General Electric Advanced Technology Manual

Chapter 4.2

Anticipated Transient Without Scram

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4.2 ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

Learning Objectives:

1. Define the term “anticipated transient without scram” (ATWS).

2. Recognize the potential consequences of an ATWS event and the expected plant response for the worst case ATWS event with no operator actions.

3. Recognize the regulatory requirements for BWRs contained in 10CFR50.62, to reduce the risk from an ATWS event.

4. Identify potential causes of an ATWS event in BWR design and the indications of each.

5. Recognize the strategies used in the EOPs to limit core power during an ATWS event and ultimately to shut down the reactor.

4.2.1 Introduction

In general, the term reactor transient applies to any significant deviation from the normal operating value of any of the key reactor operating parameters. Transients may occur as a consequence of an operator error or the malfunction or failure of equipment.

Anticipated transients are deviations from the normal operating conditions that may occur one or more times during the service life of a plant. Anticipated transients range from trivial to significant in terms of the demands imposed on plant equipment. Anticipated transients include such events as a turbine trip, EHC failure, MSIV closure, loss of feedwater flow and loss of feedwater heating. More specifically, all situations (except for LOCA) which could lead to fuel heat imbalances are anticipated transients.

Many transients are handled by the reactor control systems, which return the reactor to its normal operating conditions. Others are beyond the capability of the reactor control systems and require reactor shutdown by the Reactor Protection System (RPS) in order to avoid damage to the reactor fuel or coolant systems. If such a transient should occur and if, in spite of all the reliability built into the Reactor Protection System, a scram does not result, then an Anticipated Transient Without Scram (ATWS) event has occurred.

4.2.2 ATWS Historical Background

ATWS became a possible source of concern in nuclear power plants in 1968 during discussions between Advisory Committee on Reactor Safeguards (ACRS), the regulatory staff, and reactor instrument designers about the safety implications of interactions between normal control system circuitry and protection system circuitry in the instrument systems of power plants. After considerable discussion, it was determined that separation of control and protection functions was being achieved to a reasonable degree, either by physical separation or by electrical isolation. The focus of interest with regard to instrument systems then shifted to the ability of the shutdown system to function with the needed reliability considering common mode failures. Common mode failures have to do with design or maintenance errors that might be made for similar redundant portions of a protection system. One of the difficult aspects of deciding whether or not common mode failures were being adequately accounted for in shutdown system design was that techniques to analyze a system for common mode failures were not as well developed. In 1969, the efforts to evaluate the safety concerns of ATWS events went in two general directions. The first was concerned with attempting to evaluate the likelihood of common mode or other failures of the reactor protection system that could lead to ATWS events. The second was to assume, simply as a basis for discussion, that ATWS was possible and to examine the consequences of various postulated ATWS events.

The ATWS event was analyzed in combination with different initiating conditions. The results showed that the worst-case ATWS initiating event for a boiling water reactor is a closure of all Main Steam Isolation Valves (MSIVs). A closure of all MSIVs results in an increase in primary system pressure and temperature. The pressure increase would decrease the volume of steam bubbles in the reactor core and this, in turn, would increase the reactivity and cause a surge in reactor power. The power surge would cause a further increase in system temperature and pressure, with the pressure rising to values above acceptable limits.

Another transient that has the potential for significant damage as an ATWS initiating event is a loss of condenser vacuum. A loss of main condenser vacuum causes the automatic closure of the turbine stop valves and turbine bypass valves. The turbine stop valves are fast acting valves, so there is an abrupt interruption of steam flow from the reactor. The sole difference in severity between the closure of the turbine stop valves versus the closure of all MSIVs is that, for the turbine stop valve closure there is a large steam line volume that can act to buffer the pressure rise. Although the turbine stop valve closure transient proceeds more slowly than the MSIV closure transient, the result will still be an excessively high reactor coolant system pressure.

The debate concerning the safety significance and plant susceptibility to an ATWS event continued through the 1970’s between the ACRS, the NRC, and utility licensees. During this time, reactor vendors submitted analyses on ATWS to the NRC. Then, in 1978, the NRC issued NUREG-0460 (Anticipated Transients Without Scram for Light Water Reactors). The NUREG included the following:

- ATWS Acceptance Criteria

  The staff recommends that all nuclear power plant designs should incorporate the designs features necessary to assure that the consequences of ATWSs would be acceptable. The primary criterion for acceptability is that the calculated radiological consequences must be within the dose guidelines values set forth in 10 CFR Part 100. In addition, more specific acceptance criteria have been developed for primary system integrity, fuel integrity, containment integrity, long-term shutdown and cooling capability, and the design of mitigating systems.

- Containment Integrity

  The calculated containment pressure, temperature and other variables shall not exceed the design values of the containment structure, components and contained equipment, systems or components necessary for safe shutdown. For boiling water reactor pressure suppression containments, the region of relief or safety valve discharge line flow rates and suppression pool water temperatures where steam quenching instability could result in destructive vibrations shall be avoided.

- Long-Term Shutdown and Cooling Capability

  The plant shall be shown to be capable of returning to a safe cold shutdown condition subsequent to experiencing an ATWS event, i.e., it must be shown that the reactor can be brought to a subcritical state without dependence on control rod insertion and can be cooled down and maintained in a cold shutdown condition indefinitely.

- Fuel Integrity

  Damage to the reactor fuel rods as a consequence of an ATWS event shall not significantly distort the core, impede core cooling and prevent safe shutdown. The number of rods which would be expected to have ruptured cladding shall be determined for the purpose of evaluating radioactive releases.

