CHAPTER 12 SAFETY EVALUATION

12.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

12.1.1 PLANT ABNORMALITIES AND TRANSIENTS CONSIDERED

For the following plant abnormalities and transients, the reactor control and protection system is relied upon to protect the core and reactor coolant boundary from damage:

a) Rod Cluster Control Assembly (RCCA) Withdrawal,
b) Rod Cluster Control Assembly (RCCA) Drop,
c) Chemical and Volume Control System (CVCS) Malfunction,
d) Loss of Reactor Coolant Flow,
e) Startup of an Inactive Reactor Coolant Loop,
f) Loss of External Electrical Load,
g) Turbine Trips Due to Steam and Power System Malfunctions,
h) Loss of All AC Power to the Station Auxiliaries, and
i) Turbine Overspeed.

During the detailed design of the plant, analyses will be performed to verify that the safeguards and safety margins embodied in the design are adequate to protect the core and reactor coolant boundary and to prevent hazardous conditions from arising as a consequence of any of these abnormalities and transients. These analyses require the use of data which are developed only after engineering of the reactor core and related systems has progressed to an advanced stage. Experience with designs of this type, however, provides a general knowledge of the course of events to be expected and gives assurance that the reactor control and protection system will be able to satisfy the criteria for protection of the core and reactor coolant boundary.

All reactor protection criteria are met presupposing the most reactive RCC assembly in its fully withdrawn position. Therefore, trip is defined for analytical purposes as the insertion of all RCC assemblies except the most reactive assembly, which is assumed to remain in the fully withdrawn position. This is to provide margin in shutdown capability against the remote possibility of a stuck RCC assembly condition existing at a time when emergency shutdown is required.

Means will be provided for continuously monitoring all individual RCC assemblies together with their respective group position. This may be in the form of a deviation alarm system or a continuous analog or digital display of all RCC assemblies, both shutdown and control group. The operator would thus become readily aware of any malfunction. The reactor would then be shut down in an orderly manner and the condition corrected. Such occurrences are expected to be extremely rare based on operation and test experience to date.

In summary, reactor protection is designed on a fail-safe basis to prevent cladding failure in all transients and abnormalities listed above. The most probable modes of failure in each protection channel results in a signal calling for the protective trip. Coincidence of two out of three (or two out of four) signals is required where single channel malfunction could cause spurious trips. A single component or channel failure in the protection system itself coincident with one stuck RCCA is always permissible as contingent failures and will not cause violation of the protection criteria. The reactor control and protection system will be designed using the proposed IEEE "Standard for Nuclear Plant Protection Systems" as a guide.

12.1.2 ROD CLUSTER CONTROL ASSEMBLY (RCCA) WITHDRAWAL

12.1.2.1 General

An RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in a power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor either subcritical or at power.

12.1.2.2 Uncontrolled RCCA Withdrawal from a Subcritical Condition

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low power level during startup by RCCA withdrawal. Although the initial startup procedure uses the method of boron dilution, the normal startup is with RCCA withdrawal. RCCA motion can cause much faster changes in reactivity than that which could be provided by changes in boron concentration.

Should the rate of reactivity addition be excessive, an abnormally high rate of increase in neutron flux would initiate an alarm in the control room. If corrective action is not taken by the operator to reduce the startup rate, an automatic "RCC-stop" signal deactivates the cycling mechanism on the cluster drive control, causing cluster withdrawal to cease. In addition, trip relays would open automatically on a high startup rate signal from the intermediate range nuclear instrumentation channels. Trip relays would open also when the nuclear power level detected by any two of four power range channels exceeds a preset fractional power limit established for startup. With only the power range trip occurring (the last trip in the sequence of trips), the startup accident is terminated with a DNB ratio of 1.30 or greater.

The maximum possible number of RCC assemblies which can be moved and their maximum withdrawal speed are established by detailed plant design. This information and the maximum incremental RCCA reactivity worth, will be used to verify that the protection afforded by the startup rate and the fractional power trip settings is adequate to terminate the transient safely. Protection in this case is adequate if DNB is not reached, thus ensuring that no fuel damage or fission product release will result. A practical combination of limiting RCCA speed and trip set point will be selected.

12.1.2.3 Uncontrolled RCCA Withdrawal at Power

If RCC assemblies are improperly withdrawn when the reactor is operating in the power range, an approach to an unsafe operating condition might result. Protection is first provided by automatic RCC-stops from the power range channels when indicated nuclear power exceeds a preset limit. Should this device fail to limit the transient, a nuclear overpower signal from any two of four power range channels will cause automatic trip. Additional protection is provided by the overpower-overtemperature trip. The set point for this trip is continually calculated from reactor coolant system temperatures and pressure and prevents an unacceptable combination of power, temperature and pressure.

These trips will terminate the transient in time to prevent DNB. This conclusion will be verified for a range of reactivity insertion rates, including the maximum rate, by computer simulation methods. The required trip time will be determined and demonstrated to be within the capability of the protection system.

12.1.3 ROD CLUSTER CONTROL ASSEMBLY (RCCA) DROP

The four power range nuclear channels and the rod bottom detection from the rod position indication system serve as a means of sensing a dropped rod cluster. A dropped RCCA will rapidly depress the neutron flux in one region of the core. A sudden reduction in flux at any one of the four power range detectors will actuate the RCCA drop protection. Either protection system (ion chambers or rod position) will initiate protective action in the form of a turbine load cutback and blocking of automatic rod withdrawal. This action compensates for possible adverse core power distributions and permits an orderly retrieval of the dropped RCCA.

Criteria for design of these countermeasures will be based on preserving a safe margin against DNB; hence, no fuel damage or fission product release to the coolant system will result from a dropped RCCA.

12.1.4 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

Changes of the reactor coolant boron concentration and reactor coolant makeup are accomplished by means of the chemical and volume control system.

Reduction of boric acid concentration by any process is a means of reactivity insertion. When this occurs by operation of the reactor makeup control while the reactor is critical, the protection described in the preceding cases for RCCA withdrawal is applicable. The maximum reactivity insertion rates attainable by the charging and letdown operation are much less than by RCCA motion, hence the protection is adequate.

When reactivity is added by reducing boric acid concentration while the reactor is shut down, the margin of shutdown is reduced. Since the rate of change of k_{eff} attainable under these circumstances is quite slow, there is time for corrective action to be taken before criticality can be achieved.

Under certain abnormal conditions, the charging pumps could introduce cold water into the reactor. However the rate at which reactivity changes can occur during operation of the charging pumps is far less than that possible with control rod cluster motion. Therefore, the reactor protection system is capable of providing adequate protection.

If either the boric acid flow or the demineralized water flow deviates from the control set point during reactor coolant boration, dilution or normal leakage makeup, alarms will warn the operator to deactivate the makeup system manually.

Should a condition ever arise when reactor coolant boron is changing without the operator's knowledge, RCC group position indication is a positive means of detecting any significant change when the reactor is critical.

Subcritical count rate from the source range nuclear detectors would warn of an undetected malfunction in the subcritical condition. Section 7.3 has a more detailed description of the Nuclear Instrumentation.

12.1.5 LOSS OF REACTOR COOLANT FLOW

The reactor is tripped if flow is below a preset limit in one loop or when power is lost to one reactor coolant pump when operating above a preset power level.

Simultaneous loss of electrical power to all reactor coolant pumps at full power is the most severe loss-of-coolant flow condition. In this case, however, reactor trip together with flow sustained by the inertia of the coolant and rotating pump parts will be sufficient to prevent fuel failure and reactor coolant system overpressure. Detailed analysis of this type of accident will be performed during the design phase to verify system performance and clad conditions for various loss-of-flow accidents.

12.1.6 STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

Because there are no isolation or check valves in the reactor coolant loops, there will be reverse flow through an inactive reactor coolant loop. This back flow inherently prevents excessive mismatch in the coolant temperature and boron concentration between the inactive loop and the operating loops. In addition, a temperature interlock is provided to prevent starting a reactor coolant loop if the reverse ΔT is greater than a preset amount. The change in reactivity in the core associated with the starting of an inactive loop is therefore well within the capability of the control and protection system.

12.1.7 LOSS OF EXTERNAL ELECTRICAL LOAD

The plant will be designed to accept 40% loss of external load without a reactor or turbine trip. No unique or unproven features are required in the reactor control system to accomplish this. The automatic system which dumps steam to the condenser is designed for use in this situation as a short term supplemental load to provide time for the reactor control system to reduce the thermal output of the reactor without exceeding acceptable core and coolant conditions. The amount of steam dump is reduced as the control is effective in reducing the reactor thermal output to the desired steady state turbine load. The condenser hotwell provides an adequate supply of water which is delivered to the steam generators by the normal feedwater system.

12.1.8 TURBINE TRIPS DUE TO STEAM AND POWER SYSTEM MALFUNCTIONS

There are various steam and power system malfunctions which could cause a turbine trip. These include such things as improper response of the turbine control valves to a loss in external load, loss of condenser vacuum, and loss of feedwater flow. A turbine trip is accompanied by a direct reactor trip and a controlled short term release of steam to the condenser which removes sensible heat from the reactor coolant system while avoiding steam generator safety valve actuation. In the unlikely event that the automatic

dump system failed, and no reactor trip occurred, the steam generator safety valves would open for overpressure protection, and the reactor would attempt to meet the apparent load demand. In order to prevent DNB in the core in this case, a reactor trip from two of three high pressurizer pressure signals is provided. The pressurizer safety valves are sized to protect the reactor coolant system against overpressure in this accident.

Vacuum loss in the main condensers may be precipitated by loss of circulating water flow or by mechanical failures which may permit air to enter the condensate system. Should the vacuum loss be small, a reduction in turbine load may be necessary, with a corresponding decrease in reactor loading, until corrective action is taken to minimize the vacuum loss. A large vacuum loss below a preset limit will cause the turbine to trip automatically. The resulting effect of turbine trip on plant operation has been described previously. For this condition, the automatic steam dump to the condenser is blocked and the atmospheric safety valves would provide the necessary heat release.

A complete loss of normal feedwater flow to all steam generators is unlikely and could occur only if steam is lost to the feedwater pumps or power is lost to the condensate pumps or if rupture of non-duplicated lines occurs. Assuming no protective action is taken and full power generation in the reactor continues, the secondary side of the steam generators would be emptied in a very short period of time. Protection is provided by a low feedwater level reactor trip actuated by a difference in steam and feedwater flow from any steam generator in coincidence with a low water level in the same steam generator. Additional protection against core damage is given by the overpower-overtemperature reactor trip which would initiate a trip because of increasing coolant temperatures as heat transfer to the secondary system is reduced.

In the event of a loss of normal feedwater flow, a protective circuit is also actuated to provide emergency feed flow from the condensate storage using either the steam driven or motor driven emergency feedwater pumps and a duplicate feedwater line. Either pump will prevent excessive temperatures

in the reactor coolant system resulting from continued addition of core residual heat and will prevent the steam generator tube sheets from being uncovered. The core residual heat will provide enough heat to the steam generators to supply steam for the emergency steam driven pump and the emergency generators will be sized to power the motor driven emergency feedwater pumps.

Protective circuits to prevent reactor coolant system overpressurization consist of:

a) Loss of feedwater supply trip
b) Fixed pressurizer high pressure trip
c) Overpower-overtemperature trip
d) Actuation of emergency feedwater system

Feedwater flow sufficient to handle residual heat generation (4 to 8% of full load flow) will be restored before the reactor coolant system pressure increases to the relief valve set point which would cause loss of coolant through the valves.

12.1.9 LOSS OF ALL A. C. POWER TO THE STATION AUXILIARIES

In the unlikely event of a complete loss of all auxiliary a.c. power while the reactor plant is at power, the reactor and the turbine will be tripped in accordance with the following sequence:

a) The turbine trip and reactor coolant average temperature program deviation signals will cause the automatic initiation of steam dump to the main condenser.

b) Because of the loss of all outside a.c. and turbine trip, there is simultaneous loss of the condenser circulating water pumps and the condenser dump will be blocked.

c) As the steam system pressure subsequently increases, the atmospheric steam dump valves will be automatically tripped open.

d) If the steam flow rate through the atmospheric dump valves is not sufficient, the steam generator safety valves will temporarily lift to augment the steam flow until the rate of heat dissipation through the atmospheric dump valves is sufficient to carry away the stored heat of the fuel and coolant above no-load temperature plus the residual heat of the reactor.

e) As the no-load temperature is reached, the atmospheric steam dump valves will be used to dissipate the residual heat and to maintain the plant at the hot shutdown condition.

The loss of normal feedwater supply will signal for the start of the emergency feedwater pumps. The emergency turbine utilizes steam flow from the secondary system, in parallel with the atmospheric dump valves, to provide makeup water for the steam generators. The turbine driver exhausts the secondary steam to the atmosphere. The two motor driven emergency feedwater pumps are supplied by the emergency generators. The emergency feedwater pumps takes suction directly from the condensate storage tank for delivery to the steam generators.

If at the time of complete loss of normal auxiliary power and turbine trip, the plant is operating at full load, there will be a rapid reduction of steam generator water level. This is due partly because of the reduction of steam bubble void fraction on the secondary side of the steam generator and partly because steam flow continues after normal feedwater stops. During the first two minutes, the level will drop to about 35% of normal. By the end of this time flow will be established from the emergency feedwater pumps and further reduction of water level will be slow. The capacity of the emergency feedwater pumps is selected so that either the steam driven pump or the two motor driven pumps will prevent the water level in the steam generators from receding below the lowest level within the indicator range during the transient. This will prevent the tube sheet from becoming uncovered at any time during the transient. The reactor operator in the control room will monitor the steam generator water level and will control the feedwater addition with remote operated emergency feedwater control valves. The condensate storage tank capacity is sufficient to dissipate residual heat and maintain the plant at hot shutdown for in excess of twenty-four hours.

The steam driven feedwater pump can be tested at any time by admitting steam to the turbine driver. The electrically driven emergency feedwater pumps can also be tested at any time. The emergency feedwater pumps will circulate water from the condensate storage tank through a test line connection to the main condenser. The emergency feedwater control valves and atmospheric dump valves can be operationally tested whenever the plant is at hot shutdown and the remaining valves in the system will be operationally tested when the turbine driver and pump are tested.

Upon the loss of power to the reactor coolant circulating pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. The natural circulation capability has been calculated for the conditions of equilibrium flow and maximum loop flow impedence. The computer code model used to calculate the natural circulation flow has given results within 15% of the measured flow values obtained during natural circulation tests conducted at the Yankee-Rowe plant. The natural circulation flow ratio as a function of reactor power is given in Table 12-1.

Table 12-1

NATURAL CIRCULATION REACTOR COOLANT FLOW

VS REACTOR POWER

REACTOR POWER % Full Power	REACTOR COOLANT FLOW % Nominal Flow
0.5	2.3
1.0	3.0
2.0	3.9
5.0	5.4
7.0	6.2

12.1.10 LIKELIHOOD AND CONSEQUENCES OF TURBINE-GENERATOR UNIT OVERSPEED

The likelihood of a turbine-generator unit overspeeding is very remote because of the reliability and redundancy of the turbine control and protection system.

Should all the main steam admission valves fail to close on a full load rejection, a unit runaway would take place, causing some damage of the rotating parts. However, no missiles are anticipated to leave the unit, for analysis of the potential mode of failure shows that all parts would be contained within the unit casings.

This system is completely hydraulic. There are two low pressure oil control systems, i.e., auxiliary governor system and emergency trip system.

These two systems and the 300 psi system are interconnected through orifices.

The control and protection system is fail-safe: any loss of oil pressure causes closure of the steam valves.

The main governor normally controls the unit. Should an overspeed take place, the auxiliary governor system will be actuated first, the auxiliary governor dome valve will open, the 300 psi pressure oil will drain, and the control valve will close.

Should the unit overspeed reach the mechanical overspeed trip set point, the overspeed trip valve will open, the 300 psi pressure oil will drain, and the throttle valves will close. At the same time, a second drain path will be provided for the 300 psi oil system that controls the first set of valves, so that the control valves will trip too, in case they did not trip.

Assuming, for the purpose of analysis, that all the above mentioned valves fail to close, a turbine runaway occurs. The first disks to fail are the low-pressure turbine disks closest to the steam admission. As these disks burst, the unit will be decelerated because the steam flow between blades of the emaining disks will decrease significantly. The analysis of the energy required to violate the integrity of the low pressure turbine casings shows that there is a large margin between the kinetic energy of the broken parts of the burst disks at their bursting speed and the energy required to penetrate the low pressure turbine cylinders.

The stress analysis of the remaining disks shows that their bursting speed is at most 15% higher than the bursting speed of the first disks. The assumption is made that even these disks might fail at the bursting speed of the first disks. A plastic analysis of the energy required to penetrate the low pressure turbine casings shows that even the broken parts of these disks will not have enough energy to be ejected outside the casing.

The stress analysis of the high pressure turbine spindle shows that its bursting speed is at least 50% higher than the maximum speed at which the turbine can rotate. This large margin keeps the probability of a high pressure turbine spindle bursting to practically zero.

It is worthwhile to point out that due to conservative design, very careful rotor forging procurement and rigid inspection, Westinghouse turbine-generator units have never experienced such a massive failure.

A survey of the available literature on turbine-generator unit failure shows that the last massive failure of a turbine generator unit occurred about eight years ago. The causes of failure were identified at that time, and provisions were adopted to prevent the recurrence of massive failures. The record since that time demonstrates the soundness of these provisions and correct design.

The no-failure record of Westinghouse turbine generator units, plus the experience gained from the referenced incidents, together with the improvement in the design and inspection techniques in the past eight years indicates that the likelihood of massive turbine-generator failure is extremely remote.

