ENCLOSURE 1

BACKGROUND INFORMATION FOR

WESTINGHOUSE EMERGENCY RESPONSE GUIDELINES (E-2)

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

BACKGROUND INFORMATION
FOR
WESTINGHOUSE OWNERS GROUP
EMERGENCY RESPONSE GUIDELINE

E-2
FAULTED STEAM GENERATOR ISOLATION

LP-Rev.1
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1. INTRODUCTION

Guideline E-2, FAULTED STEAM GENERATOR ISOLATION, provides actions to identify and isolate a faulted steam generator (SG). The guideline is entered from E-0, REATOR TRIP OR SAFETY INJECTION, or E-1, LOSS OF REACTOR OR SECONDARY COOLANT, when any SG pressure decreases in an uncontrolled manner or any SG completely depressurizes. Other guidelines have a transition to E-2 whenever a faulted SG is identified and faulted SG isolation is not verified. After taking the required actions in this guideline, the operator is directed to either E-1, LOSS OF REACTOR OR SECONDARY COOLANT or E-3, STEAM GENERATOR TUBE RUPTURE, depending on whether a SGTR is identified. If all SGs are determined to be faulted, the operator is directed to ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS.

2. DESCRIPTION

Guideline E-2, FAULTED STEAM GENERATOR ISOLATION, is intended to identify and isolate a loss of secondary coolant resulting from a fault in a main steamline, main feedwater line or in any piping system that interconnects with the secondary side pressure boundary (e.g., auxiliary feedwater system, blowdown piping). The consequences of a break in any of the above piping systems vary considerably depending upon several system parameters: initial power level; location of break; size of break; safety systems that are operational; control systems that are operational; and any failures that occur.

A summary review of the following three categories of faults will be presented in order to describe the variance in system transient characteristics an operator may encounter following a loss of secondarv coolant event:

1) Small secondary break
2) Intermediate size secondary break
3) Large secondary break

Small Secondary Break

For this category of breaks, normal plant control systems are capable of maintaining nominal or near nominal operating conditions. For a small steamline break, the system transient response would be similar to a step load increase. The secondary system would indicate an increase in load with a resultant decrease in primary system average temperature and pressure. The control rods would withdraw from the core in an effort to restore the primary average temperature if the rod control system was in an automatic mode of operation. Due to the apparent increased load, the steam flow from the steam generators would be increasing in at least one loop, depending upon the location of the break. If the break occurred in the steam header, all loops would experience increased steam flow. Due to the increased steam flow, the feedwater control valves would modulate to a more open position in an attempt to maintain steam generator water level. As a result, the main feed flow in at least one loop (all loops if break is in steam header) would be increased. Another indication of this type of break would be a decreasing water level in the condenser hotwells.

Similar system characteristics would be obtained if a small feedline break occurred such that the normal plant control systems could maintain near nominal operating conditions. For this break, the feedwater control valve in at least one loop, depending upon the location of the break, would modulate to a more open position in an attempt to compensate for the flow out of the break and to maintain steam generator water level. The steam flow in all other loops would remain approximately normal. Again the water level in the condenser hot wells would decrease slowly.

For either of the above scenarios, a containment temperature and/or pressure increase may be observed if the break occurred inside containment. If the break was outside containment, an audible or visual confirmation of the break may be possible.

For this size break it is not expected that an automatic reactor trip or safety injection would occur and, therefore, guideline E-2 would not be implemented.

Intermediate Size Secondary Break

An intermediate size break is lower bounded by those sizes in which the normal plant control systems are unable to maintain approximate nominal plant operating conditions and upper bounded by those sizes in which the protective functions do not occur within approximately five minutes following initiation of the event.

The intermediate steamline break is categorized by a slowly decreasing steamline pressure in at least one loop depending upon the location of the break. If the break occurs in the steam header, all loops will experience decreasing pressure. Due to the increased steam load for which the control systems are unable to compensate, a slowly decreasing steam generator water level will result and also a slowly decreasing primary average temperature. The control rods will commence stepping out of the core in an attempt to maintain nominal primary system average temperature.

However, due to the decreasing primary temperature, a primary pressure decrease occurs. These trends will continue until such a time that the operator manually trips the reactor or until a low steamline pressure setpoint or a low pressurizer pressure setpoint is reached. In either case, a safety injection signal and a reactor trip signal are generated. This results in a subsequent turbine trip, feedwater isolation, steamline isolation and auxiliary feedwater initiation. If the break occurs upstream of the main steamline isolation valves (MSIVs), the steam generator associated with the faulted loop will blow down to atmospheric pressure. If the break occurs downstream of the steamline isolation valves, the transient is terminated following MSIV closure. The system process parameter trends that are used to identify a faulted SG are an uncontrolled pressure decrease in at least one steamline or a SG that is completely depressurized. Other symptoms include increased main feed flow to at least one steam generator, slowly decreasing primary average temperature and slowly decreasing steam generator water level in at least one steam generator.

Some of the system process parameters necessary to describe an intermediate size secondary break are plotted in Figures 1 through 4. In these figures, loop 1 is the faulted loop and loop 2 represents the intact loops. The transient data and analysis assumptions are identified in Table 1. The break is big enough to cause the primary system temperature to slowly decrease until a reactor trip and a subsequent turbine trip occur due to an overpower delta temperature condition at 23 seconds as shown in Figure 1. The reactor coolant system pressure continues to decrease as shown in Figure 2 and safety injection initiates at 55 seconds. Due to the decreasing primary temperature and pressure, the pressurizer level drops rapidly as shown in Figure 3 and eventually empties. Steam generator pressure, as shown in Figure 4, decreases in all loops initially until steamline isolation occurs at 56 seconds due to low steamline pressure. The steam generator associated with the faulted loop will blow down to atmospheric pressure. A detailed description of the subsequent recovery actions, including SI termination, are discussed in the background document for guideline E-1, LOSS OF REACTOR OR SECONDARY COOLANT.

