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Vogtle Project

April 21, 1986

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood
PWR Project Directorate #4
Division of PWR Licensing A
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

File: X7BC35
Log: GN-879

REF: YOUNGBLOOD TO CONWAY, 4/2/86

NRC DOCKET NUMBERS 50-424 AND 50-425
CONSTRUCTION PERMIT NUMBERS CPPR-108 AND CPPR-109
VOGTLE ELECTRIC GENERATING PLANT - UNITS 1 AND 2
REQUEST FOR ADDITIONAL INFORMATION: GENERIC LETTER 85-12

Dear Mr. Denton:

Attached for your staff's review is the requested response to the referenced letter. If your staff requires any additional information, please do not hesitate to contact me.

Sincerely,

J. A. Bailey

J. A. Bailey
Project Licensing Manager

JAB/sm
Attachment

xc: R. E. Conway
R. A. Thomas
J. E. Joiner, Esquire
B. W. Churchill, Esquire
M. A. Miller (2)
B. Jones, Esquire
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0447V

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1/1

VEGP SUPPLEMENTAL INFORMATION CONCERNING RESPONSE TO GENERIC LETTER 85-12

ITEM NUMBER

A. Determination of RCP Trip Criteria

The telephone conference information contained in the enclosure to the referenced letter for this identical item is confirmed as accurately portraying VEGP's process of selecting the RCP trip parameter.

A1. No additional comment.

A2. STAFF RESPONSE - The only additional information that is needed is instrument response time. What is the lag time associated with the long pressure lines leading from the RCS to the transmitters?

VEGP RESPONSE - VEGP RCS wide range pressure instrument sensing line is approximately 310 feet (180' inside containment, 130' outside) of 3/16" armored capillary tubing per instrument. Response time caluclated by Westinghouse using analytical methods with 410 feet of capillary tubing is 3.8 seconds. Response time for VEGP wide range RCS pressure indication, including the hydraulic lag time for sensing line, is evaluated to be less than 10 seconds which is sufficient for use as indication for RCP Trip.

A3. No further comment.

B. Potential Reactor Coolant Pump Problems

B1. The information contained in this section in the referenced letter is confirmed as accurately portraying VEGP RCP cooling water and seal injection configuration.

B2. STAFF EVALUATION - Components associated with RCP trip are identified, but there is no discussion of their location. The applicant should determine that none of the components will be affected by accident conditions such as high energy line breaks inside or outside containment, or other accident conditions which could create an adverse environment in the vicinity of the components.

There is no information pertinent to assurance that RCP trip will occur when necessary, nor is an alternate operator response provided if there is a failure to trip upon operation of control switches in the control room.

RCP operation in a voided system is not mentioned.

VEGP RESPONSE - Components associated with RCP trip consist of inseries 1E and NON-1E breakers and control circuitry. The 1E breakers are located on Level A of the Control Building and do not have any high energy lines or other sources of adverse environmental conditions located nearby. The NON-1E breakers are located in the Turbine Building.

In response to assuring that RCP trip will occur when necessary, VEGP submits the following:

METHODS TO TRIP THE RCPs FROM THE CONTROL ROOM

There are several means to trip the RCPs from the main control room. There is a non-1E 13.8 kv breaker switch in series with a 1E 13.8 kv breaker switch on the main control panel for each RCP. Tripping either of the series breakers will trip the RCP. Additionally, in the main control room at the Electrical Auxiliary Board, the Operator can trip the train A and/or B train 13.8 kv buses. Tripping either train will trip two RCPs. Using any of the control room means to trip the RCPs takes only a few seconds once the trip criteria is reached.

In additional to the control room means to trip the RCPs, the operator can also trip the RCPs from the Remote Shutdown Panels or locally in a matter of a few minutes.

RCP operation in a voided system is addressed in Attachment 1.A.

C. Operator Training and Procedures (RCP Trip)

C1. STAFF EVALUATION - References for training are provided as is a list of training topics. However, the applicant response is general and does not address the above identified points, nor does the response establish that the applicant has an understanding of the need to trip RCPs as contrasted to keep them running.

VEGP RESPONSE - Additional detail concerning operator training on RCP trip is presented in Attachment 1.B.

RCP trip timing is addressed in Attachment 1.C.

C2. STAFF EVALUATION - The licensee has presented a listing of selected procedures, most of which are stated to be based upon the WOG Guidelines. This list appears to be sufficient. It would be helpful to briefly describe the technical requirements to be met for restart.

VEGP RESPONSE - Attachment 1.D. is a copy of VEGP generic RCP restart and hot seals EOP attachments. The RCP restart attachment (Procedure Attachment A) provides these generic technical requirements, in addition to the procedure specific requirements such as RVLIS level, RCS subcooling, and PRZR level, which must be satisfied prior to restart of a RCP. The hot seals attachment (Procedure Attachment D) provides guidance for re-establishing seal injection to a RCP's seal package following the loss of seal injection flow and thermal barrier cooling. This attachment allows for the gradual cooling (1°F/minute) of the seal package to temperatures within the operating limit prior to restart of the RCP.

ATTACHMENT 1.A. -
RCP Operation in
a voided system.

RCP Restart with Void in the Upper Head of the Reactor Pressure Vessel

Under certain conditions, it is possible to postulate that steam or non-condensible voids will collect in the upper head of the reactor pressure vessel. Depressurization of the reactor coolant system during natural circulation cooldown of the plant may generate a steam bubble in the upper head region of the reactor vessel. This bubble could rapidly condense during RCP startup, drawing liquid from the pressurizer and reducing reactor coolant subcooling. If pressurizer inventory is not sufficient, level may decrease offspan. The normal indication of coolant inventory would not be available if this occurred and pressurizer heaters would not be available for pressure control. In addition, local flashing of reactor coolant could result in erratic system response. These conditions would make plant control more difficult and may confuse the operator if such behavior was unexpected.

An evaluation of the reactor coolant pump restart has been performed to assess the potential for coolant flashing and loss of pressurizer pressure control during RCP startup. The RCS pressure response to the collapse of an upper head void was calculated by Westinghouse modelling the pressurizer as a single, stratified node with thermodynamic equilibrium between phases to determine pressurizer inventory and RCS temperature requirements. Water was assumed to be displaced from the pressurizer to accommodate an instantaneous collapse of a steam bubble occupying 1) the entire upper head region and 2) the entire upper head region plus the upper plenum volume above the hot leg nozzles.

Figure 3 presents the minimum indicated pressurizer level required for Vogtle before starting an RCP to ensure that an indicated level will remain after pump restart with the upper head region voided. The level necessary to maintain pressurizer heaters operational is also shown in Figure 1. The results in Figure 1 include all an allowance for the pressurizer level instrument uncertainty. The effects of pressure on the calibration of the level indication are also included in these results. As demonstrated, pressurizer level would remain on span for all primary pressures following

RCP restart with an indicated level greater than 57 percent. An initial level greater than 67 percent would be required to ensure that the pressurizer heaters would remain available.

The minimum RCS subcooling required to maintain subcooling following RCP restart, is presented in Figure 2. The required subcooling is consistent with the initial pressurizer level required to keep the level on span following RCP restart. From Figure 2 it is seen that the primary pressure will remain above saturation for any primary pressure with subcooling greater than 22°F.

Starting an RCP results in a primary pressure decrease due to the collapse of the upper head void and filling of the upper head region with primary coolant. Figure 3 shows the minimum RCS pressure following RCP restart.

If a noncondensible void exists in the Reactor Pressure Vessel upper head and the RCS is depressurized, the void will continue to expand. Whether or not the RCPs are operating, this noncondensible gas can be introduced into the loops. If the upper head bubble is small enough to stay above the level of the hot leg nozzles and the RCPs are operating, the gas should not be entrained in the coolant and distributed around the RCS. If the upper head bubble grows large enough to enter the hot leg piping, the coolant mixture will be circulated and core heat removal will continue if the RCPs remain in operation. Even if only one RCP remains in operation, flows through idle loops must fall below approximately 2 feet/second to allow the noncondensible gas to collect in the SG tubes.

If the noncondensible gas enters the pressurizer, it will undoubtedly reduce the effectiveness of the pressurizer spray in controlling RCS pressure by making the pressurizer vapor space behave like a "hard" bubble rather than a steam bubble. If RCP failure or trip occurs after the noncondensible gas enters the RCS, primary-to-secondary heat transfer may be degraded if natural circulation steam velocities become low enough to fail to sweep the noncondensible gas through the SG tubes. Thus, if the presence of a large noncondensible bubble is known or suspected, RCP restart is delayed until venting of the bubble can be completed unless it is required to provide forced coolant flow for effective core heat removal.

Post Accident RCP Restart Criteria

Prior to restarting an RCP in the EOPs, it must be ensured that there is no SBLOCA concern which would require RCP trip. In the Optimal Recovery Procedures (ORPs), RCP restart criteria have been established that ensure restart of the RCPs without further aggravating any inventory loss through a postulated break. If RCS subcooling exists and pressurizer level is on span, than there is not a SBLOCA concern of excessive inventory loss and RCP operation is permitted. These two criteria (subcooling and pressurizer level) are found in the RCP restart steps when SI is in service. They are also implicit in the RCP start steps when SI is terminated, since subcooling and pressurizer level are part of the SI termination and reinitiation criteria. It should be noted that if a steam void in the upper head of the Reactor Pressure Vessel is possible, then the more stringent criteria discussed in our response to RCP Restart with Void in the Upper Head of the Reactor Pressure Vessel are imposed for RCP restart. In 19133-1, ECA-3.3 SGTR WITHOUT PRESSURIZER PRESSURE CONTROL, only subcooling is required since pressurizer pressure control is not available to restore and control pressurizer level and starting an RCP is one means of restoring pressure control to recover from the tube rupture. In the Function Restoration Procedures (FRPs), there are two places where RCP restart is desirable even if subcooling and pressurizer level requirements are not met. In 19221-1, FR-C.1 RESPONSE TO INADEQUATE CORE COOLING, RCPs are restarted if core exit thermocouples are above 1200°F and secondary depressurization does not alleviate the symptoms of inadequate core cooling. In this case the RCPs are used to provide a means of cooling the core since other mechanisms have not been successful. In 19241-1, FR-P.1 RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, an RCP is started if SI cannot be terminated to help mix the SI flow with other RCS fluid in an attempt to protect the reactor vessel from cold SI water.