With regard to design and inspection techniques, it is worthwhile to mention that a technical committee of forging suppliers and equipment manufacturers was formed about ten years ago under ASTM to study turbine and generator rotor failures. This group developed the high-toughness NiCrMoV material, now used in all turbine rotors and disks. This Task Force[1] has been very active in making additional improvements in quality and soundness of large forgings and is still in force.

The survey of the literature on massive turbine failures in the last 20 years indicates that all of them occurred between 1953 and 1958.

This survey has pointed out that the rare events of a catastrophic failure of turbines fell into one of two categories:

1) Failure by overstressing arising from accidental and excessive overspeed, and
2) Failure, due to defects in the material, occurring at about normal speed

No failure falling in the first category occurred in the USA. The only two documented examples occurred in the United Kingdom. Both accidents were caused by the main steam admission valves sticking in the open position after full load rejection, because of impurities in the turbine control and lubrication oil. The probability of this occurrence in this plant is very remote as previously pointed out.

Besides the provisions in the design of the turbine control and protection system during plant operation, valves will be exercised on a periodic basis, to further preclude the possibility of a valve stem sticking. Analysis of oil samples will be performed as required.

The turbine is periodically overspeeded to check the tripping speed. The remaining tripping devices are periodically checked.

The causes of the failures that fall in the second category, i.e., failures due to defects in the material occurring about normal speed, were completely identified and, if the ultrasonic test were used as one of the bases for rejection or acceptance of forging, many of them would not have occurred. Further, the stress concentration points that initiated failure in some units are strictly correlated to the peculiar design characteristics of those units. These discontinuities are not present in Westinghouse units.

Westinghouse specifies the quality and method of manufacturing of the purchased forgings. Written specifications cover the manufacturing process, the chemical and mechanical properties, the test to be performed, etc. Specifically, the tests performed are both destructive and non-destructive in nature. The destructive tests include tension tests, impact tests, and transition temperature measurement tests. The tension specimens are taken in a radial and/or longitudinal direction. The tensile properties are determined in accordance with ASTM A-370 on a Standard Round 1/2 inch Diameter 2 Inch Gage Length Test specimen. The yield strength is taken as the load per unit of original cross section at which the material exhibits an offset of 0.2 per cent of the original length. The Charpy impact specimens are taken in a radial direction, and the minimum impact strength at room temperature measured. The transition temperature is determined from 6 specimens tested at different temperatures in accordance with ASTM A-443. The specimens are taken in a radial direction and machined in such a manner that the V-notch is parallel to the forging axis. Two specimens are machined from each test bar. All specimens are taken following all heat treatment. Curves of impact strength and per cent brittle failure versus test temperature are drawn.

The non-destructive tests include bore inspection, sulfur printing, magnetic particle test, thermal stability test, and ultrasonic test.

The bores are visually inspected and the walls of the finished bores shall be free from cracks, pipe shrinkage, gas cavities, non-metallic inclusions, injurious scratches, tool marks and similar defects.

A magnetic particle test is made on each forging to demonstrate the freedom from surface discontinuities. The end faces of the main body and down over and beyond the fillets joining the main body to the shaft portions are magnetic particle tested. The bore is also magnetic particle tested at a high sensitivity level in accordance with ASTM A-275. These inspections are done by Westinghouse inspectors prior to Westinghouse accepting these forgings. After final machining by Westinghouse, rotors are again magnetic particle inspected on the external surfaces by Westinghouse.

The face of the test prolongations at each end of the rotor body or an area on the end faces of the rotor body equivalent to the test prolongations is sulfur printed to determine the freedom from undue ingot corner segregation and excessive sulfide inclusions.

A thermal stability test is performed on the forging at the place of manufacture after all heat treatment has been completed.

The forgings are ultrasonically inspected at the place of manufacture by Westinghouse inspectors.

Based on conservative design, reliable turbine control system, careful rotor forging procurement and rigid inspection, the probability of a combination of excessive overspeed, new-born large forging defects, and operating temperature below the transition temperature is considered practically zero, as confirmed by years of major-failure-free operation of the many Westinghouse units.

12.2 STANDBY SAFEGUARDS ANALYSIS

12.2.1 SITUATIONS ANALYZED

Adequate provisions have been included in the design of the plant and its standby engineered safeguards to limit potential exposure of the public to well below the limits of 10 CFR 100 for situations which could conceivably involve uncontrolled releases of radioactive materials to the environment. The situations which have been considered are:

a) Fuel Handling Accidents
b) Accidental Release of Waste Liquid
c) Accidental Release of Waste Gases
d) Rupture of a Steam generator Tube
e) Rupture of a Steam Pipe
f) Rupture of a Control Rod Drive Mechanism Housing - Rod Cluster Control Assembly (RCCA) Ejection

12.2.2 FUEL HANDLING ACCIDENTS

12.2.2.1 Fuel Handling and Storage Facilities

a) Fuel Transfer System

This system provides for underwater transfer of the fuel assemblies and control rod clusters from the reactor by means of a manipulator crane, fuel transfer carriage, and spent fuel hoist to the spent fuel storage pit, a second fuel handling device removes the individual spent fuel assemblies from the transfer carriage and places them in the spent fuel storage racks. They remain in the storage racks for a sufficient time to allow decay to a level which permits shipment to a reprocessing plant. Before fuel assemblies containing reuseable control rod clusters leave the reactor cavity, they are place in the control rod cluster changing fixture where the control rod clusters are removed for insertion into fresh assemblies. Spent control rod clusters are transferred within a spent fuel assembly where they remain when the assemblies are placed in the storage racks.

The following fuel handling accidents will be evaluated during the course of design to ensure that no hazards are created:

1) A fuel assembly becomes stuck inside reactor vessel.
2) A fuel assembly or control rod cluster is dropped onto the floor of the reactor cavity or spent fuel pit.
3) A fuel assembly becomes stuck in the penetration valve.
4) A fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck.
5) A fuel assembly is dropped onto the bottom of the transfer tank under the carriage.

Since all the operations are under water and since there is always ample cooling for the fuel assembly, regardless of the accident postulated, there is no danger of overheating of the fuel. Further, since such circulation is available naturally, it is possible to prevent excessive local accumulations of the activity in the water resulting

from defective fuel. Necessary shielding during all fuel handling operations is ensured by designing the fuel transfer equipment and facilities such that a fuel element is always immersed in water to a depth sufficient to protect personnel.

If the spent fuel assembly or RCC assembly were to come loose from the manipulator crane or spent fuel hoist, it would be easy to see the item through the water and retrieve it remotely.

The simplicity of the system and the ability to establish the water level in the spent fuel storage pit outside the containment at the same level as inside tends to prevent accidents from misoperation or human error.

During refueling, reactivity accidents are prevented by control of boric acid concentration in the refueling water. The reactor coolant and refueling water are borated sufficiently to yield k_{eff} of approximately 0.90 for cold conditions with all rods in. This concentration is also sufficient to prevent criticality even with all control rods removed. The boron concentration in the refueling water is periodically sampled and continuous mixing is maintained through the reactor vessel by utilizing a residual heat removal pump. During this period, the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by BF_3 counters and count rate indicators. Any appreciable increase in the neutron source multiplication including that caused by the maximum physical boron dilution rate (approximately 600 ppm per hour) is slow enough to give ample time to start corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical. The maximum dilution rate is based on the abnormal condition of both charging pumps operating at full speed delivering unborated water to the reactor coolant system at a particular time during refueling when the boron concentration is at the maximum value and the water volume in the system is at a minimum.

b) Spent Fuel Storage

The spent fuel storage area is sized for the normal storage of one and one-third complete core and associated control rod clusters. The fuel is spaced to prevent criticality even with unborated water. If an accident were to damage the cooling system, the heat capacity of the water affords ample time for countermeasures. Even if an entire day were to elapse, no serious condition would arise.

Radioactivity and clarity of the water are controlled by means of an ion exchanger and filter built into the spent fuel pit water recirculating system. Any radioactivity which escapes into the atmosphere above the pit is carried off by the ventilating system.

12.2.2.2 Fuel Handling Accident Analysis

Potential off-site consequences of a fuel handling accident have been examined. The only credible cause of cladding failure during handling would be severe mechanical loading of the fuel rods. The chance of such damage is minimized by the design of the spring clip fuel assembly. Since the remote possibility exists of damage occurring during handling of an assembly, this has been selected as the maximum credible fuel handling accident.

The upper limit of this accident is judged to be the sudden release of the gaseous fission products held in the voids between the pellets and the cladding of one row of elements in the fuel assembly. The low temperature of the fuel during handling precludes a further significant evolution of gases from the pellets themselves after the cladding is breached. Halogen release is also greatly minimized due to their relatively low volatility at these temperatures. The tendency for iodine in vapor and particulate form to be scrubbed out of the bubbles of gas during their ascent to the surface of the water also minimizes the inhalation hazard.

The most significant aspect of the incident is the release of the noble gases Xe-133 and Kr-85 which leave the surface of the pit water and are discharged through the ventilation system to the plant vent.

The maximum exposure to a person off-site was estimated, based on approximately 100 hours cooling prior to rupture. The noble gas release from one row of 15 fuel rods in the assembly is equivalent to about 10^4 curies of Xe-133. This quantity of gas discharged from the plant vent under inversion conditions (TID-14844 dispersion) with allowance for dilution in the wake of the building and a 1 meter/sec wind, results in an integrated whole body dose of less than 0.6 rem at the site boundary which essentially meets the yearly exposure limit set in 10 CFR 20 for normal operation.

12.2.3 ACCIDENTAL RELEASE OF WASTE LIQUIDS

Accidents in the Waste Disposal or Chemical and Volume Control Systems which would result in the release of liquid wastes are those which may involve the rupture or leaking of the system pipe lines, the waste storage tanks, the volume control tank and pumps. Spillage or leakage or any liquids will be retained within the auxiliary building housing the waste disposal system. Accumulation of any released liquids is accomplished through the building floor drains which flow by gravity into a drain tank. The ventilation system will collect any gaseous radioactivity and discharge it via the plant vent.

The liquid waste hold up tanks will be used to collect the normal liquid wastes produced. The contents of the tanks will be passed through the waste evaporator prior to analysis and discharge.

All liquid waste processing components are located within the building and any leakage from tanks or piping will be collected in sumps to be pumped back into the liquid waste system. The holdup tank vault volumes are sufficient to hold the full capacity resulting from rupture of any holdup tank without overflowing to areas outside the vault.

Piping external to the containment running between the containment and the auxiliary building and the holdup tank area will be run below grade in concrete trenches. Where necessary, because of radiation level, manual valves in radioactive lines will be operated by extension stems run through shield walls.

No conceivable mechanism exists for accidental release of waste liquids to the river. A river diffusion analysis was performed, however, to determine the concentrations which would result at the Chelsea reservoir if a release were assumed. The results of the analysis show that even the instantaneous release of the entire primary coolant system maximum allowable activity during normal operation would not result in peak concentrations at Chelsea in excess of 10 CFR 20 MPC limits. Drought conditions were assumed to exist at the time of and for a period following the spill limiting the total run off flow to 4000 cfs. The mean longitudinal diffusion coefficient corresponding to this flow was 8.74 square miles per day. These data represent a drought similar to conditions existing in late summer of 1964, which can be verified by data in Paragraph 1.5, Volume 1.

12.2.4 ACCIDENTAL RELEASE OF WASTE GASES

12.2.4.1 Situations Considered

Gaseous activity which could be released in the unlikely event of a tank rupture will not result in an off-site whole body or inhalation dose substantially in excess of 10 CFR 20 yearly exposure limits. The main sources of gaseous radioactivity are in the volume control tank, holdup tanks and gas decay tanks.

12.2.4.2 Volume Control Tank Rupture Analysis

In the event that a rupture should occur in the volume control tank, caused by undetermined means, the integrated whole body dose at the site boundary during passage of the cloud of escaped gases would be less than 0.8 rem.

This is based on a release from the plant vent with wind velocity of 1 meter/second, and assuming TID-14844 inversion weather conditions and allowing for dilution in the wake of the building. The calculations also assumes 100% release of the noble gas isotopes Xe-133, Xe-135, Kr-85, Kr-85m, Kr-87 and Kr-88 in the tank. The fission product inventory in the tank is based upon operation with defects in 1% of the fuel rods and the activity level in the tank at its maximum.

The inhalation hazard at the site boundary from this accident is negligible due to low concentrations, low volatility and high solubility of the halogens.

12.2.4.3 Gas Decay Tank Rupture Analysis

A sudden failure of a gas decay storage tank and release of its contents by an unspecified mechanism would yield a maximum integrated whole body dose at the site boundary (350 meter) of less than 0.8 rem as compared to the 25 rem set forth in 10 CFR 100 for accidental exposure. This is based on a release from the plant vent with a 1 meter/sec wind and assuming

TID-14844 dispersion conditions. It is also assumed that the accident occurs at the end of a 500-day operating period, with a coolant activity equivalent to 1% defective fuel resulting in maximum Kr-85 and Xe-133 in the gas decay tank. This is equivalent to 13,500 curies of Xe-133. The maximum activity level occurs shortly after a cold startup at low boron concentrations with maximum Xe in the reactor coolant.

Safety considerations based on an assumed release of the gaseous activity of the coolant, therefore, would not preclude operation with as many as 1% defective fuel elements. The expected incidence of fuel defects, based on experience to date, is lower than this value by a factor of at least 10.

12.2.5 RUPTURE OF A STEAM GENERATOR TUBE

A preliminary analysis has been performed for the case of double-ended steam generator tube rupture, occurring while the coolant contains fission products equivalent to continuous operation with 1% defective fuel rods. As stated above, it is expected that the activities actually encountered in operation will be less than one tenth of these values.

A conservative evaluation was made by assuming that in the course of cooling down and depressurizing the reactor coolant system, requiring about five hours following detection of a tube failure, that the steam generator containing the faulty tube is not isolated. A volume equal to slightly less thann 1/2 of the reactor coolant system is blown down to the secondary system prior to cooldown. Noble gas activity is volatilized and discharged to the air ejector which, upon sensing high activity in this effluent, will immediately divert the stream to the containment. Assuming operation with 1% defective fuel, the divertion could be delayed 1/2 hour without causing off-site exposure in excess of 0.5 rem to the whole body. Because of the tendency to approach equilibrium partition of I_2 between the condensate and steam phases, in the main condenser only a small fraction (about 10^{-4}) of the I_2 activity would be vented to the atmosphere along with the noble gases and therefore the inhalation thyroid dose would be negligible.

It should be noted also that the planned procedure for recovery from a tube rupture incident would call for isolation of the steam line leading from the affected steam generator after the primary pressure is reduced below the secondary safety valve setting (about 1100 psia). This action could be completed about ten minutes after rupture, even in the event the incident remains undetected until reactor trip occurs due to a safety injection signal. The discharge of radioactive vapors would be greatly reduced in this case, since contamination of the secondary system would then be terminated in ten minutes instead of five hours as assumed in the analysis. As the radioactivity in the steam is principally volatile, it does not present a contamination problem in the steam systems. Also, the accumulated radioactivity in the condenser hot well for the period of the accident does not interfere with maintenance.

12.2.6 RUPTURE OF A STEAM PIPE

A break in the steam piping between the steam generator and the turbine will not constitute an environmental hazard because the steam is not radioactive.

Regardless of the location of the steam line break, i.e., between a steam generator and its associated stop valve or beyond the stop valve in the steam line common to the steam generators, alternate paths and adequate feedwater capacity are available by use of the main or emergency feedwater pumps to remove heat from the steam generators.

Core protection is provided in the same manner as for reactivity insertion accidents by the combination of overpower and overtemperature trips. The criteria for the hypothetical steam break accident are as follows:

1. With a stuck rod, no off site power and minimum engineered safeguards there will be no consequential damage to the primary system and the core will remain in place and intact.

2. With no stuck rod, with off site power and all equipment operating at design capacity, there will be insignificant clad rupture.

The accident requires protective action, however, because the reactor control system, sensing the loss of steam pressure as an increase in load, will tend to increase reactor power. If the increase is not excessive, an appropriate manual shutdown procedure can be executed. If reactor power exceeds the overpower setpoint, automatic trip will occur.

The steam line break constitutes an uncontrolled heat removal from the reactor coolant system which is limited by the steam line non-return and trip isolation valves. These valves cannot preclude the blowdown of one steam generator e.g. a break upstream of the isolation valve. In this case there is a rapid cooldown of the reactor coolant system which, particularly at the end of core life, results in a reduction in shutdown reactivity margin after trip. The resultant coolant contraction has the characteristics of the beginning of a loss of coolant and results in the initiation of engineered safeguards as the pressurizer is emptied. These systems compensate by boron addition for the temperature effect on reactivity in all anticipated cases. If the flux distribution resulting from a single stuck RCCA is sufficiently distorted, DNB may occur at relatively low power, causing clad deformation and possible clad rupture in a limited core region. Only in the extreme case, where the combined effects of a stuck RCC assembly, largest steam line rupture and the most adverse reactivity coefficients are imposed, would the possibility exist for sufficient power generation to cause clad damage. The extent of possible damage, if any, will be determined during detailed design. The duration and magnitude of the peak power generation is limited such that conditions are significantly below those which could lead to fuel clad melting and there is no possible mechanism for primary system pressure pulses. Public and plant personnel safety is assured by the fact that the transient is self-limiting as rising coolant temperatures, blowoff of available coolant in the secondary system and continued boron addition act to terminate the transient, and by the fact that any fission products released are confined to the reactor coolant system.

The steam piping is adequately anchored at the containment wall and is routed so that any whipping of the ruptured pipe will not result in compounding of the break. The rupture of secondary piping has been assumed to impose a maximum pressure differential of 2250 psi across the steam generator tubes and tube sheet from the primary side to the secondary side. The design of the steam generators is such that under these conditions there shall be no rupture of the primary to secondary boundary, thus preventing any violation of the reactor coolant system integrity.

