425 psig system pressure. At this point, since an additional relief path has been made available to the RCS, the special operator stationed prior to attaining solid conditions is dismissed.

Once the low pressure setpoint feature of the pressurizer solenoid-operated relief valve is implemented, the requirement to station the special operator during solid conditions may be removed.

Periods of Extended Shutdown

Operating procedures require that each main coolant pump be "jogged electrically" at least once every thirty days when the plant is shutdown and the reactor coolant loops are filled. A "jog" is the energization of the pump motor for three seconds. The event of concern during this operation is the possible expansion-produced overpressurization resulting from circulating cool reactor coolant through the steam generator tubes where heat can be transferred to the reactor coolant from warmer secondary coolant. When a MCP is jogged in accordance with OP-2140, its coolant loop is completely isolated from the reactor vessel and the loop bypass valve is open. Since this operation is performed only during an extended shutdown, the probability is high that there will be no significant energy in the steam generator secondary water to produce expansion when reactor coolant circulation begins. In addition, a mere three second jog will not develop any significant reactor coolant flow. Furthermore, should the operator fail to stop the MCP after three seconds, the only flow path available is through the loop bypass line which restricts flow to 2500 gpm, approximately 1[illegible]% of full loop flow. Any expansion produced, however, would not be transmitted to the vessel because both loop isolation valves are closed. An additional safety precaution has been added to OP-2140. It requires that when MCP's are to be jogged, the RCS shall not be in a solid condition.

Plant Heatup from Cold Shutdown to Hot Conditions

Reactor Coolant System Fill and Vent

Initial conditions are:

a. Shutdown cooling system in service making available the 202 gpm relief capacity of its passive code safety valves.

b. All ECCS and main coolant pumps are tagged inoperable.

c. The reactor coolant system may be partially voided or filled and vented to the low pressure vent header.

The entire reactor coolant system is filled (or drained) to establish an initial pressurizer level of approximately 40% full. Filtered station air at 80 psig is admitted to the pressurizer air space and the vent is isolated. Water level is then increased until air bubble compression increases RCS pressure to 225 - 250 psig: the pressure required to operatue the MCP's. All hot leg loop isolation valves are opened, conncenting each loop to the reactor vessel and pressurizer; but since a MCP starting interlock requires the cold leg loop isolation valve to be closed, it is not opened. RCS water temperature at this point will be $\geq$ 100°F.

The venting procedure begins with short jogs followed by venting of the pump casing. Since the cold leg loop isolation valve is shut, flow is limited to a maximum of 2500 gpm bypass flow. The repeated jogs tend to equalize any mismatch of primary and secondary temperature in the steam generator in small segments. Although any pressure surges are reflected back to the reactor vessel through the open hot leg loop isolation valve, the brief jogs together with the air cushion in the pressurizer tend to minimize the peak of any excursion.

Following the jogs described above for pump venting, a pump is re-started, its cold leg isolation valve is opened, flow is allowed to increase to approximately 80% of full flow then the pump is tripped. Further venting occurs. This process is repreated for each loop.

Subsequently, each pump is run at full flow for three minutes followed by additional venting. Finally, all four pumps are operated together for periods of ten and sixty minutes at full flow with required venting in between. From the above it is evident that reactor coolant flow is initiated in many brief jogs so that any pressure surges resulting from reverse heat transfer across the steam generator tubes will be limited to many small excursions rather than one large one. Furthermore, the presence of an air bubble cushion in the pressurizer will help absorb any pressure spikes.

Upon completion of the venting process, the main coolant pumps are stopped. The RCS is de-pressurized by venting the pressurized air bubble to the low pressure vent header. Pressurizer level is increased, expelling all remaining air. Since at this point the RCS will be placed in a solid condition, an operator is stationed at the main control board with duties restricted to pressure monitoring and control as was described for the cooldown process.

Plant Heatup

The pressurizer bubble is formed at approximately 30 - 50 psig RCS pressure. RCS temperature at this time is less than 200°F. After pressurizer bubble formation, level is maintained such that the steady state steam volume occupies approximately 200 of the pressurizer's 295 cubic feet.

After bubble formation, RCS pressure is increased to 225 - 300 psig for MCP operation and the RCS heatup is commenced. Although no pressure excursions are expected as a result of these MCP starts, the pressurizer steam bubble is available to absorb any transient.

All pumps, valves and other components of the ECCS system are placed in service in the reverse order as removal. This procedure protects the vessel against any adverse effects of inadvertent initiation of high or low pressure safety injection.

The shutdown cooling system is maintanined such that its passive safety valves remain available as relief protection for the vessel up to a temperature of 300°F. This procedure provides required protection against an over-pressurization event caused by either continuous charging or pressurizer heater operation without bleed flow.

APPENDIX B

Plant Design Features

The following is a discussion of the existing design features as well as those to be added in the near future to protect against reactor vessel overpressurization.

Existing Design Features

a. Shutdown cooling system passive code safety valves are available to relieve expansion or mass addition at all reactor coolant temperatures below 300°F. Each of two safety valves has the capacity to relieve 101 gpm of 400°F water at a set pressure of 425 psig plus ten percent accumulation. These passive, spring loaded safety valves are capable of providing relief without dependence on either an air system or an electrical power supply.

   The water relief capacity stated above for the shutdown cooling safety valves presumes no flashing in the valve. The discharge from these safety valves is directed via closed piping to a terminal tank. Flashing, if it occurred in the discharge piping, could result in a buildup of back pressure at the safety valve discharge port. This back pressure, if sufficiently high, could reduce the relieving capacity of the valve to a value below that given above.

   The maximum temperature of water that could be relieved by the safety valves is 300°F. Therefore, the probability of flashing causing a reduction in relieving capacity exists for the shutdown cooling system safety valves.

   The question of the effect of possible flashing on safety valve capacity can be answered only by a detailed and quite complex analysis of the specific discharge piping configuration and termination tank. This analysis will be completed for the shutdown cooling system safety valves. If the results of the analysis indicate a significant reduction in the relief capacities assumed to the extent that conclusions arrived at herein are invalidated, a supplementary submittal will be made.

b. The pressurizer solenoid operated relief valve is available under dedicated operator control to relieve water when the reactor coolant system is solid. The calculated water relief capacity of the valve is plotted versus pressure in Figure A-1 of the model description contained in Appendix C. At a setpoint of 425 psig plus ten percent accumulation, the relief capability is 120 gpm of 400°F water, more than equivalent to a single shutdown cooling system safety valve.

   It is appropriate to mention that the operation of the pressurizer solenoid relief valve depends only on electrical power to operate. This is not an air operated valve. The solenoid actuates a pilot which then allows fluid pressure to open the valve. Therefore, loss of station or instrument air pressure will have no effect on the operability of this valve.

Proposed Design Features

a. A low pressure setpoint feature is to be added to the actuation circuitry for the pressurizer solenoid relief valve. This setpoint is to be established in the range of 425 to 500 psig. A separate pressure transmitter to provide the pressure signal for this feature will be installed in an existing tap located in the cold leg of a reactor coolant loop. The position of this tap is unisolable from the reactor vessel by closure of the cold leg loop isolation valve. The low pressure setpoint feature will be enabled and disabled at a specific pressure by key switch on the main control board. This feature when enabled will parallel, not override the existing circuitry which causes the valve to open at the high pressure setpoint.

b. A new full range pressure recorder is to be installed on the main control board to continuously monitor reactor vessel pressure. The pressure transmitter providing the signal to this recorder will be the same one that provides the input to the pressurizer relief valve low pressure setpoint. The recorder will be capable of indicating pressures up to 3000 psig; but pressures in the range of 400 to 600 psig will also be accurately measured.

c. A new reactor coolant pressure alarm is to be added to the main control board annunciator. This alarm will be enabled by the same key switch that will switch in or out the pressurizer relief valve low pressure setpoint. The alarm will warn the operator that reactor coolant pressure is approaching Technical Specifications limits. An existing pressure transmitter separate from that utilized in (a) and (b) above is to provide the signal for this alarm.

d. A new pressurizer solenoid operated relief valve of increased capacity is to be installed to replace the existing valve. This is a long term improvement item, since eighteen months is the expected delivery lead time. However, when installed, the steam and water relief capacities of the pressurizer will have been increased. Steam relief capacity will improve from 50,000 to 72,690 pounds mass per hour; water relief capacity improvement is shown in Figure A-1 of the transient analysis model description contained in Appendix C.