- Primary System Integrity

  The calculated reactor coolant system pressure and temperature shall be limited such that the calculated maximum primary stress anywhere in the system boundary, except steam generator tubes, is less than that permitted by the "Level C Service Limit" as defined in Section III of the ASME Nuclear Power Plant Components Code.

  In addition, the deformation of reactor coolant pressure boundary components shall be limited such that the reactor can be safely brought to cold shutdown without violating any other ATWS acceptance criterion. The integrity of steam generator tubes may be evaluated based on a conservative assessment of tests and the likely condition of the tubes over their design life.

- Mitigating Systems Design

  Mitigating systems are those systems, including any systems, equipment, or components, normally used for other functions, relied upon to limit the consequences of anticipated transients postulated to occur without scram. These systems shall be automatically initiated when the conditions monitored reach predetermined levels and continue to perform their function without operator action unless it can be demonstrated that an operator would reasonably be expected to take correct and timely action. These systems shall have high availability and in combination with the reactor protection system shall provide two independent, separate and diverse reactivity shutdown functions. The mitigating systems shall be independent, separate and diverse from the reactor trip and control rod systems, including the drive mechanisms and the neutron absorber sections. The mitigating systems shall be designed, qualified, monitored and periodically tested to assure continuing functional capability under the conditions accompanying ATWS events including natural phenomena such as earthquakes, storms including tornadoes and hurricanes, and floods expected to occur during the design life of the plant.

In 1981, based on the analyses submitted to the NRC by reactor vendors, the following proposed rules were filed in federal register Vol. 46, No. 226:

Early Operating Reactors

a. Modify the control rod drive scram discharge volume.
b. Provide actuation circuitry that is separate from the reactor protection system (i.e., ATWS recirculation pump trip)

Operating Plants with Construction Permits Issued Prior to 1/1/78

a. Provide automatic initiation of the Standby Liquid Control system and increase its flow capacity (or increase SLC boron concentration).

New Plants and Plants with Construction Permits Issued on or After 1/1/78

a. Addition of high capacity neutron poison injection systems.

4.2.3 Bases for ATWS Rules

In large, modern boiling water reactors, a transient with failure to scram from full power is very likely to cause or may follow the isolation of the reactor (i.e., turbine trip or main steam isolation valve closure). If the recirculation pumps continue to run, the power level will remain high and a severe pressure excursion will take place. Even if the reactor coolant system survives the pressure surge, the very high steam flow will rapidly heat the suppression pool and pressurize the containment. In addition, as the reactor does not depressurize, the High Pressure Coolant Injection (HPCI) System may not suffice to cool the core: overheating and core damage may follow. Ultimately, the containment is expected to rupture due to over pressurization while the core sustains damage. Continued core coolant replenishment is questionable after containment rupture. A large radiological release is a plausible outcome. A necessary mitigating feature is thus a prompt automatic trip of the recirculation pumps to avoid the pressure excursion and diminish the power and the consequent steam flow to the suppression pool. Given a trip of the recirculation pumps, the reactor power will stabilize at roughly 30% power until the reactor coolant boils down and steam bubbles (void formation) in the core throttle the chain reaction. Thereafter, an oscillatory equilibrium will be maintained in which the reactor sustains the average power necessary to boil off the amount of reactor coolant that is delivered up to about 30% power. Analysis shows that HPCI or main feedwater can adequately cool the core to avoid extensive core damage. However, the power delivered to the suppression pool will be greater than the pool cooling system can dissipate. Therefore, containment over pressure failure remains a distinct possibility unless the reactor is shutdown, either by control rod insertion or by liquid reactivity poison injection (SLC). Well before the containment is significantly pressurized, the suppression pool will approach saturation and steam condensing will become unstable. Chugging steam condensing may threaten containment integrity or pressure suppression and thus shorten the time available to shutdown the reactor without unacceptable consequences.

The fault or human error that precipitates the initial transient might also disable HPCI. In addition, system reliability analyses have indicated that HPCI may fail or be unavailable in as many as from 1% to 10% of the cases in which a demand is made of the system. This may be insufficient reliability for the mitigation of a potentially serious accident having a frequency of occurrence that might be as high as once in a thousand reactor years. A second diverse system, the Reactor Core Isolation Cooling (RCIC) System should be expected to auto start and run, delivering coolant to the reactor. If RCIC is the sole operative means of replenishing reactor coolant, the adequacy of core cooling, rather than the heat deposited in the suppression pool, is likely to be the factor limiting the time allowed to shut down the reactor without unacceptable consequences. The RCIC can successfully cool the reactor once it is shut down, and it can slow the boil off of reactor coolant in the reactor.

The NRC has concluded that the liquid reactivity poison injection system in large modern BWRs must have a start time and poison injection rate such that either of two redundant trains of high pressure systems, HPCI or RCIC can successfully mitigate ATWS transients.

Concern has been expressed that RCIC, though capable of meeting these success criteria, does not prevent the automatic depressurization of the reactor coolant system. Operator action is necessary in less than ten minutes to override automatic depressurization. The NRC staff does not wish to force an alteration of the logic governing the Automatic Depressurization system (ADS) which might compromise the reliability of ADS in non-ATWS events.

Several factors complicate the analysis of the ATWS tolerance of BWR plants. The delivery of main feedwater which may be available in some ATWS accident sequences may dilute liquid poison and increase the power level in ATWS events, thus threatening successful mitigation. In some sequence variants, operators might be tempted to depressurize the reactor to enable low pressure reactor coolant injection but, in so doing, disable turbine driven coolant injection systems or otherwise compromise possible avenues of successful ATWS mitigation.