12.2.7 RUPTURE OF A CONTROL ROD DRIVE MECHANISM HOUSING - ROD CLUSTER CONTROL ASSEMBLY (RCCA) EJECTION

The reactor protection system provided for the pressurized water reactor is capable of terminating any credible reactivity excursion without DNB. Thus, even if a group of RCC assemblies having maximum reactivity worth are withdrawn from the core at the maximum drive speed, there would be no significant temperature rise in the cladding and no zirconium water reaction. The only conceivable way of exceeding the maximum reactivity insertion rate for which the protection system prevents DNB is for a rupture of a control rod mechanism housing to occur. Such a rupture is not considered credible for the following reasons:

a) The housings are of conservative design and initially hydrostatically tested to approximately 4100 psi prior to operation.

b) Stress levels in the housing are not affected by system transients at power or by thermal movements of the coolant loops.

c) The housing is constructed of Type 304 stainless steel which shows excellent notch toughness at all temperatures that will be encountered during the operation of the plant. A significant margin of strength in the inelastic range is thus provided and gives additional assurance that gross failure of the housing will not occur.

If a rupture of a control rod mechanism housing were assumed to occur, the resulting loss of coolant would be small compared with the hypothetical accident for which the containment is designed. The significance of this accident derives from the possibility that the RCCA could be ejected from the core in a very short time by full system pressure differential acting on the drive shaft. The resultant power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals. Due to the extremely low probability of the incident, fuel damage is considered acceptable. However, the plant is designed such that there is no possibility of a consequential pressure surge causing further rupture of the reactor coolant system.

The amount of fuel damage that could result from such an accident is governed mainly by the peak power attained in the transient which in turn depends on the worth of the ejected rod and the power distribution attained with the remaining control rod pattern.

An analysis will be performed in the course of detailed design to determine the spatial distribution of excess energy in the fuel, the resulting fuel temperature and the heat transfer to the coolant as a function of reactivity insertion in this transient. If there is any likelihood of fuel dispersal due to fragmentation of the fuel pellet-clad system, a detailed analysis will be made to demonstrate that the rapid dissipation of this energy into the water does not create a significant pressure surge to jeopardize either the integrity of the vessel or that of the reactor coolant system.

The core design does not represent a significant departure from previous plants for which detailed rod ejection analyses have been made. By utilizing the flexibility in the selection of control rod groupings, radial locations, and axial position as a function of load, the final design will limit the maximum fuel temperature for the highest worth ejected rod to a value which will preclude any further rupture of the primary system.

The results of the sensitivity analysis performed for the Indian Point Unit #2 and reported in the "Second Supplement to Preliminary Safety Analysis Report" Docket No. 50-247 are applicable. The expected maximum ejected rod worth for the zero power case is 0.6% as for Indian Point Unit #2 and using a conservative analysis there would be no fuel melting and no potential rupture of the reactor coolant system for this case. The expected maximum ejected worth at full power is considerably less than 0.5% and as reported for Indian Point Unit #2, even this case resulted in no fuel dispersal and hence no potential rupture. The increased thermal rating for Indian Point Unit #3 above Indian Point Unit #2 is accompanied by lower hot channel factors resulting in a lower peak specific power. The initial conditions for the hot spot are therefore no worse for Indian Point Unit #3 than for Indian Point Unit #2 with respect to the rod ejection analysis. The upper limit for potential insertion from the positive moderator coefficient is also comparable to Indian Point Unit #2 e.g., approximately 1.5% δk.

12.3 MAJOR RUPTURE OF A REACTOR COOLANT PIPE

12.3.1 GENERAL

A loss-of-coolant accident may result from a rupture of the reactor coolant system or of any line connected to that system up to the first closed valve. Ruptures of very small cross section will cause expulsion of coolant at a rate which can be accommodated by the charging pumps. Should such an incident occur, these pumps would maintain an operational level of water in the pressurizer, permitting the operator to execute an orderly shutdown. A moderate quantity of coolant containing such radioactive impurities as would normally be present in the coolant, would be released to the containment.

Should a larger break occur, resultant loss of system pressure and pressurizer liquid level will cause reactor trip and initiation of safety injection. These countermeasures will limit the consequences of the accident in two ways:

a) Reactor trip and borated water injection will supplement void formation in causing rapid reduction of the nuclear power to a residual level corresponding to delayed fissions and fission product decay.

b) Injection of borated water ensures sufficient flooding of the core to prevent excessive temperatures.

The safety injection system, even when operating on emergency power, prevents any melting of fuel cladding and limits metal-water reaction to a negligible amount for reactor coolant piping ruptures up to and including the double ended rupture of a reactor coolant loop. Consequences of these ruptures are well within the exposure limits of 10 CFR 100 with only partial effectiveness of the engineered safeguards and essentially meet 10 CFR 20 yearly exposure limits with the expected performance of the engineered safeguards systems.

Supports for the reactor vessel and the other main reactor coolant piping will be sufficient to prevent rupture or failure of any safety injection line which is not connected to the pipe assumed to rupture. All piping which is connected to the reactor coolant system and which penetrates the containment will be provided with sufficient anchorage and load limit controls to prevent violation of the containment at the containment penetration as a result of a reactor coolant pipe break or a seismic disturbance. The design of the steam generator is such that the maximum secondary to primary differential pressure of 1100 psi acting on the tubes and tube sheet after reactor coolant blowdown will not cause failure of the primary/secondary boundary.

The steam generators will be supported in a manner which will prevent rupture of the secondary side of a steam generator and/or main steam and feedwater piping as a result of the forces created by the rupture of a reactor coolant pipe or by a seismic disturbance. In general, the forces due to pipe rupture are considerably larger than seismic forces. As necessary, each steam generator will be provided with additional supports which will supplement the gravity supports and will limit the motion of the steam generators under the reaction forces due to the reactor coolant pipe break to a distance that is compatible with the flexibility of the main steam and feedwater piping. The support system design will be sufficient to resist the forces from an assumed break at any point in the reactor coolant system.

Whenever possible, piping and valves, except root valves and their connections to the coolant piping, will be run in the annular space between the primary shielding and the containment wall. Thus, these components will be completely outside the area occupied by the primary equipment and the reactor coolant loops and will be protected from possible effects of a loss-of-coolant accident.

Special precautions will be taken with the pressurizer relief valves (which must be located above the operating floor) to protect these valves from mechanical damage which could lead to a loss-of-coolant accident.

Valves necessary for operation under accident conditions will be specified and selected so that they will not be rendered inoperative by the temperature and pressure conditions existing at the time of an accident. Organic material used for electrical insulation and other uses, such as valve stem packing, will be selected so that prolonged exposure to radiation and high temperatures will not cause excessive deterioration.

The design criteria for the safety injection system are presented in Chapter 6 along with a description of the system.

12.3.2 LOSS OF COOLANT ACCIDENT EVALUATIONS

To determine the requirements of the containment structural design and to evaluate the engineered safeguards systems, analyses were performed to describe the transient condition inside the containment following a postulated loss of coolant accident. The analysis is based on mathematical models which reflect the phenomena occurring within the Reactor Coolant System, the Reactor Core, and the Containment during and after a loss of coolant accident. Conservative approaches and parameter values have been assumed for these mathematical models.

Six groups of delayed neutrons were used. The total effective fraction was 0.0061, a conservative minimum, for the positive moderator coefficient cases. A conservative maximum of 0.0072 was used for the zero or negative coefficient cases to slow down power decay.

Core Thermal Analysis

The LOCTA-R2 transient digitial computer program was developed for evaluating fuel pellet and cladding temperatures during a loss of coolant accident. It determines the extent of the Zircaloy-steam reaction and the magnitude of the resulting energy release in Zircaloy clad cores.

The transient heat conduction equation is solved by means of finite differences, considering only heat flow in the radial direction. A lumped parameter method is used; the fuel containing three nodes and the cladding one node.

Internal heat generation can be specified as a function of time, or the decay heat from any initial power level can be calculated by the code. The decay heat is based on the heat generated from:

a) Fission Products,
b) Capture Products, and
c) Delayed Neutrons

It is assumed that the core has been irradiated for an infinite period of time.

In addition to decay heat, the code calculates the heat generated due to the Zircaloy-steam reaction, if any. The $Zr-H_2O$ reaction is governed by the parabolic rate law unless there is an insufficient supply of steam available, then a " steam limited" evaluation is made. The buildup of the Zircaloy-oxide film is calculated as a function of time, and its effect on heat transfer is considered. An isothermal clad melt is considered based on the heat of fusion of Zircaloy. Once the Zircaloy metal melts, it is retained by the Zirc-oxide, and slumps against the fuel. The Zircaloy-steam reaction may continue until the oxide melts. If the oxide melts, the remaining Zircaloy is assumed to fall, and 10% of this metal is assumed to react with additional water which is available in the vessel.

The code has been developed to stack axial sections and thereby describe the behavior of a full length region as a function of time. A mass and energy balance is used in evaluating the temperature rise in the steam as it flows through the core.

The initial conditions of the fuel rod are specified as a function of power. The following core conditions are also introduced as a function of time as determined by the Flash Code:

a) Mass flow rate through the core
b) Coolant quality
c) Pressure

Heat transfer coefficients during the various phases of the accident are evaluated in the following manner:

a) Nucleate boiling film coefficients on the order of 20,000 Btu/hr ft^2 °F are used until DNB.

b) When DNB occurs, it is assumed that the fuel rods can immediately develop a condition of stable film boiling. No credit is taken for higher transition boiling coefficients that exist prior to the establishing of a stable film on the fuel rods. The correlation used during this period is

$$h = .023 \frac{k_v}{D_e} \left(\frac{\rho_v D_e}{\mu_v} \frac{Q_1 + Q_v}{A_c} \right)^{0.8} \left(\frac{c_p \mu}{k} \right)^{0.4}$$

c) During the time the core is uncovered (period of steam flow through the core), laminar or turbulent forced convective coefficients and radiative coefficients are evaluated.

For laminar forced convection to steam:

$$\left(\frac{hD}{k}\right)_{iso} = 3.66$$

$$h/h_{iso} = \left(\frac{Tb}{Tw}\right)^{.25}$$

For turbulent forced convection to steam;

$$\frac{hD}{k} = 0.020 \, (Re_b)^{0.8} \, (Pr_b)^{0.4} \left(\frac{Tw}{T_b}\right)^{-0.5}$$

d) Conservative heat transfer coefficients of the order of 25 Btu/hr ft^2 °F is all that is needed to turn back the rising clad temperature when the core is recovered.

Information generated by LOCTA-R2 as a function of time includes:

a) Fuel Temperature,
b) Clad Temperature,
c) Steam Temperature,
d) Amount of metal-water reaction,
e) Volume of core melt, and
f) Total heat released to coolant.

Symbols for Equations

h - Heat transfer coefficient on outer surface of fuel rod (Btu/hr-ft^2-°F)
De - Equivalent diameter of flow channel - (ft)
ρ - Density (lbs/ft^3)
μ - Viscosity (lbs/ft-hr)
Q - Volumetric flow rate (ft^3/hr)
Ac - Area of flow channel (ft^2)
Cp - Specific heat (Btu/lb-°F)
k - Thermal conductivity (Btu/hr-ft-°F)
T - Temperature°F

Subscripts

v - Evaluation of the property at the saturated vapor condition
1 - Evaluation of the property at the saturated liquid condition
b - Evaluation of the property at the saturated bulk fluid condition
w - Evaluation of the property at the saturated bulk clad condition.

12.3.2.3 Blowdown and Recovery Transient - Large Breaks

The operation of the Safety Injection System with accumulators was analyzed for a range of reactor coolant pipe breaks up to the double ended severance for a typical four loop PWR (2758 MWt) in Supplement 6 for Indian Point Unit No. 2, Docket No. 50-247. The performance of the core cooling system will be re-evaluated on the basis of the expected stretch rating of Unit No. 3, 3217 MWt, to verify that the criteria of Section 12.3.2.1 are met. The effect of the higher power level on the analyses presented is not expected to be significant. The blowdown and recovery transient will be essentially unchanged. The margin to hot spot recovery may be reduced slightly as a result of the increase in heat generation rate.

12.3.2.4 Core and Internals Integrity Analyses

a) Integrity Requirements

The basic requirement of any loss of coolant accident, is that sufficient integrity exist to permit the safe and orderly shutdown of the reactor. To insure that the basic requirement is met the following three sub-requirements must be met;

1) The basic configuration must be maintained. This implies that gross fracture and/or deformation of core and internals must not occur.

2) The ability to move control rods must be maintained so that they can be used to provide shut down even though insertion is not necessarily required following an accident.

3) Internals deformation must be sufficiently small so that primary loop flow, and particularly, adequate safety injection flow, is not impeded.

b) Analysis of Forces and Pressures on Internals and Core

The forces exerted on reactor internals and core, following a loss of coolant accident, are computed by employing the SATAN digital computer program developed for the space-time-dependent analysis of multi-loop PWR plants.

This computer program generally provides a means for the study of the nuclear plant dynamic behavior for a variety of conceivable plant transient operations and accidents. In particular, it can be used to determine the reactor coolant system dynamic characteristics after the occurrence of a rupture in the main cooling piping. SATAN is generally similar to, yet more elaborate than, the FLASH code. It can be employed to determine the shock-wave effects in the reactor coolant system over short time periods in great detail. It can also handle a larger number of elements along the reactor coolant loops.

The SATAN mathematical formulation employs a three-dimensional approach (length along the flow path, radius and time) in the reactor core; a two-dimensional approach (length along the flow path and time) in the reactor coolant loops and a lumped-parameter representation for the secondary system. The reactor coolant system is assumed to consist of one or two parallel loops. The reactor core and the primary loops are sectionalized into elements with variable spatial mesh size. The present memory capacity of the IBM-7094 allows a maximum of 70 spatial elements for the system, which may be distributed between the core and the reactor coolant loops in any desirable fashion. Two typical schemes for the simulation of the reactor coolant system are shown on Figures 12-1 and 12-2. Figure 12-1 portrays a one-loop representation which can accommodate leaks over an appropriate size range in the hot leg (element 7) or in the cold leg (element 11). Figure 12-2 is a two-loop representation. In this arrangement three parallel loops are lumped as one-loop and double-ended complete severance ruptures can be accommodated on the fourth loop either in the hot leg (between elements 6 and 7) or in the cold leg (between elements 13 and 14).

The thermodynamic state of the coolant in any element is defined by:

a) a set of main dependent variables; and

b) a set of auxiliary dependent variables,

The main dependent variables are those variables computed by the integration of a differential equation. The main dependent variables are: coolant pressure, enthalpy, element liquid mass, element vapor mass, axial mass flow rate and radial (or branch) mass flow rate. The auxiliary dependent variables are: coolant steam quality, density and temperature. To compute the main and auxiliary dependent variables, the following fundamental equations are first expressed for each element:

a) The continuity equation;
b) The energy equation;
c) The momentum equation;
d) The state equation

These equations are then rearranged so that the time derivative of the main dependent variables and the values of the auxiliary dependent variables are explicitly defined in terms of the present system variables. These explicit equations form a system of simultaneous differential equations which is integrated numerically on the computer to obtain the time-variation of all dependent variables.

The SATAN program has several distinctive features summarized as follows. The code can accommodate:

a) A leak or a complete-severance double-ended rupture of a desired size with zero or a given rupture time for any element in the hot or the cold leg.
b) Flow reversal for any element.
c) Subcooled, two-phase, or superheat flow in any element,
d) Critical choking flow at the rupture point.
e) Heat transfer in the core and the steam generator,

To verify the correctness of the mathematical formulation and the various numerical techniques employed, the code was checked against LOFT experimental results presently available. The LOFT semi-scale blowdown test facility consists of a 12 inch dia. 10 ft. long cylindrical reservoir connected to a 4 inch

dia. pipe plugged at the end by rupture discs. The reservoir is filled with high enthalpy, high pressure water. When the rupture discs are broken, the pressure and temperature time-variation in the reservoir and the discharge pipe are then measured.

The SATAN program was used to simulate the LOFT semi-scale blowdown test numbers 530 and 522. Figures 12-3 through 12-6 show that the pressure and temperature time history obtained from SATAN are in good agreement with LOFT experimental data from test 530. The discrepancies observed at the onset of two-phase blowdown are due to metastability phenonenon when the fluid undergoes a non-equilibrium flow and predicts higher pressures with subsequently larger and more conservative blowdown discharge and reduced blowdown time.

Having gained confidence in the validity of the SATAN program results, the code was applied to the Indian Point Unit No. 3 blowdown analysis. The plant was first brought to steady-state from initially uniform pressure and enthalpy distributions throughout the system. The computed steady-state variables were then checked against the available plant thermal and hydraulic characteristics and found to be in a very good agreement. This steady-state verification provided a further proof for the validity of the digital program. Starting from these steady-state values, the following blowdown analyses were performed.

1) One loop system with large leak (area equal to double-ended rupture for one actual loop) on hot leg with zero rupture time;

2) One loop system with large leak (area equal to double-ended rupture from one actual loop) on cold leg with zero rupture time;

3) Two-loop system with double-ended rupture (complete severance) to the hot leg of one actual loop with zero rupture time;

4) Two-loop system with double-ended rupture (complete severance) on the cold leg of one actual loop with zero rupture time.

From the pressure time history of core inlet and outlet plenums and the vessel inlet and outlet plenums the pressure difference across the core and the upper plenum was respectively computed and used to determine forces on the reactor internals. Figures 12-7, and 12-9 show typical pressure-time histories.