Design Criteria

A. Operator Action

The criteria for operator action is consistent with that suggested at the recent meetings between PWR owners and the NRC. The plant is operated in accordance with established operating procedures; therefore protection afforded by normal operating procedures is a vital part of the overall plan for protection against overpressurization. In the analysis of postulated events, once the event has occurred, operator action to mitigate its consequences is conservatively assumed to not occur for ten minutes (with one exception discussed below).

Normal operating procedures provide the following specific action which protects against overpressurization events:

a. They insure that the relief capacity of the shutdown cooling system safety valves (202 gpm) is available at all temperatures below 300^{o} F.

b. During solid water conditions, an extra operator is stationed at the main control board; his sole responsibility is to monitor reactor coolant pressure and take action to terminate any pressure surge that threatens to exceed Technical Specifications limits. In this particular case, an operator response within ten minutes to mitigate the consequences of an overpressure initiating event is assumed. Justification for this assumption is based upon the fact that this special operator is dedicated to the limited responsibility of monitoring system pressure and taking required action if a pressure surge occurs. The required action is simple: open the pressurizer solenoid operated relief valve if reactor coolant pressure exceeds a specified value. Further supplementary action by this operator is not mandatory. Therefore, since this dedicated operator's responsibility is restricted and simple, we feel justified in expecting a prompt response well within the ten minute period assumed for all other, post-event, mitigating action. In any event, the above operator action, expected within ten minutes, is totally redundant to adequate protection provided by either of the following:

1. The 202 gpm relief capacity afforded by the shutdown cooling system safety valves at all reactor coolant temperatures below 300° F.

2. Operator action performed at the ten minute point that is effective in protecting against overpressurization.

c. ECCS components are systematically removed from service during heatup and cooldown periods when the potential for overpressurization due to their inadvertent activation exists.

d. Main coolant pump operation is avoided during plant conditions which could produce overpressurization from adverse primary to secondary temperature differences.

e. On a plant cooldown, steps are taken to reduce the steam generator secondary fluid temperature following termination of steam dump.

f. The usage of an air or steam bubble in the pressurizer is maximized while the duration of solid water conditions is minimized.

B. Single Failure

The single failure criterion has been applied to the consideration of postulated overpressurization events. Either a single equipment failure or a single erroneous operator control manipulation has been assumed to initiate each of the postulated overpressurization events discussed in Appendix C. Single failure of the pressure limiting equipment has been assumed to a degree in that below 300°F, the plant has available three relief valves, two of which are sufficient to limit transient pressure in the worst case event to below the reactor vessel overpressurization limit. The replacement pressurizer relief valve discussed previously in this Appendix will posses sufficient relief capacity to be fully redundant to both of the shutdown cooling safety valves. In the temperature range between 300°F and 360°F, operator action after a response delay interval of ten minutes provides the required overpressure protection. Above 360°F, the reactor coolant can be pressurized to full system design pressure without exceeding a reactor vessel pressure limit. The redundant steps in immoblizing the ECCS components at low temperatures meet the single operator error criteria. In summary, every means practicable has been employed to provide the margin of protection afforded by the single failure criterion in both preventing an event initiation and limiting the effects of an event once initiated.

C. Testability

Limited opportunity exists for testing the components being relied upon to provide low temperature overpressurization protection. The shutdown cooling system code safety valves are welded to the system piping and are, therefore, not testable. The solenoid of the pressurizer relief valve can be tested prior to plant cooldown; but due to the design of the valve, its total operability cannot be conveniently determined. At the present time, it is planned to modify operating procedures to require a pilot only actuation test of the pressurizer relief valve during cooldown for each refueling shutdown.

D. Seismic Considerations

The Yankee Rowe Plant was not designed to specific seismic criteria. Thus the purchase specifications for the shutdown cooling safety valves or the pressurizer solenoid relief valve did not address seismic requirements. However, the design of these valves is identical to valves that are seismically qualified. The seismic design criteria of any mechanical or electric equipment to be added in the design changes previously described will be in accordance with current YAEC specifications for additions or modifications to plant equipment.

E. IEEE 279

The requirements of IEEE-279 will not be blanketly applied to the electrical equipment to be installed as design changes. Instrumentation and other electrical equipment will be specified to meet the applicable requirements of classification 1E. However, the separation and single failure criteria are not considered applicable to this installation and therefore will not be met. In addition, these instruments are not required to function in a post-LOCA environment.

F. Common Mode Failure

There is no single failure of a component or system or single event that has been identified in our analysis as capable of both causing an over-pressurization event and defeating the protection afforded against such events.

APPENDIX C

Postulated Overpressurization Events

This Appendix presents the results of our analyses of all postulated events having the potential of causing reactor vessel overpressurization. The specific events are listed below and discussed in order:

a. Inadvertent ECCS operation

b. Charging without bleed flow

c. Pressurizer heater operation without bleed flow

d. Loss of shutdown cooling heat removal capacity

e. Reactor coolant flow initiation transients

Inadvertent ECCS Operation

Inadvertent operation of two ECCS components was considered: the opening of the safety injection accumulator isolation valve and the startup of one or more trains of ECCS pumps. (One train of ECCS pumps consists of a high and a low pressure safety injection pump, associated tanks, piping and valves).

Inadvertent opening of the safety injection accumulator isolation valve poses no threat to vessel overpressurization, because of the low pressure (347 psig) maintained in the accumulator.

Each ECCS pump train is intended to operate in tandem, and therefore, has the potential for producing pressures in the range of 1,200 to 1,500 psig if initiated inadvertently. Actual pressure depends upon relief capability available. In terms of reactor vessel maximum pressure limits, this pump head capability presents a potential overpressurization threat at temperatures below 250° F.

In order to eliminate the potential for overpressurization, timely steps are taken as described below to remove the possibility of inadvertent start of ECCS pump trains during that portion of heatup or cooldown when the plant is vulnerable.

Present technical specifications limits allow the reactor vessel to be pressurized to full system pressure at all temperatures above 360°F. At 1,800 psig during a plant cooldown, automatic initiation of safety injection is blocked and the control switches for all high and low pressure safety injection pumps are placed in the trip pull-out position. At this time, reactor coolant temperature is above 360°F. At 1,000 psig with reactor coolant temperatures still above 360°F, a motor operated isolation valve in the safety injection header to each reactor coolant loop is closed. Also, the safety injection accumulator isolation valve is locked closed and made electrically inoperable. The above actions, taken during cooldown before protection against low temperature overpressurization is required, places both the ECCS pump trains and the accumulator in such a condition that neither a single equipment malfunction nor a single erroneous operator control manipulation can result in reactor vessel pressurization from these components. Furthermore, safety injection accumulator maximum pressure is 347 psig, which is insufficient to pressurize the reactor vessel above allowable limits.

On a plant heatup, the ECCS components are placed in service in the reverse order that they were removed, thus affording the same protection as during cooldown.

It is concluded that, since plant procedures effectively eliminate the potential for low temperature overpressurization from inadvertent actuation of ECCS components, further protection against this event is not required.

Charging Without Bleed Flow

At Yankee Rowe, the charging and bleed path is similar in function, although not identical, to the charging and letdown portion of the chemical and volume control systems of later vintage Westinghouse plants. Since several of the reactor vessel overpressurization events that have occurred in other operating plants were initiated by inadvertent closure of the letdown path while charging, this incident has been evaluated for the Yankee Rowe plant.

