General Electric Advanced Technology Manual

Chapter 4.2

Anticipated Transient Without Scram

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

4.2 ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

Learning Objectives:

1. Define anticipated transient without scram.

2. Explain the expected plant response for the worst ATWS event.

3. List the scram signals received during the initial ATWS event.

4. Explain why an ATWS event is safety significant.

5. List various ways to limit core power during an ATWS event.

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 would 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 should not result, then an ATWS event would have occurred.

4.2.2 History

ATWS became a possible source of concern in nuclear power plants in 1968 during discussions between 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 and some design changes, 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 as techniques to analyze a system for random failures.

4.2.2.1 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.

4.2.2.2 1970

After analyzing vender supplied information it was concluded that several anticipated transients in BWRs would require prompt action to shutdown the reactor in order to avoid serious plant damage and possible offsite release. The resulting list of transients considered for boiling water reactor plants is as follows:

- Primary Pressure Increase

  These transients include loss of load events such as generator trip, turbine trip, and loss of condenser vacuum. Also considered are such transients as closure of one or all of the main steam line isolation valves and malfunction of the reactor primary system pressure regulator, causing increasing pressure.

- Reactor Water Inventory Decrease

  These transients include events leading to a decrease in the inventory of reactor primary coolant such as loss of auxiliary power, loss of feedwater, pressure regulator failure in a direction to cause decreasing reactor system pressure, inadvertent opening of a safety or relief valve, and opening of condenser bypass valves.

- Reactor Coolant Flow Increase

  These transients include events that might increase the recirculation flow and thus induce a positive reactivity increment. They include a malfunction of the recirculation flow controller in a manner to cause increasing primary coolant flow and the start-up of a recirculation pump that had been on standby.

- Reactor Water Temperature Decrease

  These transients include events that might cause a power surge by reduction of the reactor primary coolant water temperature. They include malfunction of feedwater control in a direction to increase feedwater flow, loss of a feedwater heater, shutdown cooling malfunction, and inadvertent activation of auxiliary cold water systems.

- Reactivity Insertions

  These transients include control rod withdrawal transients from the zero reactor power, hot, critical condition and from full power; fuel assembly insertion; control rod removal; and control curtain removal errors during refueling.

- Reactor Coolant Flow Decrease

  These transients include failure of one or more recirculation pumps or malfunction of the recirculation flow control in a direction to cause decreasing flow.

The transients having the greatest potential for significant damage are those leading to a reactor primary coolant system pressure increase. The most severe of these are the loss of condenser vacuum and the closure of all main steam isolation valves. A loss of condenser vacuum causes automatic closure of the turbine stop valves and the turbine bypass valves. The turbine stop valves are fast acting valves, so that there is an abrupt interruption of steam flow from the reactor. The main steam isolation valves are slower in closing, but in this case the large steam line volume is not available to buffer the pressure rise. The result in either case would be 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. The other transients that lead to primary system pressure increase are less severe. Generator or turbine trips are less severe because the turbine bypass valves can be assumed to open and the condenser to be operative. Although the transient proceeds more slowly in these cases, the result still would be an excessively high reactor coolant system pressure.

4.2.2.3 1971

The ACRS and the regulatory staff concluded that a design change to the proposed Newbold Island (now Hope Creek) BWR/4 (Public Service of New Jersey) was appropriate to limit the possible consequences of ATWS. The same design change was applied to other BWR/4s. The design change consisted of tripping of the recirculation pumps.

4.2.2.4 1972

The ACRS recognizes ATWS as a low probability event. Nevertheless, it believed that, in consideration of the large number of BWRs expected eventually to be in operation, and in view of the expected occurrence rate of anticipated transients, experience with scram systems of current design is insufficient to give assurance of an adequately low probability for an ATWS event with possible serious consequences. Accordingly a set of positions and actions is implemented and was published as WASH-1270.

4.2.2.5 1973

The regulatory staff amends licensing position setting October 1, 1973 as the effective date of the position. Analyses for older operating plants should be provided by October 1, 1974, and the need for any changes would be considered on a case-by-case basis. Plants recently started in operation, now under construction, or for which applications for construction permits are filed before October 1, 1976, should have any equipment provided and any changes made that are necessary to make the consequences of ATWS acceptable. Analyses of the effects of ATWS and plans and schedules for any changes found necessary should be provided for these plants by October 1, 1974, or at the time of submission of an application for a construction permit, whichever is later. Plants for which applications for construction permits are filed after October 1, 1976, should have improvements in the protection system design that make an ATWS event negligibly small.