Sensitivity studies were performed to evaluate the effect of rupture time on pressure tradients. It was found that rupture times of the order of 0.3 sec. considerably reduces the values of pressure gradients on reactor internals by at least a factor of two as compared with the pressure gradients obtained with zero rupture time used in the present analysis.

12.3.2.5 Force and Pressure Transient Results on Core and Internals

a) Core

During a hot leg break, the difference in pressure across the core is oscillatory for approximately 2 seconds (Figures 12-7 to 12-9) and later on is approximately constant. The largest longitudinal force on the fuel assembly will occur during the initial transient and will reach a value of 6,700 pounds per fuel assembly in compression.

During a cold leg break the longitudinal compressive force on the fuel assembly has a peak value of approximately 9,900 pounds.

The force to buckle a fuel assembly is of the order of 85,000 pounds. The inconel grids connecting the fuel rods are able to maintain the rods in position during the transient.

b) Core Support Structure

As a consequence of the dynamic effect during the initial transient following a hot leg break, the upper core support structure has a maximum deflection upward of 0.120 inches and the maximum total stresses (approximately 12,000 psi) occur at the grid and upper support plate ligaments.

After the first 2 seconds, the force on the upper core support structure becomes approximately constant; each fuel assembly exerts a force of approximately 2920 pounds. The upper core support structure deforms 0.057 inches under this load. Maximum total stresses are at the grid and upper support plate (approximately 12,700 psig).

Following a cold leg break, there will be no significant effects on the upper core support structure because the external forces on the core are predominantly downward.

c) Lower Core Support Structure

Following a hot leg break the maximum total stress occurs at the lower girth weld during the initial transient and is approximately 15,000 psi.

Following a cold leg break, the maximum total girth weld stress is approximately 11,000 psi.

d) Thermal Shield

The thermal shield is rigidly connected to the core barrel and will not be adversely affected by pressure and flow transients following either a hot or cold leg break.

e) Upper Core Barrel

The pressure across the upper core barrel becomes oscillatory during an accident and, to establish its behavior, buckling and radial natural frequency as a short cylinder were computed. The results are:

Buckling Pressure	- 850 psi $< p_{cr} <$ 2400 psi
Extensional Natural Frequency	- $f >$ 1180 cps
Bending Natural Frequency	- $f =$ 60 cps - 100 cps

During the first 0.5 seconds following a hot leg break, the difference in pressure across the core oscillates between +400 and -400 psi at a frequency of 15 cps which is small compared with the natural frequencies listed above. The corresponding stress level is $\sigma = \pm$ 13,500 psi. After this initial transient, this difference in pressure remains approximatley constant and much smaller ($\sim$75 psi inward).

The maximum pressure oscillation following a cold leg break is $\Delta p = \pm$ 600 psi at a frequency of f = 17 cps producing a stress level of $\pm$ 20,300 psi assuming completely elastic behaviour.

f) RCC Guide System

The RCC guide system will not be adversely affected following either a hot or cold leg break. Compressive stresses (approximately 16,800 psi), which are below the yield strength of the material will be present in the fuel assembly thimbles without affecting the integrity and/or the stability of the system.

During a hot leg break, cross flow in region above the core will stress the guide column near the outlet nozzle leading to the break producing a maximum bending stress of 6,000 psi for the initial flow peak immediately after the break. Then the flow reduces to a value below the normal steady-state flow..

During a cold leg break the effect of transverse flow across the guide columns is much smaller than for the hot leg break.

In each of the cases described above, the stresses and deformations which would result following either a hot or cold leg break are less than those which would adversely affect the integrity of the structures. Also, the natural and applied frequencies are such that resonance problems would not occur. Therefore, it is concluded that:

a) The forces imposed, due to either a hot or cold leg break, can be sustained by the internals system,

b) The integrity of the internals system is maintained, and,

c) The basic requirements described in Section 12.3.2.3 are met.

12.3.2.6 Containment Pressure and Temperature Transient Analysis

Calculation of containment pressure and temperature transients is accomplished by use of the digital computer code, COCO. The analytical model is restricted to the containment volume and structure. Transient phenomena within the reactor coolant system affect containment conditions by means of convective mass and energy transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into two systems. The first system consists of the air-steam phase, while the second is the water phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam, and subcooled or saturated water. Switching between states is handled automatically by the code. The following are the major assumptions made in the analysis:

i) Discharge mass and energy flow rates through the reactor coolant system break are established from the coolant blowdown and core thermal transient analysis (described in the preceding paragraphs).

ii) At the break point, the discharge flow separates into steam and water phases. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment.

ii) Homogeneous mixing is assumed. The steam-air mixture and the water phase have uniform properties. More specifically, thermal equilibrium between the air and steam is assumed. This does not imply thermal equilibrium between the steam-air mixture and the water phase.

iv) Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

During the transient, there is energy transfer from the steam-air and water systems to the internal structures and equipment within the shell.

Provision is made in the containment pressure transient analysis for heat transfer through, and heat storage in, both interior and exterior walls. Every wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite difference form accounts for transient conduction into and out of the node as well as temperature rise of the node. The film heat transfer coefficients between the containment contents and a wall surface may be specified either as a function of time or temperature difference. For this study, test values obtained by Kolflat[6] were used to set inner surface film coefficients for condensing steam. Initial values of 620 and 240 BTU/hr-ft^2-°F were assumed for condensation on steel and concrete, respectively, as recommended in the proposed ASA Section N-6.7, Proposed Design And Pressure Decay Requirements. The coefficients were then linearly decreased with time to a value of 40 BTU/hr-ft^2-°F at the end of the blowdown.

The latter value is typical of steady state heat transfer coefficients for steam condensing through a stagnant surface air layer where the ratio of air to steam is high. The former values are reasonable since the initial blowdown will produce large amounts of steam in the containment together with high turbulence. Thus the ratio of air to steam will be low and a turbulent boundary layer will exist leading to a high film coefficient of heat transfer. The assumed reduction in film coefficient is conservative

since, as predicted by Jacob and demonstrated by Uchida[7] the film coefficient will be a function of the ratio of air to steam in the containment which will not be reduced significantly until pressure reduction due to steam condensation occurs.

Provision is made in the computer analysis for the effects of several engineered safeguards, including internal spray, fan coolers, and recirculation of sump water. The heat removal from containment steam-air phase by internal spray is determined by allowing the spray water temperature to rise to the steam-air temperature. This procedure has been experimentally verified[8].

Finally, hot metal surfaces not cooled by safety injection water are simulated as hot walls in contact with the containment steam-air mixture. A small film heat transfer coefficient is employed to reflect actual condition since these surfaces are covered by stagnant steam inside the reactor coolant system.

12.3.2.7 Containment Pressure Transient Results

a) Containment Pressure Transient for a Reactor Coolant Break Assuming Partial Safeguards

Figure 12-10 shows the containment pressure transient resulting when only partial engineered safeguards are operating on emergency power. In addition to the three of four accumulators delivering to the core, the engineered safeguards started automatically are:

1) Two high head safety injection pumps (one assumed in the analysis)
2) One low head safety injection pump (residual heat removal pump)
3) Three air recirculation fan cooling units
4) One containment spray pump
5) One service water pump

One component cooling water pump and one residual heat exchanger and one auxiliary building vent fan are placed into service manually during the recirculation phase.

A 10 second delay in initiation of safety injection water flow via pumps and a 45 second delay in the start of containment spray flow is assumed. The residual heat exchangers are used for long-term recirculation cooling, and are not operated until 75% of the water in the refueling water storage tank is used.

The Reactor Coolant System blowdown is essentially completed at 10 seconds, at which time the containment pressure has reached an initial peak of 42 psig. Continued heat absorption by the steel and concrete in the containment and internal structure leads to a slight decrease from the peak pressure. For the reactor coolant pipe break, the assumption is also made that the injection lines feeding into the affected loop spill one third of the total injection water flow through the pipe break. As cool water from the accumulators fills the vessel, a portion of the injection water boils (due to core stored heat, residual heat generation, and stored heat in the core structure and vessel), and flows into the containment. The heat sinks more than compensate for the heat addition, and containment pressure remains below the blowdown pressure peak. The prompt injection and core cooldown limits the amount of zirconium-water to an insignificant amount.

After 106 seconds, the rate of residual heat generation plus stored heat release is not sufficient to raise the safety injection water temperature to the saturation point. Hence, the flow to the containment remains subcooled and has little influence on containment pressure. The containment pressure then decreases as steam is condensed by the action of the fan coolers and containment spray, as well as by continued energy absorption of the concrete and internal structure.

When the refueling water storage tank is depleted, containment spray and safety injection using refueling water is stopped. Hot water is then drawn from the bottom of the containment, and pumped back into the vessel through the residual heat exchangers by means of the recirculating water pumps. The flow spilling out the break continues to remain subcooled, and containment pressure decreases further.

The design philosophy requires that the cooling capacity with one spray pump operating be equal to that of 2-1/2 air recirculation coolers. Therefore virtually the same pressure transient shown by Figure 12-10 would be obtained for the following combinations of containment engineered safeguards.

1) Two containment spray pumps only, or
2) Three air recirculation coolers and one spray pump.

b) Containment Pressure Transients for Smaller Pipe Breaks

The double-ended rupture of the 29-inch I.D. piping near the reactor vessel outlet nozzle causes the most rapid blowdonw and initial containment pressurization rate. The initial discharge in slightly smaller breaks carries more residual heat and stored heat into the containment due to longer blowdown times. The analysis of a range of break sizes shows the greatest initial pressure peak occurs with a 3 square foot break.

Figure 12-11 compares the maximum hypothetical accident curve with partial operation of the engineered safeguards on emergency power to curves for three representative breaks.

a) A break area of 3 square feet (smaller than the double-ended 29-inch break but larger than the severance of the largest connecting line).

b) A break equivalent to the surge line (a 14" pipe).

c) A break equivalent to a 4-inch connecting line.

These results show that the containment pressure remains well below safe limits for any break. All of these cases were studied for the same conditions, which are:

1) Partially effective safeguards operating from the emergency power supply.

2) Three accumulators delivering to the core.

3) One low-head, residual heat removal pump effective after a delay when reactor coolant pressure falls below the pump shutoff head but no less than 20 seconds (the estimated starting time delay with only emergency power).

4) One high-head safety injection pump effective after 10 seconds delay (the estimated starting time delay with only emergency power).

5) Three of the five ventilation fan cooling units after 45 seconds delay.

6) One containment spray pump effective after 45 seconds delay (the estimated starting time delay with only diesel power).

7) One service water pump for fan coolers.

For an assumed rupture opening of three square feet, the pressure transient resulting from these conditions is depicted on Curve A of Figure 12-11. A longer time is required to reach the pressure peak because of the increased blowdown time, and the peak is slightly higher than that obtained for a reactor coolant pipe break.

Although there is more heat absorption by the containment structure, this is offset by the additional residual and stored heat transported into the containment during the lengthened blowdown.

For each of the smaller sized breaks, the analysis is the same. Each case shows a correspondingly lower initial peak.

In all cases, the engineered safeguards systems even when operating with partial effectiveness and operating on emergency power are adequate to prevent overpressuring the containment and to significantly reduce the pressure within half an hour.

12.3.3 ENVIRONMENTAL CONSEQUENCES OF LOSS OF COOLANT

Chapters 5 and 6 describe the protective systems and features of the unit which are specifically designed to limit the consequences of a major loss of coolant accident. The capability of the safety injection system for preventing melting of the fuel clad and the ability of the containment and containment cooling systems to absorb the blowdown resulting from a major loss of coolant have been discussed in Section 12.3.2. The capability of the safeguards in meeting dose limits set in 10 CFR 100 is demonstrated in this section.

For the purpose of evaluating radiation protection, a double-ended rupture of a reactor coolant loop is considered with partial effectiveness of the engineered safeguards, operating from the Unit No. 3 emergency power system. As shown in Section 12.3.2, the safety injection system, with only emergency power, will prevent clad melting and will limit zirconium-water reaction to an insignificant extent. As a result of the cladding temperature increase and the rapid system depressurization, however, cladding failure may result in the hotter regions of the core. Release of the inventory of the volatile fission products in the pellet cladding gap is assumed to occur during the time interval required to flood the core flooding by the Safety Injection System.

Containment leakage through unblocked paths at a rate of 0.1% of the containment volume per day is assumed during the one minute interval prior to isolation and seal water injection. The continuity of pressurization of the double penetrations and weld channels and operation of the isolation and seal water injection systems are not affected by loss of outside power. The total leakage is calculated to be 3.07 dose equivalent curie of iodine 131, and some noble gas and other fission products which are much less important radiologically than the iodine. The corresponding off-site dose from the leaking activity based on meteorology discussed later in this section is 1.64 rem to the thyroid and less than 0.25 mrem to the whole body at the nearest site boundary. These doses are well within 10 CFR 100 limits for accident conditions and essentially meet the greatly exposure limits set in 10 CFR 20 for normal operation.

If it is postulated further that leakage continues possibly because the Isolation Valve Seal Water System does not function in a leaking line, the iodine removal afforded by the spray system serves as adequate backup in meeting regulatory site limits. In Section a) which follows, the expected performance of the spray system is described. In Section b), the analysis for the resulting doses is presented.

a) Effectiveness of Spray System for Iodine Cleanup

The basis used for predicting iodine removal by the sodium thiosulfate spray follows the method described by Griffiths[14]. The fundamental assumption is that elemental iodine (I_2) absorbed by the liquid drop is rapidly reduced to the highly soluble iodide by the reagent sodium thiosulfate, with the result that the partial pressure of I_2 at the drop surface is very small compared with that in the atmosphere. The experimental work of Taylor[15], performed in a column of controlled geometry, confirms this assumption for the aqueous thiosulfate absorption process. These tests show the overall mass transfer rate of I_2 to be independent of liquid flow rate a parameter which would be expected to alter liquid film resistance. By contrast the transfer rate was found to be gas-velocity dependent, indicative of the fact that gas film is controlling. The HI present in the atmosphere is readily absorbed by the aqueous spray without chemical reaction; hence consideration is directed in this discussion to the behavior of elemental iodine (I_2).

In a gas film controlled process, the dependence of the mass transfer rate on film conditions is expressed by

$$V_g = \frac{D_v}{d} (2 + 0.6\ Re^{1/2}\ Sc^{1/3}) \qquad (1)$$

Where

V_g = transfer coefficient expressed moles transferred per unit time, area, and concentration differential, cm/sec.

D_v = diffusion coefficient for iodine in air, cm^2/sec.

d = drop diameter, cm.

Re = Reynolds number, $\frac{d\,v\,\rho}{\mu}$, where ρ and v and μ are the density, velocity, and viscosity of the vapor, respectively (consistent units).

Sc = Schmidt number, $\frac{\mu}{\rho\ D_v}$ (consistent units).

Equation (1) is well substantiated by experiments in a variety of systems in which the gas film resistance is controlling, as reported by Ranz and Marshall[16].

The removal capability for the spray is calculated on the basis of an exponential removal of the inorganic vapor form iodine as the spray passes through the containment atmosphere.

The removal of iodine is then calculated by the equation:

$$A_t = A_o\ e^{-\lambda_s t} \qquad (2)$$

Where

A_t = The amount of elemental iodine in the containment atmosphere at time t - curies

A_o = The amount of elemental iodine initially in the containment atmosphere - curies

λ_s = Elemental iodine removal coefficient as defined by equation (3) - sec^{-1}

The removal coefficient, λ_s, is calculated by the following relationship:

$$\lambda_s = V_g \frac{Sd}{V_c} \qquad sec^{-1} \qquad (3)$$

Where

V_g = Previously defined - cm/sec

S_d = Total surface area of drop in the containment volume - cm^2

V_c = Containment volume being covered by the spray - cm^3

The total surface area of the spray drops in the containment atmosphere is calculated on the following basis;

$$S_d = \frac{A_d \; F \; \bar{t}}{V_d} \qquad cm^2 \qquad (4)$$

Where

A_d = Surface area per drop - cm^2

V_d = Volume per drop - cm^3

F = Spray flow rate - cm^3/sec

$\bar{t}$ = Average residence time of the drop in the atmosphere - sec

For spherical drops, the ratio A_d/V_d is equal to 6/d where d is the drop diameter so that equation (4) now becomes;

$$S_d = \frac{6 \; F \; \bar{t}}{d} \qquad cm^2 \qquad (5)$$

and the removal factor λ_s is not

$$\lambda_s = \frac{6 \; V_g \; F \; t}{V_c \; d} \qquad (6) \qquad sec^{-1}$$

For the Indian Point Unit #3, the removal coefficient can be evaluated using the following parameters

$V_c = 2.6 \times 10^6 \text{ ft}^3$ - Free Volume of the containment.

$t = 2.0$ sec - Average fall time for the drops assuming a maximum fall velocity of 70 feet/sec and average fall distance of 140 feet. The fall velocity considers the exit velocity from the spray nozzles and gravity effects. The fall distance is assumed to be the minimum distance available (from the spray headers to the operating deck).

$F = 2600$ gpm - Design flow rate for one spray pump ($5.8 \text{ ft}^3/\text{sec}$)

$V_g = 14$ cm/sec - Based upon the data of Griffiths which gives a V_g/d ratio of 200 for a drop diameter of 700 microns.

The resulting removal coefficient is

$$\lambda_s = \frac{6 \times 5.8 \times 14 \times 2.0}{0.07 \times 2.6 \times 10^6} \quad \text{sec}^{-1}$$

$$\lambda_s = 5.35 \times 10^{-3} \quad \text{sec}^{-1} \quad \text{or}$$

$$\lambda_s = 19.3 \text{ hr}^{-1}$$

b) Off-Site Dose

Sources

All of the pellet-clad gap activity is assumed released to the containment atmosphere with no holdup in the fuel rods or coolant and coolant system components. It is assumed to mix homogeneously in the containment atmosphere where it is available for cleanup and also for leakage. The gap activity is computed based on buildup in the fuel from the fission process and diffusion to the fuel rod gap at rates dependent on the operating temperature. The fuel pellets are divided into five concentric rings each with release rate dependent on the mean fuel temperature within that ring. The diffusing isotope is assumed present in the gas gap when it has diffused to the boundary of its ring.