Figure 1. CORE AVERAGE TEMPERATURE-INTERMEDIATE SIZE SECONDARY BREAK

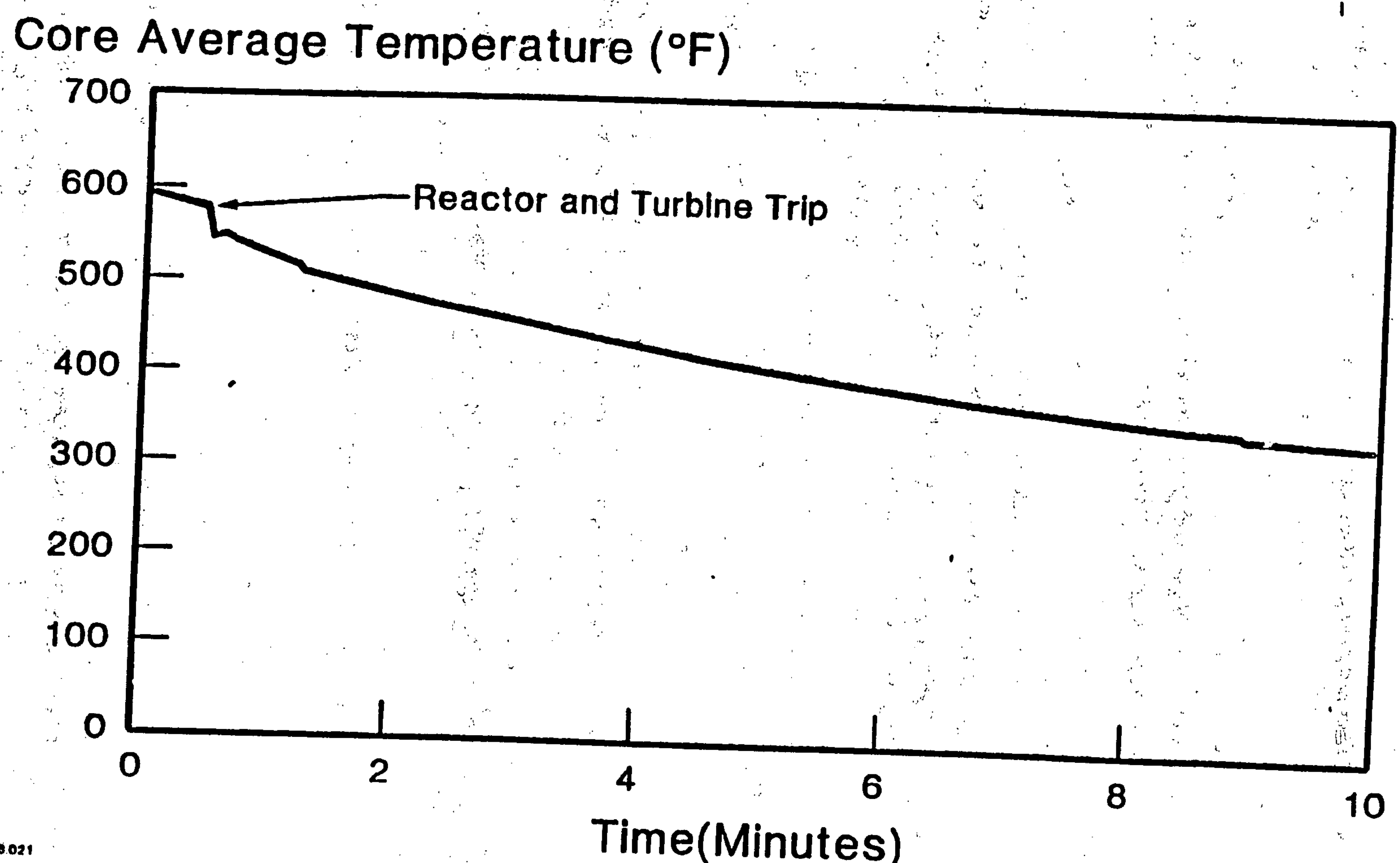

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Figure 2. RCS PRESSURE - INTERMEDIATE SIZE SECONDARY BREAK