Each of the three positive displacement charging pumps is capable of 33 gpm. Although three pumps are almost never operated simultaneously, 100 gpm was taken as the assumed charging flow rate in the analysis of this event. Plant procedures require that, whenever the reactor coolant system is closed and reactor coolant temperatures below $300^{o}F$, two shutdown cooling system safety valves are available. The capacity of each safety valve under these conditions is at least 101 gpm at a 425 psig setpoint plus 10 percent accumulation. Above $300^{o}F$, there will also be a pressurizer steam bubble with a steady state volume of $\geq$200 cubic feet.

Analysis of this event is done for two conditions:

Case A: Reactor Coolant Temperature $\leq 300^{o}F$; Shutdown Cooling System Safety Valves Available

Case B: Reactor Coolant Temperature $\geq 300^{o}F$; Shutdown Cooling System Safety Valves Unavailable

In Case A, overpressure protection is provided by the shutdown cooling system safety valves, either of which is capable of relieving the required 100 gpm at all temperatures $\leq 300^{o}F$. Backup protection is afforded by at least one of the following means, depending upon specific plant conditions:

1. Pressurizer steam bubble
2. Pressurizer air bubble
3. Pressurizer solenoid operated relief valve
4. Pressurizer vent line to the low pressure vent hea er

In Case B, overpressure protection is provided by operator action no sooner than 10 minutes after event initiation. The pressurizer steam bubble provides the required time delay. At 300°F, the overpressure limit is about 1300 psig. Analysis of this event indicates that, over the ten minute period, pressure increase is on the order of 100 to 200 psi. Plant operating procedures have incorporated an operational pressure limit that provides an adequate margin to the Technical Specification pressure limit in the temperature range of 300°F to 360°F. This margin precludes vessel overpressurization from this event.

It is concluded that protection is afforded under all conditions against the hypothetical occurrence of 100 gpm continuous charging into the reactor coolant system without attendant bleed flow.

Pressurizer Heater Operation Without Bleed Flow

Pressurizer heater operation without bleed flow produces expansion which in turn, increases the pressure of the reactor coolant system. Depending upon the time of occurrence of the postulated event, a variety of plant conditions affecting the outcome can prevail. Specifically:

1 The pressurizer may be solid or contain a steam bubble.
2. The shutdown cooling system may be in service or isolated.
3. All or a fraction of the total pressurizer heater capacity may be energized.

It is unrealistic to arbitrarily combine the most limiting of the above variable conditons, since that hypothetical combination would not normally occur during proper plant operation. Furthermore, such an arbitrary condition might be brought about only by the assumption of more than a single failure or operator error. Therefore, this event will be analyzed by considering four separate cases which could normally arise. It is assumed throughout the discussion that the single equipment failure or operator error is that which causes isolation of the bleed path. No further failures or errors occur.

Case A: Pressurizer heaters are energized in a solid system; fluid temperature is below the saturation temperature for existing pressure. The bleed path becomes isolated.

This situation could only arise when all heaters are energized in preparation for forming apressurizer bubble. The following pertinent conditions would prevail:

1. All heaters energized.
2. Reactor coolant system solid.
3. Shutdown cooling system in service.
4. Main coolant pumps stopped.
5. Special, dedicated operator stationed.

Since steam is not formed in this case, system expansion arises only from the slight expansion of heated water in the pressurizer. The expanding water, however, would increase the pressure of the solid system. Analysis indicates that at the design relief pressure of the shutdown cooling safety valves, the expansion rate is less than 33 gpm, the capacity of a single charging pump. This is less than the capacity of three charging pumps for which adequate overpressurization protection has been previously shown to exist.

Case B: All pressurizer heaters are energized with saturation conditions existing in the pressurizer. The bleed path becomes isolated.

This condition could only arise when in the actual process of forming a bubble. All conditions would be as in Case A except that a pressurizer bubble would be forming or formed.

The system expansion rate would be significantly higher than in Case A due to the conversion of water to steam. The expansion rate is inversely proportional to system pressure: and analysis has shown that, prior to pressure exceeding 500 psig, the expansion rate is reduced below the relief capacity of both shutdown cooling system safety valves.

The pressurizer relief valve would be available under operator control to relieve steam as an additional measure. The following actions are available to the operat as further backup to terminate the pressure transient:

a. De-energize pressurizer heaters.
b. Initiate pressurizer auxiliary spray.
c. Attempt to re-open the bleed path.
d. Open a drain path from a main coolant loop to the low pressure surge tank.

Therefore, it can be concluded from the above discussion, that adequate protection exists against overpressurization from this event. The shutdown cooling system safety valves provide sufficient relief capacity; while a variety of operator-initiated back-up actions are available to the special, dedicated operator, stationed during this event.

Case C: The partial, steady state load of pressurizer heaters is energized with a steam bubble and normal water level established. The shutdown cooling system is in service. The bleed path becomes isolated.

It is readily apparent that, with less than the full complement of pressurizer heaters energized, this event results in a significantly lower expansion rate than in Case B. This case further differs in that the special, dedicated operator is not stationed. Therefore, the protection afforded by this operator as detailed in Case B is assumed to be unavailable. However, the relief capacity of the available shutdown cooling system safety valves provides the required overpressure protection.

Case D: The same conditons prevail as in Case C except that reactor coolant temperature is above 300°F. The shutdown cooling system is unavailable.

In this case, no low pressure relief protection is available. When the bleed path is isolated, system pressure increases. Protection against overpressurization is provided by operator action, assumed to take place no sooner than ten minutes after event initiation. Analysis indicates that even when the unrealistic assumption is made that all pressurizer heaters are energized, the resultant pressure increase in the ten minute period is on the order of only 500 psi.

Plant operating procedures have established an operational pressure limit that provides an adequate margin to the Technical Specifications pressure limit in the temperature range of 300°F to 360°F. This margin precludes vessel overpressurization from this event.

It is concluded that protection is afforded under all conditions against the postulated event of pressurizer heater-induced expansion without accompanying bleed flow.

Loss of Shutdown Cooling Heat Removal Capacity

Loss of shutdown cooling heat removal capacity could cause expansion of the reactor coolant by forcing it to absorb both core decay heat and main coolant pump thermal energy. In this case, the bleed path is available.

Loss of shutdown cooling is a potential cause of overpressurization only at temperatures below 300°F. Above this temperature, sufficient steam pressure exists in the steam generators to render them effective heat removal devices. The magnitude of the total heat source would depend upon plant conditions. On a cooldown, the decay heat level could be high, but only one or two main coolant pumps would be operating. On a plant heatup, four main coolant pumps could be operating, but core decay heat would be at a lower level than on cooldown. Pressurizer heaters could also be energized under conditions of the first three cases of the previous discussion. Furthermore, since the normal bleed path is available, only expansion above that being passed by the bleed path need be handled by available relief valves.

The maximum expected contribution to the system expansion rate from core decay heat plus main coolant pump thermal energy is below 33 gpm, the flow rate of one charging pump. This expansion rate could be coupled with that produced by the pressurizer heaters under previsouly discussed conditions. The total expansion rate in any case would not exceed 202 gpm. Therefore, overpressure protection would be afforded by the shutdown cooling system safety valves.

The inadvertent closure of either of the two shutdown cooling system inlet isolation valves could eliminate both the heat removal and relief valve capacity of this system. However, the bleed path would not be isolated, since it is independent of the shutdown cooling system. However, this event is considered an unreasonable postulation for Yankee Rowe because of the administrative control existing over the system inlet isolation valves. Push button controls for each valve motor operator are located on a local plant panel. The push buttons are ineffective unless a key is inserted and turned in an accompanying lock. Administrative control over the keys is maintained by the duty Shift Supervisor. Because of this controlled key interlock, the inadvertent closure of the shutdown cooling system inlet isolation valves has not been postulated as a potential overpressurization initiating event.