The diffusion coefficient, D', for Xe and Kr in UO_2, varies with temperature through the following expression:

$$D'(T) = K'(1400)\ e^{-\frac{E}{R}\left[\frac{1}{T} - \frac{1}{1673}\right]}$$

where

E = activation energy

D'(1400) = diffusion coefficient at 1400°C

T = temperature in °K.

The above expression is valid for temperatures above 1100°C. Below 1100°C fission gas release occurs mainly by two temperature independent phenomena, recoil and knock-out, and is predicted by using D' at 1100°C. The value used for D'(1400°C) is based on data at burnups greater than 10^{19} fissions/cc to account for possible fission gas release by other mechanisms and pellet cracking during irradiation.

The diffusion coefficient for iodine isotopes is assumed to be the same as for Xe and Kr. Toner and Scott[19] observed that iodine diffused in UO_2 at about the same rate as Xe and Kr and had about the same activation energy. Data surveyed and reported by Belle[20] indicates that iodine diffuses at slightly slower rates than do Xe and Kr.

For a full core cycle at the 3217 Mwt expected stretch capability of the plant, the above analysis results in a pellet-clad gap activity of less than 3% of the dose equivalent equilibrium core iodine inventory. The noble gas activity present in the pellet-clad gap and assumed released to the containment is about 2.5% of the core inventory.

Calculations are based on 0.2% of the released gap iodine issuing in the organic form as methyl iodide, and subsequent buildup at the rate of 0.05% per hour by deposition of elemental iodine, conversion to methyl iodide and desorption of the methyl iodide.

The assumption of 0.2% CH_3I is believed to be pessimistic because of the tendency of the molecules to decompose under the influence of heat and radiation in the fuel. If 0.2% of the iodine were to exist in the fuel as CH_3I, the first-order rate constant for decomposition of CH_3I molecules would have to be less than 500 times the radioactive decay constant for the isotope in question (that is, less than 5×10^{-4} sec^{-1} for the biologically dominant isotope I-131). The thermal decomposition rate of CH_3I in the high beta and gamma fluxes attained in the fuel would be expected to add substantially to the total decomposition rate. This assumption of 0.2% as the relative inventory of I-131 as CH_3I is therefore conservative.

The rate of formation of CH_3I by desorption from surfaces was estimated from experiments simulating the conditions for such a process. These experiments, performed at Zenith[11], at Hanford[12], and at Oak Ridge[13] measured CH_3I formation in enclosures following release and deposition of I_2 in a moist air environment. The Zenith and Hanford data were attained in painted enclosures, while the ORNL data were obtained in stainless steel systems. The yields, when corrected for the surface to volume ratios of the respective surface materials in the containment were in every case less than 0.005% of the iodine inventory converted to CH_3I per hour of exposure. For the present analysis, the assumed formation rate of 0.05% per hour is therefore conservative.

Off-site Inhalation Dose

The inhalation dose rate is expressed as

$$Dr(t) = A(t) \times L(t) \times \frac{X}{Q}(t) \times B(t) \times DCF \quad \text{rem to thyroid per unit time}$$

where A - is the activity in the containment vapor space available for leakage (curies) - dependent on decay and halogen removal by safeguards

L - is the containment fractional leak rate - dependent on containment pressure

X/Q - is the dispersion (dilution) factor from the containment to the point of interest - dependent on wind conditions and velocity as well as distance from containment.

B - is the breathing rate of a person assumed immersed in the leakage plume

DCF - Is the dose conversion factor - the rem incurred per curie of iodine inhaled.

Inorganic and organic iodine activity present in the containment atmosphere following the assumed accident is evaluated with the following balance on source and removal terms.

$$\frac{d\,A_{inorg}}{dt}(t) = -(\lambda_S + \lambda_D)\,A_{inorg}(t)$$

$$\frac{d\,A_{org}(t)}{dt} = F\,A_{inorg}(t) - \lambda_D\,A_{org}(t)$$

where the λ's represent removal rate coefficients for the processes of - spray - and decay. F is the formation rate (5×10^{-4} hr^{-1}) of methyl iodine from elemental iodine by the process of plate out, conversion and desorption discussed previously.

The dose equivalent curie source of I-131 in the pellet clad gap and assumed released to the containment atmosphere during the depressurization is 4.42×10^6 curies. Of this, 0.2% is assumed in the form of methyl iodine with the remainder elemental iodine.

Dispersion

The meteorological dispersion of the leakage from the containment has been calculated using the Sutton dispersion model and dispersion parameters measured at the Indian Point site. The Sutton model has been modified to account for additional dispersion of the leakage due to turbulence in the wake of the containment building. Conservative dispersion characteristics applicable to three time periods were selected and the doses calculated for each period.

The Sutton equation for the dispersion of a point source at ground level gives the ground level plume concentration as a function of distance:

$$X = \frac{2Q}{\pi \bar{u} C_y C_z x^{2-n}} \quad e- \left(\frac{y^2}{C_y^2 x^{2-n}} \right)$$

where C_y C_z and n are the dispersion parameters, $\bar{u}$ is the wind speed, y is the lateral distance from the plume center line, x is the downward distance and Q is the point source release term.

In order to take into account building dilution, the Sutton equation is applied to a virtual point source upwind from the containment. The distance of this source from the building is obtained by the requirements that the dispersion factors σ_y and σ_z of the gaussian distribution obey the relationships:

$$4 \sigma_y = \sqrt{A}$$

$$4 \sigma_z = \sqrt{A}$$

where A is the corss sectional area of the containment building. Thus σ_y and σ_z yield each a value for the distance; the geometric average of those values is the distance x_o upwind of the virtual source.

$$x_o = \left(\frac{A}{8\ C_y\ C_z} \right)^{\frac{1}{2-n}}$$

The modified Sutton equation becomes:

$$x = \frac{2Q}{\pi\ \bar{u}\ C_y\ C_z\ (x + x_o)^{2-n}} \quad e^{-\frac{y^2}{C_y^{\ 2}}} \ (x + x_o)^{n-2}$$

The first and second periods of the dose calculation utilized this modified dispersion formula, a building area of 2000 square meters, and the inversion parameters assumed in TID 14844 which are conservative for the Indian Point site.

Category	C_y	C_z	n	$\bar{u}$	x_o
Inversion-I	$.4m^{n/2}$	$.07\ m^{n/2}$	.5	1 m/sec	430 m

The first period comprises the first two hours after the accident. The direction of the 1 meter per second wind is assumed to be constant throughout the period.

The second period is the next 22 hours following the accident, during which the same inversion condition exists, but the average wind speed from the same direction is assumed to be 2 meters per second.

The third period is from 24 hours after the accident to 30 days after the accident. During this period, the meteorological conditions are assumed to be randomly distributed among the categories listed below.

Category i	Fraction Fi	$1/\bar{u}$	C_z	C_y	n
Lapse - L_1	0.137	0.575	0.48	0.6	0.2
Lapse - L_2	0.061	0.191	0.43	0.53	0.3
Neutral - N	0.378	0.358	0.39	0.47	0.4
Inversion - I	0.424	0.493	0.07	0.40	0.5

The parameters $\bar{u}$, C_y, C_z, n for L_1, L_2 and N are those measured at the site, and those for I are again the TID 14844 assumptions. Because the winds are not expected to be from the same direction throughout the 30 day period, the dispersion formula was modified to account for long term variability of the mean wind direction. The most adverse distribution was assumed to result in a maximum of 35 per cent of the winds blowing in one 20° sector. The dispersion formula used is:

$$(X/Q)_{30} = \frac{2f}{\beta\sqrt{\pi}} \sum_i \frac{Fi}{\bar{u}_i \, C_{z_i} \, x^{\left(2 - \frac{ni}{2}\right)}}$$

This expression is obtained by integrating the Sutton equation from $-\infty$ to $+\infty$ in the y direction and then averaging the concentration over the desired sector, β, for the appropriate fraction of the time, f. The other parameters have been defined with F_i being the fraction of the time any particular weather category exists. As stated, $\beta = 0.353$ (2 tan 10°) and $f = +0.35$.

Based on the above data, the site dispersion factors are obtained.

SITE DISPERSION FACTORS

Distance (meters)	(X/Q) 2 hours (sec/m^3)	(X/Q) 22 hours (sec/m^3)	(X/Q) 30 days (sec/m^3)
320	1.1×10^{-3}	5.5×10^{-4}	
400	9.51×10^{-4}	4.75×10^{-4}	1.03×10^{-4}
700	5.98×10^{-4}	2.99×10^{-4}	3.87×10^{-4}
1,000	4.20×10^{-4}	2.10×10^{-4}	2.07×10^{-5}
2,000	1.90×10^{-4}	9.50×10^{-5}	6.13×10^{-6}
4,000	7.68×10^{-5}	3.84×10^{-5}	1.82×10^{-6}
7,000	3.55×10^{-5}	1.77×10^{-5}	6.79×10^{-7}
10,000	2.14×10^{-5}	1.07×10^{-5}	3.63×10^{-7}
20,000	7.78×10^{-6}	3.89×10^{-6}	1.07×10^{-7}

The containment leak rate is assumed to be at the design value of 0.1% per day for the first 24 hours and then to drop to 0.045% per day for the next 29 days.

Adult breathing rate of 1.25 m^3/hr, 0.795 m^3/hr and 20 m^3/day are assumed for the periods 0-2 hr, 2-24 hr, 1 day-30 day respectively.

RESULTS

The effectiveness of the various safeguards in limiting thyroid dose are evaluated in terms of the reduction in off-site exposure which is achieved, relative to the exposure which would result if the safeguards were not provided. The reference case where no safeguards are employed is evaluated using the fission product release and transport model described in TID-14844.[6]

The Safety Injection System by preventing core meltdown and limiting iodine activity release to the 3% of the core inventory contained in the coolant and pellet-cladding gaps achieves a reduction of 25/3 in the iodine source available for leakage. Since the Safety Injection System limits the source its dose reduction factor of 8.35 is independent of the other safeguards operating and of the location or duration of exposure.

The dose reduction effect of the Isolation Valve Seal Water System in terminating leakage in one minute would be a factor of 120 in two hours if it were the only safeguard system provided for reducing iodine leakage. However since the Containment Spray System will reduce the iodine leakage by a factor of 38.6 (based on only one of the two spray pumps) in two hours with no termination of leakage, the actual two hour dose reduction factor for the Isolation Valve Seal Water System is 120/38.6. The dose reduction factor for both the isolation system and the spray system are functions of the exposure duration while the dose reduction achieved by the spray system is also a function of the exposure distance. Dose reduction factors are summarized in Table 12-2 for the two hour period at the site boundary and the 30 day period at the low population zone.

TABLE 12-2

Dose Reduction Factors, Relative to TID-14844 Calculation Model

	Dose Reduction Factor at Site Boundary (350m) 2 Hour Exposure	Dose Reduction Factor at Low Population Zone (1100m) Duration of Accident
1. Safety Injection System Operating On Emergency power	8.35	8.35
2. Isolation Valve Seal Water System		
a) No spray removal	120	510
b) One Thiosulfate spray pump	3.1	3.4
3. One Of Two Spray Pumps	38.6	148

The dose reduction factor required to meet 10 CFR 100 limits are 5.6 at the site boundary in two hours and 8.6 at the low population zone in 30 days. The reduction in dose afforded by either the spray system or the isolation system is seen to easily meet 10 CFR 100 requirements even if no credit were to be taken for the reduction in leakage source obtained in preventing meltdown.

The closest approach to the plant site boundary is 350 meters from the surface of the reactor containment. The two hour thyroid dose at this exclusion distance is 5.2 rem without termination of containment leakage and with only one of the two spray pumps functioning. For the same condition the 30 day thyroid dose at 1100 meter low population zone is 2.1 rem.

In addition to the thyroid inhalation exposure, the whole body exposures were also evaluated, both due to direct radiation from the containment and from exposure due to immersion in the leakage plume.

The whole body gamma exposure due to immersion in the plume of leaking gaseous fission products released from the pellet-clad gap during depressurization is about 110 mrem at the site boundary in two hours with no isolation and about 1 mrem when containment leakage is terminated in one minute. With no isolation the corresponding 30 day dose at the low population zone is less then 300 mrem.

For the direct dose the sources are assumed to be homogeneously distributed within the free volume of the reactor containment. The source intensity as a function of time after the accident is determined by considering natural decay only, with no credit taken for removal by sprays or washdown. The dose is based on a point kernel attenuation model, with the source region divided into a number of incremental source volumes, and the associated attenuation and gamma ray buildup computed between each source point and the dose point. The combined direct and scattered radiation dose from the confined activity is less than 0.03 mrem for two hours and less than 0.3 mrem for the duration of the accidents at distances beyond the site boundary,

Table 12-3 presents a summary of the doses calculated for the hypothetical double ended rupture of a coolant loop.

TABLE 12-3

Offsite Exposure for Loss of Coolant Accident Resulting in Release of Pellet-Clad Gap Activity to Containment

	2 Hour Exposure at 350 Meters (Min. Exclusion Radius)	Total Exposure at 1100 Meters (Min. Low Population Zone Radius)
I. Thyroid Dose		
a. Leakage Terminated in one Minute by Isolation Valve Seal Water System.	1.6 rem	0.6 rem

b. Continuous Leakage With One Of The Two Post Accident Sprays Operating	5.2 rem	2.1 rem
c. 10 CFR 100 Limit	300 rem	300 rem

II Whole Body Dose

a. Immersion in Leakage Plume		
1. No Isolation	110 mrem	300 mrem
2. Isolation Terminates Leakage in One Min.	1 mrem	<< 1 mrem
b. Direct Dose From Confined Activity	0.03 mrem	<<0.03 mrem
c. 10 CFR 100 Limit	25 mrem	25 rem

Doses in the control area of Unit No. 3 were evaluated. The direct whole body dose from the confined activity is 9 mr in 2 hours and 25 mr in 1 week with no consideration taken for the shielding afforded by the control room structure. The combined effectiveness of the Isolation Valve Seal Water System and the Penetration and Weld Channel Pressurization System in terminating containment leakage in one minute, limits the thyroid dose to 4 rem assuming only building wake dilution. The corresponding doses in the control areas of Unit No's. 1 and 2 are less because of the greater separation distance.

Recirculation Leakage

Subsequent to the emptying of the refueling water storage tank during the initial phase of safety injection and containment spray, water from the containment sump is recirculated by the recirculating pumps and cooled via the residual heat exchangers and then returned to the reactor system. The recirculation pumps and the residual heat exchangers are located inside the containment hence for a big break coolant would not be circulated outside the containment. Provisions will be made, however, to recirculate through the safety injection and residual heat removal pumps if needed. Because the loss of coolant accident may cause the sump water to contain radioactivity, the potential off-site exposures due to operation of these external recirculation paths have been evaluated.

As shown in Section 6.2, the maximum estimated leakage to the auxiliary building from the components and joints of the safety injection system components during recirculation is approximately 1000 cubic centimeters per hour.

During the recirculation phase of post-accident cooling, the sump water temperature is calculated to be about 200°F at the initiation of recirculation so that essentially no leakage will flash to vapor. For conservatism in the analysis, it is assumed that approximately 10 per cent of the leakage or 100 cc/hr will vaporize and carry the entrained iodine to the atmosphere of the auxiliary building for a period of 1 hour following initiation of recirculation. The auxiliary building ventilation system will discharge the vapor through absolute filters and out the plant vent.

It was also assumed that all of the released iodine activity is in the water in the sump which has a total volume of about 1.3×10^9 cc including reactor coolant and injection water. The combined leakage from the recirculation subsystem results in a dose of about 0.1 rem to the thyroid in the one hour at the site boundary. The actual dose would be negligible as the temperature of the recirculated water will be substantially reduced so that little or no vaporization should occur. Also, the iodine in the sump will be combined with the thiosulfate from the containment spray and will remain trapped in any leakage.

d) Population Center Considerations

The preceding discussion showed that engineered safeguards adequately protect persons at the site exclusion radius and at the low population zone radius from accidental exposure in excess of the limits for these distances set forth in 10 CFR 100. This regulation also requires that the reactor be so situated that there is no population center (defined as a city of 25,000) having its nearest boundary closer than 1-1/3 times the low population zone radius. The nearest boundary of the population center (the town of Peekskill) lies at a minimum of distance of 1.4 km of the reactor which is 1-1/3 times the low population zone radius of 1.1 km.

To illustrate further the population dose factors at this site, it can be shown that in no sector affected by a release at Indian Point would the population dose exceed the man-rem limit implicit in the population center provision of 10 CFR 100. The residents of a hypothetical town of 25,000 located at 1-1/3 times the low population zone distance would receive an integrated thyroid dose of about 4×10^6 man-rem from a cloud which gives the limiting dose of 300 rem per man at the low population zone radius. The calculated man-rem dose due to passage of the same cloud into either of two highly populated 20° sectors from Indian Point are presented in the following tables. The South sector, including parts of lower Westchester County and New York City would receive about 1.4×10^6 man-rem. The Northeast sectors, which takes in the more proximate populated areas including Peekskill, receives only 0.86×10^6 man-rem. Both sectors receive much less than the reference population group representing the regulatory guidelines.

In this evaluation, projected 1980 population figures were used. The simplifying assumption was made that cloud dosage varies as the -1.8 power of downwind distance.