The tables and figures at the end of this chapter provide timelines of significant ATWS event milestones with (and without) operator action, as well as provide graphical representations of the effects of an ATWS on key reactor plant parameters.

4.2.4 10CFR 50.62

The Code of Federal Regulations requires all BWRs to have an alternate rod insertion (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves.

Each BWR must have a standby liquid control system (SLC) with the capability of injection into the reactor vessel of a borated water solution with a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from the injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel for a given core design. The SLC system and its injection location must be designed to perform its function in a reliable manner. The SLC initiation must be automatic for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

Each BWR must have equipment to trip the recirculation pumps automatically under conditions indicative of an ATWS.

4.2.5 ATWS Event Mitigation

4.2.5.1 ATWS Causes and Indications

A failure of some (or all) of the control rods to fully insert into the reactor core when a Reactor Protection System (RPS) signal demands an automatic reactor scram (or a manual scram is demanded) could be caused by one or more of the following:

- Failure of the RPS relays to open

This condition would be indicated to the operator in the Control Room by the RPS Scram Group lights remaining lit when an RPS trip setpoint has been exceeded. Removing RPS fuses or opening circuit breakers in the scram solenoid valve power supply circuit are both methods that can be used to de-energize the RPS relays.

- Failure of the scram air header to vent to atmosphere

This condition would be indicated to the operator in the Control Room by lack of receipt of the Scram Pilot Air Header Pressure Low Alarm (red light on Panel 603). Local manual venting of the scram air header in the Reactor Building will depressurize the header and cause scram valves being held closed by air pressure to subsequently open.

- Stuck control rods

This condition could be diagnosed by numerous indications but would most likely be exhibited by partial control rod insertion. The condition could be caused by mechanical binding of control rods, reactor pressure and HCU accumulator pressure values not sufficient to affect a full control rod scram, or a scram system malfunction which did not result in a full control rod insertion. There are numerous methods to mitigate this condition including: resetting the scram, defeating RPS logic trips, draining the scram discharge volume, and initiating subsequent manual reactor scrams; attempting to insert control rods one at a time using the individual scram test switches; increasing CRD cooling water differential pressure; venting control rod drive overpiston volumes; or driving control rods using the Reactor Manual Control System (RMCS).

4.2.5.2 ATWS Mitigation Strategies

In order to effectively mitigate the consequences of an ATWS event, licensees take actions as specified in their respective Emergency Operating Procedures (EOPs) that are derived from the Boiling Water Reactor Owners Group Emergency Procedure Guidelines (BWROG EPGs). The BWROG EPG strategy guidelines have the operator initiate a reactor scram and, if the scram is not successful, reduce reactor power and shut down the reactor by manual control rod insertion and boron injection.
The BWROG EPG mitigating actions for an ATWS event are as follows:

- <u>Confirm or place the Reactor Mode Switch in “SHUTDOWN”</u>. This action causes a diverse and redundant reactor scram signal to be generated by the RPS logic and also prevents MSIV closure on low main steam line pressure (assuming the ATWS event is not combined with a full MSIV closure), which maintains main condenser availability and minimizes the heat load on the containment.

- <u>If Alternate Rod Insertion (ARI) has not initiated, initiate ARI</u>. This action provides an independent and redundant means of depressurizing the reactor scram air header and operating the scram discharge vent and drain valves as required to affect a reactor scram.

- <u>If the main turbine generator is on-line (and a main steam line is not isolated) or if feedwater, HPCI, or RCIC are operating, confirm or initiate recirculation flow runback to minimum</u>. This action causes an immediate and rapid reactor power reduction. The most rapid power reduction is achieved by tripping the recirculation pumps, however, if the pump trip is initiated from a high power level, the resultant rapid changes in steam flow, RPV pressure, and RPV water level may cause a trip of the main turbine generator and trip of RPV injection systems, making the ATWS more severe.

- <u>If reactor power is above 3% (APRM downscale trip) or cannot be determined, trip the recirculation pumps</u>. This action affects a prompt reduction in reactor power.

- <u>When periodic neutron flux oscillations in excess of 25% peak to peak commence and continue OR before suppression pool temperature reaches the Boron Injection Initiation Temperature (BIIT), inject boron into the RPV with SLC and prevent automatic initiation of ADS</u>. Injecting boron into the RPV adds negative reactivity to help achieve reactor shutdown conditions. Preventing automatic initiation of ADS prohibits automatic injection of a large volume of relatively cold, un-borated water into the RPV which could result in significant core damage due to the substantial positive reactivity addition. Initiating boron injection when neutron flux oscillations in excess of 25% peak to peak are sustained helps to minimize the chances of localized fuel clad damage due to dryout and rewetting of the clad surfaces. Initiating boron injection prior to suppression pool temperature reaching the BIIT helps to ensure that the Hot Shutdown Boron Weight can be injected prior to reaching the Heat Capacity Temperature Limit in the suppression pool (which would require Emergency Depressurization of the RPV).

- <u>Insert control rods using any and all methods available</u>. Fully inserting all control rods reduces reactor power and helps to place the reactor in a shutdown condition.

- <u>Deliberately lower RPV water level by terminating and preventing injection from all systems except RCIC, CRD, and SLC</u>. This action lowers RPV water level below the level of the feedwater spargers which reduces core inlet subcooling and increases the steam volume in the core, both of which in turn help to lower reactor power. Lowering RPV water level also reduces the driving head through the reactor core which will also help to lower reactor power.