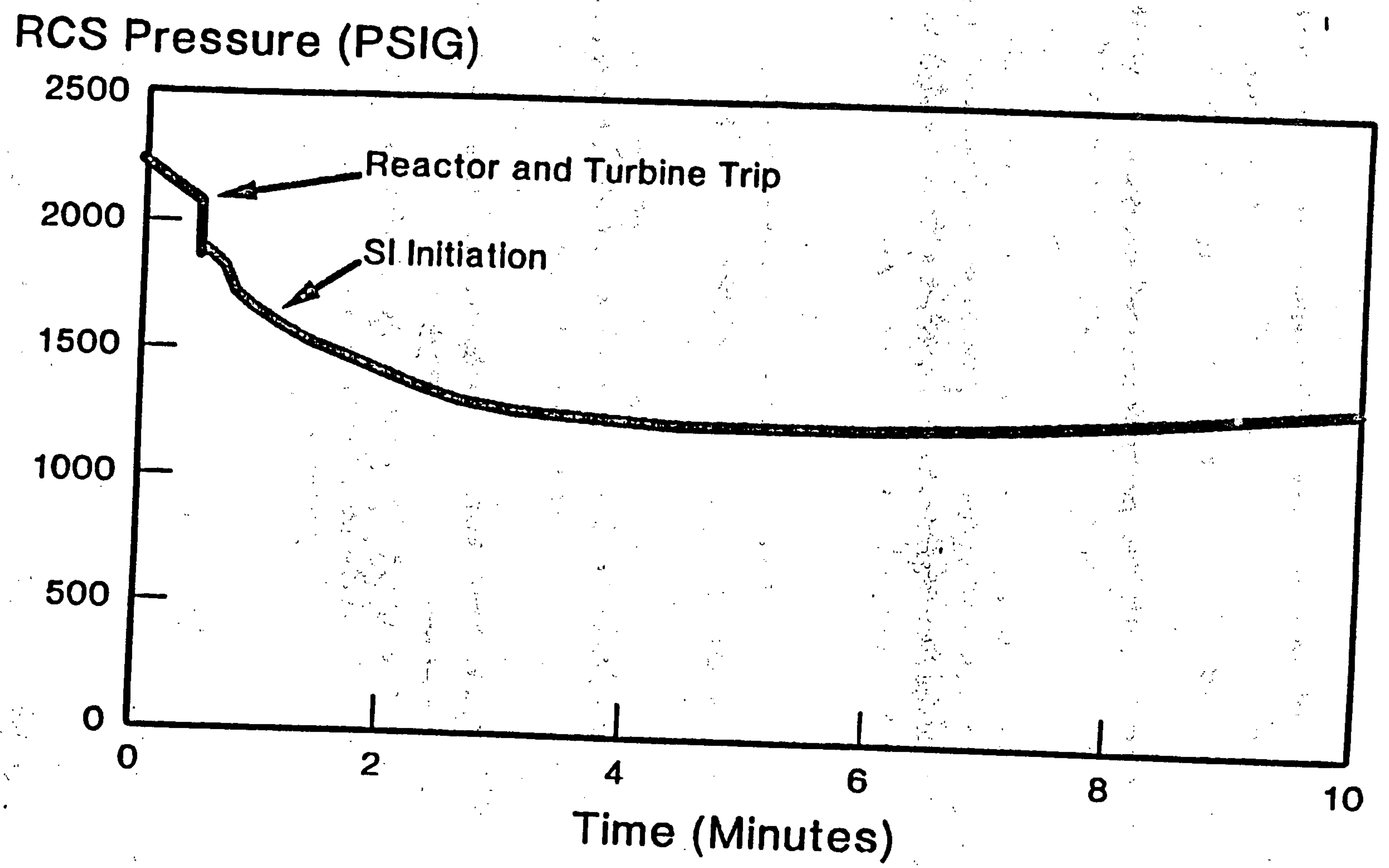

Figure 3. PRESSURIZER LEVEL - INTERMEDIATE SIZE SECONDARY BREAK

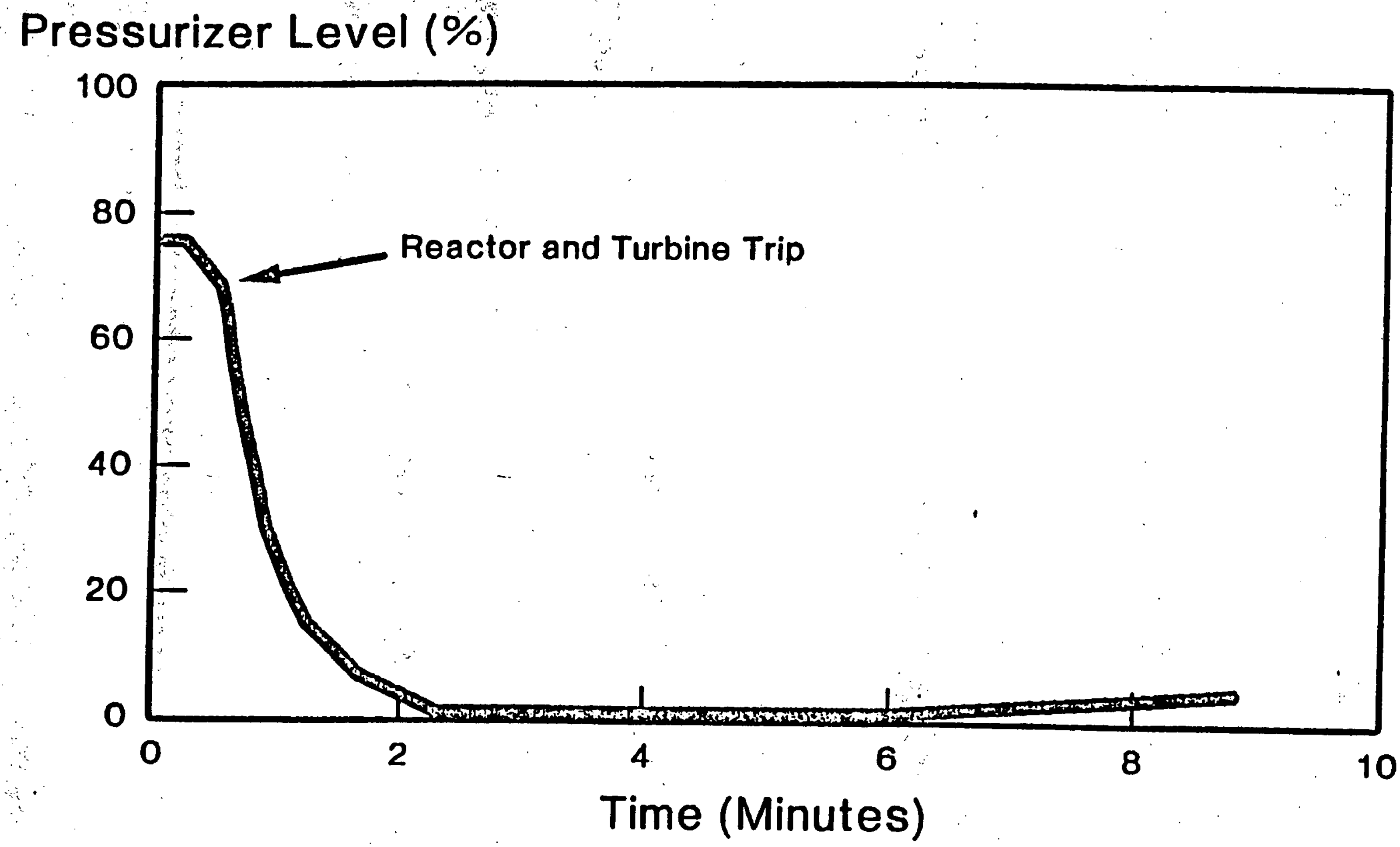

Figure 4. STEAM GENERATOR PRESSURE - INTERMEDIATE SIZE SECONDARY BREAK

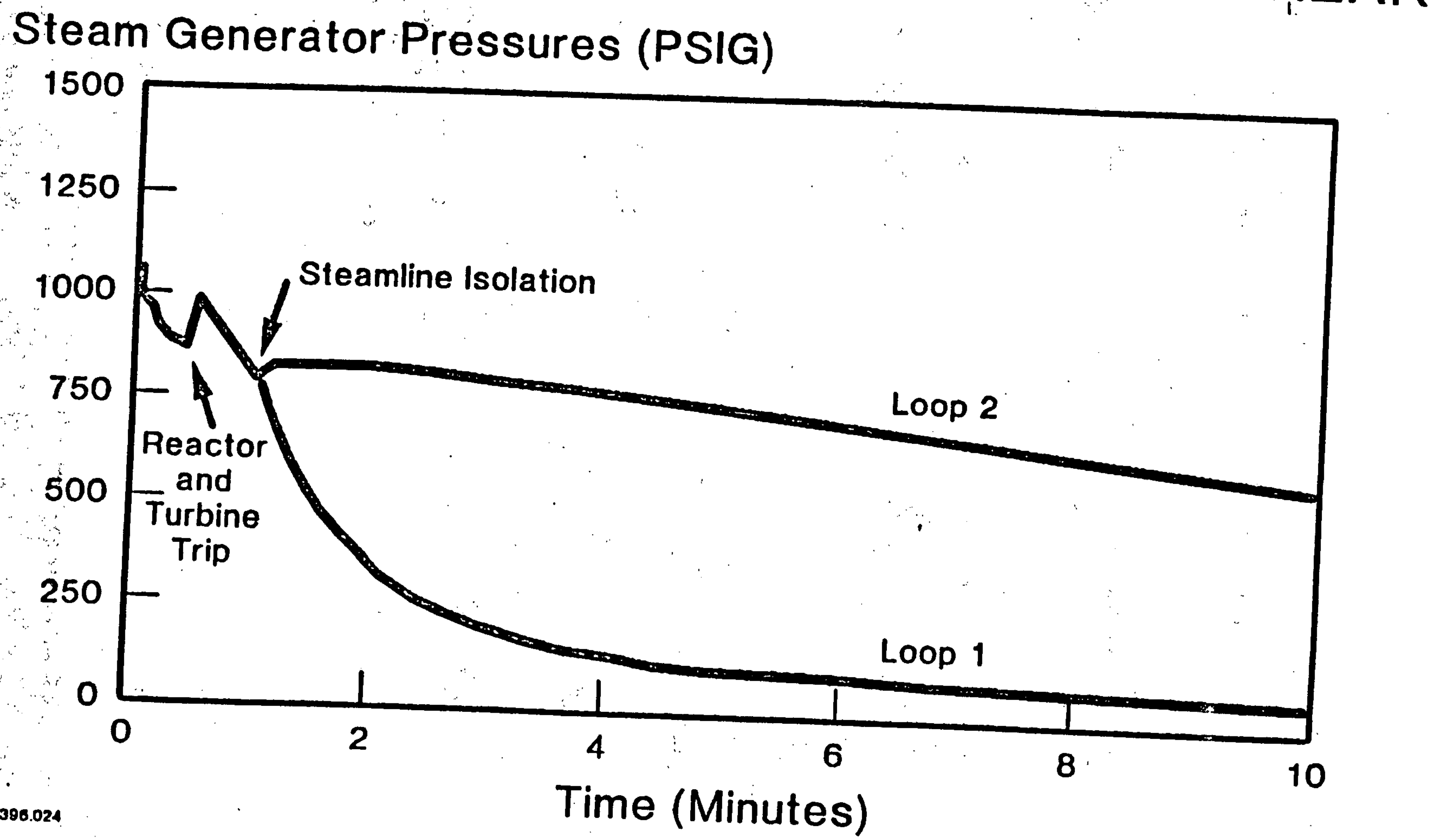

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TABLE 1

TRANSIENT DATA--INTERMEDIATE SIZE STEAMLINE BREAK

Parameters*	Data
Break Size (ft^2)	0.6
Initial Power Level (%)	100
Analysis Assumptions	Best Estimate
Trip Function	Overpower ΔT
Time Setpoint Reached (Sec)	23
SI Initiation Function	PRZR Pressure
Time Setpoint Reached (Sec)	55
Steamline Isolation Function	Steamline Pressure
Time Setpoint Reached (Sec)	56
AFW Termination Time (Min)	10
Limiting Parameter For SI Termination	PRZR Level RCS Pressure

* Typical 3-loop plant with Model D steam generators

For an intermediate feedline break in which the control systems are incapable of compensating for the loss of flow, the secondary side would experience a slowly decreasing steam generator water level in at least one steam generator. Depending upon the height of the low or low-low level trip setpoint in the steam generator and size of the break, a slowly increasing primary average temperature prior to reactor trip may occur due to the loss of main feedwater and degraded steam generator heat transfer. The transient is eventually terminated by manual reactor trip or when the low or low-low level trip setpoint is reached in any one steam generator. This results in a reactor trip and auxiliary feedwater initiation. A subsequent turbine trip occurs due to reactor trip. If the break occurs downstream of the main feedline non-return valves, all steam generators continue to experience a reverse blow down through the steam generator associated with the faulted loop until a low steamline pressure setpoint is attained resulting in a safety injection initiation and steamline and feedline isolation. The faulted steam generator will then blow down until atmospheric pressure is reached. If the break occurs upstream of the feedline non-return valves, the feedwater spillage is terminated and the auxiliary feedwater system is sufficient to mitigate the consequences of the resultant loss of normal feedwater transient. The system parameter trends that are used to identify a faulted SG are an uncontrolled pressure decrease in at least one steamline or a SG that is completely depressurized. Other symptoms include decreasing water level in at least one steam generator and slowly rising primary system average temperature prior to reactor trip.