TABLE 12-4
INDIAN POINT SITE
POPULATION DISTRIBUTION
AND INTEGRATED MAN-REM*

South 20° Sector Through Manhattan

Distance (miles)	Population (1980)	Man-Rem
0-5	5200	119,200
5-10	45,600	165,800
10-15	50,320	77,300
15-25	498,740	336,000
25-35	1,488,435	345,300
35-45	2,130,400	375,000
45-55	1,380,110	168,400
Total	5,598,805	1,587,000

*Based on 300 Rem ∞ Dose at 1100 Meters

TABLE 12-5
INDIAN POINT SITE
POPULATION DISTRIBUTION
AND INTEGRATED MAN-REM*

Northeast 20° Sector Through Peekskill

Distance (miles)	Population (1980)	Man-Rem
0-5	30,140	691,000
5-10	7,250	26,400
10-15	6,650	10,200
15-25	24,650	18,100
25-35	20,515	4,800
35-45	15,375	2,700
45-55	47,880	5,800
Total	152,460	759,000

*Based on 300 Rem ∞ Dose at 1100 Meters

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e) Effects of Rainout after the Hypothetical Loss of Coolant Accident

An evaluation has been performed to determine the effects of rainout after the hypothetical accident. This evaluation has been done for two reservoirs within the five-mile radius which have limited alternate water suppliers. They are the Campfield reservoir of the City of Peekskill and the Queensboro Lake which supplies the Bear Mountain park facilities.

	Campfield Reservoir	Queensboro Lake
Direction from site	NE	NW
Distance from site	4,700 m	8,000 m
Average area	44,500 m^2	150,000 m^2
Average capacity	200,000 m^3	210,000 m^3
Average depth	4.5 m	1.4 m

Rainout has been considered for two typical weather conditions. The first covers the case of continuous rain lasting 24 hours with a rainfall of 0.05 inch per hour over an area including the reactor site and the reservoir; the second assumes a thunderstorm lasting one hour just above the reservoir and no rain between the site and the reservoir.

Washout and maximum first day concentrations are computed first. Yearly average concentrations are then obtained on the basis of a 8.05 day half-life; they are compared with the Maximum Permissible Concentration of 10 CFR 20 for I^{131}, 3×10^{-7} $\mu c/ml$.

Dose calculations were then made, assuming: (1) that the washout was totally and uniformly mixed in the reservoir water during the first day; (2) that an individual with a 20 gram thyroid consumed 1200 ml/day of reservoir water and another with a 2 gram thyroid consumed 300 ml/day; (3) that 30% of the ingested I^{131} reached the thyroid.

The yardstick by which these doses can be assessed is given in Report #5 of the Federal Radiation Council which provides guidance as to the level of exposure at which protective measures should be taken. The Federal Radiation Council proposes that protective action against contamination resulting from a single release of I^{131} need not be taken unless the thyroid dose is expected to exceed 1/10 of 10 CFR 100 guidelines in some individuals. The Council assumes that 1/10 of 10 CFR 100 will not be exceeded in any individuals if the per capita dose does not exceed 1/30 of 10 CFR 100. Thus, the yardstick to be used here is 10 rems.

1. Continuous rain -

   It is assumed that

   a) a steady rain (.05 in/hr) persists for the first twenty-four hours after the accident.
   b) the wind direction remains in a fixed 20° sector for 24 consecutive hours. The hourly plume centerlines are uniformly distributed in the 20° sector. Wind speed is 1 m/sec. ($\bar{u}$)
   c) the stability conditions are neutral.
   d) the rate of removal from the cloud, Λ, is 10^{-4} sec^{-1} (see AECU-3066 Meteorology & Atomic Energy, Fig. 7.5).

The average washout in curies per square meter resulting from these assumptions is:

$$\text{Washout} - \frac{(57°)}{20°} \frac{\Lambda Q}{x u} e^{-\frac{\Lambda x}{u}}$$

Q is the total number of curies released. Assuming that a leak rate of 0.1% per day exists for the first 1 minute and that the gap activity is immediately available for leakage, Q is equal to 1.65 cuires of I^{131}. The results are:

	Campfield	Queensboro
Washout ($\mu c/m^2$)	6.2×10^{-2}	2.5×10^{-2}
First day concentration ($\mu c/ml$)	1.4×10^{-8}	1.9×10^{-8}
Yearly aver. concentration ($\mu c/ml$)	4.3×10^{-10}	5.9×10^{-10}
Yearly aver. concentration /(MPC)	1.5×10^{-3}	2.0×10^{-3}
Dose -20 gr thyroid (rem)	4.5×10^{-4}	6.1×10^{-4}
Dose -2 gr thyroid (rem)	1.2×10^{-3}	1.5×10^{-3}

For the case where the containment is assumed to be leaking at the rate of 0.1 percent of the contained volume per day, the total amount of iodine-131 leaking in 24 hours, considering one of the two containment spray units operating, is approximately 10 curies, the corresponding doses are 6 times larger.

The consequences of a continuous rain after the accident would thus be well below the doses at which the Federal Radiation Council recommends that preventive measures be taken.

2. Thunderstorm -

The following model is used: the thunderstorm extends over a three square mile area centered at the reservoir and lasts one hour. All the iodine released during the first hour after the accident travel to and are entrained in the thunderstorm circulation.

A the rate of iodine removal from the cloud (2×10^{-4} sec^{-1})

Q = cuire of I^{131}

$$\text{Washout} = \frac{Q\,(1 - e^{-\Lambda t})}{7.68 \times 10^{6}} \quad \text{curies/m}^2$$

This formula yields doses of 2 mrem and 6.5 mrem at Campfield and Queensboro respectively for the 2 gm thyroid.

REFERENCES - CHAPTER 12

1. Curran, R. M., "History of the Special ASTM Task Force on Large Turbine and Generator Rotors" ASTM Meeting, Purdue University, 1965

2. Baker, L., "Studies of Metal-Water Reactions at High Temperatures, III Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548

3. Epstein, L. F., "Metal-Water Reactions, VIII," GEAP-3279

4. Lustman, B., "Zirconium-Water Reactions," WAPD-137

5. Margolis, S. G. and Redfield, S. A., "FLASH: A Program for Digital Solution of the Loss of Coolant Accident," WAPD-TM-534, May, 1966

6. A. Kolflat and W. A. Chittenden, "A New Approach to the Design of Containment Shells for Atomic Power Plants," Proceedings of American Power Conference, Pg. 651, Volume XIX, 1957

7. Uchida, "Evaluation of Post-Incident Cooling Systems of Light Water Power Reactors," 3rd U. S. Int'l Conference on Peaceful Uses of Atomic Energy, Vienna, May, 1964

8. D. Hasson, D. Luss, U. Navon, "An Experimental Study of Steam Condensation on a Laminar Water Sheet," Int. J. Heat Mass Transfer 7, 983-1001 (1964)

9. M. B. Zabetakis, "Research on the Combustion and Explosion Hazards of Hydrogen-Water Vapor-Air Mixture," AECU-3327, Sept. 1965

10. TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites," J. J. DiNunno, et. al. USAEC, March, 1962

11. J. F. Croft, et. al. "Experiments on the Deposition of Airborne Iodine of High Concentration," AEEW-R265, June, 1963

12. J. D. McCormack and R. K. Hilliard, "Natural Removal of Fission Products Released from UO_2 Fuel in Condensing Steam Environments," International Symposium on Fission Product and Transport Under Accident Conditions, CONF-650407, April, 1965

13. G. W. Parker, et. al. "Behavior of Fission Products Released from Simulated Fuels in the Containment Mockup Facility," p. 83, Nuclear Safety Program Semiannual Progress Report for Period Ending June 30, 1965, ORNL-3843

14. V. Griffiths, "The Removal of Iodine from the Atmosphere by Sprays," AHSB (S) R 45, Dec. 3, 1962.

15. R. F. Taylor, "Absorption of Iodine Vapor by Aqueous Solutions," Chemical Engineering Science, 1959, Vol. 10, pp. 68-79

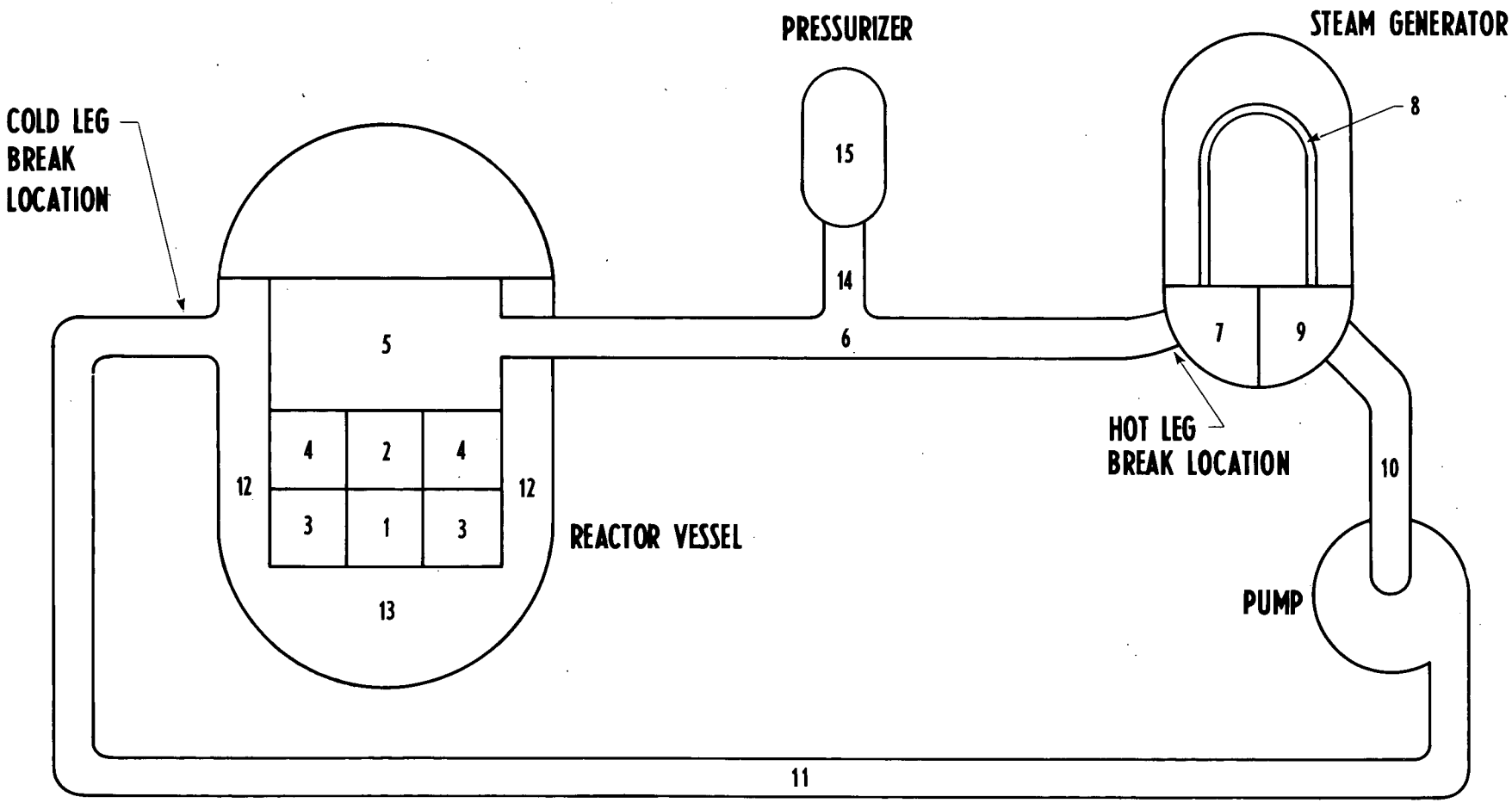
16 W. E. Ranz and W. R. Marshall, Jr., "Evaporation from Drops," Chemical Engineering Progress, Vol. 48, No. 3, p. 141 and No. 4, p. 173

17. Isletzer, N. F., "Aerodynamic Effect of Large Reactor Complexes upon Atmospheric Turbulence and Diffusion," IDO-12041.

18. Redfield, V. A., "CHIC-KIN A Fortran Program for Intermediate and Fast Transients in a Water Moderated Reactor" WAPD-TM-479, Jan. 1, 1965.

19. Toner, D. F., and Scott, J. L., "Fission Product Release from UO_2," Nuclear Safety, Vol. 3, No. 2, Dec. 1961.

20. Belle, J., Uranium Dioxide: Properties and Nuclear Applications, Naval Reactors, Division of Reactor Development United States Atomic Energy Commission, 1961.



REACTOR COOLANT SYSTEM ELEMENT DESIGNATION FOR SATAN CODE

FIG. 12-1

REACTOR COOLANT SYSTEM ELEMENT DESIGNATION FOR SATAN CODE 2 LOOP ANALYSIS

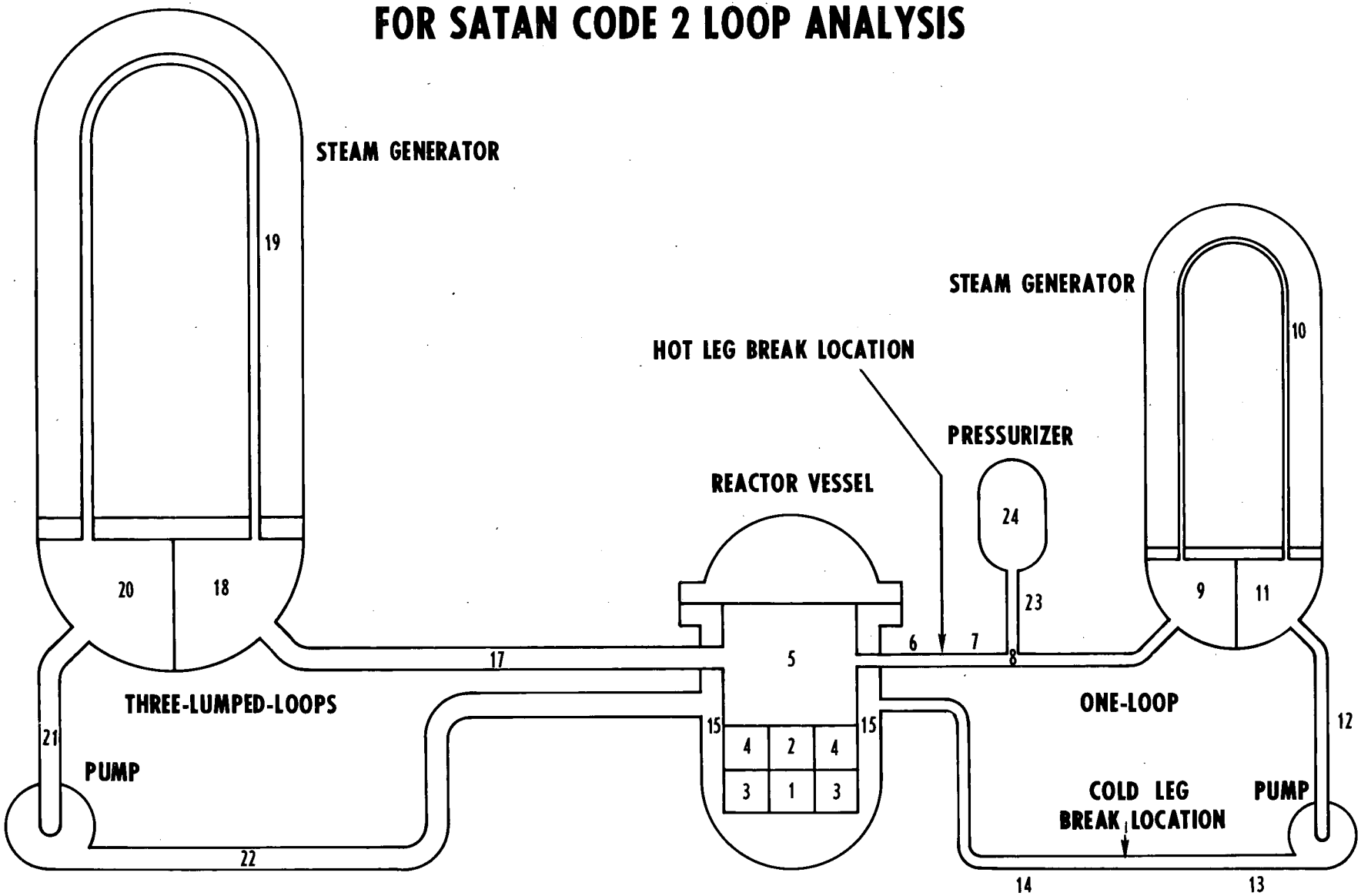


FIG. 12-2

A COMPARISON OF SATAN CODE RESULTS WITH LOFT EXPERIMENT (RESERVOIR BLOWDOWN)