4.2.6 Operational Occurrences

4.2.6.1 Browns Ferry ATWS

On June 28, 1980, a manual scram of the Browns Ferry Unit 3 reactor was attempted in conjunction with a planned shutdown for repairs of a feedwater line in the turbine building. With the exception of the need for the turbine building feedwater line repairs, plant conditions were normal. Browns Ferry Unit 3 operators were shutting down the reactor by first lowering reactor power level to 36 percent by reducing recirculation flow and inserting a number of control rods to decrease the neutron chain reaction; and secondly by depressing the manual scram pushbuttons to insert all control rods completely in order to terminate the neutron chain reaction. At Browns Ferry Unit 3 the plant is designed such that complete control rod insertion is normally accomplished in less than 3 seconds after both scram pushbuttons are depressed.

In this incident, normal control rod insertion did not occur when the scram pushbuttons were depressed (77 control rods failed to insert completely). Of the 185 control rods, 10 were fully inserted prior to the manual scram attempt. 77 control rods failed to insert fully upon the manual scram request, with insertion ranging from position 02 to position 46. Observing the incomplete control rod insertion, the operator reset the scram which allows recharging the scram accumulators and discharging the scram discharge volume. The manual scram procedure was then repeated. Rod insertion progressed somewhat, but 59 control rods remained only partially inserted. A subsequent scram reset and manual scram resulted in 47 control rods partially inserted. Accumulator recharging and draining of the scram discharge volume was repeated and the scram instrumentation automatically initiated a fourth scram. All control rods were then fully inserted, which placed the reactor in the normal shutdown condition. This was accomplished within approximately 14 minutes of the first manual scram.

Upon investigation following the Browns Ferry ATWS event, it was found that the ATWS was caused by an accumulation of drainage into the scram discharge volume associated with the east bank of CRD Hydraulic Control Units which prevented the respective control rods from fully inserting during a reactor scram.

4.2.6.2 Salem ATWS

On February 25, 1983, Salem Unit 1 experienced a low-low steam generator level in one steam generator. The unit had just been synchronized to the grid following a turbine overspeed test performed the previous evening and was operating at 14 percent reactor power. The reactor protection system responded as designed to the low-low steam generator water level condition and generated a reactor trip signal. This signal de-energized the undervoltage trip attachment on each reactor trip breaker to release the spring loaded latching mechanism. Both breakers failed to open. The reactor was tripped about 25 seconds later when the operator opened the breakers using the manual trip switch in the control room. The plant was then stabilized and placed in cold shutdown pending further evaluation.

In subsequent testing, the ‘A’ trip breaker failed to open two out of three times and the ‘B’ breaker failed to open in all three attempts. The cause of the failures of the RPS trip breakers was determined to be due to binding of the undervoltage trip mechanisms due to a lack of proper lubrication.

4.2.7 PRA Insight

The NRC staff evaluation of ATWS in NUREG-0460 was one of the first applications of PRA techniques to an Unresolved Safety Issue (USI). The evaluation highlighted the relative frequency of severe ATWS events for various reactor types and estimated the expected reduction in frequency for various postulated plant modifications. The study also proposed quantitative goals for resolving this issue. Other notable examples of PRA applications to the ATWS issue are the NRC sponsored survey and critique of reactor protection system (SAI, 1982), and the ATWS Task Force report summarized in SECY-83-293. The RPS survey reviewed 16 reliability studies; most of them published PRAs, to compare the predicted failure probability per unit demand, the anticipated transient frequency, and the primary influences on RPS unavailability. There was a surprising degree of agreement among the 16 studies. The second study quantified the relative improvement to be gained by implementing a set of recommendations proposed by the utility consortium in an ATWS petition to the NRC. The third study, a value impact evaluation of the risk reduction of generic plant classes, provided the basis for a final rule on ATWS (SECY-83-293).

NUREG-1150 looked at several accident sequences which include a failure of the reactor protection system. One of the major sequences is initiated by a transient that requires a reactor scram. The mechanical RPS fails which eliminates any possibility of scramming the reactor or manually inserting control rods. The recirculation pumps are tripped and the SRVs properly cycle to control reactor pressure. The standby liquid control system is initiated manually to inject borated water into the reactor to reduce reactivity. The ADS valves are not inhibited and the reactor depressurizes which allows low pressure cooling systems to operate. The RHR system is placed in the suppression pool cooling mode or containment spray mode for containment overpressure protection, resulting in a safe core and containment. As a result of the analysis completed for the NUREG-1150 final report, it was shown that ATWS events do in fact have a significant contribution to a plant’s core damage frequency, but in most cases the contribution is not as significant as from a Station Blackout event

An ATWS does have the possibility of leading to a core damage situation if the operator does not follow the Emergency Operating Procedures and initiate corrective actions like SLC initiation. However, the total contribution to the core damage frequency may not be very large (31% at Peach Bottom to 6% at Fermi). With respect to its negative impact on core damage frequency during an ATWS, operator action to manually initiate injection of boron using the Standby Liquid Control system is of the utmost importance.