Applicants should be required to:

- demonstrate that with their present designs the consequences of anticipated transients without scram (ATWS) are acceptable, or

- make design changes which render the consequences of anticipated transients without scram acceptable, or

- make design changes to improve significantly the reliability of the scram system.

It is necessary to establish acceptable consequences of ATWS in order to implement either option 1 or option 2 of the recommended position. Acceptable conditions are defined as follows:

- Radiological Consequences

  The radiological consequences shall be within the guideline values set forth in 10 CFR Part 100.

- Primary System Pressure

  The maximum acceptable transient primary system pressure shall be based on the primary system pressure boundary limit or the fuel element limit whichever is more restrictive. Primary pressure boundary limits transient pressure shall be limited to less than that resulting in a maximum stress anywhere in the reactor coolant pressure boundary of the "emergency conditions" as defined in the ASME Section II Nuclear Power Plant Components Code.

  Fuel pressure limits transient pressure shall not exceed a value for which test and/or analysis demonstrate that there is no substantial safety problem with the fuel.

- Fuel Thermal and Hydraulic Effects

  The increase in fuel enthalpy shall not result in significant cladding degradation or in significant melting of fuel even in the hottest fuel zones.

- Containment Conditions

  Calculated containment pressures shall not exceed the design pressure of the containment structure. Equipment which is located within the containment and which is relied upon to mitigate the consequences of ATWS shall be qualified by testing in the combined pressure, temperature and humidity environment conservatively predicted to occur during the course of the event.

- Analyses of Possible Detrimental Effects of Required Modifications

  Any modifications made to comply with option 2 of the recommended position shall be shown not to result in violations of safety criteria for steady state, transient, or accident conditions and shall not substantially affect the operation of safety related systems.

- Diversity Requirement for Implementing Option 2 of the Recommended Position

  Design changes to make the consequences of ATWS acceptable should not rely on equipment or system designs which have a failure mode common with the scram system. The equipment involved in the design change shall, to the extent practical, operate on a different principle from equipment in the scram system. As an absolute minimum, the equipment relied on to render acceptable the consequences of the ATWS event shall not include equipment identical to equipment in the associated scram system.

- Diversity Requirement for implementing Option 3 of the Recommended Position

  Improvements must reduce considerably the potential for common mode failure of the scram system. Failures of identical equipment from a common mode should not disable sensing circuits, logic, actuator circuits or control rods to the extent that scram is ineffective. The addition of a separate protection system utilizing principles diverse from the primary protection system is indicated in order to meet this requirement.

4.2.2.6 1974

Reactor vendors submitted analyses on ATWS in general response to the following requirements set forth in WASH-1270:

- Trip of the reactor recirculation pump upon high reactor vessel pressure or low low reactor water level.

- Logic for automatic initiation of the liquid control system.

- Add piping to supply some of the liquid control flow through the HPCI system.

4.2.2.7 1975

ATWS is almost resolved. With the issuance of WASH-1400 (Assessment of Accident Risks), reactor vendors turned to the results which demonstrated that ATWS was not a major contributor to the risk from LWRs and as such no modifications are required.

4.2.2.8 1976

ATWS remained a controversial issue between the NRC and the industry.

4.2.2.9 1977

NRC formed a task force on ATWS in an effort to finally resolve the matter. The report sent to ACRS, reiterated the general position of scram unreliability which could not be shown to be acceptable low and measures were required to mitigate the consequences of ATWS. The year 1977 passed without issuance of a new NRC position on ATWS.

4.2.2.10 1978

The NRC issues NUREG-0460 (Anticipated Transients Without Scram for Light Water Reactors). The NUREG includes 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.

4.2.2.11 1979

The TMI accident forced deferral of all NRC work on ATWS and most industry work was halted or delayed as well.

4.2.2.12 1981

Proposed rules filed in federal register Vol. 46, No. 226:

Rule #1

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., 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.