Starting an RCP in a 2-Phase RCS

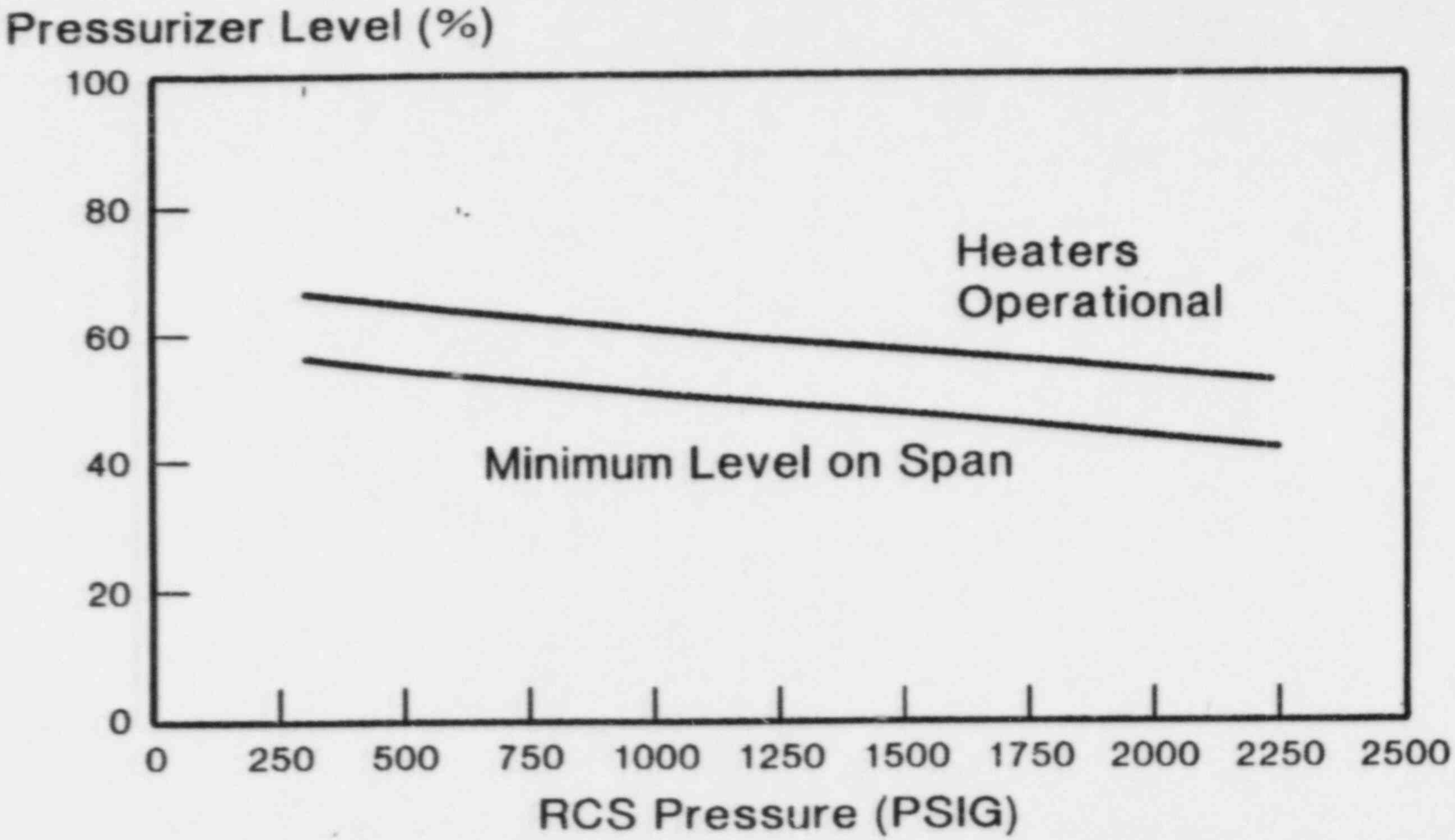
If all RCPs are tripped during an accident and RCS voiding occurs to the point where virtually empty hot legs, SG tubes and cold legs exist, fluid in the crossover leg pipe will undoubtedly exist in a 2-phase condition. Steam will flow along the upper portion of the pipe with water collecting in the lower portion of the pipe. This water may be generated by condensation of steam in the SG and/or from RCP seal injection flow. Steam velocities in the piping will not be sufficient to entrain the water and carry it to the core.

If restart of an RCP is made under these conditions, the pump will experience a relatively short period where there will be very high density variations in the fluid entering the pump suction. This fluid would be homogenized rapidly but the starting conditions may approximate slug flow. While full scale tests under conditions like these have not been performed, tests in LOFT *(according to Westinghouse) have indicated only minor effects on the pumps, with no mechanical damage. Pump current changes, believed to be consistent with fluid density changes, were the predominant effect.

The only places in the ERGs where an RCP is restarted under these conditions are in 19221-1, FR-C.1 RESPONSE TO INADEQUATE CORE COOLING, and 19241-1, FR-P.1 RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION.

* P.D. Bayless, Experiment Results Report for LOFT Nuclear Experiments L3-5, L3-6 and L8-1, EGG-LOFT-5471, July 1981.

Figure 1. MINIMUM INDICATED PRESSURIZER LEVEL FOR RCP RESTART



ATTACHMENT 1.A.

Figure 2. MINIMUM RCS SUBCOOLING FOR RCP RESTART

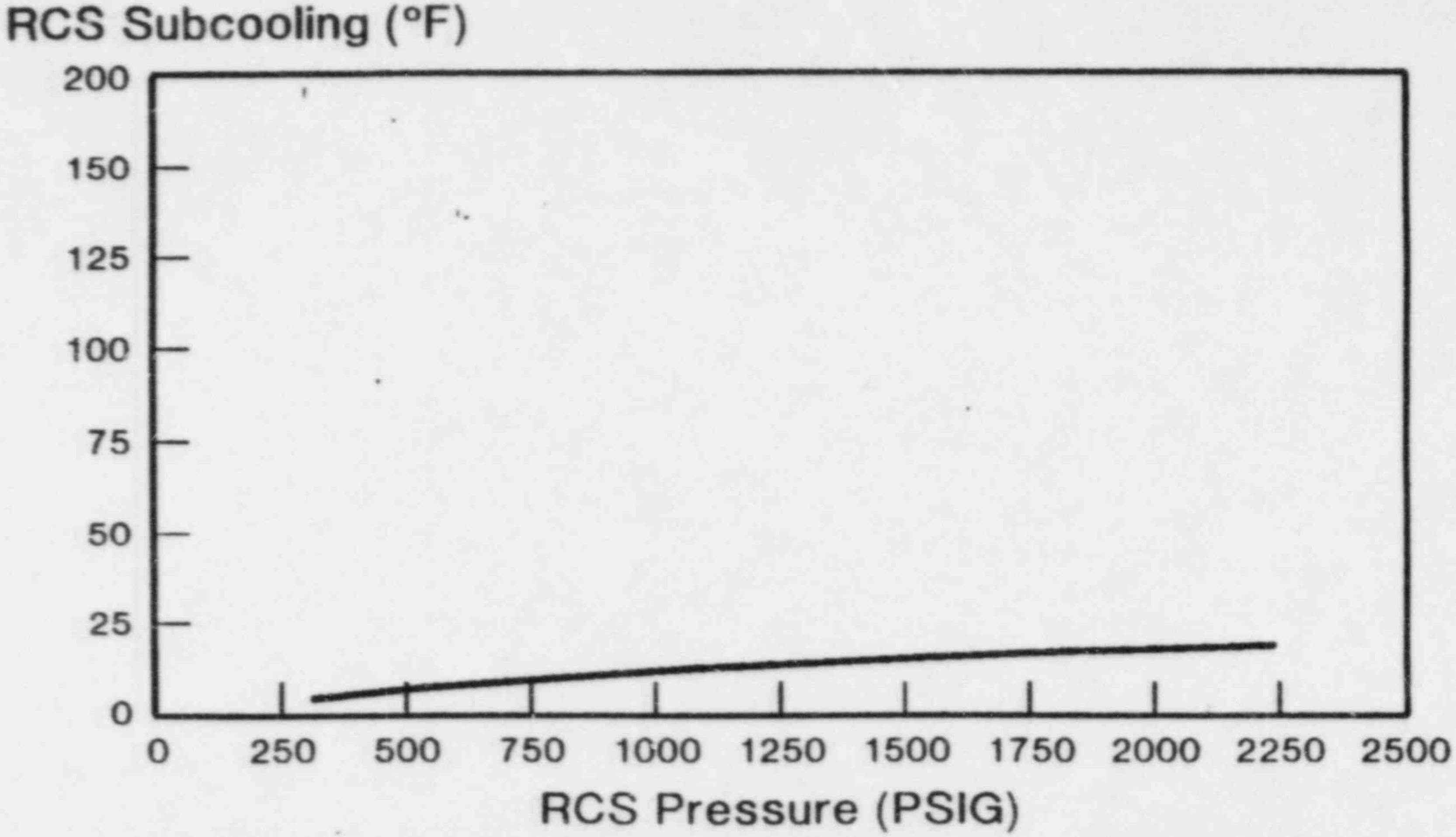
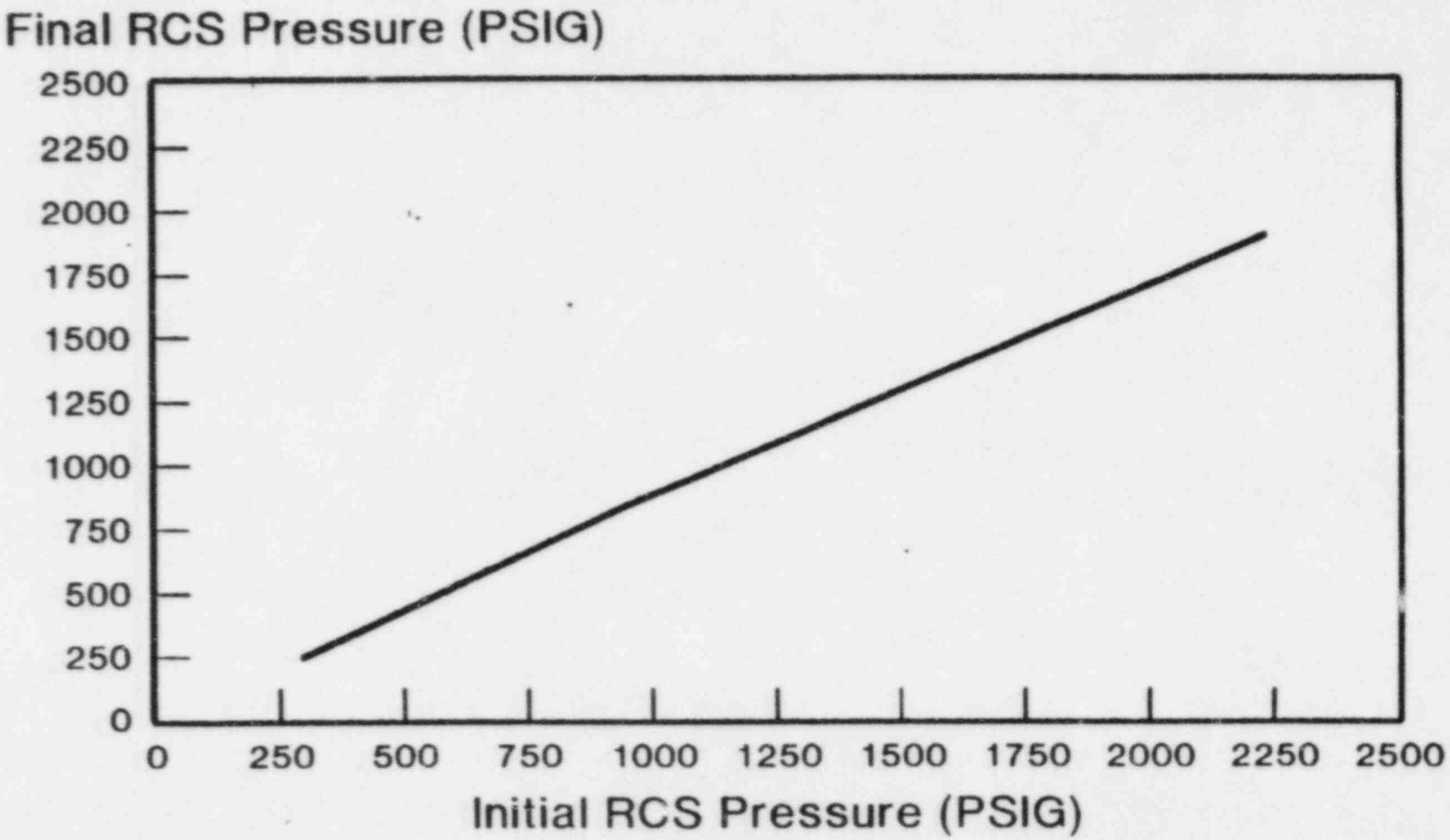


