BWR OWNERS' GROUP

T. J. Dente, Chairman

P.O. Box 270 • Hartford, Connecticut 06101 • (203) 666-6911 X 5489

BWROG-8252

August 12, 1982

U. S. Nuclear Regulatory Commission
Division of Licensing
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Darrell G. Eisenhut, Director

Gentlemen:

SUBJECT: NEDO-22155, "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containments"

References: 1) "Interim Requirements Related to Hydrogen Control," Final Rule, published in the Federal Register, Vol. 46, No. 231, on December 2, 1981, pp 58484- 58486

2) Letter, BWROG-8224 from T. J. Dente (BWR Owners' Group) to D. G. Eisenhut (NRC), June 21, 1982, same subject

Enclosed for your use and internal distribution are sixty copies of NEDO-22155, "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containments." The subject report was prepared by the BWR Owners' Group in response to Reference 1, and is identical in content to the prepublication version previously transmitted to you under Reference 2.

This report has been endorsed by a substantial number of the members of the BWR Owners' Group; however, it should not be interpreted as a commitment of any individual member to a specific course of action. Each member must formally respond to the rule (Reference 1) and endorse the BWR Owners' Group position in order for that position to become the member's position.

G002
1/60 on shelf

Should you have any questions on the enclosed material, please contact David R. Helwig (Philadelphia Electric Company), (215) 841-4542, or James F. Schilder (General Electric), (408) 925-5772.

Very truly yours,

Thomas J Dente

T. J. Dente, Chairman
BWR Owners' Group

TJD:JFS:na

Enclosures

cc: BWR Owners' Group
J. F. Schilder (GE)
S. J. Stark (GE)
C. J. Anderson (NRC)
W. R. Butler (NRC)
R. J. Mattson (NRC
T. H. Novak (NRC)
J. J. Shea (NRC)

NEDO-22155
82NED069
CLASS I
JUNE 1982

GENERATION AND MITIGATION OF COMBUSTIBLE GAS MIXTURES IN INERTED BWR MARK I CONTAINMENTS

F. R. HAYES
L. B. NESBITT
P. P. STANCAVAGE

Dupe
8206230103

GENERAL ELECTRIC

NEDO-22155
82NED069
Class I
June 1982

GENERATION AND MITIGATION OF COMBUSTIBLE
GAS MIXTURES IN INERTED BWR MARK I CONTAINMENTS

F.R. Hayes
L.B. Nesbitt
P.P. Stancavage

APPROVED BY: James F. Schilder
J. F. Schilder, Manager
BWR Generic Programs

APPROVED BY: R J Brandon 6/18/82
R. J. Brandon, Manager
Nuclear Services
Engineering Operation

APPROVED BY: for J F Quirk
J. F. Quirk, Manager
BWR Systems Licensing

APPROVED BY: L. F. Fidrych
L. F. Fidrych, Manager
Plant Design & Analysis

This document contains 46 pages.

DISCLAIMER OF RESPONSIBILITY

This document was prepared by or for the General Electric Company. Neither the General Electric Company nor any of the contributors to this document:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information disclosed in this document may not infringe privately owned rights; or

B. Assumes any responsibility for liability or damage of any kind which may result from the use of any information disclosed in this document.

TABLE OF CONTENTS

TABLES

FIGURES

SUMMARY

Analyses of oxygen generation in inerted boiling water reactor (BWR) Mark I containments for a range of transient and accident events have been performed to evaluate existing combustible gas control capability. The purpose of this study is to determine if the existing inerted containment design adequately controls combustible gas concentrations to below combustible gas limits without requiring the use of hydrogen recombiners or containment purging.

The containment oxygen concentration as a function of time was calculated for a limiting plant. Conservative values of the key input parameters were selected based on available test data and analysis using codes developed by U.S. National Laboratories. In addition, parametric analyses were performed to determine the sensitivity of the results to variations in these input parameters.

Results of tests on operating BWR plants support an oxygen generation rate of 0.1 molecules per 100 eV for boiling water. Test data and extensive analysis indicate that under non-boiling conditions of high radiation and excess hydrogen, radiolysis is a self-limiting reaction whereby negligible net oxygen is produced. Therefore, oxygen generation rates of 0.1 and 0.0 molecules per 100ev were chosen for boiling and non-boiling water, respectively.

The results of the analysis show that, for all BWR plants with inerted Mark I containments, peak containment oxygen concentrations are maintained below the combustible gas limits at all times without requiring containment venting or hydrogen recombiners.