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

a. Addition of high capacity neutron poison injection systems.

Rule # 2

Proposed Hendrie Rule

The essence of the Hendrie rule is that power reactor licensees would be required to implement a reliable assurance program to seek out and rectify reliability deficiencies in those functions and systems that prevent or mitigate ATWS accidents.

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, 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 pressure 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, a oscillatory equilibrium will be maintained in which the reactor sustains the average power necessary to boil off however much reactor coolant 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. 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 HPCI is a single-train system.

The fault or human error that precipitates the initial transient might also disable the

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 reactor coolant replenishment systems, either of which may be expected to be available under ATWS conditions, can successfully mitigate ATWS transients. The two trains may be the HPCI and RCIC.

Concern has been expressed that the 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 the 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 the 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.

4.2.4 10CFR 50.62 (3) (4)

The Code of Federal Regulations requires all BWRs to have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

Each BWR must have a standby liquid control system (SLC) with the capability of injection into the reactor vessel of a borated water solution at such 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 prior to 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 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.

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).

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 reactor vessel level of 476.5 inches.
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	Reactor vessel level 413.5 inches
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	Vessel water level 476.5 inches
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	Vessel level 413.5 inches
Level below TAF	6.8	Emergency system range, normal level indication off scale low
Vessel level 2/3 core height	9.5	
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