Figure 3. MINIMUM RCS PRESSURE FOLLOWING RCP RESTART



Chapter 1. LOSS OF COOLANT ACCIDENT

After reading this chapter you will be able to:

1. Describe the physical basis for establishing equilibrium temperature and pressure in the RCS.
2. Describe the effect of various size breaks on the primary system with respect to temperatures and pressures.
3. Describe the four stages of post-loca core conditions and evaluate potential core damage.
4. Explain why a small break LOCA is a concern for propagation of cracks in the reactor vessel.
5. Describe how a LOCA is initially detected and the proper procedure entered.
6. State the RCP trip criteria and tell why it is especially important in the case of a small break LOCA.
7. Explain the purpose for the hot leg recirculation mode of post LOCA core cooling.
8. Describe the basic decision criteria to determine the proper emergency classifications for different sized LOCA events.

The large break loss of coolant accident is presented in the Vogtle FSAR as a design basis accident. That is, it represents one of the most severe challenges to the integrity of the core of any postulated accident. Extensive calculations are run to assure the NRC and the people of this area that in no case will hazardous amounts of radiation be released from the plant. Although the designers of the plant systems have made many provisions to handle this event automatically the operator must be prepared to verify Emergency Core Cooling System proper response. More importantly, the action of the operator will be required in the less hazardous, but more likely, small break LOCAs to bring the plant down in an orderly fashion. Once the repairs are made, then the plant can go back on line producing electricity. It is not really the operator's concern as to the size and location of the break. His main decisions are focussed around identifying the best way to recover control of the plant. Vogtle emergency procedures 19010,11,12,13,14 were developed from Westinghouse Owner's Group Guidelines E-1, ES-1.1, ES-1.2, ES-1.3, and ES-1.4. Additionally, procedure 19000(E-0) is used to verify proper automatic response of the ECCS and to select the 19010 procedure for the LOCA symptoms.

1. Mass and Energy Relationships

To understand the response of the plant to a loss of coolant accident the operator should have a solid basis in relating key plant parameters such as temperature and pressure to their physical basis. The phenomena observed in Safety Analysis transients are useful for looking at extreme conditions but more realistic transients may be significantly different. Before discussing several important transient scenarios, the reactor's mass and energy relationship following a loss of coolant accident will be reviewed.

Reactor coolant system pressure after the LOCA is indicative of the balance of mass flow rates. See figure 1. That is, a relatively stable pressure will occur when the liquid phase break flow equals the safety injection flow into the system. If the break is large enough then the break flow will eventually become a two phase mixture, perhaps alternating with pure steam flow. In this case, RCS pressure will oscillate around a value where the SI flow matches the break flow on a time-averaged, or integral, basis. If pressure is rising or falling then there is a net gain or loss of mass to the RCS. Also, to the extent that the operator can control the flow out of or into the system he can control the pressure of the RCS.

ECCS pump operation of pushing the water from the RWST into the reactor coolant system is described by the pump characteristic curve such as in figure 2. This particular curve is a composite representation for several different combinations of high head charging pumps and the intermediate head safety injection pumps. Similarly, the resistance to the escape of the fluid by a given size break/leak can be drawn on the same flow vs. pressure curve. Initially, when system pressure is at 2235, the pressure will force a maximum amount of reactor coolant fluid out through the break. This loss of fluid will cause system pressure to decrease and as pressure decreases the amount of flow drops off. On the other hand, it will be hard for the ECCS pumps to force fluid into a system that is at an initially high value. As the system pressure decreases, the pumps can force more and more fluid into the system. Eventually the break flow drops and the pump flow rises until the two values are equal to each other. Two leak flows are represented on figure 2. Intersections of the leak resistance curve and the particular combination of operating ECCS pumps define the equilibrium pressure for the RCS after a LOCA. For example with leak rate 1 which is fairly small, if two charging pumps are running the pressure will equilibrate at about normal operating pressure, 2235 psig. The big difference is that 500 gpm is flowing into and out of the RCS to places unknown. If one of the charging pumps is turned off or had never started then one would expect a pressure of 1700 psig with a reduced break flow of 400 gpm. For a break about twice as large with all charging and SI pumps on, equilibrium will be at 1300 psig and 1150 gpm. Turning off a CCP will drop pressure by 100 psig and break flow by 150 gpm.

The second indicator of stability as shown in figure 3 is core temperature. If a LOCA occurs at power a great amount of decay heat energy will be present as well as the stored heat in the metal and fluid. When the pressure drops quickly, the fluid reaches saturation conditions and flashes to steam. This serves to reduce the rate at which pressure drops due to the expanding steam bubble. Temperature stability is reached when the heat produced by the core is equal to the heat being removed from the system. For large breaks this heat removal is accomplished by the cool ECCS water from the RWST being injected into the core and the heated water being removed from the system via the break. For small breaks this may not represent a significant amount of heat flow and the heat removal by the steam generator is of primary importance. There are two methods for the steam generator to remove heat from the reactor coolant. In both cases, natural circulation is used since the reactor coolant pump trip criteria usually forces this condition. The first is a continuous single or two-phase flow through the core and the RCS piping. If the break is large enough to cause draining of the RCS, then heat will be removed by condensation of steam in the U-tubes with the condensate falling back down the hot legs. This second method is called reflux cooling and is almost as efficient as two-phase natural circulation. If the heat removal rate of the steam generator and the rate caused by temperature change in the SI fluid exceeds that of the decay heat input the plant will cooldown which in certain cases may be excessive.

The temperature stability affects the pressure stability in several ways. If the system is still cooling down the shrink will represent another volumetric outflow from the mass balance and further decrease pressure. Temperature changes are also important where generation of steam in the primary system is controlling the pressure. RCS pressure will stay above the steam generator pressure as long as heat must be removed from the system through the steam generator. The delta T across the tubes determines the heat transfer rate. Since both systems will be at saturation conditions for this case, the pressure in the RCS with respect to that of the steam generator is indicative of the heat transfer rate. This is seen in the case of reflux cooling as a higher delta in pressure. Since reflux cooling has a lower heat transfer co-efficient, a higher temperature difference is required to maintain the heat transfer rate. This causes the primary pressure to remain that much higher than the secondary.

2. Plant Response to Breaks of Various Sizes.

The phenomena described for each category of break size is more important to the operator than knowing what size of break causes what type of plant response. A standard reference analyzing loss of coolant accidents uses six different scenarios to illustrate plant response to primary system breaks. The first category is calculated to be a break smaller than 3/8 in. and is referred to as a leak rather than a LOCA. The normal charging system can maintain reactor coolant inventory so that RCS pressure and pressurizer level do not decrease. Slight system depressurization may occur but no automatic trip or safety injection signal would be generated. If the leak is within Tech Spec limits or it can be isolated, the plant could remain in power operation. Otherwise the plant would be shutdown using normal plant procedures.

The second and third categories use a break size greater than 3/8 in but less than 1 in. for an equivalent diameter. Category 2 was developed assuming minimum safety injection and some key plant response curves are shown in figure 4. For these break sizes the normal makeup system cannot maintain level and pressure. The RCS will depressurize and an automatic reactor trip and safety injection signal will be generated. Provided that the level is maintained in the steam generators, the RCS will reach an equilibrium pressure which corresponds to the pressure at which the liquid phase break flow equals the high pressure pumped safety injection flow. Early in the transient a loss of subcooled liquid in the RCS occurs which results in a moderate depressurization to the pressure which corresponds to the saturation pressure in the core and hot legs. At this point the upper head, upper plenum, hot legs, and core begin to experience some slight voiding, but more than enough liquid flow exists to keep it covered and cooled. During this period of voiding however, RCS depressurization occurs at a much slower rate than during the time when the entire system was subcooled. Eventually the RCS depressurizes to the point of reactor trip signal. Immediately following reactor trip, the RCS rapidly depressurizes, since only a fraction of the heat previous to the trip is now being transferred to the primary fluid. This causes the SI to occur and within a few minutes an equilibrium pressure is established which is above the steam generator pressure. Core heat is removed through the steam generators by continuous single or two-phase natural circulation. Abnormal indications should be present in the containment for this size of break although the response will be slower and milder than for larger size breaks. The curves shown in fig. 4 show a 45 lb/sec(340gpm) flow equilibrium being established for the 1 in. break. The pressurizer emptied in about 10 minutes and never refilled. At 14 hours into the transient hot leg flow became subcooled indicating a gradual cooling trend for this minimum SI flow case. In order to depressurize to a cold shutdown condition it is necessary to cool the primary fluid further while stepping down the SI flowrate.

The category 3 LOCA is the same size as the category 2 but considers the effect of a maximum response from the ECCS pumps. This causes the RCS to repressurize above normal system pressure. The response for a half-inch diameter break is shown in figure 5. The system behavior is identical to the category 2 response until the time when the RCS depressurizes to the SI setpoint. At that time when SI flow reaches the RCS, the RCS has lost some liquid inventory, the pressurizer level has dropped, and some voiding in the core, upper plenum and upper head exists. As soon as the SI flow enters the system, the RCS begins to repressurize, and system void fraction rapidly decreases. The pressurizer begins to refill and, if the pressurizer completely refills, the RCS will be water solid and the repressurization rate will increase dramatically. For the case analyzed for the W Owners Group Guidelines, the SI termination criteria were assumed to be met at approximately 78 minutes when the RCS hot leg subcooling reaches 50°F. At this time pressurizer level is in span, total feedwater flow is greater than required and RCS pressure is increasing. Further recovery of the plant would be performed using the SI termination procedure 19011. If SI were not terminated no core uncovery would occur and the RCS would remain water solid in a stable condition with the pressurizer PORV's cycling open and closed.