1. INTRODUCTION

1.1 BACKGROUND

The BWR containment for Mark I plants is provided with an inerted atmosphere to preclude the possibility of a hydrogen combustion event within the containment. The oxygen deficient atmosphere assures that hydrogen build-up due to metal-water reaction is not a concern for these plants. Combustible gas control for these plants is based on control of oxygen, which is produced in more limited quantities than hydrogen following a LOCA or transient event.

Following a postulated LOCA, both oxygen and hydrogen may be produced by the radiolytic decomposition of primary coolant and suppression pool water. Decomposition would occur due to the absorption of gamma and beta energy released by fission products into reactor coolant and suppression pool water. Radiolysis is the only significant reaction mechanism whereby oxygen, the limiting combustion reactant, is produced within the containment. Therefore, radiolysis is the primary foc s relative to combustible gas control for contairments with inerted atmospheres.

The purpose of this report is to determine if the peak oxygen concentration in the inerted BWR Mark I containment atmosphere will be below the maximum allowable concentration specified in Regulatory Guide 1.7 using conservative assumptions. The Regulatory Guide 1.7 oxygen concentration limit is below that which could support hydrogen combustion or detonation which might threaten containment structural integrity or the survivability of essential equipment.

1.2 EXISTING COMBUSTIBLE GAS CONTROL FEATURES

The current BWR Mark I plants contain several design features for prevention of a hydrogen combustion event.

The generation of significant quantities of hydrogen due to a metal-water reaction from high fuel cladding temperatures is prevented by assurance of adequate core cooling. During normal operation, there are several systems, including feedwater and control rod drive (CRD), which inject water directly to the reactor pressure vessel. A reliable, automatic means of cooling the core is provided by the emergency core cooling system (ECCS). This system is designed to provide sufficient core cooling in accordance with 10CFR50.46 limits assuming any single failure in addition to loss of offsite power.

Accumulation of noncondensible gases (including hydrogen and oxygen) in the primary coolant during normal operation is prevented by continuous removal of noncondensible gases from the main condenser. The continuous removal of noncondensible gases ensures that the initial primary coolant hydrogen and oxygen concentrations will contribute negligibly to the hydrogen and oxygen concentrations in the containment following the LOCA.

Prevention of hydrogen combustion is assured by maintaining an inert nitrogen atmosphere in the containment. Typical inerted BWR's maintain oxygen concentration below an allowable technical specification limit of 4% by volume, which is well below that which would support hydrogen combustion or detonation under conditions of excess hydrogen. An inert atmosphere is maintained in both the drywell and wetwell. Capability is provided for makeup of nitrogen to the containment during normal operation to maintain the containment oxygen concentration below the technical specification limit. As noted earlier, this inerting feature assures that hydrogen combustion will not occur regardless of the amount of hydrogen produced by the metal-water reaction, since there is no oxygen produced by this reaction.

Although the recently published interim hydrogen control rule[1] is based on the undesirability of containment venting, this technique was considered by the NRC to be an acceptable method of controlling combustible gases for several operating plants based on the November 1978 version of 10CFR50.44. For all of those plants, it was required that such venting not result in offsite doses greater than all or a portion of 10CFR100 limits depending upon the date for which a notice of hearing for a construction permit was issued. If these dose criteria could not be met, then an additional hydrogen control system, such as hydrogen recombiners, would be required.

In response to this November 1978 version of 10CFR50.44, these operating plants provided a venting capability as a means of hydrogen control. This venting capability meets the dose criteria as provided in that version of 10CFR50.44. Consequently, it is an existing feature for control of combustible gases for these plants. It is expected, however, that due to the other BWR Mark I design features for mitigating combustible gases, this venting would not be needed to prevent oxygen concentrations from exceeding the Regulatory Guide 1.7 limit. Consequently, it is available as a back-up. Section 5.0 provides more details on the radiological impact of controlled venting.

Many BWR plants have a nitrogen containment atmosphere dilution (CAD) system. The purpose of this system is to provide the capability for injection of nitrogen into the containment following a LOCA. The added nitrogen serves as a diluent further lowering the volumetric concentration of hydrogen and oxygen in the containment. The CAD systems generally consist of a liquid nitrogen storage tank, vaporizers, pressure-reducing valves with controllers, and associated instrumentation, valves, and piping.