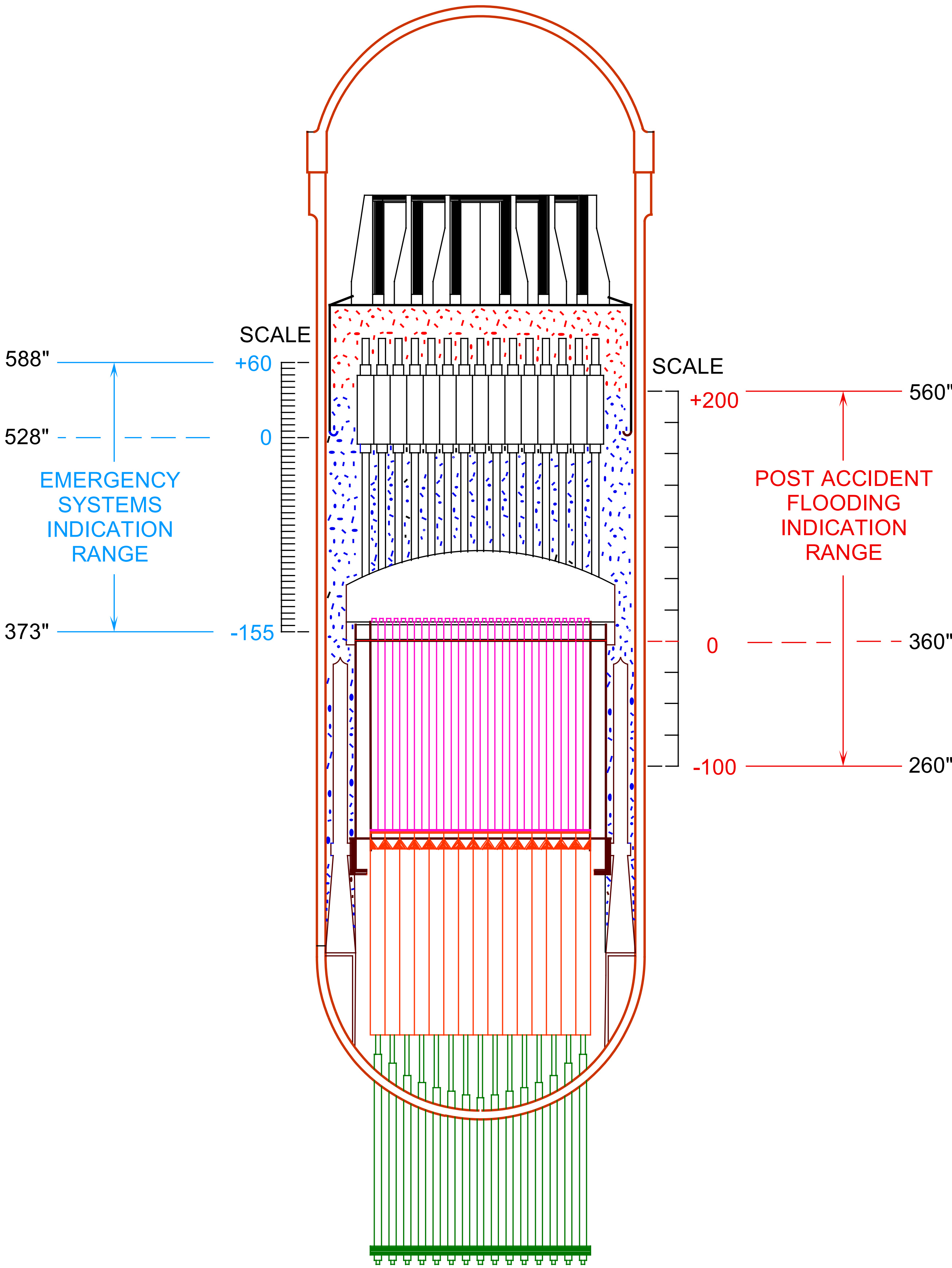

Figure 4.2-1 Level Instruments Available During ATWS Event