The fourth category of LOCA's encompasses break sizes from 1 in of equivalent diameter to 1 sq. ft. The pressure response to breaks of 2 and 3 in. equivalent diameter are shown on figure 5. For breaks of 1 to 2 in size the RCS will rapidly depressurize early in the transient with the automatic actuation of the protective systems. During the early stages, when the system is still full of two-phase liquid, the break flow, which will also be mostly liquid, is not capable of removing all the decay heat. Therefore, RCS depressurization will temporarily hang up above the steam generator safety valve set pressure (assuming no dumps or ARV's). The RCS pressure stays at this level in order to provide a temperature difference from primary to secondary so that core heat may be removed by the steam generator. At this energy-balance controlled pressure, however, pumped SI flow is less than break flow and there is a net loss of mass in the RCS. Voiding throughout the primary side occurs and eventually the RCS begins to drain, starting from the top of the steam generator tubes. As draining continues the heat removal to the steam generators changes from two-phase natural circulation to reflux methods. As soon as the break flow becomes all steam flow for breaks in this range of size, steam generated in the core can exit out the break and further depressurization occurs. Safety injection flow increases to greater than the break flow, and there is no longer a net loss of mass from the RCS. No further core uncovery will occur beyond that necessary to create a vent path for core steam which applies only to cold leg breaks. Once the break flow has become all steam flow, the volume removed through the break is greater, so that the RCS depressurizes. Because of the RCS depressurization the safety injection flow increases and results in additional cooling.

The last category of LOCA's is the design basis event and is a break from 1 sq. ft to the Double Ended Cold Leg Guillotine (DECLG) shear break. The sequence of events for the large break LOCA is shown on figure 8. This transient has four characteristic stages: blowdown, refill, reflood, and long-term recirculation. Blowdown starts with the assumed initiation of the LOCA and ends when the RCS pressure falls to essentially that of the containment atmosphere. Refill starts at the end of blowdown and ends when the addition of ECCS water fills the bottom of the reactor vessel and reaches the elevation of the bottom of the fuel rods. Reflood is defined as the time from the end of refill until the reactor vessel has been filled with water to the extent that core temperature rise has been terminated and core temperatures subsequently have been reduced to their long term steady state levels associated with dissipating decay heat. These time divisions are established mainly for analytical convenience but are widely used. As contrasted with the large break, the blowdown phase of the small break occurs over a longer period and does not result in reduction of the effective water level in the reactor vessel below the bottom of the core. Thus, for a small break LOCA there are only three characteristic stages, i.e. a gradual blowdown in which the decrease in water level is checked before the bottom of the core is uncovered, reflood, and long-term recirculation.

The pressure response of the plant to the large break LOCA is also shown on figure 5 although it is barely noticeable. The primary pressure drops rapidly from the initial value to a low value of 40-50 psig by the end of blowdown. This will take approximately 30 seconds. Accumulator flow is initiated approximately 16 seconds after the break occurs. The containment pressure transient is shown in figure 9. The containment pressure reaches a peak value early in the transient during the blowdown phase. During reflood, the ECCS cooling water from SI accumulators and safety injection pumps enters the top of the reactor vessel downcomer annulus and starts to fill the reactor vessel lower plenum, which is filled after 45 seconds. Figure 10 shows the fluid temperatures for the 7.5 foot and the 5.75 foot level of the core following a DECLG LOCA, the temperature peaks at about 1300°F at 100 seconds but is reduced to relatively normal levels by 400 seconds. The time period is short enough to prevent violating ECCS design criteria.

3. Pressurized Thermal Shock Concerns

The safety significance of pressurized thermal shock (PTS) events is the potential for propagation of a flaw in and potentially through the reactor vessel wall. Depending on the size and location of the penetration, through-wall vessel cracking could reduce the effectiveness of the core cooling system. Even without a degradation in core cooling, vessel repair and requalification, following the development of even a partial penetration crack, would require an extensive shutdown of the plant. Furthermore, any event suspected of causing pressurized thermal shock may result in shutdown to demonstrate, by inspection, that no significant RPV flaws had been created. The PTS phenomena will be discussed in detail in volume VII.

The severity of a PTS event is dependent upon the temperature and pressure during an overcooling transient and the degree of embrittlement of the reactor vessel.

The basic mechanisms for rapidly cooling the inner surface of the reactor vessel of a PWR include (1) depressurization of the primary system, (2) injection of cold fluid, and (3) rapid removal of energy through the steam generators. Of all types of accident scenarios considered large break LOCA events produce the greatest thermal shock but also very low system pressures. Due to the size of the break, repressurization is precluded. The potential for a flaw to penetrate through the vessel wall for large break LOCA events is therefore extremely low.

Smalll break LOCA events, including stuck-open pressurizer relief and safety valves as well as small breaks in the primary coolant system boundary, cool the system less severely than large break events, but maintain a higher system pressure. Small break LOCA transients may thermally shock the reactor vessel by primary system depressurization and injection of cold fluid from the high pressure centrifugal charging pumps, CCPs. The severity of the thermal shock is dependent on the degree of mixing that is attained with warmer fluids. Following an initial system depressurization and thermal shock, the charging pumps can repressurize the reactor coolant system; the degree of repressurization is dependent on the LOCA break size as already discussed and whether it can be isolated.

To date, no known pressurized thermal shock event has caused preexisting flaws to propagate through a reactor pressure vessel. However, transients have occurred that demonstrate the potential for overcooling at pressure. One of the small break LOCA transients with PTS potential occurred in 1980 at the Crystal River Unit 3 plant. The transient began when a power operated relief valve (PORV) was opened inadvertently. The resulting transient caused a decrease in RCS temperature of about 90 degrees F in 30 minutes (approximately 200 F/hr) with a system pressure of about 2400 psig. The transient was initiated when an electrical short in a DC power supply caused a pressurizer PORV to open, a loss of most control room instrumentation, and the generation of erroneous signals to the plant's Integrated Control System(ICS). The ICS caused a reduction in feedwater flow and a withdrawal of control rods. RCS pressure initially increased, tripping the reactor on high pressure, and then decreased as coolant discharged through the open PORV. The high pressure injection pumps started at 1500 psig and repressurized the RCS to about 2400 psig. The PORV block valve was closed, but flow out of the RCS continued through the pressurizer safety valves. After approximately 30 minutes, the high pressure injection pumps were throttled back, but RCS pressure was maintained at about 2300 psig for the next one and a half hours. The RCS temperature decreased by about 90 degrees F in the first 30 minutes and was thereafter brought to cold shutdown conditions by normal operating procedures since the DC power had been restored.

4. EOP Reactor Coolant Pump TRIP PARAMETER

Analysis has shown that for certain cold-leg small break LOCAs, RCP trips within different time frames in the event produces different effects on final peak clad temperature. The lowest final clad temperatures were the result of allowing the RCPs to continue to run throughout the transient. However the analysis must assume a loss of off-site power as a possibility anytime during the accident, because continued RCP operation cannot be guaranteed. Therefore, studies were conducted to determine the best and worst time frame for an RCP trip in terms of final peak clad temperature. The analysis was set up with a 3 inch diameter cold leg break and all RCPs in operation. The analysis was run with the RCP trip occurrence at different time intervals after the initiation of the transient (see Fig. 11). The results show that if the RCP trips occur early in the transient, peak clad temperatures are lower. The bounding time frame appears to be about 10 minutes. If the RCPs are tripped later in this accident than about 10 minutes, the peak clad temperatures are higher. Figure 11 shows the results of the study for a 3 loop plant. Results for a 4 loop plant are similar.

To understand the reasons for this, we have to understand what is occurring in the RCS during the transient. RCS inventory is lost through the break. When RCS inventory is depleted to the level of the break, a phase state change occurs and the Quality of the break exit flow approaches 1 if the RCPs are not running. If the RCPs are running, water as well as steam will be pumped out the hole. Fluid inventory loss causes a more rapid mass inventory loss. The core is partially uncovered due to the loss of inventory upon subsequent RCP trip. Clad temperatures increase but stabilize at a higher temperature (but less than 2200°F) due to the steam-water mix flow rate if the RCPs continue to run. If the RCPs were tripped after break is uncovered, this higher level of clad temperatures would cause a higher peak clad temperature after the pumps are tripped due to the additional inventory loss caused by the forced circulation. Conclusions drawn from this analysis show that when symptoms indicate that the accident is a SBLOCA, the RCPs should be tripped early to minimize core uncovery.

During an SGTR or other non-LOCA event it is desirable to maintain a responsive heat sink. For the SGTR event, a responsive heat sink is necessary to allow rapid cooldown of the RCS prior to RCS depressurization down to the ruptured SG pressure. Natural circulation is a slow process and less responsive than forced circulation. It is desirable to control RCS pressure using normal pressurizer spray, rather than PRZR PORVs. For these reasons, when conditions indicate that the transient is an SGTR or other non-LOCA event, the RCPs should not be tripped.

It is often difficult to distinguish between a small steam break or small SGTR and a SBLOCA during the initial stages of the event. Radiation monitors are the primary means of early differentiation. During BOL conditions, with a clean RCS, the radiation monitors may not be the first indication of an SGTR.

In order to provide the operator with a definitive RCP Trip Criteria which could be applied unambiguously for a specified set of conditions, Westinghouse proposed three different RCP Trip Criteria for evaluation.

1. RCS Pressure - Indication of saturation pressure being reached at the top of the SG tubes, including instrument uncertainties, based on the most restrictive conditions.

2. RCS Subcooling - Indication of void formation in the core, including instrument uncertainties as a function of RCS pressure.

3. RCS/SG Delta Pressure - Indication of saturation pressure being reached at the top of the SG tubes, including instrument uncertainties and based on actual highest SG pressure.

All three trip criteria are indicators of the beginning of void formation in the core. Westinghouse analyzed non-LOCA events at different plants to determine the minimum values for the three RCP Trip Criteria parameters for these non-LOCA events. VEGP's non-LOCA RCS Pressure Parameter was calculated to be 1738 psig.; the RCS Pressure Parameter for SBLOCA events is calculated at 1373 psig. This pressure of 1373 is substantially less than that of non-LOCA events and therefore provides adequate discrimination between a LOCA and non-LOCA event.

The final Vogtle RCP Trip Criteria using the RCP Trip Parameter is:

1. ECCS Pumps - AT LEAST ONE RUNNING

   AND

2. RCS Pressure - LESS THAN 1373 PSIG

Applicability of the RCP TRIP PARAMETER

The conditions where the RCP Trip Parameter applies are:

1. Following reactor trip and safety injection actuation initiated from power operation.
2. During recovery actions or at hot standby conditions, before initiation of an operator controlled RCS cooldown.

The conditions where the RCP Trip Parameter does not apply are:

1. Following a safety injection actuation initiated from cold shutdown, hot shutdown or startup conditions. Refueling is not considered in the context of applicability.
2. During recovery actions or at hot standby conditions, following initiation of an operator controlled RCS cooldown.
3. Following any RCP restart specified in EOP instructions.
4. When the EOPs specifically state that the RCP Trip parameter does not apply.