Reactor Coolant Flow Initiation Transients

Pressure transients resulting from the initiation of reactor coolant flow can develop if the steam generator secondary fluid is warmer than the reactor coolant accelerating through the tubes. In the reactor coolant flow initiation transient of a PWR having no loop isolation valves, the reactor coolant pump is started and flow accelerates to its full value in 20 to 30 seconds. The existence of a temperature difference between the primary and secondary fluids, results in heat transfer into the reactor coolant during flow acceleration. If solid conditions prevail, the resulting expansion can lead to a very rapid pressure increase. If the pressurizer has a steam or air bubble, the total expansion remains unchanged, but is absorbed to some degree by the bubble, and consequently, the pressure transient is less severe. Since this incident is of generic concern, and is being analyzed by all PWR owners, it is appropriate to point out design features of the Yankee Rowe plant affecting this transient, that differ from those of other PWR's. These differences center around the reactor coolant loop isolation valves and are as follows:

A. A main coolant pump cannot be started unless the associated cold leg loop isolation valve is closed and the bypass valve is open. When a main coolant pump is started under these conditions, flow circulates between the pump and the steam generator through the bypass line which limits flow to 2500 gpm, one tenth the full value. The loop hot leg isolation valve can be (and usually is) open; therefore, any pressure surges created are transmitted to the reactor vessel.

B. Full reactor coolant flow can be initiated only by opening the cold leg loop isolation valve once the main coolant pump is running. (The opening of the cold leg isolation valve is affected by a temperature interlock designed to protect the core against a cold water reactivity addition accident. However, this interlock, because of its particular design, provides no protection against vessel overpressurization; and therefore, it is not considered in this discussion.)

Although it has not occurred, it is expected that a reactor coolant flow initiation transient such as described above, would be less severe at Yankee Rowe than at a similar PWR plant not equipped with loop isolation valves because:

A. Reactor coolant flow must be initiated in a two step process. First, the main coolant pump is started and flow can accelerate to only 2500 gpm on the bypass line. This low flow serves to gradually reduce any existing temperature differences between the primary and secondary fluids. Secondly, when full reactor coolant flow is initiated by opening the cold leg loop isolation valve, the valve opening time extends the duration of flow acceleration as compared to that of flow initiated by pump start alone.

B. Plant operating procedures specify that there will always be either an air or steam bubble in the pressurizer whenever a main coolant pump is started.

The reactor coolant flow initiation transient was analyzed to determine the behavior of pressure vs. time. The results of this analysis appear in Figures B-1 and B-2 of the analysis model description. The following assumptions have been made in the analysis of this transient:

A. The transient is initiated by reactor coolant flow acceleration in response to the opening of the cold leg isolation valve. Although it is assumed that the main coolant pump is running at the time the valve starts opening, no account is taken for the mitigating effects of prior main coolant pump operation on the bypass line.

B. An initial temperature differential of 100°F is assumed between the reactor coolant and secondary fluid in the steam generator. This conservative value is significantly greater than the maximum value expected as a result of normal operating procedures.

C. The calculated relief capacity of the pressurizer solenoid relief valve as plotted in Figure A-1 of the transient analysis model description, is assumed to be the only system relief available. The safety valves of the shutdown cooling system, which are normally available when main coolant pumps are started, provide in excess of 100 percent redundant capacity.

D. The reactor coolant system is assumed to be solid. The water relief characteristics of the pressurizer relief valve are calculated conservatively. For additional details, refer to the transient analysis model descritpion.

E. Thermal or pressure expansion of the reactor vessel, pressurizer and coolant piping is neglected.

Additional assumptions can be found in the description of the transient analysis model supplied with this Appendix.

From Figures B-1 and B-2 of the analysis model description, it can be seen that the plant has redundant relief capacity capable of limiting the consequences of this hypothetical event such that reactor vessel overpressurization does not occur.

Reactor Coolant Pump Seal Water

The possibility of reactor coolant pump seal water being instrumental in causing an overpressurization event has been identified for some PWR plants. In these plants, relatively cool seal water is supplied by auxiliary systems to the mechanical seals of each reactor coolant pump. If the pump is idle, the portion of cool seal water that enters the reactor coolant system via the pump seals can accumulate in the pump bowl. On subsequent pump start, expansion can result from the reverse heat transfer phenomenon as this cool seal water is swept through the warmer steam generator. The main coolant pumps at Yankee Rowe are of the canned rotor design and therefore are not supplied with external seal water. Therefore the seal water potential overpressure event is not applicable to Yankee Rowe.

The above discussions have shown that a combination of operating procedures and installed plant equipment protects the plant against reactor vessel overpressurization. Operating procedures maintain plant parameters within certain bounds and preclude the possibility of occurrence of certain postulated events. Installed plant equipment (safety and relief valves, associated instrumentation and alarms) functions effectively to limit system pressure to below maximum values if an event occurs.

Current pressure-temperature limitations and existing plant equipment is the basis for the above discussions. As time progresses, reactor vessel irradiation will alter the pressure-temperature limitations. As these conditions change, appropriate reviews will be made of operating procedures, administrative controls and the degree of protection existing against postulated events. These periodic reviews will verify that acceptable margins of safety continue to exist thus ensuring continued protection of the reactor vessel against overpressuration.

This concludes the discussion of postulated overpressurization events. The remaining pages of this Appendix present a description of the model used to analyze the reactor coolant flow initiation transient.

Analysis of Reactor Coolant Flow Initiation in Presence of a Differential Secondary to Primary Temperature Condition

Introduction

To determine the transient resulting from initiation of flow during solid pressurizer conditions, an analytical model was developed. Figure 1 provides a schematic of the nodal scheme utilized in this model. The model consists of two loops which, for startup of a single reactor coolant pump simulates the active loop (loop 1) and the inactive loop(s) (loop 2). System volumes modeled are:

1. Reactor core (variable number of nodes)

2. Core outlet plenum

3. Hot legs (variable number of nodes)

4. Steam generator active tube volume (variable number of nodes)

5. Cold legs (variable number of nodes)

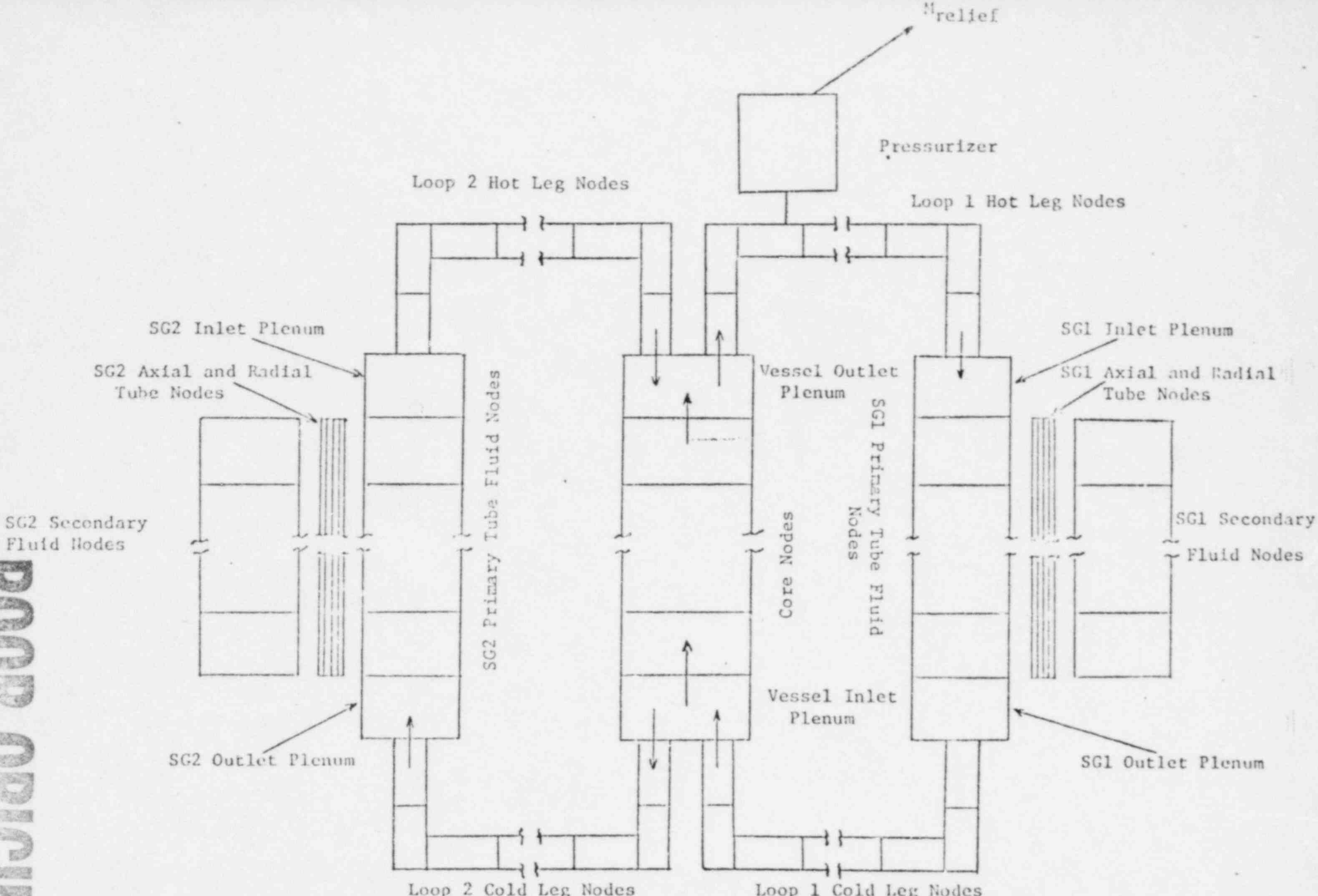
6. Core inlet plenum

7. Pressurizer

One-dimensional heat transfer across the steam generator tubes from secondary to primary and subsequently time dependent tube and secondary nodal fluid temperatures are included. To simulate a plant containing more than two loops, the loops other than the active loops must be combined.

The following discussion provides a summary of the procedures employed in the program to determine the space and time dependent fluid conditions in the primary system and subsequently the system pressure.

Figure 1


M relief
Pressurizer
Loop 2 Hot Leg Nodes
Loop 1 Hot Leg Nodes
SG2 Inlet Plenum
SG2 Axial and Radial Tube Nodes
SG2 Secondary Fluid Nodes
SG2 Primary Tube Fluid Nodes
SG2 Outlet Plenum
Vessel Outlet Plenum
Core Nodes
Vessel Inlet Plenum
SG1 Inlet Plenum
SG1 Axial and Radial Tube Nodes
SG1 Primary Tube Fluid Nodes
SG1 Secondary Fluid Nodes
SG1 Outlet Plenum
Loop 2 Cold Leg Nodes
Loop 1 Cold Leg Nodes

A. System Modeling

A.I. Primary Fluid Energy Distribution in Time

A.I.1 Cold Legs

$$h_{CL}(i)\big|_{t+\Delta t} = h_{CL}(i)\big|_t + \frac{\Delta t*\dot{m}(t+\Delta t/2)}{M_{CL}(i)} * \Big(h_{CL}(i-1)-h_{CL}(i)\Big)\Big|_t$$

where: $h_{CL}(i)$ = enthalpy of cold leg node i

t = time

Δt = time step

$\dot{m}$ = mass flow rate through loop

$M_{CL}(i)$ = mass of cold leg node i

A.I.2 Hot Legs

$$h_{HL}(i)\big|_{t+\Delta t} = h_{HL}(i)\big|_t + \frac{\Delta t*\dot{m}(t+\Delta t/2)}{M_{HL}(i)} * \Big(h_{HL}(i-1)-h_{HL}(i)\Big)\Big|_t$$

where: $h_{HL}(i)$ = enthalpy of hot leg node i

$M_{HL}(i)$ = mass of hot leg node i

A.I.3 Steam Generator Primary

$$h_{SGP}(i)\big|_{t+\Delta t} = h_{SGP}(i)\big|_t + \frac{\Delta t}{M_{SGP}(i)} * \Big(\dot{m}(t+\Delta t/2)*(h_{SGP}(i-1)-h_{SGP}(i))$$
$$+ UI(i)\big|_t * A(i,J) * (TT(i,J) - TSGP(i))\Big)\Big|_t$$

where: $h_{SGP}(i)$ = enthalpy of steam generator primary fluid node i

$M_{SGP}(i)$ = mass of steam generator primary fluid node i

$UI(i)$ = heat transfer coefficient from steam generator tubes to coolant for node i

$A(i,J)$ = steam generator tube surface area in contact with primary fluid for node i

$TT(i,J)$ = temperature of steam generator tube node i in contact with fluid

$TSGP(i)$ = temperature of steam generator primary fluid node i

A.I.4 Core

$$h_{core}(i)\big|_{t+\Delta t} = h_{core}(i)\big|_t + \frac{\dot{m}_{core}(t+\Delta t/2)*\Delta t}{M_{core}(i)}(h_{core}(i-1)-h_{core}(i))\big|_t$$

where: $h_{core}(i)$ = enthalpy of core node <u>i</u>

$\dot{m}_{core}$ = mass flow rate through core $(=\dot{m}_1-\dot{m}_2)$

$\dot{m}_1$ = mass flow rate through loop 1

$\dot{m}_2$ = mass flow rate through loop 2

$M_{core}(i)$ = mass of core node <u>i</u>

A.I.5 Core Inlet Plenum

$$h_{IP}\big|_{t+\Delta t} = h_{IP}\big|_t + \frac{\Delta t*\dot{m}_1(t+\Delta t/2)}{M_{IP}} * (h_{CL1}(NC\emptyset LD1)-h_{IP})\big|_t$$

where: h_{IP} = enthalpy of core inlet plenum

M_{IP} = mass of core inlet plenum

$h_{CL1}(NC\emptyset LD1)$ = enthalpy of loop 1 cold leg fluid node next to core inlet plenum

A.I.6 Core Outlet Plenum

$$h_{OP}\big|_{t+\Delta t}=h_{OP}\big|_t + \frac{\Delta t}{M_{OP}} * \Big(\dot{m}_1(t+\Delta t/2)*(h_{core}(NCORE)-h_{OP})\big|_t$$
$$+ \dot{m}_2(t+\Delta t/2)*(h_{HL2}(NHOT2)-h_{core}(NCORE))\big|_t\Big)$$

where h_{OP} = enthalpy of core outlet plenum

M_{OP} = mass of core outlet plenum

$h_{core}(NCORE)$ = enthalpy of core exit node

$h_{HL2}(NHOT2)$ = enthalpy of loop 2 hot leg node entering core outlet plenum

A.I.7 Pressurizer

$$h_{pres}\big|_t = h_{pres}\big|_{t=0} = \text{constant}$$

where: h_{pres} = enthalpy of pressurizer

A.II Primary System Mass

The flu'd mass of each primary node remains constant in time except for the pressurizer whose mass at any time t is determined as follows:

$$M_{pres}|_t = M_{pres}|_{t=0} - \sum_{k=1}^{K} \dot{m}_v(k)*\Delta t$$

where: M_{pres} = mass of pressurizer

K = number of time steps between t=0 and t

$\dot{m}_v$ = mass flow rate out pressurizer relief valve

The assumption that the mass of each primary fluid node other than the pressurizer remains constant in time necessitates varying the primary fluid nodal volumes in time since the total system mass assumed available for transport around the loop(s) is assumed constant in time (i.e., equal to the initial value). Thus the pressurizer volume (and any other non-loop flow node) must vary in time to compensate for the varying transport fluid nodal volume changes thus yielding a total constant volume.