For either of the above transients, if the break occurs inside containment, an increasing containment temperature and/or pressure indication could be observed. If the break occurs outside containment, audible or visual indications may assist the operator in diagnosing the transient.

Large Secondary Break

The least likely and most severe of the postulated loss of secondary coolant events is the double-ended break. (These are the transients that are generally presented in the applicant's Safety Analysis Report.)

For the double-ended main steamline break, an immediate decrease in pressure in at least one steamline occurs depending upon the location of the break. The low steamline pressure setpoint is reached (5-10 seconds) which results in safety injection initiation and steamline isolation. This yields a reactor trip, turbine trip, main feed isolation and auxiliary feedwater initiation. If the break is downstream of the MSIVs, closure of the MSIVs may terminate the blowdown to atmospheric pressure. In coincidence with rapidly decreasing steamline pressure, the primary experiences a decreasing average coolant temperature and decreasing primary pressure. The primary system transient following reactor trip and safety injection initiation is greatly dependent upon the initial power level prior to transient initiation. For a double-ended steamline break from low power levels, the primary average coolant temperature decreases below no load temperature and primary pressure initially decreases below the pressurizer pressure safety injection initiation setpoint until the high-head safety injection (HHSI) pumps begin repressurizing the primary side and restoring primary inventory. For a double-ended break from full initial power, the primary average temperature and pressure will initially decrease below no load temperature and pressurizer pressure safety injection initiation setpoint, respectively, after which decay heat generated in the core will immediately begin restoring primary temperature and pressure. The important system parameter trends for this break are an uncontrolled pressure decrease in at least one steamline or a SG that is completely depressurized, other symptoms include decreasing steam generator water level in at least one steam generator and initially decreasing primary pressure and temperature.