5. Hot Leg Recirculation

Hot leg recirculation is implemented to terminate boiling in the core and to prevent boron precipitation in the core. Following a large cold leg break in the RCS, conservative analyses have shown that the boric acid concentration limit established by the NRC would be exceeded if cold leg recirculation is maintained for an extended period. The analysis considers the increase in boric acid concentration in the reactor vessel during the long-term cooling phase of a LOCA. The calculation of boric acid concentration in the reactor vessel considers a cold leg break of the reactor coolant system in which steam is generated in the core from decay heat while the boron associated with the boric acid solution is completely separated from the steam and remains in the effective vessel volume. The cold leg safety injection flow is not effective in counteracting this boiloff from the core since for larger breaks the downcomer level is low and the injection flow is primarily refilling the downcomer as opposed to the core, and no flushingof the core occurs. If the plant is transferred from cold leg to hot leg recirculation prior to the time the boric acid concentration limit is reached in the reactor vessel, the hot leg safety injection flow will dilute the vessel boron concentration by passing relatively dilute boron solution from the hot leg through the vessel to the cold leg break location and will terminate boiloff from the core. This will prevent boron precipitation in the core along with any resultant plateout on the fuel cladding which could reduce heat transfer from the fuel to the reactor coolant.

For a large hot leg break in the RCS, the safety injection flow delivered to the cold legs during cold leg recirculation will flow through the core and spill to the containment sump via the hot leg break. With the core being flushed there would be no boron buildup problem. After transfer to hot leg recirculation, the cold safety injection flow enters the core and absorbs decay heat energy. This will prevent boiloff from the core. Charging flow will continue to be provided to the RCS cold legs and will also help preclude any boron concentration buildup in the reactor vessel for hot leg breaks.

6. Emergency Plan Implementation

The LOCA event may force the plant into any of the emergency classes depending on severity. The plant tech specs place the following limits on loss of RCS fluid.

- No pressure boundary leakage
- 1 GPM unidentifial leakage
- 1 GPM primary-to-secondary leakage through all S/G's and 500 gal per day through any one S/G.
- 10 GPM identified leakage
- Table specified values for valve leakage

Exceeding these limits places the plant in the Notification of Unusual Event (NUE) category. Higher levels of Emergency Plan classes are determined by whether the three radiation release barriers are breached or challenged. The RCS barrier is assumed breached or challenged if any of the following conditions exist:

- Containment air particulate, iodine, or radiogas monitors (RE-2526 A, B, and C) increase very rapidly to off-scale high.
- Containment vent effluent APD, iodine, or radgas (RE-2526 A, B and CO at the high setpoint
- Core Cooling CSF status tree red or orange; i.e. high thermocouples, and low RVLIS indicators
- Integrity CSFST red or orange (PTS challenge)
- Tube rupture indications

The likely high radiation alarm following a significant LOCA will elevate the emergency to the alert status. The steam pressure in containment following one of the larger breaks will challenge the integrity of the containment. A red or orange status on the containment CSFST will result when containment pressure exceeds 15 psig or the containment emergency sump level exceeds 80%. The challenge to this second barrier will elevate the plant to a site area class emergency.

If indications of a breakdown in fuel clad integrity exist, such as a high alarm from the Gross Failed Fuel Detector, then this barrier must be assumed breached or challenged. If 3 or the 3 barriers are not verifiably intact then a General class emergency must be declared.

Pressure

Equilibrium Conditions

following Loss-of-Coolant Accident

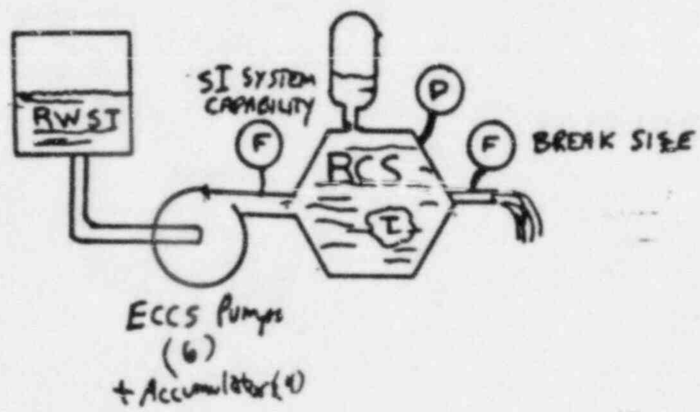


FIG. 1

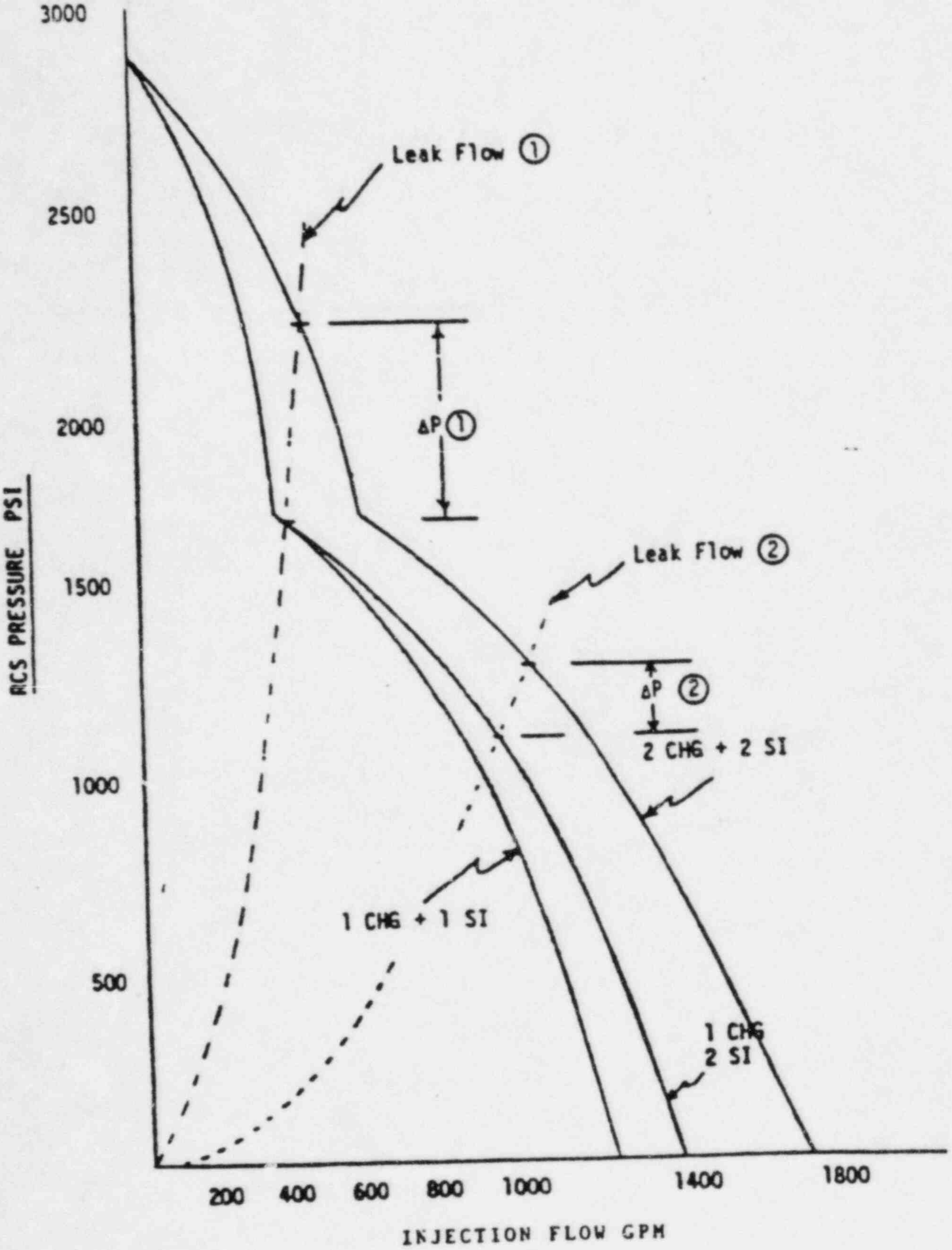
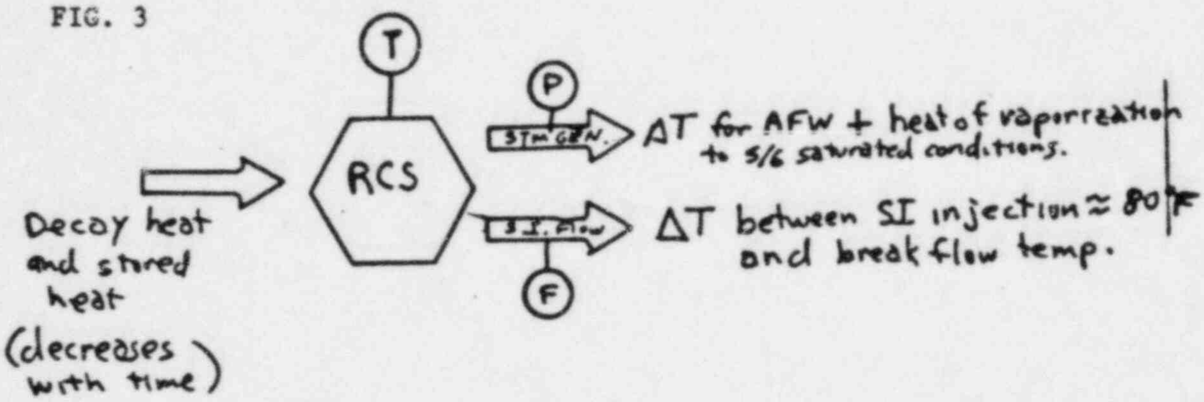