Current NRC regulations allow repressurization of the containment due to CAD system operation to 50% of design pressure.[2] For a typical BWR Mark I, this would allow the addition of nitrogen to a containment pressure of about 30 psig. As discussed later in this report, this pressure limit is conservative and pressurization to higher containment pressures would be allowable based on the ASME code.

1.3 COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT

1.3.1 Limiting Plant Determination

The calculation of combustible gas concentrations in the containment following a LOCA have been performed for a limiting plant. Consequently, the conclusions of this report will be applicable to all of the BWR Mark I plants for the utilities participating in this study.

For the determination of radiolytic oxygen concentration in the containment, the limiting plant is that plant with the highest ratio of core thermal power to containment free volume. Table 1-1 shows that there are five plants which have identical power-to-volume ratios, which are also the highest ratios. Those plants are Browns Ferry 1, 2 and 3 and Peach Bottom 2 and 3. Since the calculation for this study is intended to be applicable to all Mark I plants, the limiting parameters used for this calculation were those pertaining to Browns Ferry and Peach Bottom.

1.3.2 Limiting Event Determination

In determining the limiting event, one must consider those factors which most significantly impact radiolytic oxygen generation in the reactor water and suppression pool. For the BWR, the limiting factors would be the time duration and the extent of boiling for the reactor water in the core. As will be discussed in the following section, the oxygen generation rate in subcooled water is negligible. Consequently, the limiting event was selected in order to maximize the time duration and extent of boiling in the core water in order to maximize the total release of oxygen due to radiolysis.

A number of event scenarios were analyzed ranging from the loss of coolant design basis accident (i.e., a double-ended break of the recirculation line with the worst single active failure from the standpoint of oxygen generation) to normal shutdown from full power. These scenarios were based on those considered in the Peach Bottom FSAR analyses as well as the degraded events considered for generic BWR/4 plants in NEDO-24708A.[3]

For those scenarios which rely on the operator to depressurize the vessel, depressurization times were chosen to be conservative relative to operating plant experience and the instructions in the Emergency Procedure Guidelines.[4] It was determined that the limiting events are isolation transients with only low pressure emergency core cooling systems available for core cooling. For these isolation events, a time period of up to 12 hours may be required to achieve a subcooled condition in the reactor coolant.

The 12 hour saturation period is based on the following event sequence. At 7 hours following isolation of the vessel, the operator starts to depressurize the vessel. The 7 hour time period is conservative, because the operator would probably initiate depressurization before this time. At 7 hours following isolation, the suppression pool would reach its peak temperature assuming blowdown through the safety/relief valves to maintain system pressure and both RHR supression pool cooling loops running. Depressurization of the vessel to the pressure at which RHR shutdown cooling can be initiated is conservatively assumed to take 3 hours. This is a slower depressurization rate than the 100°F per hour rate recommended in the Emergency Procedure Guidelines. An additional 2 hours is assumed for preparing the RHR system for shutdown cooling. Following initiation in the shutdown cooling mode, the reactor core water will rapidly be subcooled (within a few minutes). Summing the time duration of each of these events in the sequence yields a total time of 12 hours to reach subcooled conditions following isolation of the vessel. It is also conservatively assumed that all of the water in the reactor vessel is boiling for this 12 hour period.

It should be noted that these isolation transients were used only to obtain a bounding condition for boiling in the reactor vessel. This bounding condition was conservatively applied to the design basis LOCA event. The LOCA event is the primary focus for these calculations because, although such an event is not expected to produce a signifiant metal-water reaction due to the high reliability of existing core cooling systems, it is the only design basis event which could potentially lead to a limited hydrogen generation for which oxygen control may be desirable.

For these analyses, the assumption of the release of 50% halogens and 1% solids from the core as recommended in Regulatory Guide 1.7 is applied even though a design basis loss of coolant accident will not produce significant fission product releases.

1.3.3 General Approach

The adequacy of the BWR Mark I design for hydrogen control is shown through the use of a conservative analysis of radiolytic oxygen generation and release to containment. Input assumptions were selected based on a conservative interpretation of applicable experimental data and Regulatory Guide 1.7 assumptions where no data were available.

The following section, 1.3.4, describes this analysis of oxygen concentration in the containment. This analysis, which uses applicable test data and conservative assumptions for the determination of oxygen generation rates, is the basis for the determination of the adequacy of the Mark I containment design for control of combustible gases.