Thus, total system volume equals a constant and is equal to the Pressurizer Volume (t) plus the volume of the fluid transport nodes (t). This is a negligible affect since the integrated fluid released from the pressurizer is very small at the time of peak system pressure. The most significant occurrences affecting the transient are:

1. Space dependent fluid energy distribution,

2. Heat transfer rate between secondary and primary, and

3. Pressurizer relief valve capacity.

A.III Steam Generator Tube Time Dependent Temperatures

For J cross-sectional nodes per steam generator primary fluid node i.

j=1, steam generator tube node is contact with secondary fluid.

$$TT(i,1)|_{t+\Delta t} = TT(i,1)|_t + \frac{\Delta t}{MT(1)C_{PT}} * \Big(UO(i)*AT(1)*(TSGS(i) - TT(i,1)) - \frac{k*AT(2)*}{\Delta X}(TT(i,1) - TT(i,2))\Big)\Big|_t$$

j=2, J-1

$$TT(i,j)|_{t+\Delta t} = TT(i,j)|_t + \frac{\Delta t*k}{MT(j)C_{PT}*\Delta X} * \Big(AT(j)*(TT(i,j-1)-TT(i,j)) + AT(j+1)*(TT(i,j+1) - TT(i,j))\Big)\Big|_t$$

j=J, steam generator tube node in contact with steam generator primary fluid

$$TT(i,J)\big|_{t+\Delta t} = TT(i,J)\big|_t + \frac{\Delta t}{MT(J)C_{PT}} * \Big(UI(i)\big|_t * AT(J+1) *$$

$$(TSGP(i) - TT(i,J))\big|_t + \frac{k*AT(J)}{\Delta X} *$$

$$(TT(i,J-1) - TT(i,J))\big|_t\Big)$$

where: TT(i,j) = temperature [illegible] am generator tube node (i,j)

MT(j) = mass of s [illegible] rator tube node (i,j)

C_{PT} = specific [illegible] city of steam generator tubes

ΔX = mesh thick[illegible]

AT(j) = outer surface area of steam generator tube node j

UI(i) = heat transfer coefficient between steam generator tubes and steam generator primary fluid for node i

A.IV Steam Generator Secondary Time Dependent Temperature

$$TSGS(i)\big|_{t+\Delta t} = TSGS(i)\big|_t + \frac{\Delta t * UO(i)\big|_t}{M_{SGS}(i) * Cp} * AT(1) * (TT(i,1) - TSGS(i))\big|_t$$

where: TSGS(i) = temperature of steam generator secondary node i

UO(i) = steam generator secondary to tube heat transfer coeffficient

$M_{SGS}(i)$ = mass of steam generator secondary fluid node i

Cp = specific heat capacity of secondary fluid

A.V Pressurizer Relief Rate Calculation

$$\dot{m}_V(t) = \dot{m}_{Vf}(P) * FRAC$$

where: FRAC = minimum of $(1.0, (P-P_S)/(P_F-P_S), (t-t_S)/DT)$

$\dot{m}_V$ = mass flow rate out valve at time t

$\dot{m}_{Vf}(P)$ = full open flow rate out valve at pressure P

P = pressure at time t

P_S = set pressure of valve opening

P_F = full open pressure for valve

t_S = time at which pressure first exceeds P_S

DT = time for full opening of valve

Thus, the ability to model valve opening as a function of either pressure difference or time is available.

A.VI Pressure Calculation

Since each nodal mass remains constant in time, with the exception of the pressurizer (and then only if flow through the pressurizer relief valve takes place), the following expression uniquely defines the system pressure at any time t.

$$V_{tot} = \sum_{\ell=1}^{L} M_\ell * V_\ell(P, h_\ell) + M_{pres}(t) * V_{pres}(P, h_{pres})$$

where: V_{tot} = total primary system volume

L = total number of primary fluid nodes

M_ℓ = mass of each primary fluid node ℓ

V_ℓ = specific volume of primary fluid node ℓ

P = pressure

h_ℓ = enthalphy of primary fluid node ℓ

M_{pres} = mass of pressurizer fluid

V_{pres} = specific volume of pressurizer fluid

h_{pres} = enthalpy of pressurizer fluid

Thus, the system pressure (assuming no flow losses or pump input) can be determined by iterating on P until the LHS and RHS of the above expression are within a desired tolerance. The program written to analyze the transient resulting from the startup of a reactor coolant pump contains a procedure for determining the pressure via this technique.

A.VII Valve Discharge Capacity During Liquid Relief Conditions

To determine the liquid water discharge capacity of the pressurizer relief valve, the Isentropic Homogeneous Expansion (IHE) model outlined in ANSI-N661 was used. This model consists of determining the critical flow rate through the valves in the following manner:

$$\dot{m} = 0.9\ K_D\ \dot{m}_C$$

where: $\dot{m}_C$ = predicted critical flow rate based on IHE considerations

K_D = discharge coefficient of valve

$\dot{m}$ = predicted critical flow rate

0.9 is ASME conservatism factor

Now, $\dot{m}_C = A_t{}^{*}\ (2g_C J(h_o - h_t))^{1/2} / ((1-X_t)V_{\ell_t} + X\ V_{g_t})$

where - the state of the fluid at the throat (t) is determined by the stagnation entropy and the critical assumed pressure.

A_t = minimum valve flow area (throat)

$g_C = 32.2\ \frac{\text{lbm-sec}^2}{\text{ft-lbf}}$

J = 778 Btu/ft-lbf

h_o = upstream stagnation enthalpy

h_t = enthalpy of the fluid at the throat

X_t = quality of fluid at the throat

V_{ℓ_t} = specific volume of saturated liquid at throat pressure

V_{gt} = specific volume of saturated vapor at throat pressure

The critical pressure and consequently the critical flow rate, is determined by choosing successive downstream pressures and corresponding specific volumes and enthalpies from steam tables with the downstream pressure yielding the maximum flow rate chosen as the critical pressure.

The discharge coefficient, K_D, was determined as specified in ASME Section VIII for steam release and consequentially used for liquid discharge. Since the discharge capacity is a strong function of the upstream stagnation enthalpy, a conservative value (equivalent to saturated liquid enthalpy at 400 psia) was used in determining the valve discharge capacity.

Figure A-1 provides the discharge capacity of the Yankee Rowe pressurizer relief valve as a function of pressure. This capacity was utilized in the transient analysis of the pump startup event.

A.VIII Heat Transfer Coefficient Calculation

A.VIII.1 Forced Convection

Forced convection heat transfer coefficient from the steam generator tubes to the primary coolant was determined using the Dittus Boelter correlation for the Nusselt number:

$Nu = 0.023\ Re^{0.8}\ Pr^{0.4}$

NOTE: Heat transfer from the heated primary fluid to the cold walled piping was conservatively assumed to be zero.

A.VIII.2 Natural Convection

In the absense of flow (i.e., the secondary fluid to steam generator tube heat transfer) the heat transfer coefficient was based on the maximum natural convection value obtained in a literature survey for vertical cylinders:

$Nu = 0.10\ (Gr\ Pr)^{1/3}$ from "Heat Transfer", J. P. Holman, Third Edition, McGraw-Hill Book Company, 1972.