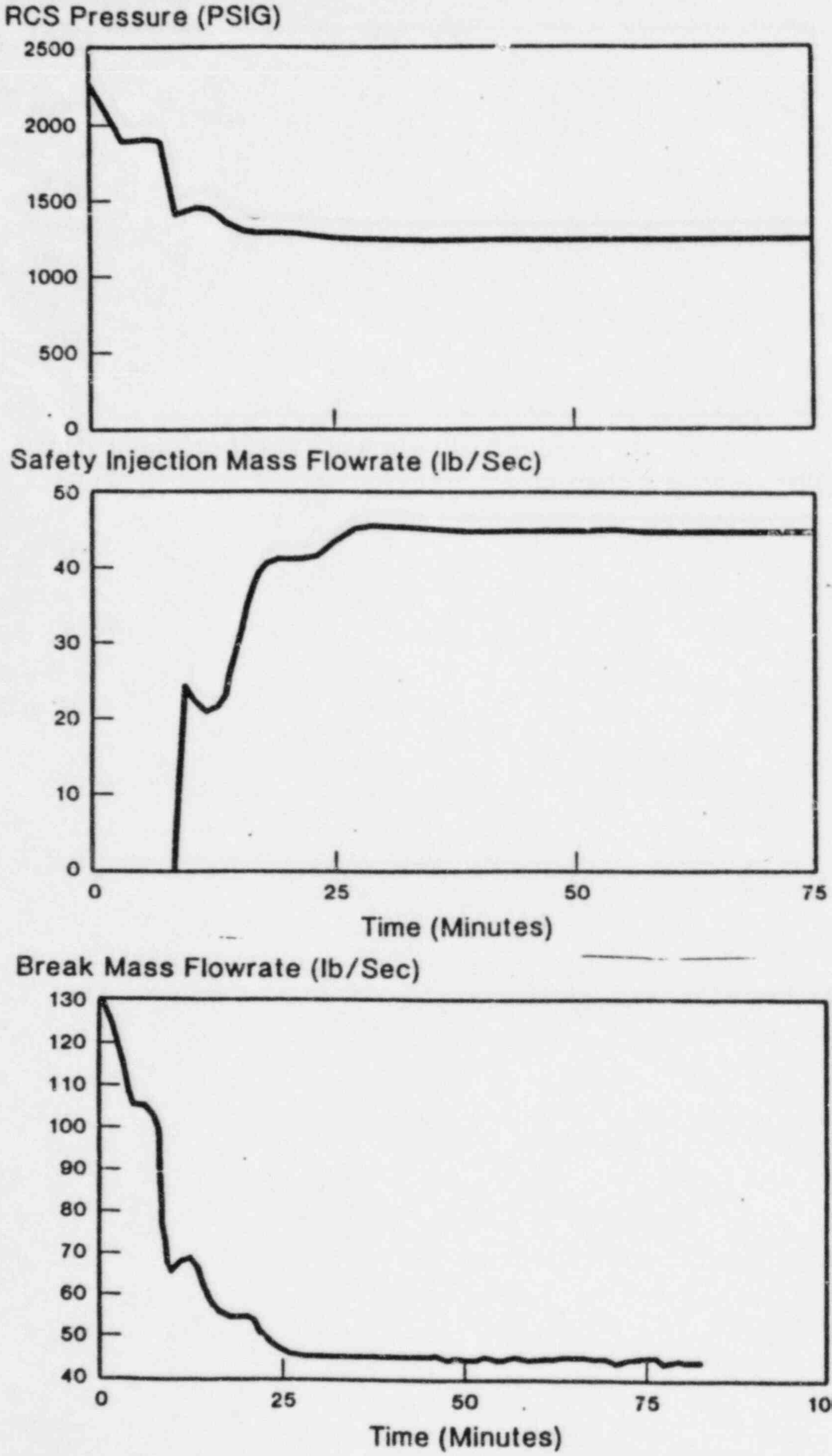
FIGURE 2 INJECTION vs FLOW CURVES

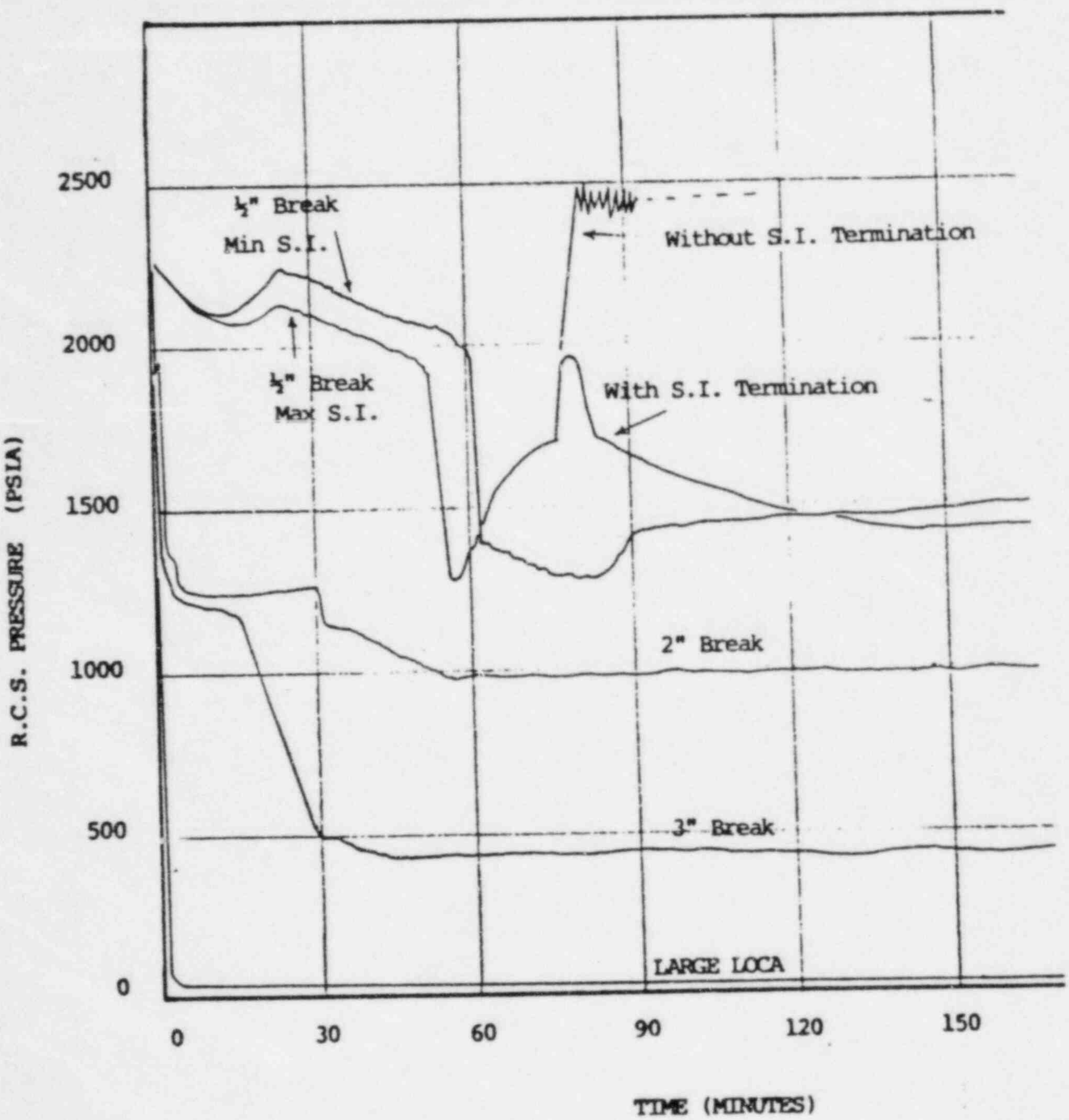
FIG. 3



Temperature Equilibrium

Fig. 4 1.0 INCH COLD LEG BREAK

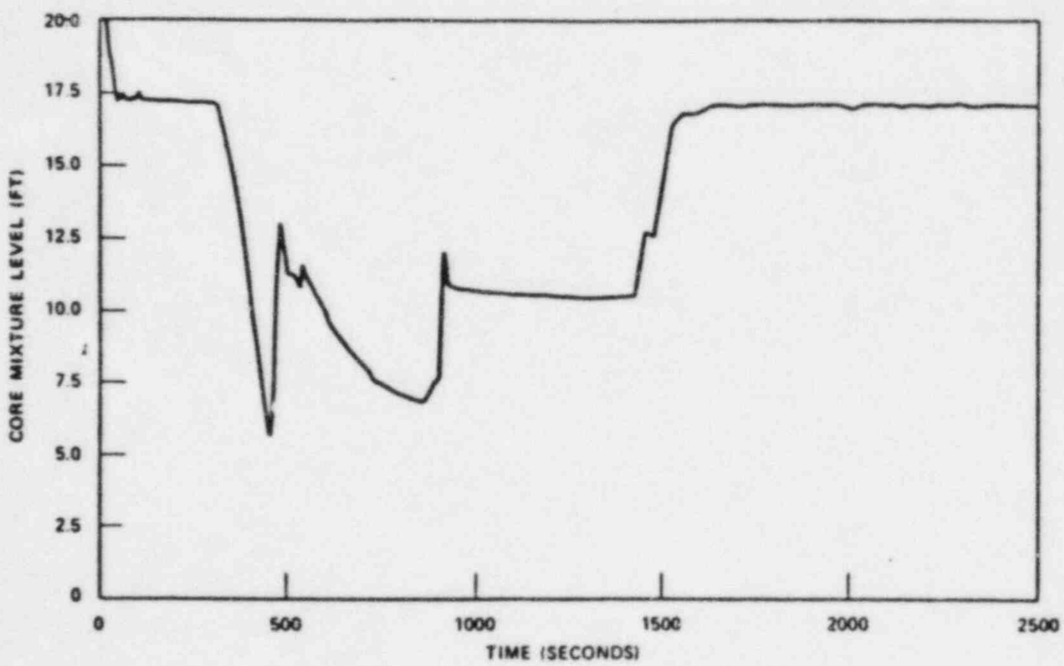




2500
2000
1500
1000
500
0
R.C.S. PRESSURE (PSIA)
½" Break
Min S.I.
½" Break
Max S.I.
Without S.I. Termination
With S.I. Termination
2" Break
3" Break
LARGE LOCA
0
30
60
90
120
150
TIME (MINUTES)

Fig. 5

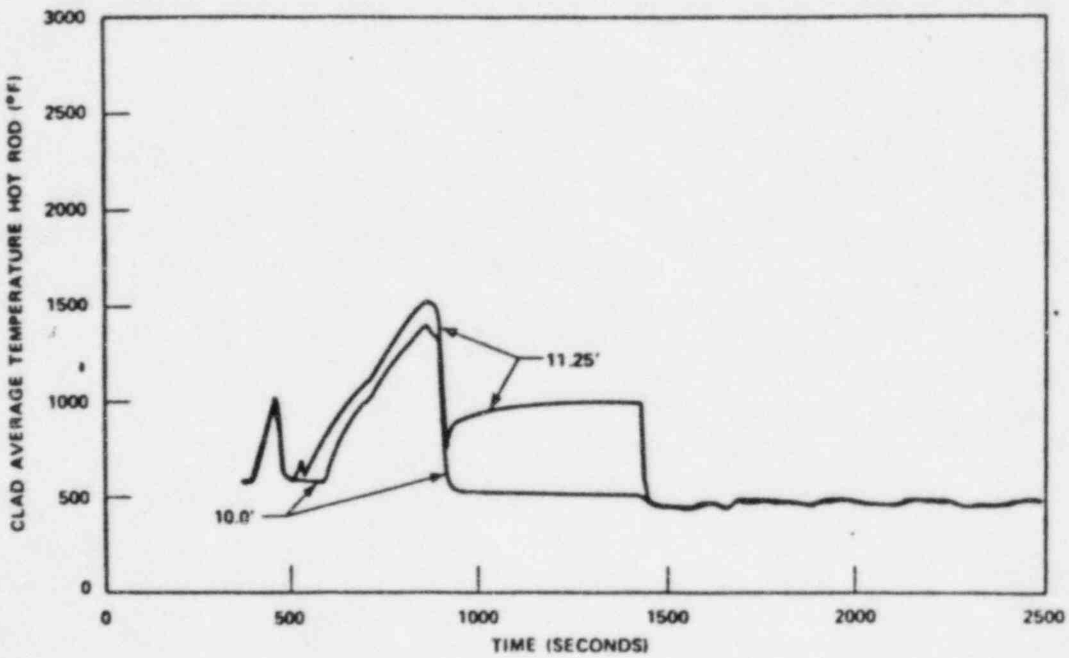
FIG. 6



CORE MIXTURE (4-in. BREAK)

Breaks in the 2 in to 1 sq. ft. range result in significant draining of the RCS and could result in some core uncovery depending on the status of safeguards equipment, break size, and break location. Figure 6 shows the Vogtle FSAR analysis for a 4 in break with the core mixture level descending momentarily to about the 6 ft level, increasing, then later dropping to that level about 15 min. into the transient. The second time the level drops the clad temperatures reach the 1800°F mark as shown in fig. 7. Some core exit thermocouples may indicate localized hot temperatures (superheating of the fluid). This response corresponds to design basis assumptions. More probable conditions for this size of break will yield less core uncovery or no core uncovery.