Table 4.2-1 MSIV Closure with no Operator Action

EVENT	TIME (MIN)	COMMENT
MSIV closure initiated	0	No scram
ATWS-RPT	0.1	At reactor 1135 psia pressure
HPCI and RCIC Start	1.0	At RPV level 2
HPCI suction shift	8.3	High suppression pool level
HPCI fails	14.8	Suppression pool temperature 190^{o}F
CS and RHR systems start, ADS timer initiated.	16.0	At RPV level 1
TAF uncovered	16.7	Vessel level 360 inches
ADS actuation	18.0	Two minutes after timer actuation
BAF uncovered	19.0	Vessel level 216 inches
RHR, CS, and Condensate booster pumps start injecting	19.6	CBPs at 418 psia; CS at 357 psia; RHR at 346 psia
First core recovery	19.9	
Water introduction by RHR, CS and CBPs stop as vessel pressure increases	20.4	
Vessel pressure at relief valve setpoint	20.7	Vessel pressure 1120 psia
First core power peak	20.7	Thermal power = 178%
Drywell coolers fail on over temperature	22.4	Drywell temperature 200^{o}F
Second core uncovery	23.1	

Table 4.2-1 MSIV Closure with no Operator Action

EVENT	TIME (MIN)	COMMENT
RHR, CS and CBPs start injecting	24.4	
Second core recovery	24.7	
Injection by RHR, CS and CBPs stops	25.2	
Vessel pressure at relief valve setpoint	25.4	
RCIC turbine trip on high exhaust pressure	26.0	
Second power peak	27.7	Thermal power = 140 %
Third core uncovery	28.6	
RHR, CS and CBPs start injecting	29.0	
Third core uncovery	29.4	
Injection by RHR, CS and CBPs stop	29.8	
Third power peak	30.0	Thermal power = 156 %
Relief valves lift	30.1	
Fourth core uncovery	32.1	
RHR, CS and CBPs start injecting	33.6	
DRYWELL FAILS	36.8	Pressure at 132 psia

Table 4.2-2 MSIV Closure with Operator Action and With Failures of SLC and Manual Rod Insertion

EVENT	TIME (MIN)	COMMENT
MSIV closure initiated	0	No scram
SRVs cycling	0 - End	No manual SRV actuation
ATWS-RPT	0.1	Reactor Vessel Press 1135 psia
HPCI and RCIC automatically start	1.0	At RPV level 2
RCIC runs at full capacity	1-End	600 gpm
Suppression pool temperature exceeds 110^{o}F	1.5	EPG criterion for operator initiation of SLC
Operator attempts to manually insert rods	3.0	No rod motion
Operator attempts to start SLC	5.0	Pumps inoperative
Operator trips HPCI	5.0	To reduce core power and prevent HPCI failure
CS and RHR pumps start	6.2	At RPV level 1
Level below TAF	6.8	Emergency system range, normal level indication off scale low
Vessel water level stable at 2/3 core height	9.5 - End	Upper 1/3 of core steam cooled
Operators initiate both loops of suppression pool cooling.	10	Containment spray select and 2/3 core coverage override hand switches actuated
Suppression pool heat capacity temperature limit exceeded	43	Operators do not depressurize
ADS two minute timer starts	50	Drywell pressure >2.45 psig, vessel level 413 inches, and Low pressure ECCS pumps running
ADS two minute reset by operators	52 - End	Prevents actuation of ADS
Suppression pool temperature at 168^{o}F	60	Temperature slowly increasing
Suppression pool approaching maximum temperature	360	206^{o}F bulk temperature