The double-ended main feedline break exhibits characteristics quite similar to the double-ended steamline break. The system response is characterized by a rapid decrease in steam generator water level in at least one steam generator. After the low or low-low water level setpoint is reached in one steam generator, a reactor trip occurs and auxiliary feedwater is initiated. A subsequent turbine trip is generated. Following reactor trip, all steam generators exhibit reverse blowdown through the faulted feedline until the low steamline pressure setpoint is reached in any steamline resulting in steamline and feedline isolation and safety injection initiation. Following MSIV closure, one steam generator blows down to atmospheric pressure. During this time period, the primary experiences a cooldown since the heat removal capability of the secondary inventory exceeds the decay heat generated in the core. A rapid decrease in primary temperature and pressure results. The trend of the remainder of the transient is highly dependent upon the initial power level of the system prior to transient initiation. For a double-ended feedline break initially from full power, the primary average temperature and pressure begin increasing whenever the secondary inventory is not capable of removing all decay heat generated in the core. The primary temperature increases until the auxiliary feedwater flow heat removal capability exceeds the decay heat generated in the core. No voiding occurs in the primary if minimum auxiliary feedwater flow requirements are met following a feedline break. If the feedline break is initiated during low power operation, the primary system heatup is minimized following the initial system cooldown. As a result, primary temperature will remain at or below no load temperature.

Typical transient responses for a double-ended feedline break are presented in Figures 5 through 9. Table 2 identifies the transient data and assumptions. The plots represent a 3-loop plant with a feedline break in loop 1. The reactor trips on steam generator low level at 11.9 seconds and auxiliary feedwater is initiated. Safety injection is initiated at 32.7 seconds on a steamline low pressure signal. Figures 5 and 6 show the T_{cold}, T_{hot}, and T_{sat}, temperatures of the faulted loop and the intact loops, respectively. Steamline isolation occurs at 39.7 seconds and, following MSIV closure, the loop 1 SG blows down to atmospheric pressure as shown in Figure 7. Primary system temperature and pressure rapidly decrease until MSIV closure as shown in Figures 5, 6 and 8. Pressurizer level also decreases due to the primary cooldown and begins to recover and stabilize after MSIV closure as shown in Figure 9.

Figure 5. LOOP 1 TEMPERATURE-LARGE FEEDLINE BREAK

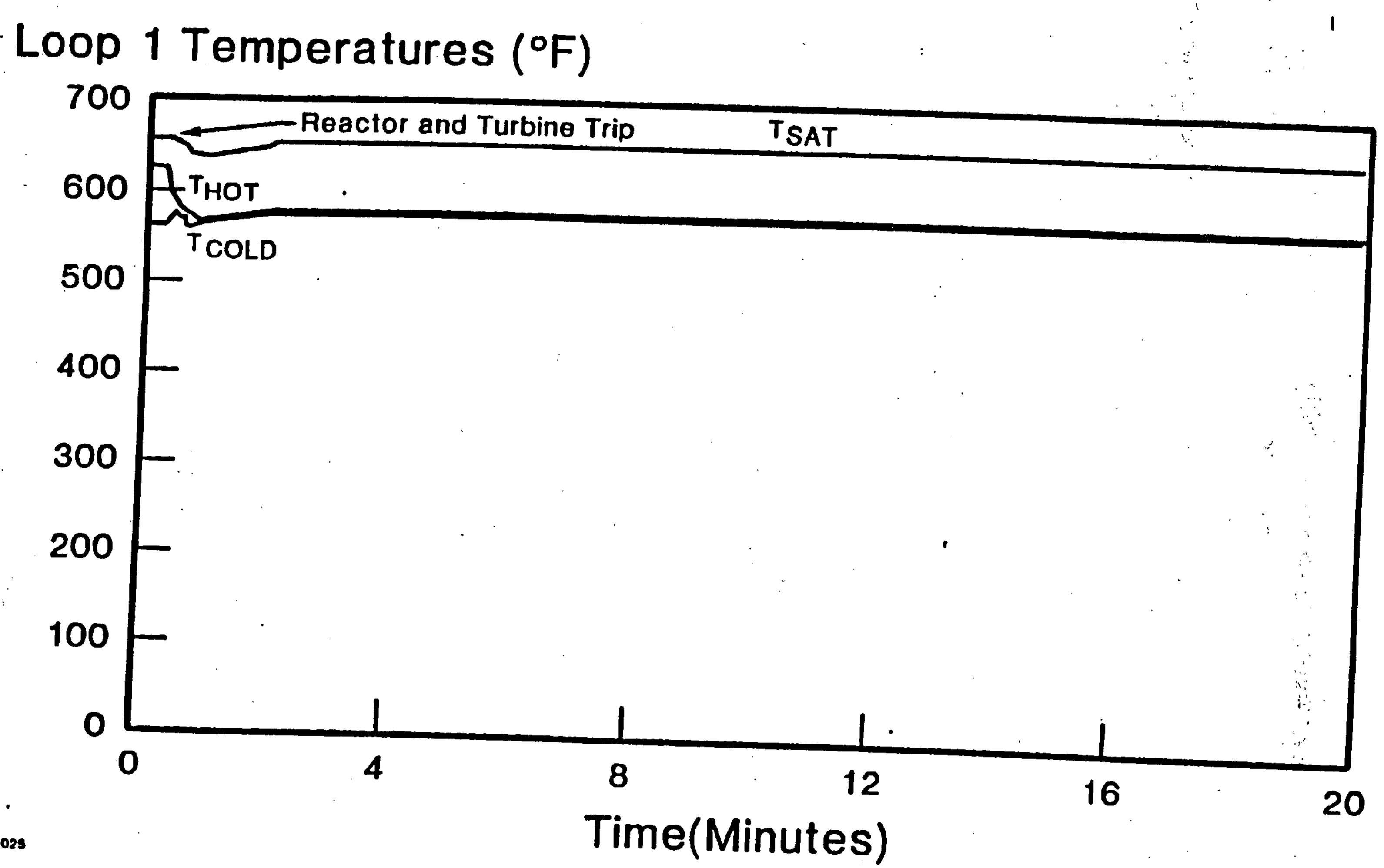

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Figure 6. LOOP 2 TEMPERATURE-LARGE FEEDLINE BREAK