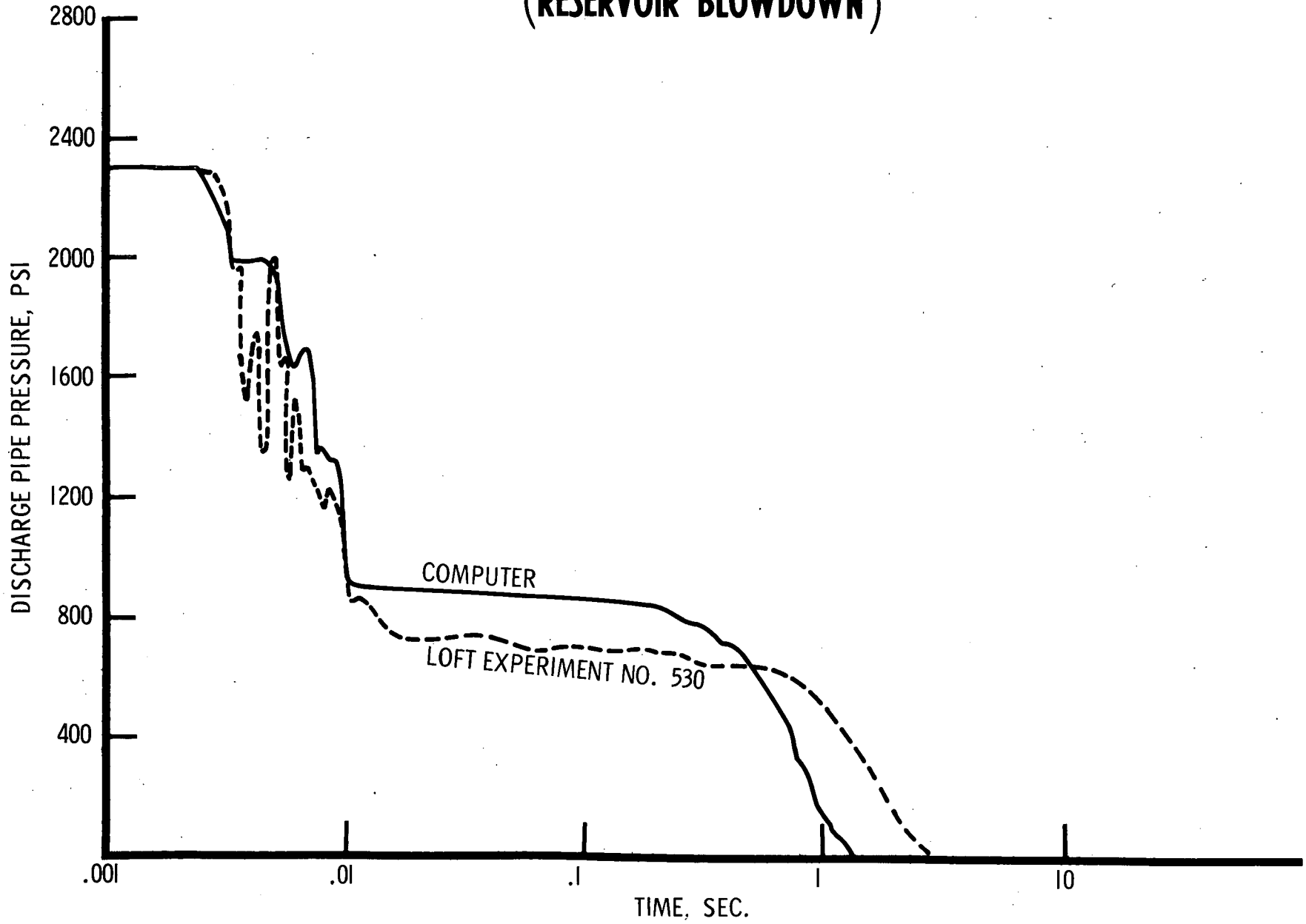


FIG. 12-3

A COMPARISON OF SATAN CODE RESULTS WITH LOFT EXPERIMENT (RESERVOIR BLOWDOWN)

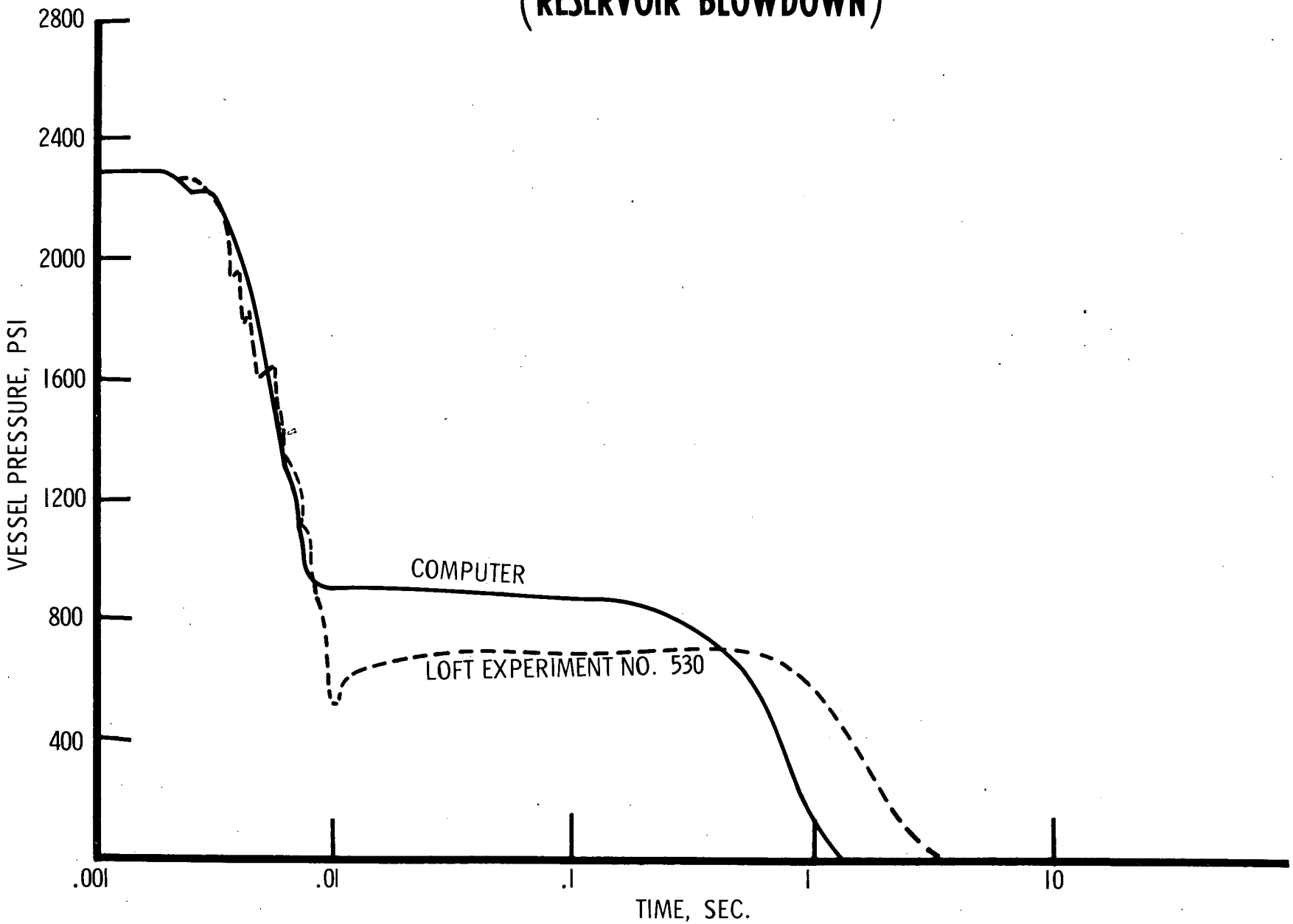


FIG. 12-4

A COMPARISON OF SATAN CODE RESULTS WITH LOFT EXPERIMENT (RESERVOIR BLOWDOWN)

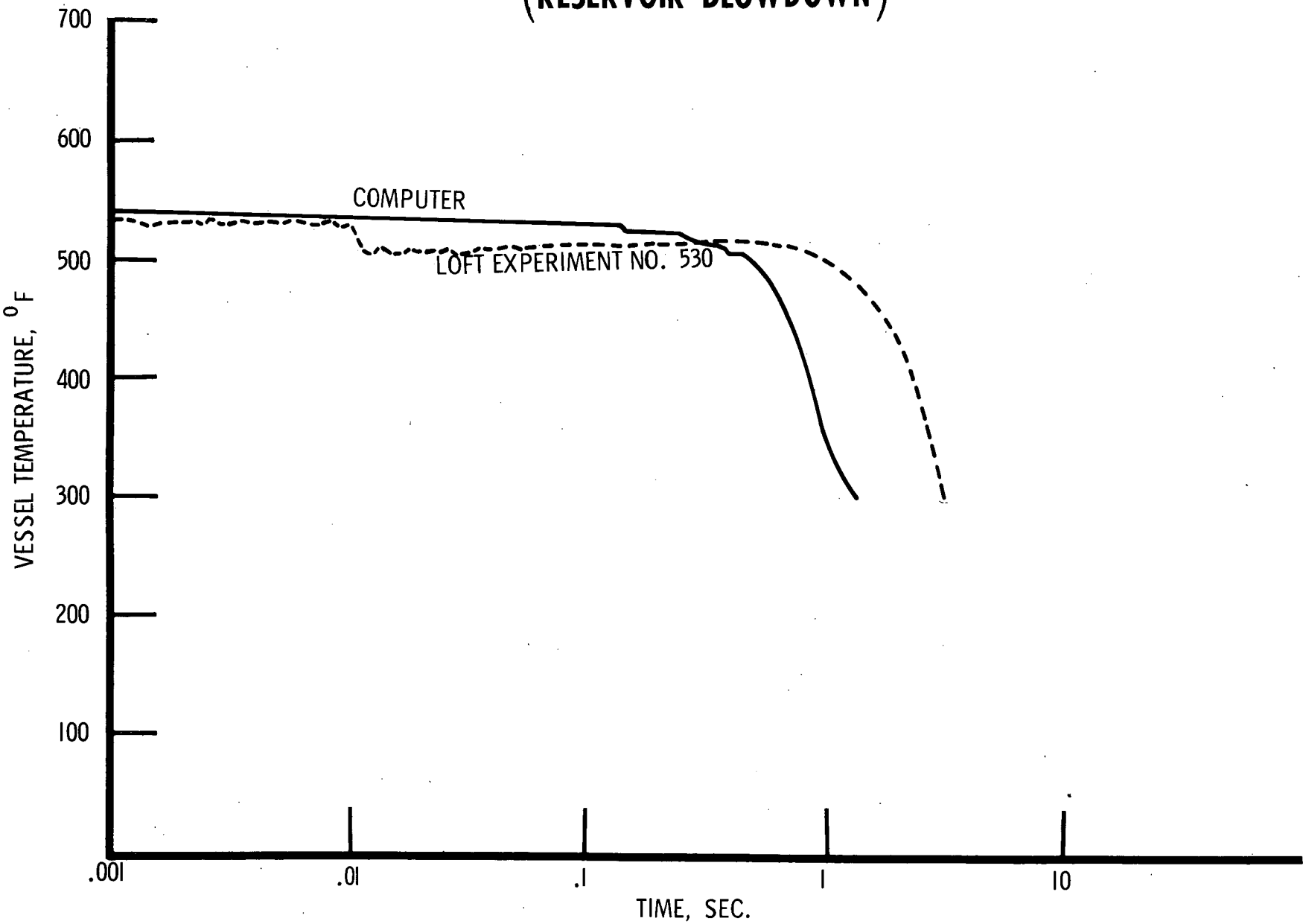


FIG. 12-5

A COMPARISON OF SATAN CODE RESULTS WITH LOFT EXPERIMENT

(RESERVOIR BLOWDOWN)

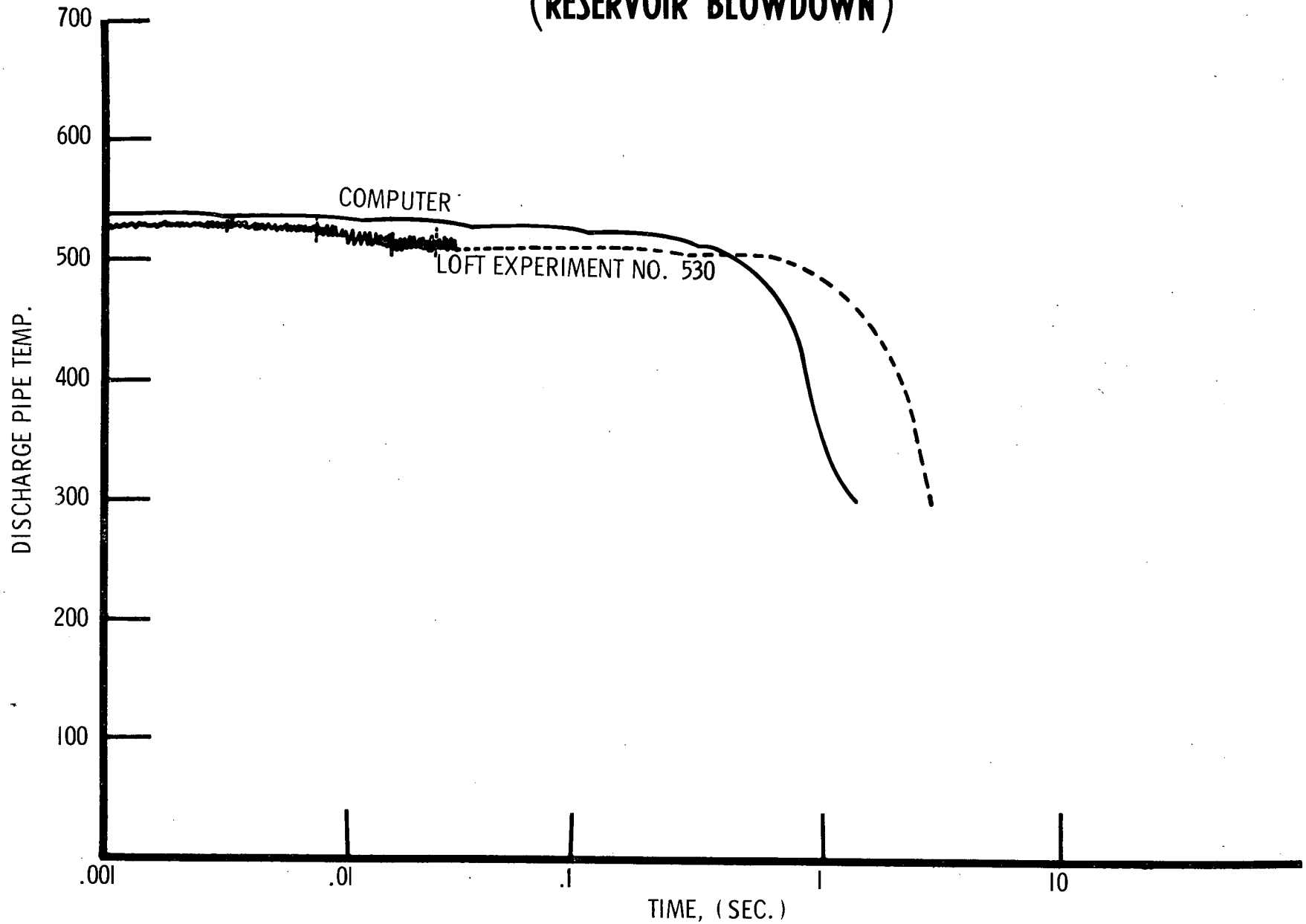


FIG. 12-6

HOT LEG BREAK–0.0 SEC. RUPTURE TIME–PRESSURE DIFFERENCE ACROSS THE CORE (POSITIVE UPWARD)