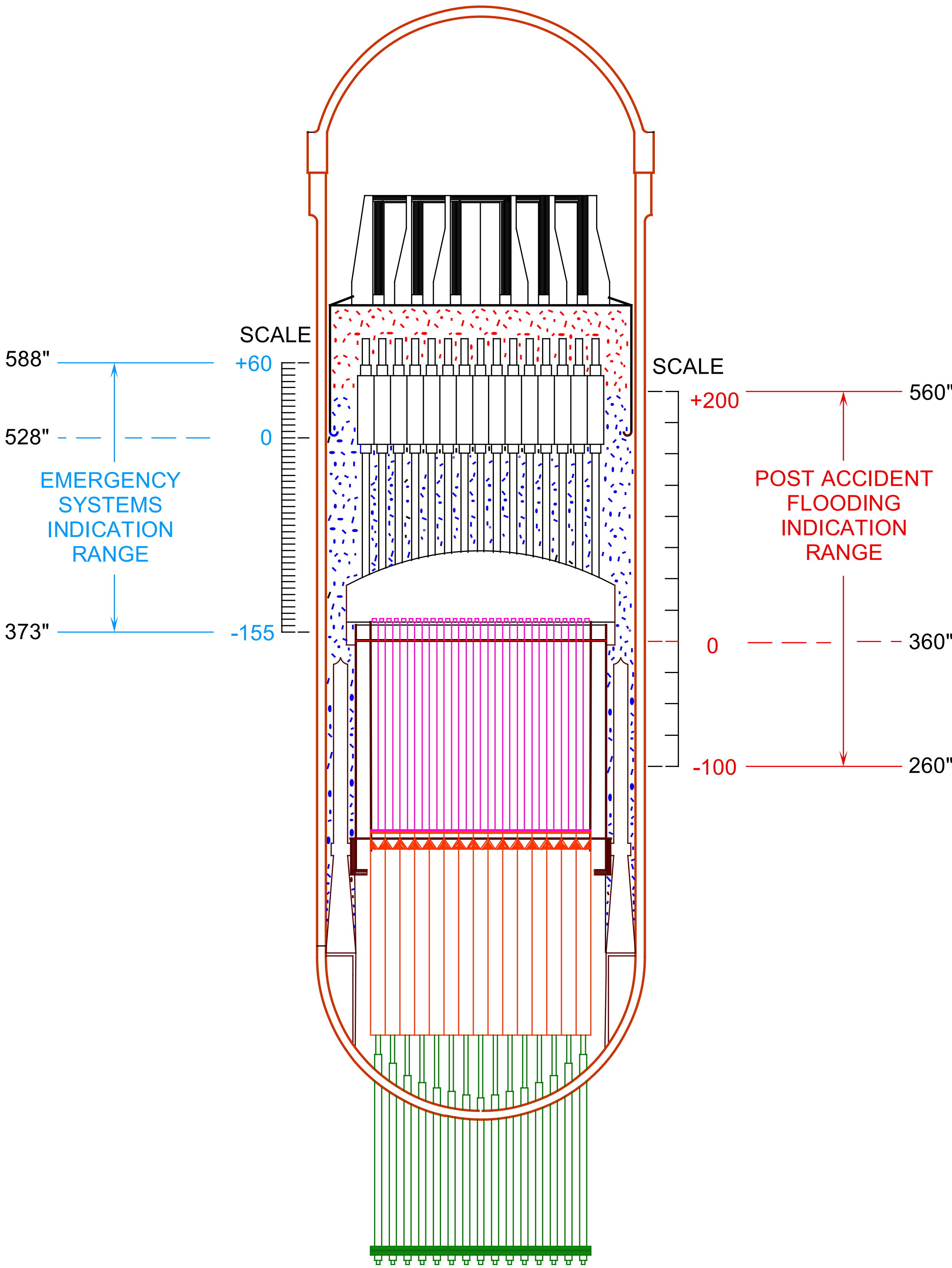

Figure 4.2-1 Level Instruments Available During ATWS Event

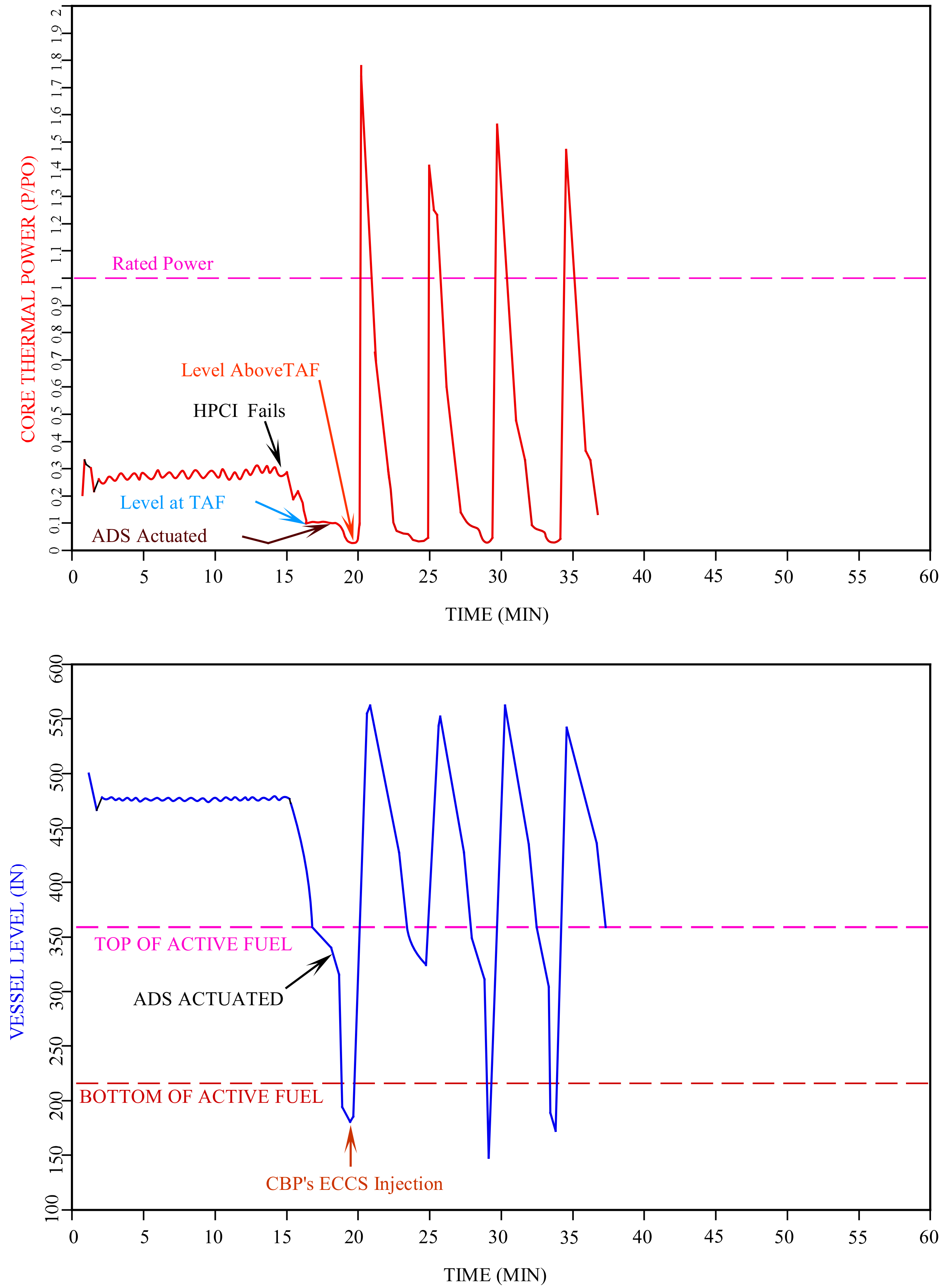