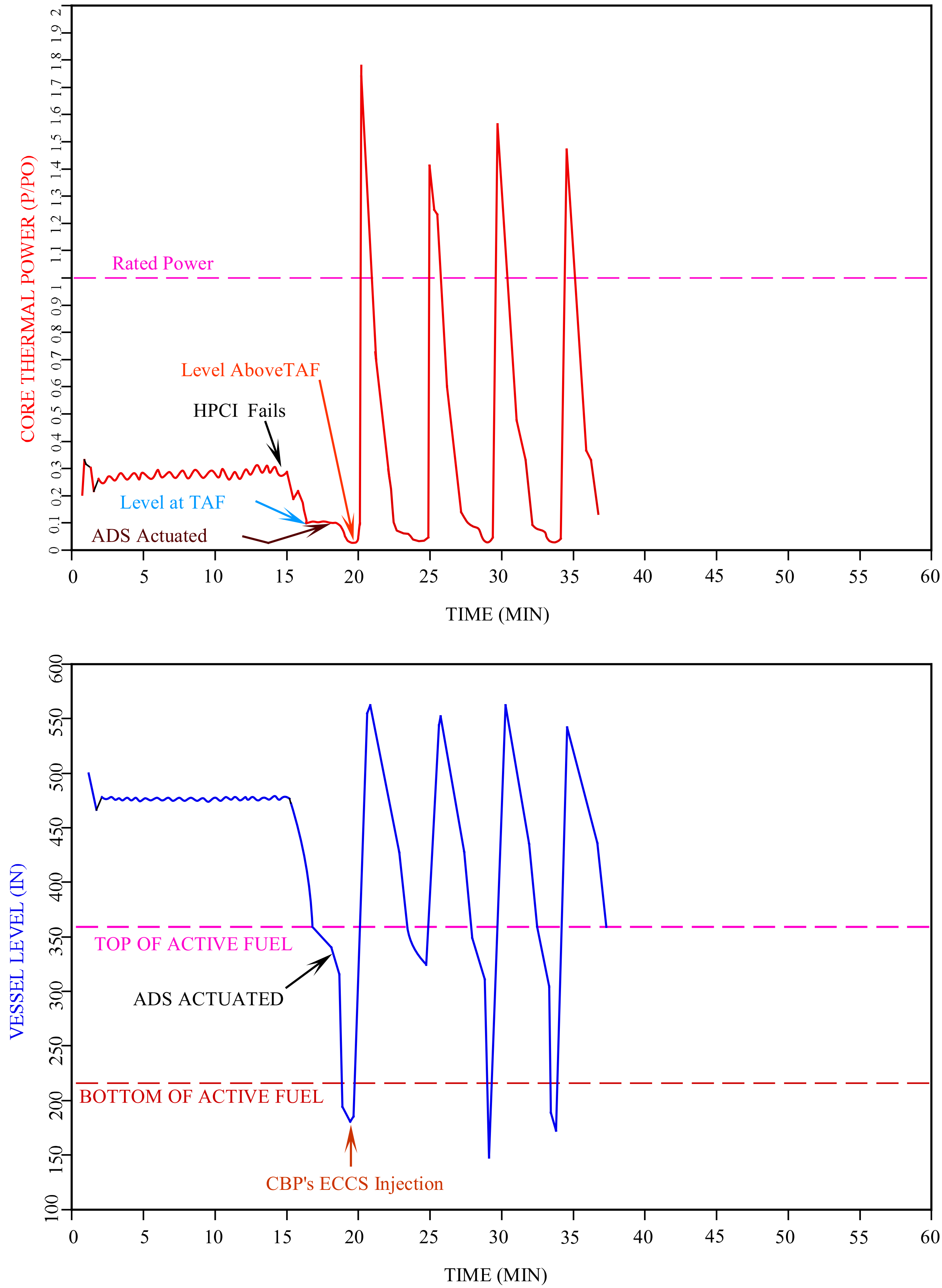

Figure 4.2-2 Reactor Vessel Level and Power

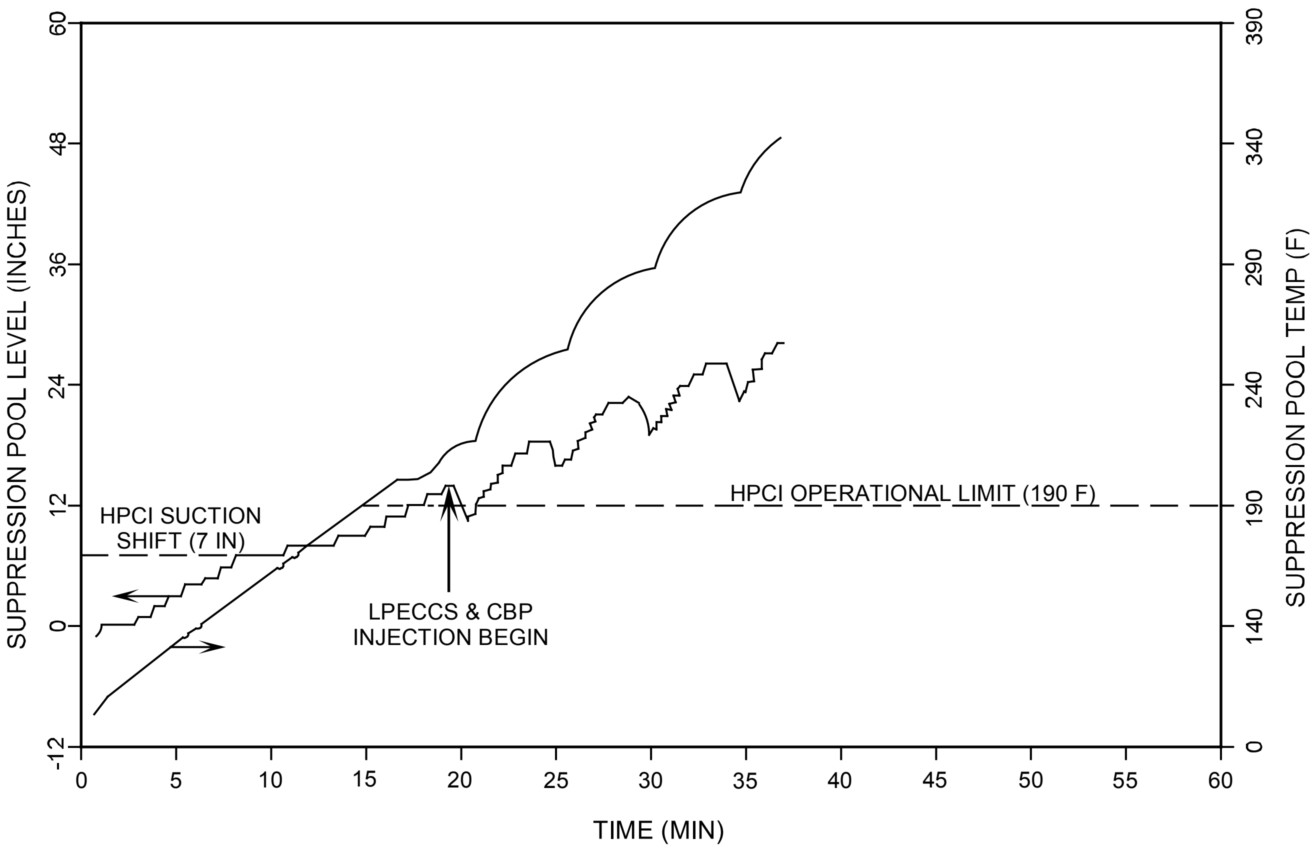