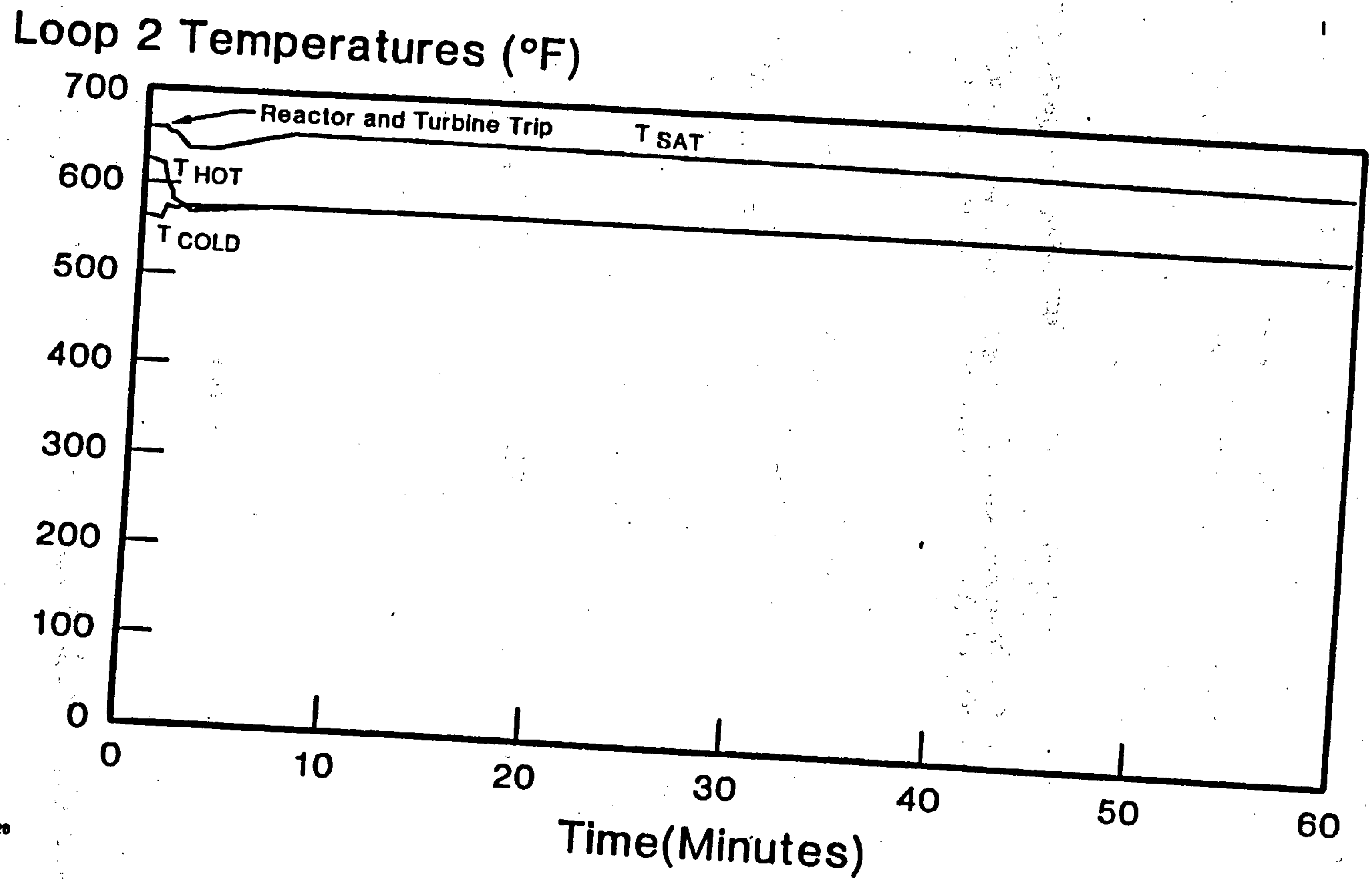

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Figure 7. STEAM GENERATOR PRESSURE - LARGE FEEDLINE BREAK