Section 2.0 provides a discussion of the key input parameters for determining the oxygen concentration in containment due to radiolysis. The basis for the values selected for these parameters for this analysis is provided. In addition, the basis for the values of these parameters considered in the sensitivity analyses is provided.

Section 3.0 provides a general description of the analytical model which was used to determine oxygen concentrations in containment due to radiolysis. A curve of containment oxygen concentration as a function of time is provided for this analysis. In addition, the results of the sensitivity studies are presented whereby the containment oxygen concentration is determined as a function of the key input parameters. The values of the parameters range from those in Regulatory Guide 1.7 to those based on available experimental data and operating plant experience as discussed in Section 2.0.

Section 4.0 identifies the major conservatisms in the analysis.

Section 5.0 addresses the radiological impact of controlled venting as a backup measure.

1.3.4 Description of Analysis

The analysis presented in this report is the basis for the determination of the adequacy of the inerted Mark I containment to control combustible gas concentrations below the Regulatory Guide 1.7 limit. The analysis will use an oxygen generation rate which is based on experimental data and analysis as described in Section 2. That oxygen generation rate is 0.1 molecules per 100 eV for boiling water. For non-boiling water, the radiolysis reaction will reach equilibrium resulting in a negligible generation of oxygen. For non-boiling water, the assumed oxygen generation rate is zero. As noted in Section 1.3.2, the core water is conservatively assumed to boil for 12 hours following the event, whereas the suppression pool is assumed not to boil.

For this analysis no repressurization of the primary containment with nitrogen is assumed even though many plants have a nitrogen CAD system. This is a conservative assumption, because such repressurization would significantly dilute both the hydrogen and oxygen concentrations in the containment. The containment leak rate is assumed to be zero. This assumption is also conservative, because it ignores a depletion mechanism for containment oxygen which will exist.

TABLE 1-1. CORE POWER TO DRYWELL VOLUME RATIOS

PLANT	NITROGEN CAD SYSTEM	102% POWER LEVEL MW_t	DRYWELL VOLUME FT^3	$\frac{MW_t}{\text{DRY. VOL.}}$
Browns Ferry 1, 2, 3	Yes	3359	159000	.0301
Brunswick 1, 2	Yes	2485	166000	.0150
Cooper	No	2429	132000	.0184
Dresden 2,3	No	2578	158000	.0163
Duane Arnold	Yes	1625	144000	.0113
Fitzpatrick	Yes	2485	150000	.0165
Hatch-1	Yes	2485	146000	.0170
Millstone 1	No	2051	147000	.0140
Nine Mile Point 1	Yes	1887	180000	.0105
Oyster Creek	Yes	1971	180000	.0109
Peach Bottom 2-3	Yes	3359	159000	.0301
Pilgrim	No	2038	147000	.0138
Quad Cities 1,2	No	2561	158000	.0162
Vermont Yankee	No	1625	134000	.0121

2. SIGNIFICANT INPUT PARAMETERS

2.1 IDENTIFICATION OF SIGNIFICANT INPUT PARAMETERS

The "significant input parameters" considered in this section are those which: (1) have a significant impact on the calculated radiolytic oxygen generation rate and (2) are parameters for which the values presented in Regulatory Guide 1.7 or existing NRC regulations are felt to be overly conservative based on test data and analysis. The "significant input parameters" are the oxygen generation rate, the containment leak rate, and the maximum allowable containment pressure for repressurization with nitrogen.

In performing the parametric analyses, these parameters were varied over a range which includes Regulatory Guide 1.7 assumptions as well as values based on experimental data and analysis.

2.2 RADIOLYTIC OXYGEN GENERATION RATE

2.2.1 Evaluation of Available Test Data and Analyses

The most significant parameter in determining the radiolytic oxygen concentration in the containment is the oxygen generation rate, $G(O_2)$. $G(O_2)$ is the net number of molecules of oxygen produced from radiolysis per 100 eV of energy absorbed. That energy would be produced from fission product decay primarily in the form of gamma and beta radiation.

The radiolytic generation of oxygen has been the subject of extensive investigation through test and analysis. There is general agreement that the production rate of radiolytic gases is a function of the water temperature, presence of impurities, physical state of the water (i.e., boiling vs. subcooled), concentration of radiolytic gases, and the amount and kind of external radiation being applied. Data and analyses exist for both a boiling and a non-boiling pool of water.