FIG. 7



CLAD TEMPERATURE TRANSIENT
(4-in. BREAK)

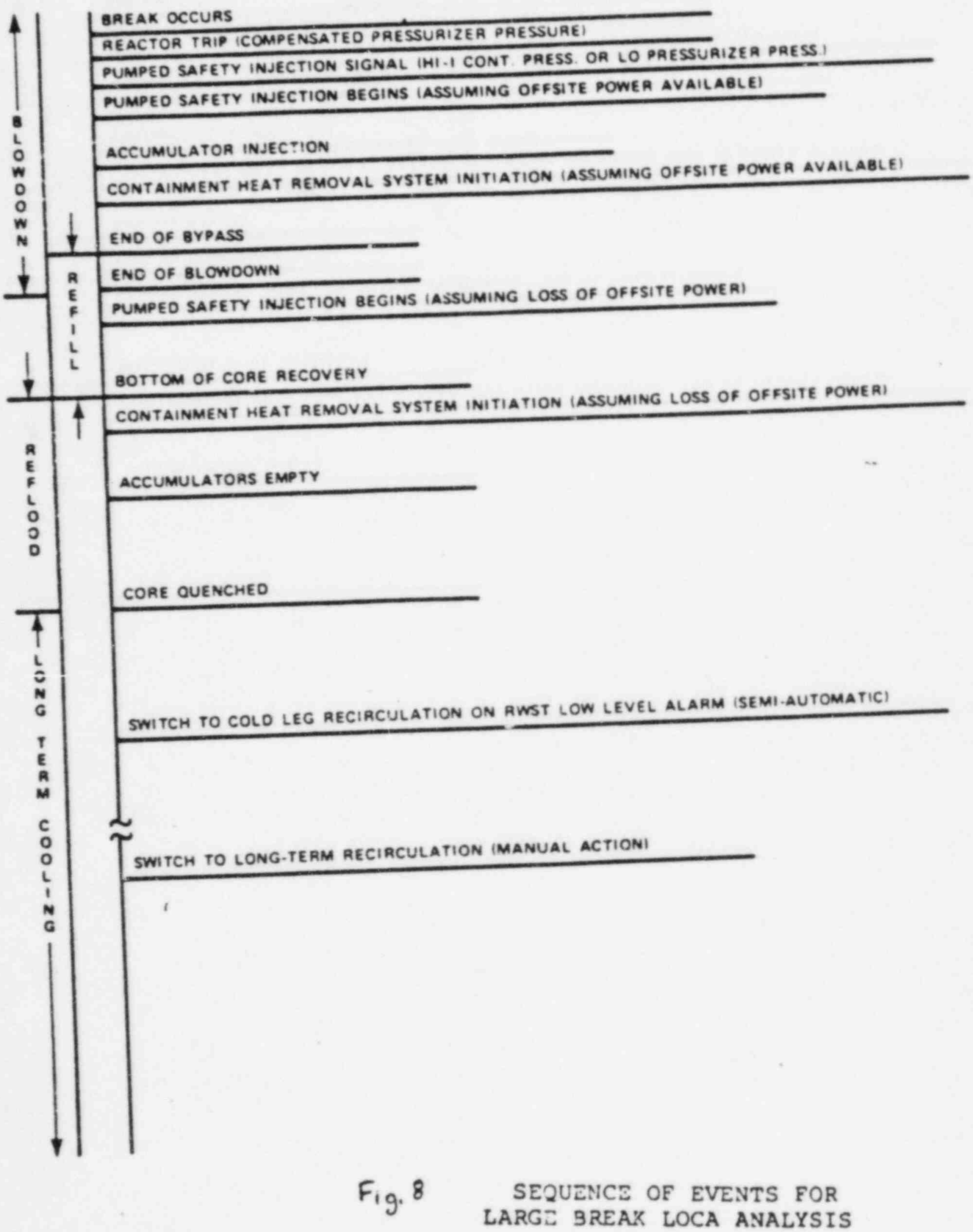


Fig. 8 SEQUENCE OF EVENTS FOR LARGE BREAK LOCA ANALYSIS

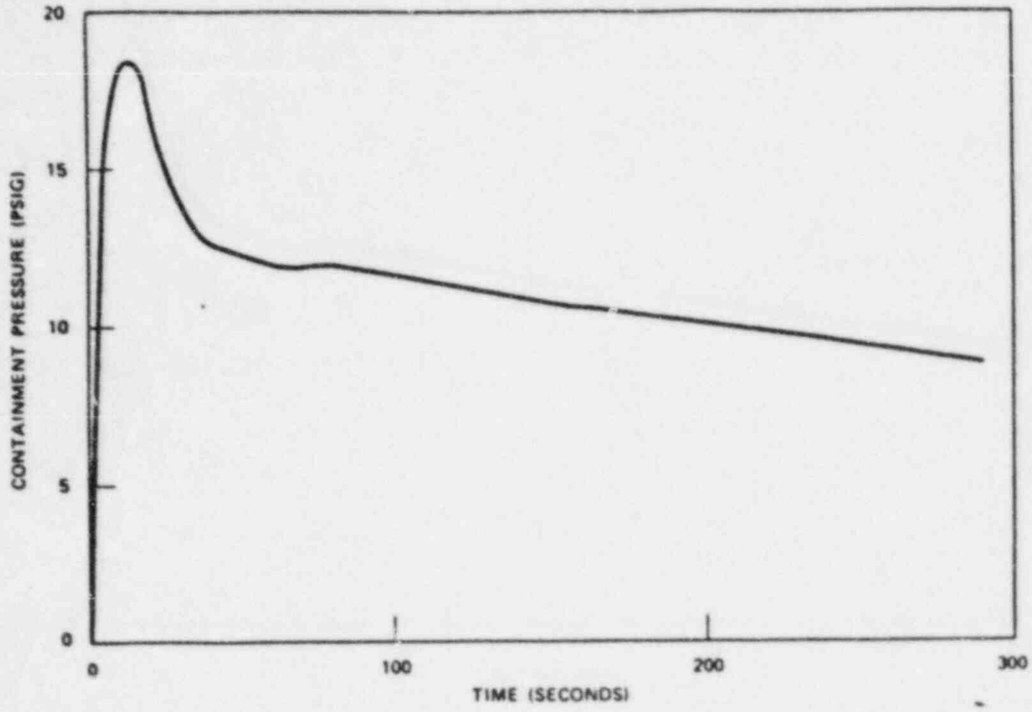


Fig. 9 CONTAINMENT PRESSURE TRANSIENT

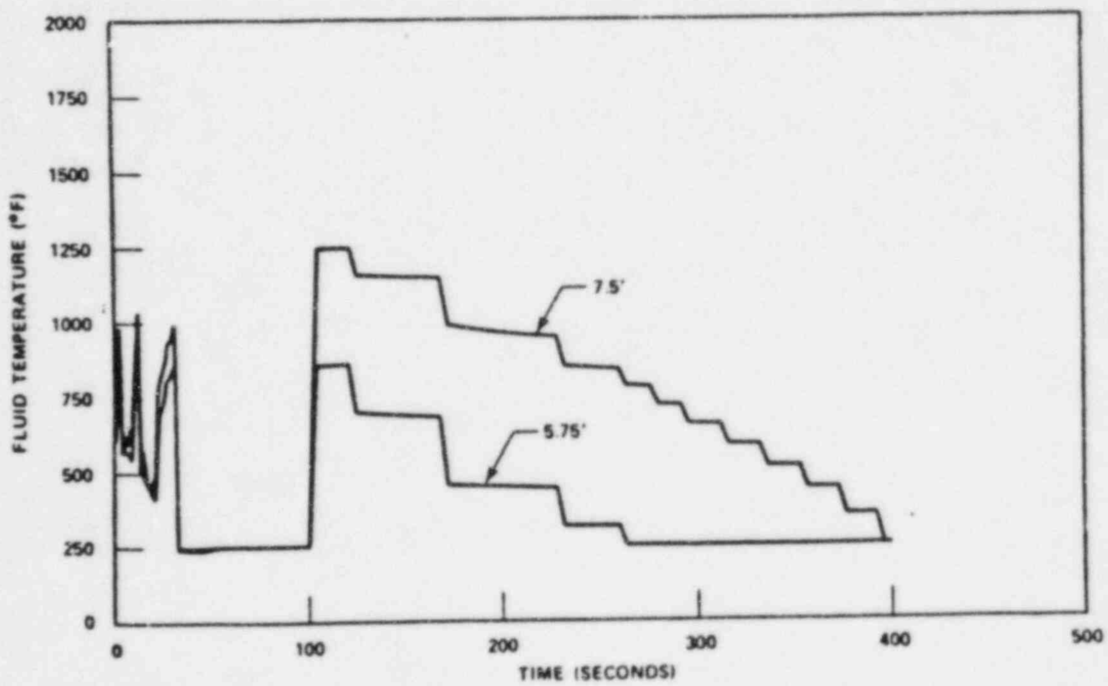
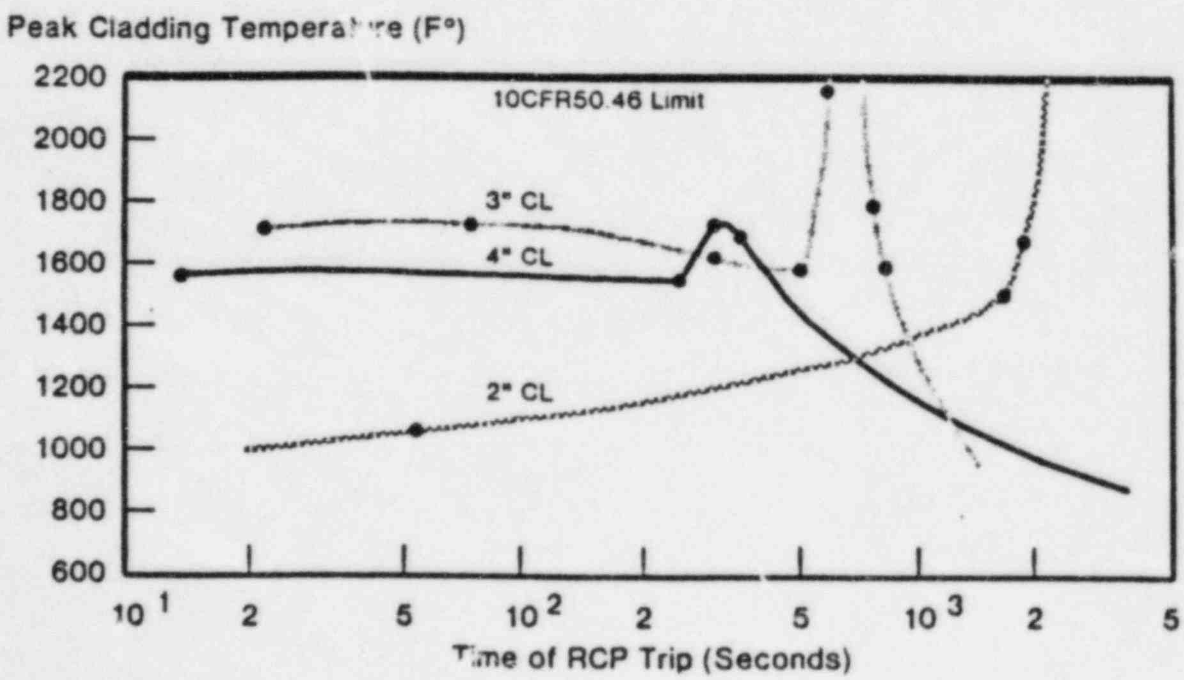


Fig. 10 FLUID TEMPERATURE
DECLG

FIG. 11.