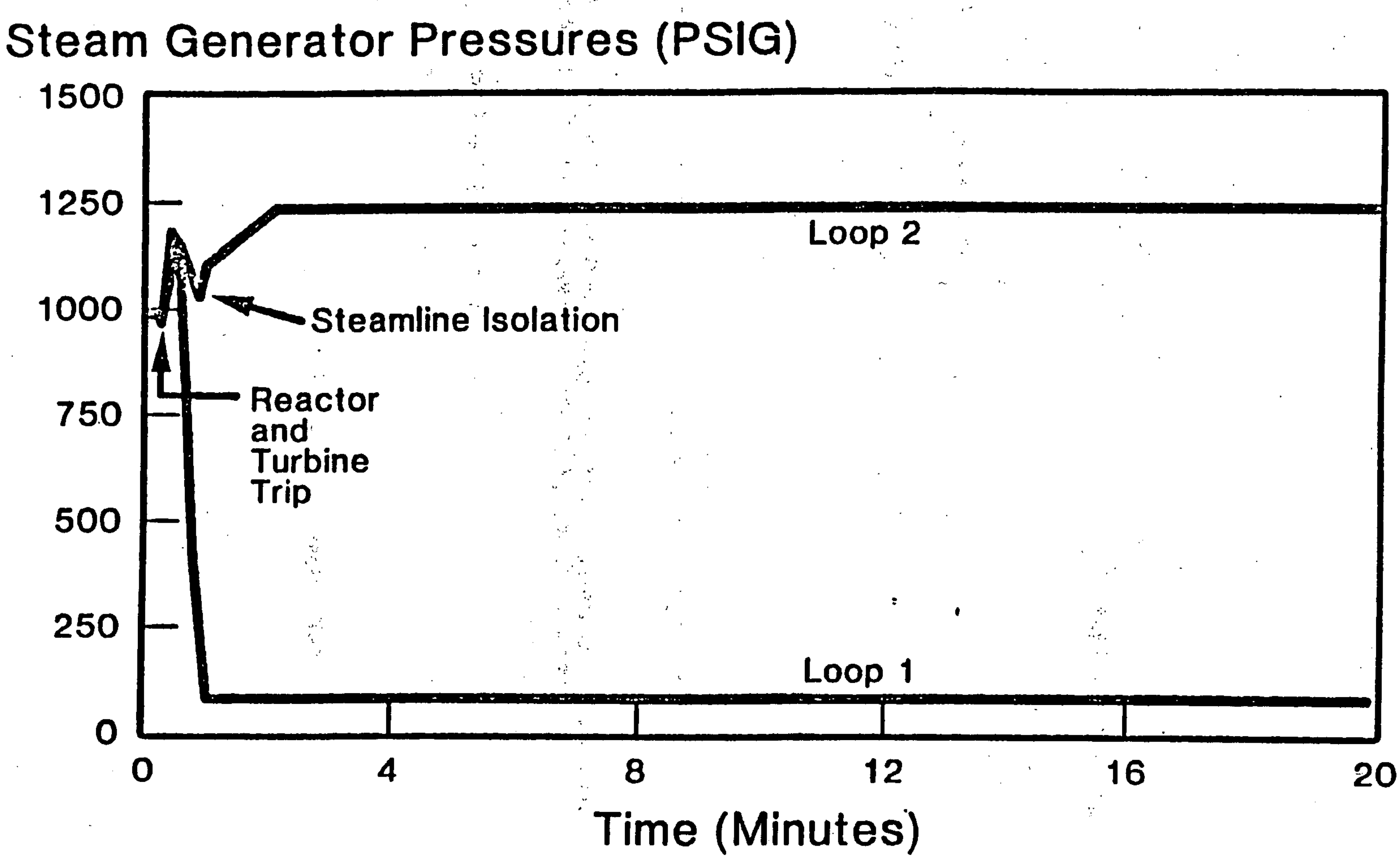

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Figure 8. RCS PRESSURE - LARGE FEEDLINE BREAK