Figure 4.2-2 Reactor Vessel Level and Power

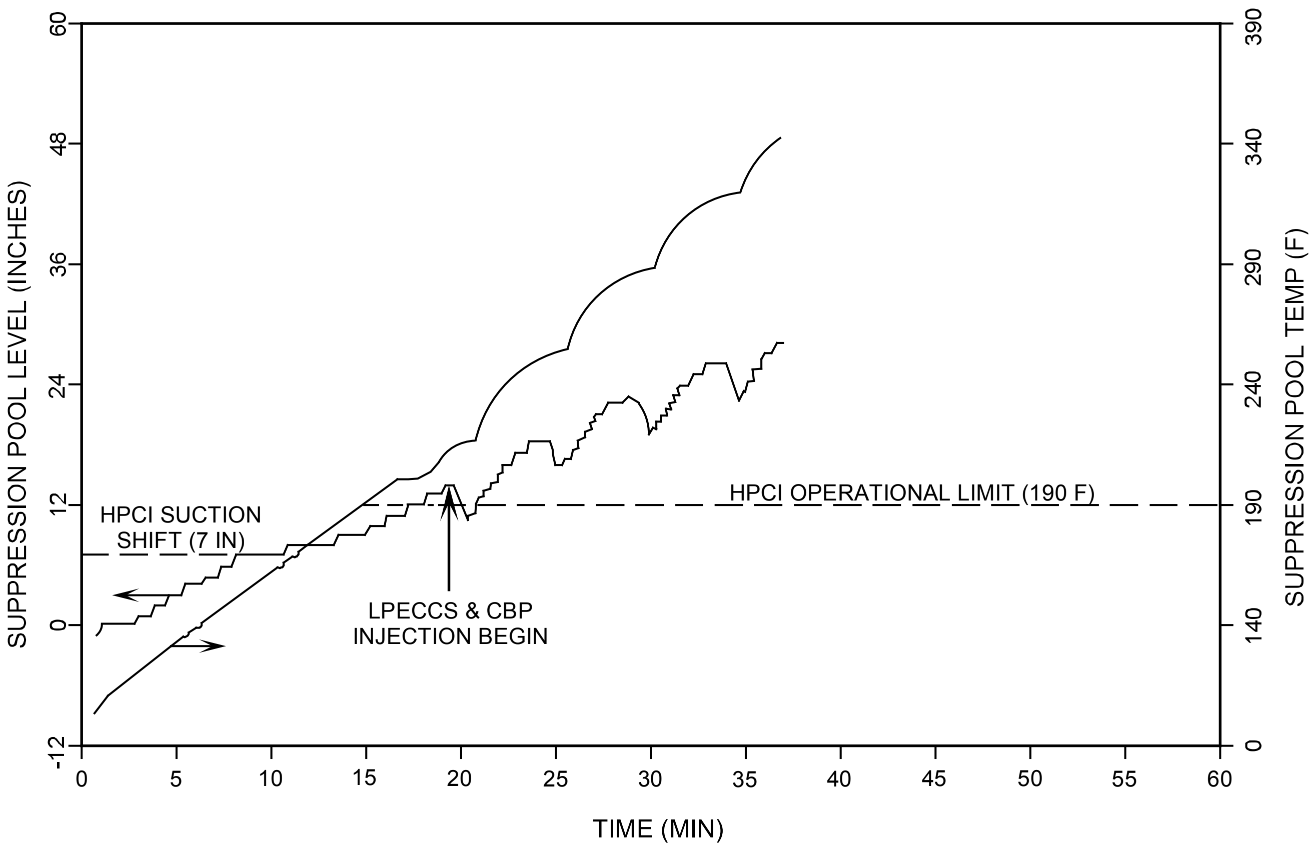

Figure 4.2-3 Suppression Pool Temperature and Level