EFFECT OF PUMP TRIP TIME ON PEAK CLADDING TEMPERATURE FOR WESTINGHOUSE 3-LOOP PLANT (PREPARED FROM TABLE 3.2-1 OF WCAP-9584)



Westinghouse has determined (WCAP-9584) that approximately 10 minutes are available for trip of the RCPs following the worse case small break LOCA. Trip of the RCPs within this time frame will prevent peak clad temperatures from exceeding 2200°F.

Per Vogtle's FSAR, Section 15.0.1.3, a small break LOCA is classified as a Condtion III event.

ANSI N660 indicates that for a Condition III event, 20 minutes is required for the operator to perform the manual protective action (in this case; trip the RCPs). This time period, it states, allows longer time intervals for the operator to 1) recover from his initial stress, 2) diagnose the event that has occurred, and 3) plan his actions. ANSI N660 also states that for a Condition III event, 2 + N minutes are required for the operator to complete the protective action, where N is the number of discrete operator actions to be performed in order to carry out the protective action.

We feel ANSI N660 does not adequately consider the RCP Trip Criteria. The time allocation for the operator to diagnose the event is unnecessary. The EOPs are designed and written to be symptom based. That is, the procedure directs the operator to look at the value of a parameter and based on that value meeting the procedural requirement, he is directed to perform a specific action. Even though it is expected that the plant operators know and understand the events in the progress, the EOPs are written such that the consequences of the events can be mitigated without a complete understanding of what has occurred.

The time allocation for the operator to plan his action is also unnecessary. There is no need to plan; the actions are specified in the procedures. RCP Trip Criteria are listed as specific procedural steps throughout the EOPs which are written for diagnostics and conditions where a small break LOCA could be the initiating event or occurring concurrent with other failure events. The RCP Trip Criteria are also listed in the foldout page for those EOPs, and are applicable as long as the procedure is in effect.

We also feel that the 2 + N minute requirement does not adequately apply itself to the discrete actions performed in tripping the RCPs. The handswitches for the two in-series RCP trip breakers are located side by side. The eight handswitches for all four (4) RCPs are located together forming two columns on the horizontal portion of the main control board adjacent to the RCS loop flow meters. The indications of RCP trip; green status light on each handswitch, underfrequency and undervoltage annunciation (UF and UV sensed at RCP motor, downstream of breakers), and RCS loop flow are located on the same section of the main control board with a maximum distance of separation of four (4) feet occurring between annunciation and furthermost breaker handswitch.

We do not feel that ANSI N660 takes into account operator training on such significant manual protective actions as the RCP Trip Criteria. This is a major area in our EOP training program. Operators are required to know the RCP Trip Criteria RCS pressure setpoint and when it is applicable (i.e. not applicable when a controlled cooldown such as the maximum rate depressurization of intact SGs following a SGTR results in the setpoint being exceeded).

Verifying that the RCPs would be tripped within the 10 minute time frame specified by Westinghouse, was accomplished by the EOP Validation performed on the Vogtle Simulator. Two different teams of operator and controllers performed various LOCA scenarios. The average time period for the operator to trip the RCPs using the trip criteria was 155 seconds (2½ minutes) from the initiation of the LOCA event, with 15 seconds being the shortest time period and 4 minutes the longest. No concerns were expressed about using the RCP Trip Criteria in the debriefings with operators following the scenario runs. No instances occurred, during the runs, where the RCP trip was inadvertently ommitted, nor was there any instance where the RCPs were unnecssarily tripped.

Tripping the RCPs after the time interval tends to prolong liquid break discharge which depletes more liquid mass out of the RCS. This results in two main effects; 1) deeper core uncovery, and 2) reduced total time of uncovery. These two characteristics have opposing effects on peak clad temperatures which give rise to the maximum function and worse time interval of RCP trip. The peak clad temperatures for RCP trip during this interval exceed those calculated FSAR PCTs and in some cases exceed 2200°F. A detailed discussion of the effects of delayed RCP trip on core cooling is included in WCAP-9584, Section 3.3.1.

In the event that a delayed RCP trip should occur, the EOPs provide guidance for restoring core cooling. The EOPs are designed to consist of two types of procedures: ORPs and FRPs. Optimal Recovery Procedures (ORPs) diagnose the event and provide symptom based guidance for mitigating the consequences of the event. These procedures are entered following a reactor trip or safety injection. The Function Restoration Procedures (FRPs) are implemented to maintain the critical safety functions which maintain the barriers against radiation release (i.e. fuel clad, RCS pressure boundary, and containment).

The EOPs utilize Critical Safety Function Status Trees (CSFSTs) to monitor the critical safety functions. These status trees directs the operators to FRPs when a critrical safety function is in jeopardy. FRPs provide actions to restore or minimize the consequences of loss of the safety function.

During emergency events the operators initiate and perform the actions of the ORPs. The STA or a designated individual monitors the CSFSTs either manually or on one of the SPDS displays. When a CSFST inicates a safety function is in jeopordy the STA informs the Shift Supervisor who implements the FRP.

Should a delayed trip of the RCPs occur at the worst case time, the Core Cooling CSFST (a copy is attached) directs the operator to the appropriate FRP. A vessel void fraction of 50%, as indicated by a RVLIS dynamic range setpoint, is used as the symptom of this Core Cooling challenge. If this setpoint is reached, the Function Restoration Procedure 19222-1, FR-C.2 RESPONSE TO DEGRADED CORE COOLING, is implemented and directs the operator to not trip the RCPs in order to ensure continued core cooling from forced flow. Core cooling should be ensured as long as RCPs continue to run. Subsequent steps direct a controlled RCS cooldown using the steam generator to cause SI accumulator water injection and increase system inventory.

If the RCPs fail or are tripped after a 50% void fraction is reached, core uncovery and clad heatup may occur. If core exit thermocouple temperatures reach 700°F or greater and the RVLIS full range indication decreases to less than 3.5 feet in the core, then Procedure 19221-1, FR-C.1 RESPONSE TO INADEQUATE CORE COOLING, is implemented and directs the operator to depressurize all steam generators at the maximum rate. If a further clad heatup occurs, exceeding 1200°F at the core exit, and SG depressurization is not effective, then RCPs are restarted one at a time to provide forced cooling flow.

In summary, the EOPs provide multiple levels of contingency actions that are symptom-based and function-related. In addition to the RCP trip parameter and setpoint, RVLIS and core exit thermocouples are used to direct operator action if a challenge to a Critical Safety Function is occurring. In this way the operator is provided with actions to maintain Critical Safety Functions that are dependent only on parameters available in the control room and that are independent of the specific event sequence. If an RCP tr p criteria step is missed by the operator and conditions degrade to the point where core cooling may be challenged if RCPs are stopped, then the operator is provided with appropriate contingency actions.

ATTACHMENT A

STARTING A REACTOR COOLANT PUMP

1. Establish Initial Conditions:

   a. 13.8KV power available to RCP.

   b. Steam bubble in PRZR

   c. #1 Seal ΔP greater than or equal to 200 psid.

   d. Seal injection flow 8 to 13 gpm.

   e. Seal leakoff flow greater than or equal to 0.2 gpm.

2. Check the following alarms clear or establish conditions to clear those alarms for the RCP to be started:

   a. RCP LOWER OIL RSVR HI/LO LEVEL.

   b. RCP UPPER OIL OIL RSVR HI/LO LEVEL.

   c. VOLUME CONTROL TANK OUTLET TEMP HI.

   d. VCT HI/LO PRESS.

   e. RCP STNDPIPE LO LEVEL.

   f. RCP STNDPIPE HI LEVEL.

   g. RCP MTR OVERLOAD.

   h. RCP NO 2 SEAL LKOFF HI FLOW.

   i. ACCW RCP CLR OUTLET HI TEMP.

   j. ACCW RCP CLR LO FLOW.

   k. ACCW RCP THERM BARRIER HX HI FLOW.

   l. ACCW RCP THERM BARRIER HI PRESS.

3. Verify all RCP ACCW thermal barrier isolation valves open.

4. Start the associated RCP oil lift pump.

5. After two minutes of lift pump operation, start the RCP.

ATTACHMENT A (Cont'd.)

6. After approximately one minute check the following alarms clear:
   a. RCP LOW FLOW.
   b. RCP SHAFT VIBRATION.
   c. RCP FRAME VIBRATION.
   d. Those alarms in Step 2.
7. After one minute of RCP operation, stop the oil lift pump.

END OF ATTACHMENT A

ATTACHMENT D

RECOVERY OF RCP SEAL INJECTION

1. Check RCP No. 1 seal temperature.

   IF less than 220°F,
   THEN open CVCS SEALS RCP SEAL INJ SUPPLY CNMT ISO valve and ACCW supply to the affected RCP. Return to Step 26.

   IF greater than 220°F,
   THEN proceed to Step 2 of this Attachment.

2. Verify seal injection supply temperature - LESS THAN 135°F.
   Verify ACCW supply temperature - LESS THAN 105°F.

3. Dispatch operator to shut CVCS SEALS RCP SEAL INJ NEEDLE VLVS TO #1 SEAL for affected RCP.

   - 1208-U4-414(RCP 1)
   - 1208-U4-415(RCP 2)
   - 1208-U4-416(RCP 3)
   - 1208-U4-417(RCP 4)

4. Verify CVCS SEALS RCP SEAL INJ NEEDLE VLVS of Step 3 of this Attachment is shut.

5. Open CVCS SEALS RCP SEAL INJ SUPPLY CNMT ISO valve for affected RCP.

6. Slowly open CVCS SEALS RCP SEAL INJ NEEDLE VLVS TO #1 SEAL to establish a 1°F per minute cooldown rate.

7. WHEN RCP No. 1 seal temperature is less than 220°F,
   THEN restore ACCW supply to the affected RCP.

8. Verify RCP seal parameters:

   o Seal injection supply temperature - LESS THAN 135°F.
   o RCP No. 1 seal temperature - LESS THAN 220°F.
   o ACCW supply temperature - LESS THAN 105°F.

   IF RCP seal parameters can NOT be verified,
   THEN secure the affected RCP.

9. Return to Step 26 of procedure.

END OF ATTACHMENT D