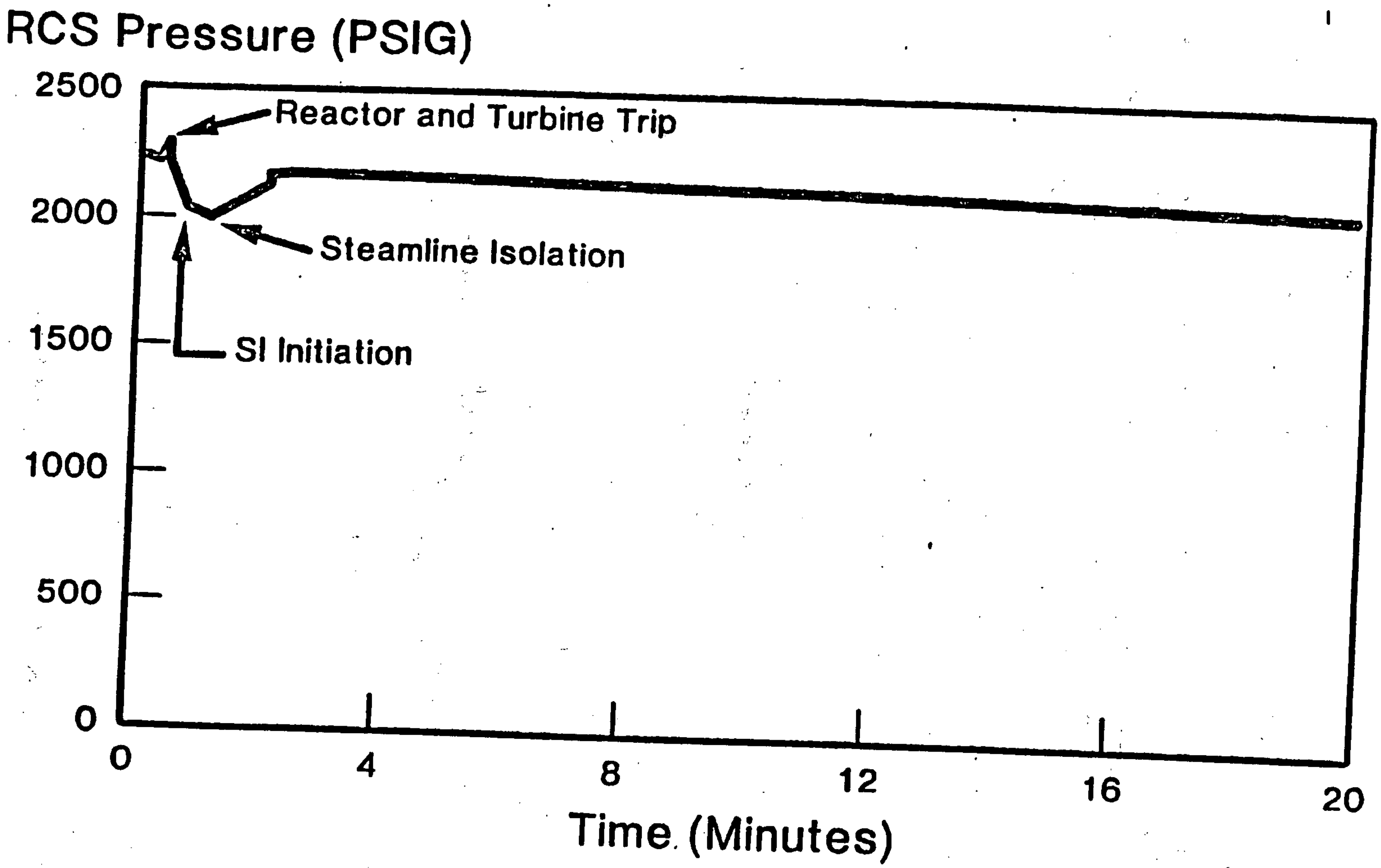

Figure 9. PRESSURIZER LEVEL - LARGE FEEDLINE BREAK

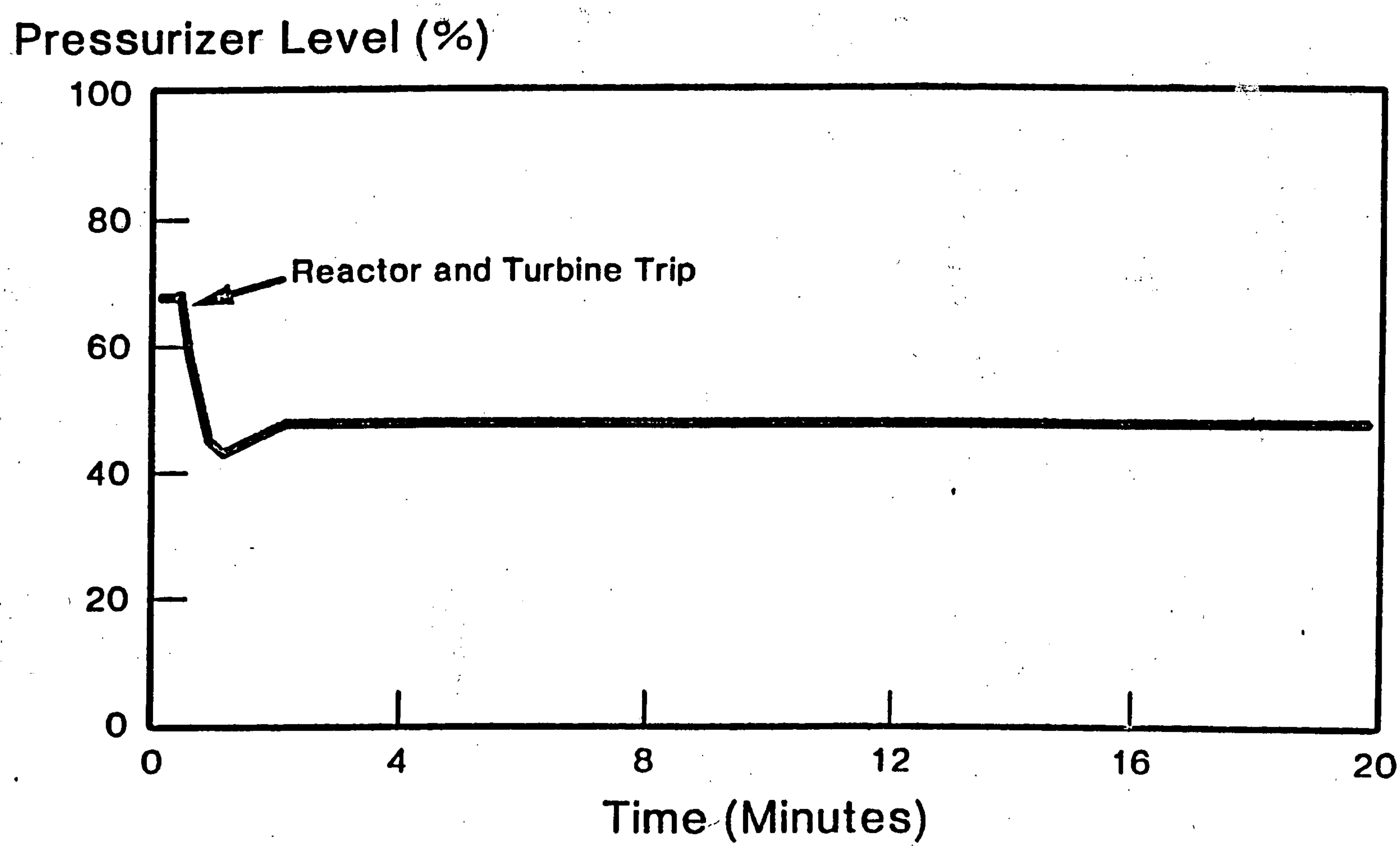

TABLE 2

TRANSIENT DATA--DOUBLE-ENDED FEEDLINE BREAK

Parameters*	Data
Break Size (ft^2)	Double-Ended
Initial power Level (%)	100
Analysis Assumptions	Best Estimate
Trip Function	SG Level
Time Setpoint Reached (Sec)	11.9
SI Initiation Function	Steamline Pressure
Time Setpoint Reached (Sec)	32.7
Steamline Isolation Function	Steamline Pressure
Time Setpoint Reached (Sec)	39.7
AFW Termination Time (Sec)	NA
SI Termination Time (Min)	10.2
Limiting Parameter for SI Termination	Operator Action Time

* Typical 3-loop plant with Model D generators

For both the double-ended steamline break and feedline break, steamline pressure in at least one steamline would be rapidly decreasing. Containment pressure and temperature increases would be observed if the break occurred inside containment. Audible and visual confirmation of the break may be possible if the break occurs outside containment. Also, for all secondary double-ended high energy line breaks, a distinct characteristic is the closure of all main steamline isolation valves.

ENCLOSURE 2

SUPPORTING DOCUMENTATION FOR RELAP5/1 AND FORCE COMPUTER CODES

1. Test Plan for RELAP5/1 and FORCE
2. FORCE Version 2 Quality Assurance, Product Test Report, BCS QA Certification (QA Section 2.3)
3. RELAP5/MOD1 Quality Assurance, Product Test Report, BCS QA Certification (QA Section 2.3)
4. National Certification: FORCE Version 2.0
5. RELAP5, Vendor Version 3.0 and BCS Version 1.0.1 National Certification, Class C and Category Regulated

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
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BOEING COMPUTER SERVICES COMPANY
A Division of The Boeing Company

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To: F. A. Hanna

cc: J. C. Jervert
B. Mukherji
J. T. Muldoon
S. J. Pruitt
C. F. Wolfe

Subject: Test Plan for RELAP 5/1 and FORCE

Reference: ETA Quality Assurance Procedure 40356-D1

The attached Test Plan reflects the Quality Assurance requirements for NUCLIB (Nuclear Library) computer codes RELAP5/Mod 1 and FORCE.

This plan fulfills BCS requirements for conditional certification to B-class and nuclear category QA regulated status.

PKonichek
D. P. Konichek
@ 11/11/82