Figure 4.2-3 Suppression Pool Temperature and Level

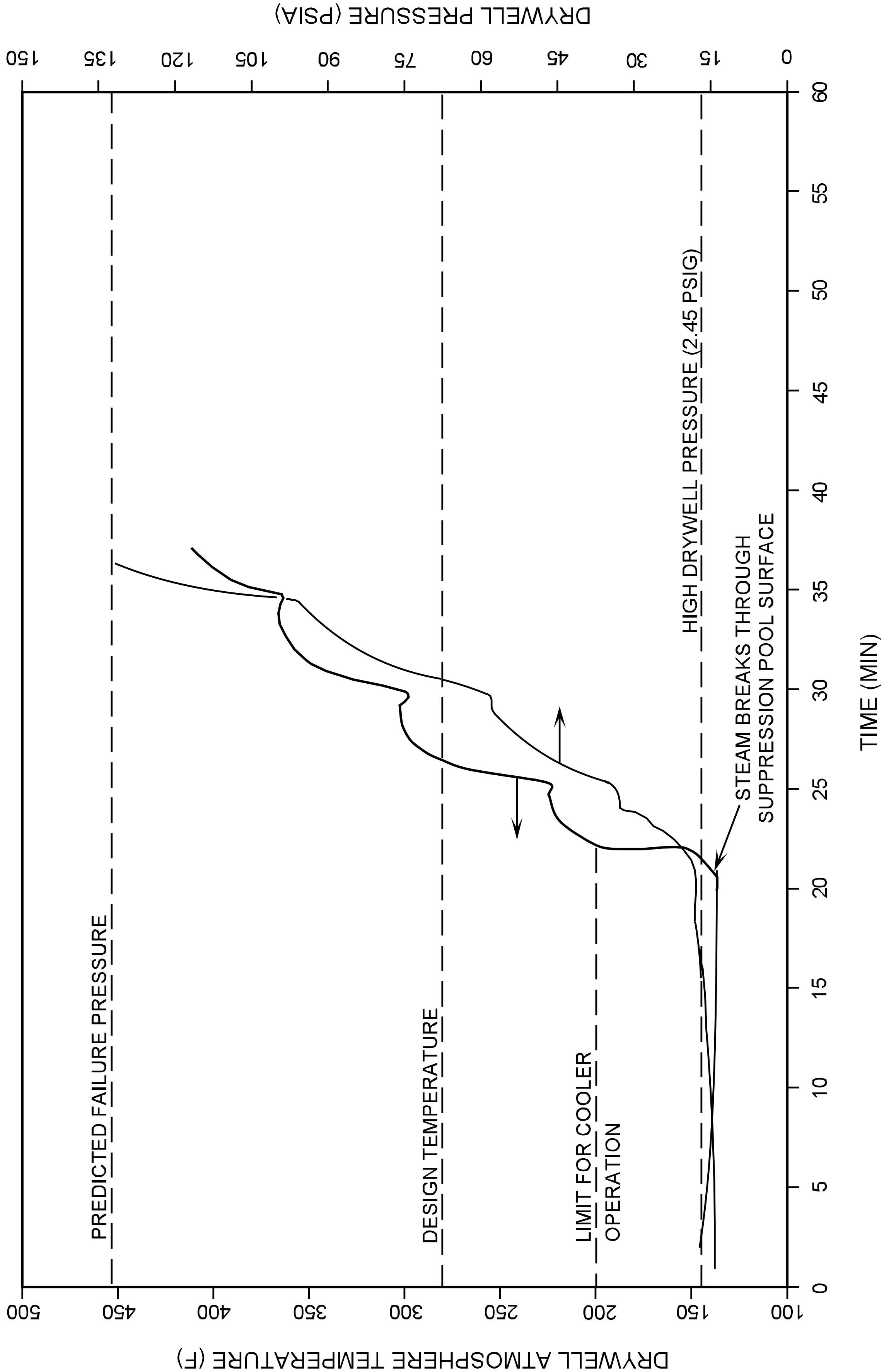