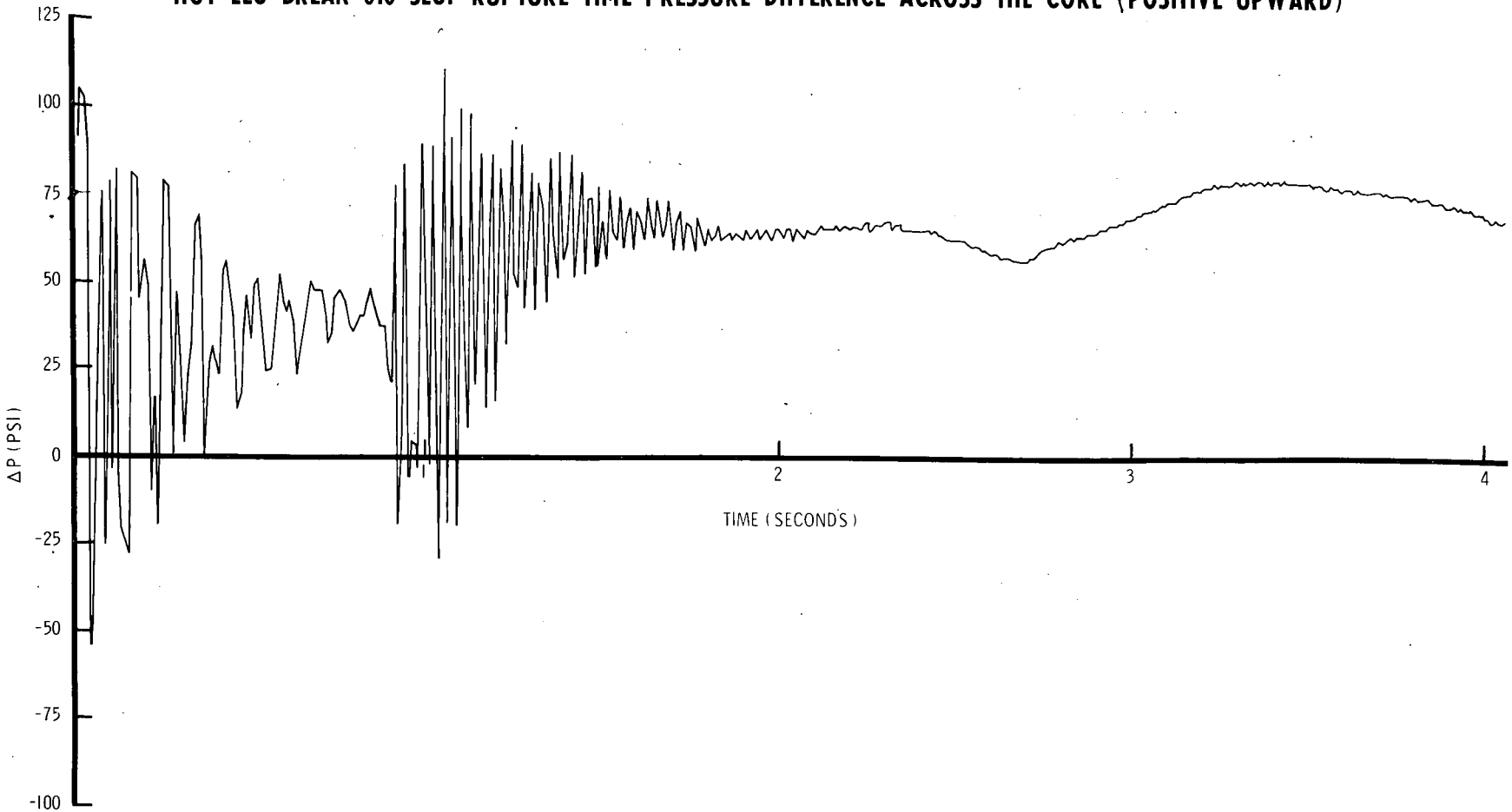


FIG. 12-7

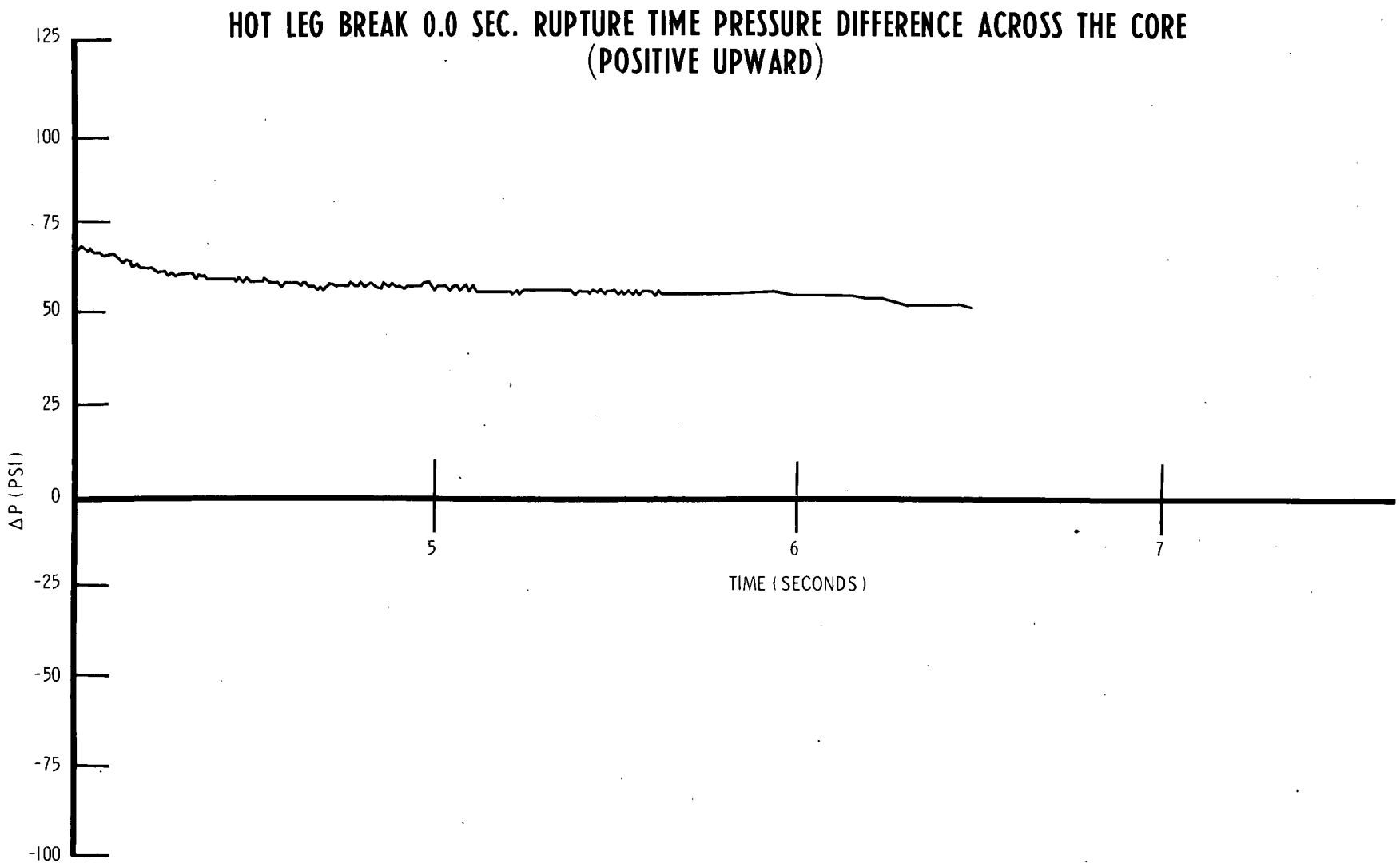

HOT LEG BREAK 0.0 SEC. RUPTURE TIME PRESSURE DIFFERENCE ACROSS THE CORE
(POSITIVE UPWARD)
125
100
75
50
25
0
-25
-50
-75
-100
ΔP (PSI)
5
6
7
TIME (SECONDS)

FIG. 12-8

HOT LEG BREAK 0.0 SEC. RUPTURE TIME-PRESSURE DIFFERENCE ACROSS UPPER BARREL (POSITIVE INWARD)

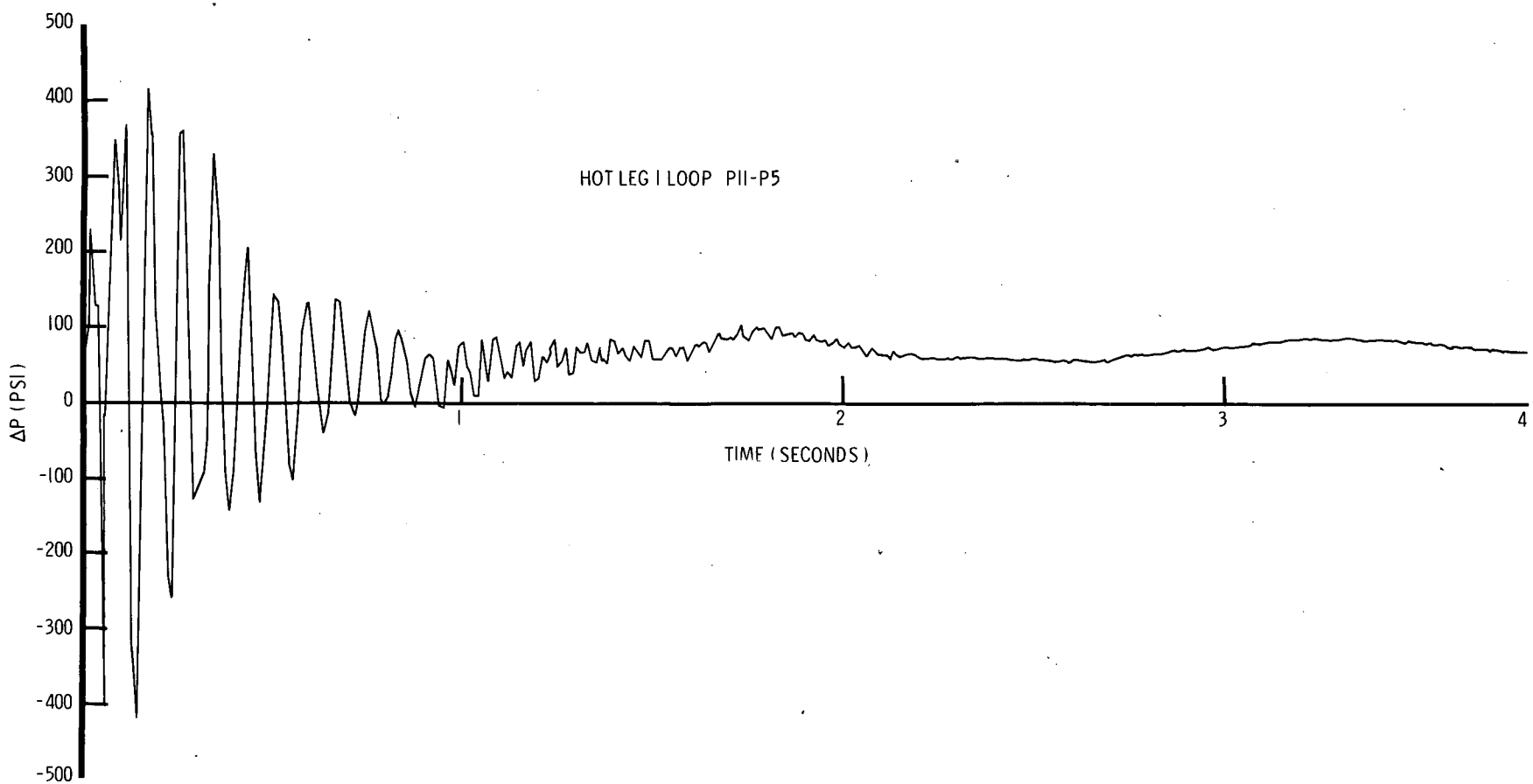


FIG. 12-9

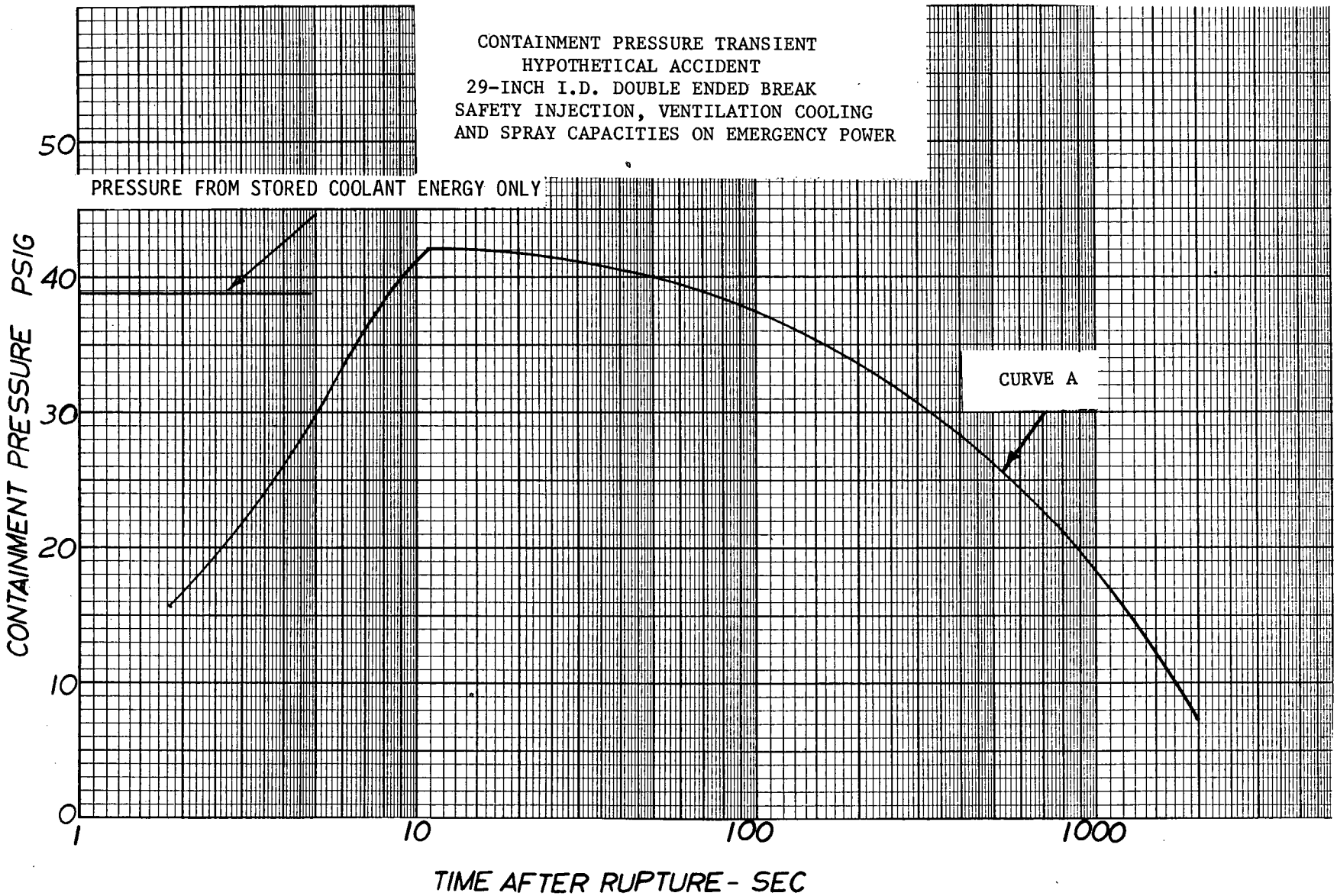

CONTAINMENT PRESSURE TRANSIENT
HYPOTHETICAL ACCIDENT
29-INCH I.D. DOUBLE ENDED BREAK
SAFETY INJECTION, VENTILATION COOLING
AND SPRAY CAPACITIES ON EMERGENCY POWER
PRESSURE FROM STORED COOLANT ENERGY ONLY
CURVE A
CONTAINMENT PRESSURE PSIG
50
40
30
20
10
0
1
10
100
1000
TIME AFTER RUPTURE - SEC

FIG. 12-10

CONTAINMENT PRESSURE TRANSIENT WITH
SAFETY INJECTION, VENTILATION COOLING
AND SPRAY CAPACITIES ON EMERGENCY POWER

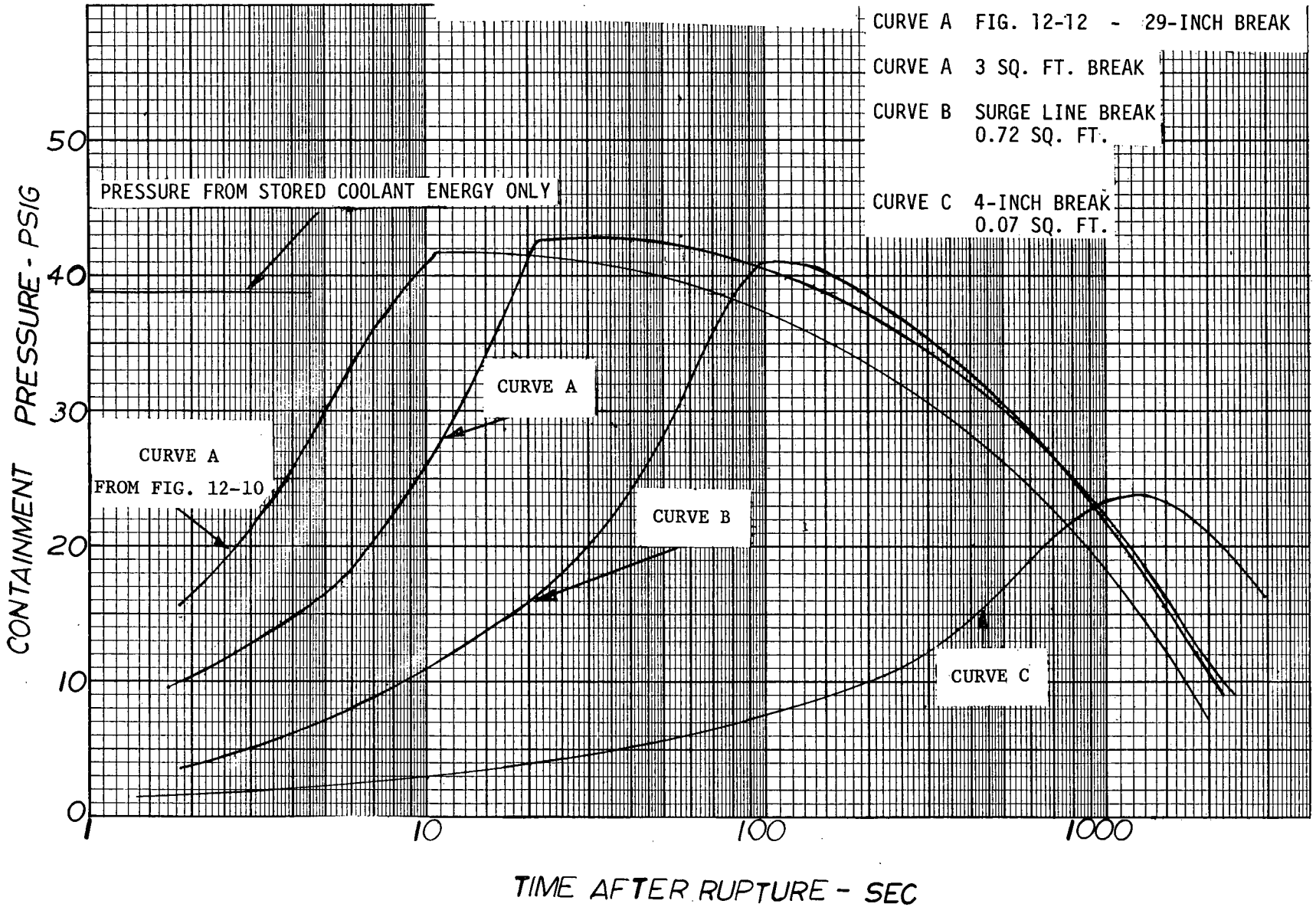


FIG. 12-11

SECTION APPENDIX A

PSAR Section	PSAR Page	Remarks
		Appendix A on "Seismic Design Criteria" has been largely superseded by Item 15 of Supplement 1 to the PSAR. The exceptions are the acceleration response spectra given in this appendix which are not superseded. Further information on seismic design is given in Supplements 2 and 4, Item 13 for the seismic limit curves.