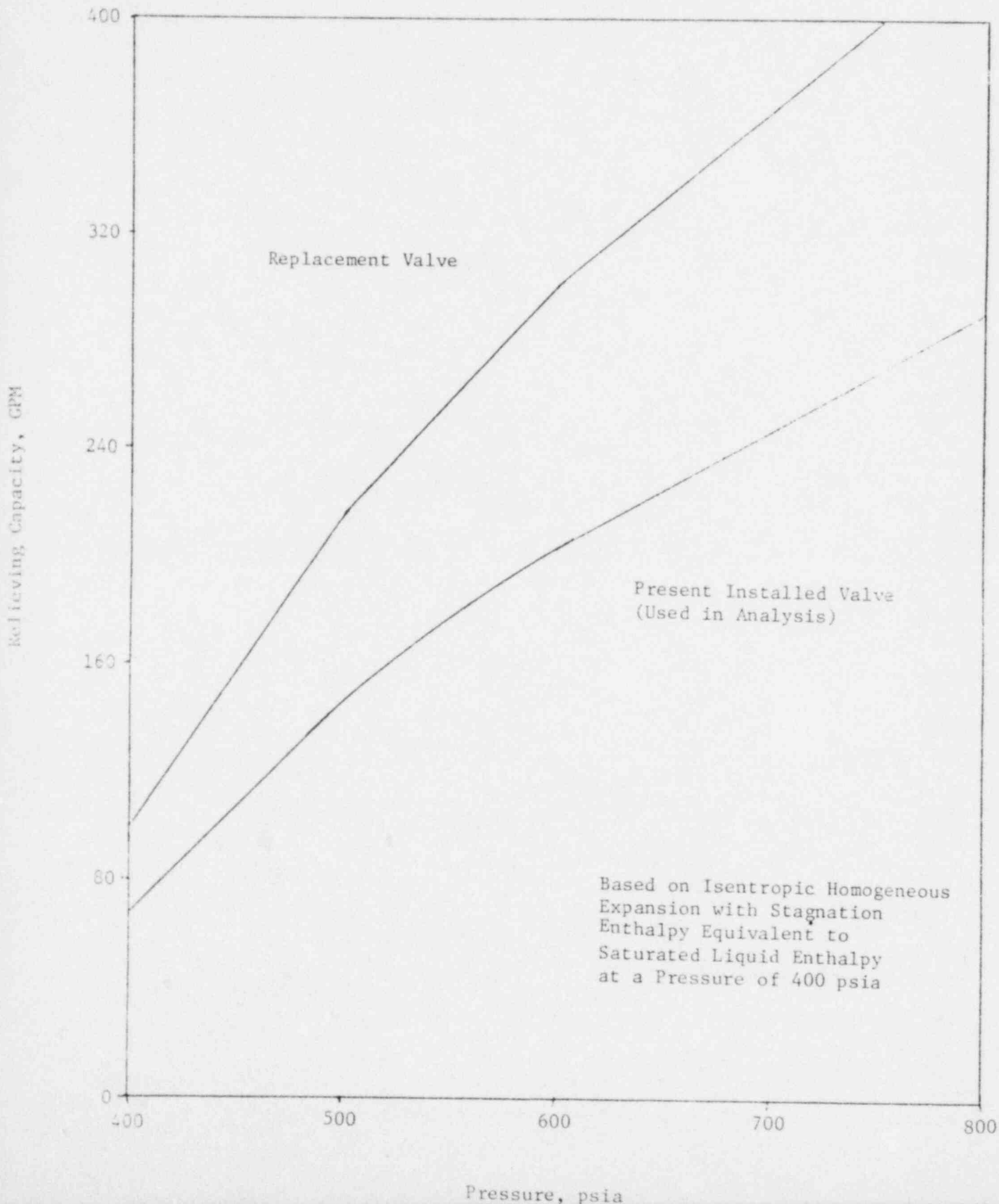
NOTE: For the initial portion of the pump start event the heat transfer coefficient from the steam generator tubes to the primary fluid was based on the maximum of the calculated natural convection value and forced convection value.

A.IX Fluid Properties

The fluid properties in the primary system (i.e., time dependent temperature and specific volume) were determined using the compressed water properties routines provided in Formulations and Iterative Procedures for the Calculation of Properties of Steam", R. B. McClintock and G. J. Silvestri, ASME, 1968.

Figure A-1

Yankee Rowe
Pressurizer Relief Valve
Liquid Relieving Capacity



B. Analysis of Flow Initiation Transient for Yankee Rowe

B.I Introduction

The transient analyzed resulting from the flow of a single reactor coolant pump during filled pressurizer conditions was initiated from the conservative conditions provided in Table B-1. These conditions were chosen as limiting following a review of plant operating procedures and allowable pressure versus temperature curves contained in the Technical Specifications. In addition to the base case conditions contained in Table B-1, a number of parametric studies were performed to determine the sensitivity of the transient to the more important of the input variables. Table B-2 provides a summary of the cases analyzed.

B.II Results

Figures B-1 and B-2 provide the system transient resulting from primary flow initiation during filled pressurizer conditions for the cases outlined in Table B-2. It is evident that a rapid pressure surge occurs during the initial portion of the transient(prior to relief valve opening) due to the transfer of energy from the initially hot steam generator tubes in the active loop to the incoming hot leg cold fluid. The energy content of the tubes drives the initial pressure surge until the pressurizer relief valve startsto open and subsequently limit the pressure excursion. After the initial rapid increase in primary fluid energy due to transport of energy from the steam generator tubes to the cold incoming primary fluid, the energy addition rate is limited by the steam generator secondary to steam generator tube heat transfer rate (which is via natural convection). Thus, this explains the requirement of cross sectional nodalization of the steam generator tubes.

The results of the parametric studies on valve discharge capacity, flow acceleration rate,and pressurizer relief valve opening characteristics show that the maximum pressure resulting from the event is not strongly dependent on these parameters (i.e., maximum pressure range from 520 psia to 538 psia).

Table B-3 provides a summary of the results for the cases analyzed.

B.III Conclusions

Based on the analysis provided above, it is concluded that operation of the pressurizer relief valve will assure that the pressure excursions resulting from the flow of a reactor coolant pump during cold filled pressurizer conditions in the presence of a differential secondary to primary temperature condition are terminated prior to exceeding vessel pressure temperature limits. This is evident even for a relief valve discharge rate less than 50 percent of that evaluated using the IHE method since the valve is not calculated to fully open even with this assumption. (Pressure limit ± 550 psig at vessel temperature of 100°F).

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TABLE B-1

Conditions Used to Determine Transient Resulting From Flow of a Reactor Coolant Pump During Cold Filled Pressurizer Conditions

Parameter	Value
Initial Pressure	400 psia
Initial Hot Leg Temperature	100 F
Initial Cold Leg Temperature	100 F
Initial Core Temperature	100 F
Initial Core Outlet Plenum Temperature	100 F
Initial Core Inlet Plenum Temperature	100 F
Initial Temperature of Primary Fluid in Steam Generator Tubes	200 F
Initial Steam Generator Tube Temperature	200 F
Initial Steam Generator Secondary Temperature	200 F
Temperature of Pressurizer	440 F
Time to reach full coolant flow consistent with one pump operation at temperature of 100 F (assumed linear)	50 sec
Full Flow Coolant Flow Rate	30900 GPM
Number of Fluid Nodes:	
1. active cold leg	5
2. inactive cold leg	1
3. core	5
4. active hot leg	5
5. inactive hot leg	1
6. active steam generator	20
7. inactive steam generator	1
Number of Cross Sectional Steam Generator Tube Nodes:	
1. active loop	5
2. inactive loop	1

TABLE B-1
(continued)

Parameter	Value
Pressurizer Relief Valve Opening Pressure	500 psia
Pressurizer Relief Valve Full Open Pressure	550 psia

NOTES:

1. Steam generator outlet plenum is combined with cold leg.
2. Steam generator inlet plenum is combined with hot leg.
3. Negligible flow exists in the inactive loops due to reverse flow check valves in the cold leg of each loop.

TABLE B-2

Summary of Cases Analyzed

Case Number	Flow Acceleration Time to Full Flow (sec)	Relief Valve Flow Rate (% of IHE Value)	Relief Valve Opening Characteristic
1	50	100	Pressure
2	50	50	Pressure
3	25	100	Pressure
4	25	100	Time

NOTES:

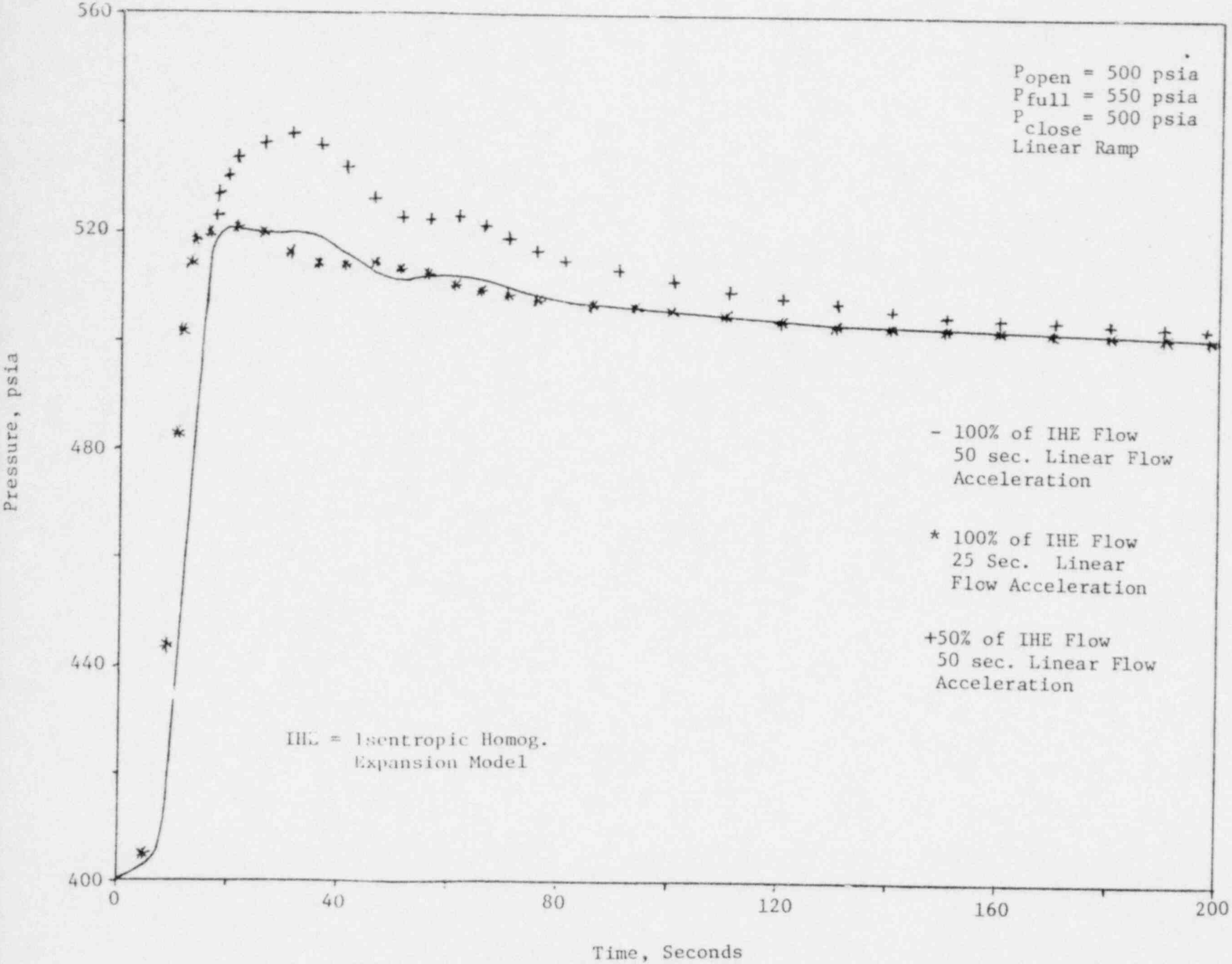
1. For the cases analyzed with pressure controlling the valve opening, the set pressure was assumed to be 500 psia and the full open pressure 550 psia. The fraction of valve throat area available for flow was assumed to follow the pressure linearly between these two points.

2. For the case analyzed with the time controlling the valve opening, a full open time of 10.0 seconds was assumed and the relation described in Section A.V used to determine the fraction of throat area available for flow.

TABLE B-3

Summary of Results

Case Number	Peak Pressure (psia)
1	520
2	538
3	521
4	532



100% and 50% of IHE, T_{cold} = 100,
T_{hot} = 200, Pressure = 400

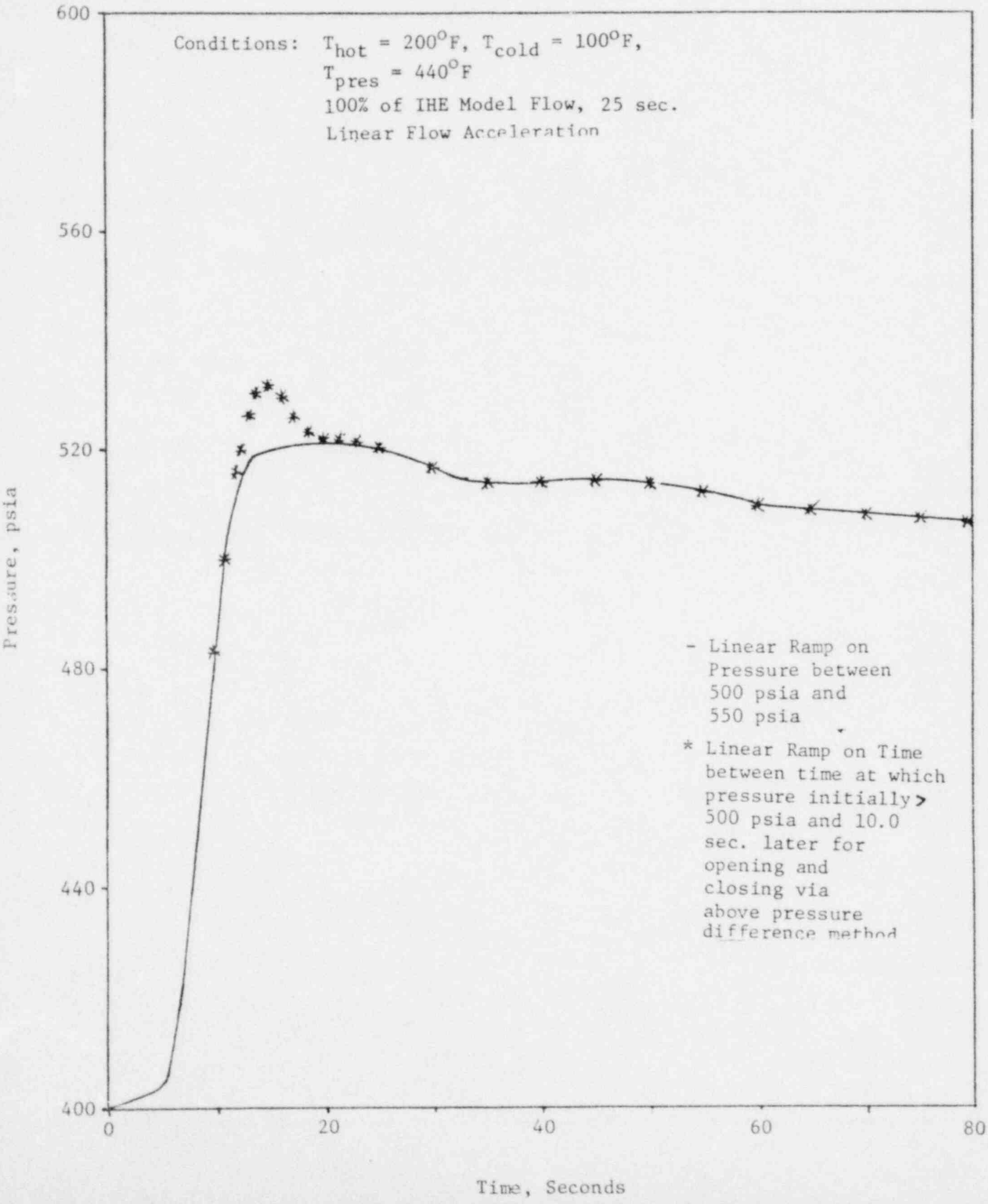
C-29

Figure B-1 Yankee Rowe Flow Initiation

Yankee Rowe Flow Initiation

Figure B-2

System Pressure versus Time for Different Valve Opening Characteristics



APPENDIX D

Schedule of Future Progress

It is estimated that the following action will be completed by the indicated dates:

a. Installation of the following design features shall be completed by June 30, 1977:

   1. Key-operated low pressure setpoint for the pressurizer solenoid-operated relief valve.

   2. Reactor vessel pressure recorder.

   3. Low temperature overpressure alarm.

b. The replacement of the existing pressurizer solenoid operated valve with a valve of higher capacity is expected to be completed by June 30, 1978.

c. A topical report covering the reactor coolant flow initiation transient model developed by YAEC will be submitted by February 28, 1977.

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