Figure 4.2-4 Drywell Pressure and Temperature

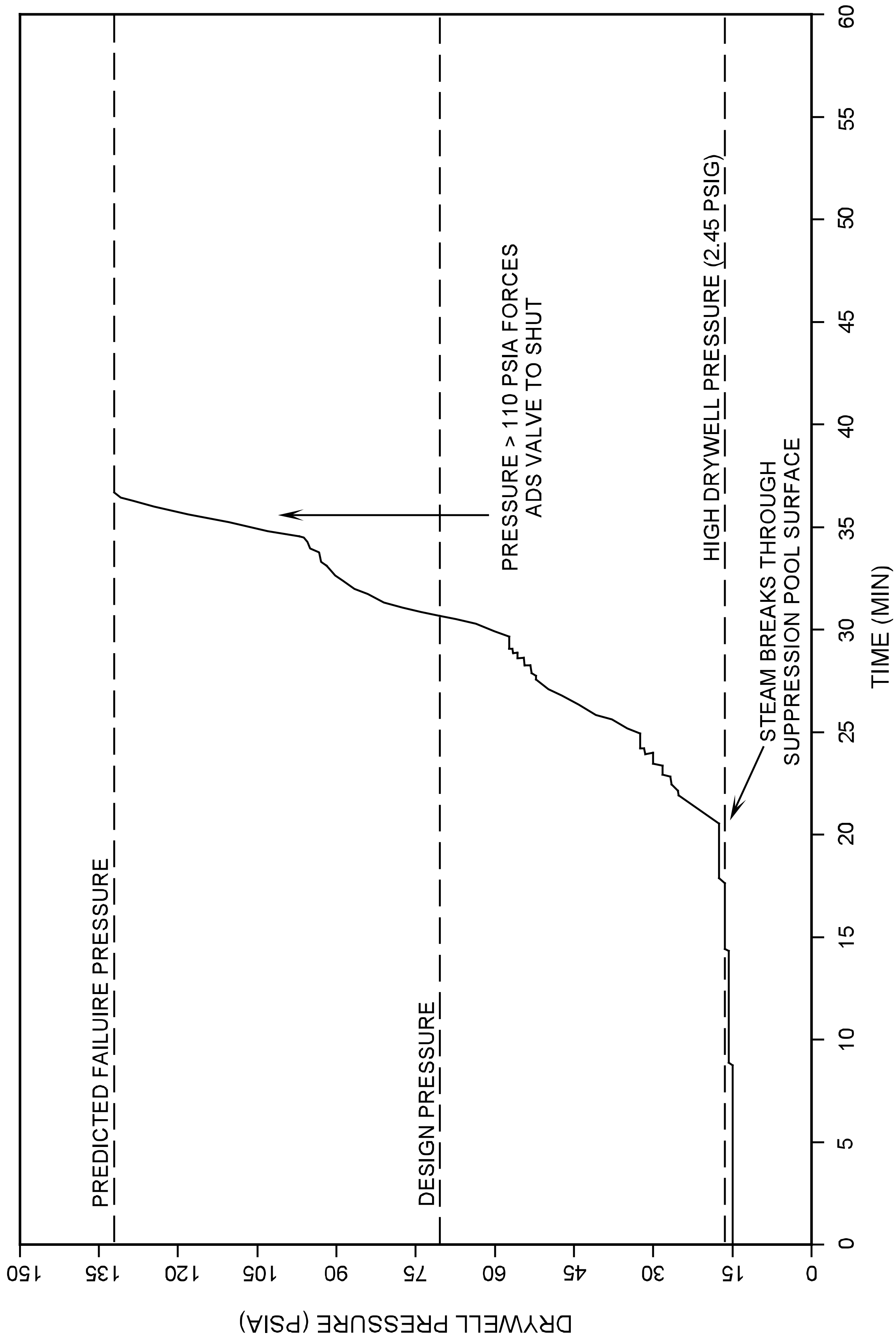

Figure 4.2-5 Drywell Pressure