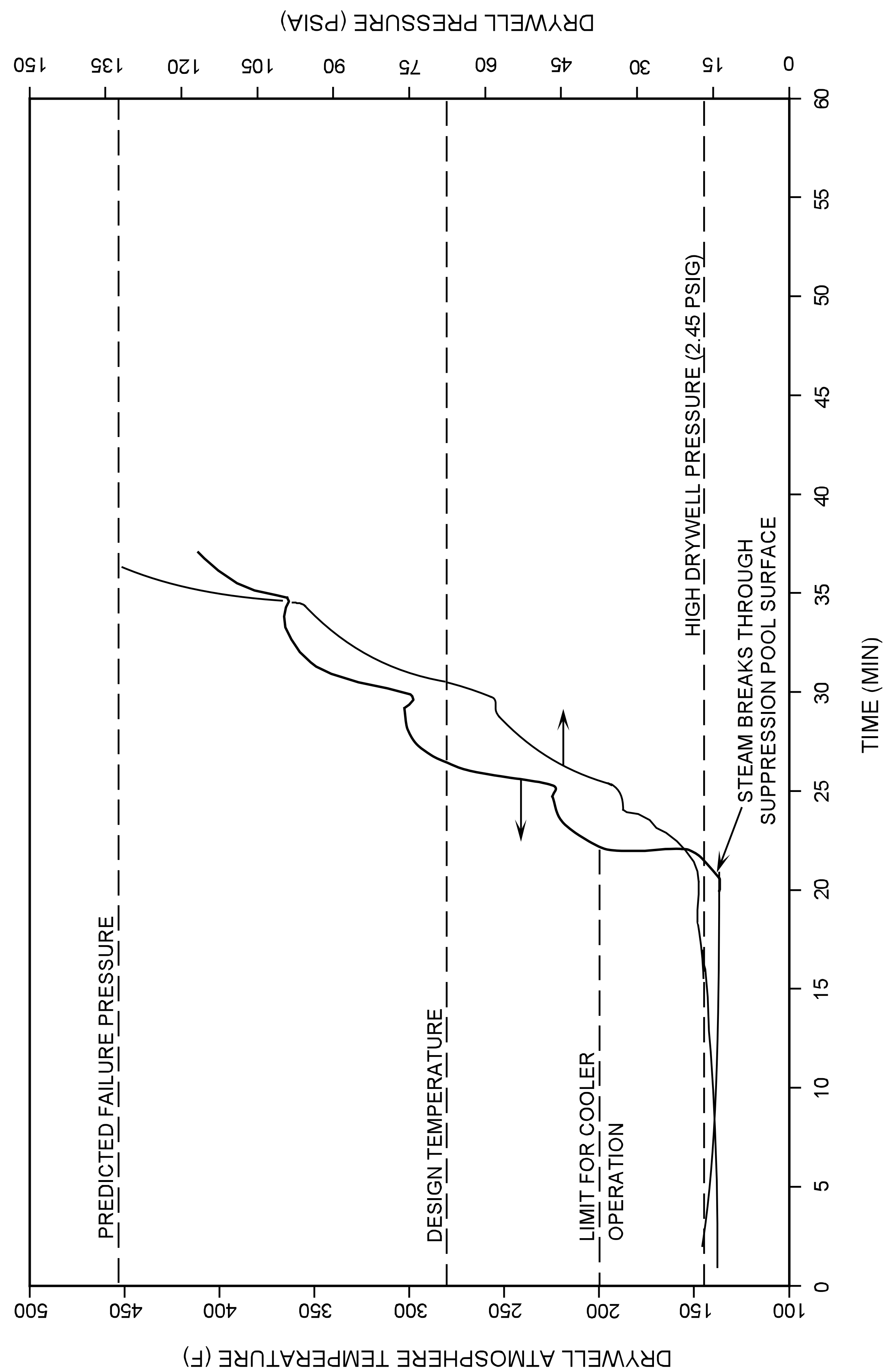

Figure 4.2-4 Drywell Pressure and Temperature

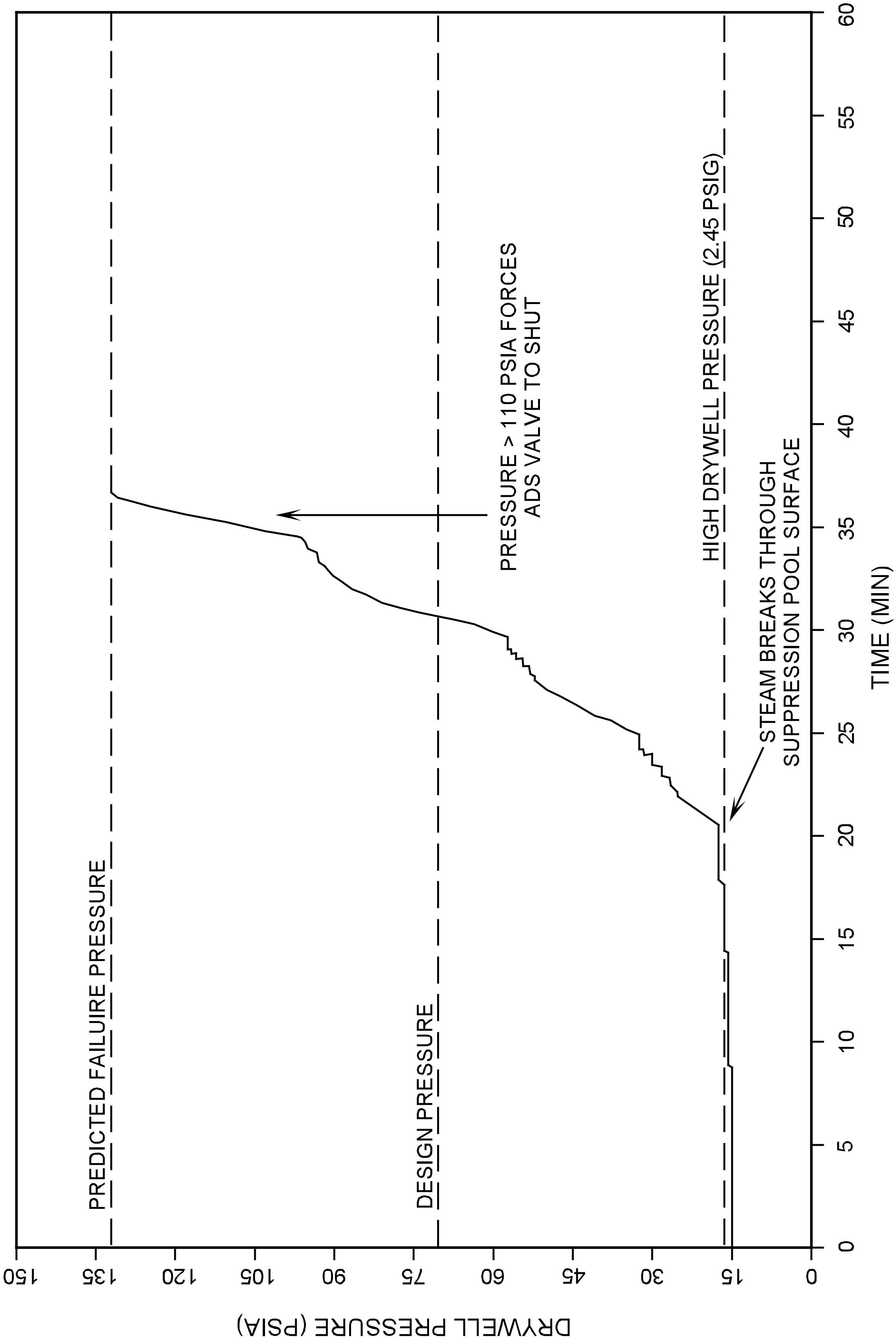

Figure 4.2-5 Drywell Pressure