There is a considerable amount of data for the determination of $G(O_2)$ in the boiling water for BWRs. Extensive measurements of the hydrogen gas evolution rate from the offgas systems during normal (boiling) operation for operating BWRs, including Mark I plants such as Dresden 2 and Dresden 3 have been made. Those measurements were used to derive values of $G(H_2)$ for those plants which were tabulated in NEDC-23856-1.[5] The range of values correspond to $G(O_2)$ values of 0.043 and 0.094. However, the $G(O_2)$ values following shutdown will be considerably lower than those during normal operation (about a factor of 5) due to: (a) the lack of any appreciable energy deposition due to neutrons, and (b) the higher absorption fraction relative to that assumed in NEDC-23856-1. Consequently, the expected range of $G(O_2)$ during shutdown for operating BWRs would be approximately 0.01-0.02 based on this measured data.

In addition to these data, further supporting data were obtained from tests conducted by General Electric during refueling shutdowns at two operating reactors.[6] Calculations based on test data obtained during the first 2-1/2 hours, i.e., during the boiling regime, of a scheduled shutdown of Humboldt Bay Nuclear Plant indicated an oxygen yield of between 0.015 and 0.10 molecules per 100 eV. Similar data from tests conducted at KRB Nuclear Power Plant resulted in a calculated value of $G(O_2)$ less than 0.1. Improved methods in experimental procedure and measurement techniques were used in the KRB tests, and the KRB test conditions are thought to be more representative of the actual conditions in a large BWR.

Based on these test results, the oxygen yield rate of 0.25 molecules per 100 eV recommended in Regulatory Guide 1.7 is an overly conservative value for BWR boiling conditions. A more appropriate but still very conservative value for the BWR would be 0.10 based on this extensive operating plant data. Consequently, a $G(O_2)$ of 0.10 was considered in the analyses for that period of time when water in the reactor vessel would be expected to boil.

Under non-boiling conditions, the rate of oxygen generation due to radiolysis is significantly reduced relative to boiling conditions. This result is expected, since, under boiling conditions, radiolytic gases (hydrogen and oxygen) are constantly being swept away from the reaction zone allowing little or no time for recombination to occur.

The radiolytic oxygen generation for a non-boiling pool has been investigated through test and analysis.

Following the Three Mile Island accident in March 1979, Knolls Atomic Power Laboratory (KAPL) performed an analysis of the potential for radiolysis in the containment and concluded that the net radiolysis should be zero.[7] Their analysis indicated that the potential for H_2/O_2 recombination in the reactor coolant water under the influence of the gamma ray field was sufficiently high to recombine essentially all of the O_2 formed by radiolysis of coolant water, as long as the water in the core is not boiling.

In addition, KAPL estimated that the gamma radiation in the TMI containment caused the recombination of the containment atmospheric hydrogen at the rate of 0.1% (by volume) per day.[8]

Results of analyses conducted by Argonne National Laboratory (ANL)[9] of the hydrogen bubble at Three Mile Island Unit 2 following the March 1979 accident were consistent with the KAPL conclusions. Those results showed that for non-boiling water conditions with 1 atmosphere of hydrogen present, no further production of oxygen could have occurred. On the contrary, oxygen was actually consumed, the rate being determined by the rate of dissolution of oxygen from the bubble. These results led the ANL investigators to conclude that long term bubble growth, particularly by oxygen production, was not possible.

In addition to these analyses and observations from TMI, results of calculations performed by Northeast Utilities for pure water at representative BWR conditions show that the radiolytic oxygen concentration will rapidly reach an equilibrium value. Those calculations show that the radiolytic oxygen generation for a non-boiling pool is negligible when the hydrogen concentration approaches a value associated with the lower limit of flammability. These results will be submitted by Northeast Utilities to the NRC.

Based on these analyses and observations, the oxygen generation rate for the base case was assumed to be 0.1 molecules per 100 eV for boiling water and zero for non-boiling water.

For the sensitivity studies, a value of $G(O_2)$ of 0.003 molecules per 100 eV was also considered for non-boiling water in order to assess the significance of a small but non-zero non-boiling source term.

As noted in Section 1.3.2, boiling in the core is conservatively assumed to be terminated at 12 hours following the event. Consequently, a $G(O_2)$ of 0.1 molecules per 100 eV was used from 0 to 12 hrs and a value of 0.0 was used for periods beyond 12 hours for core radiolysis.

Since no boiling in the suppression pool will occur, a $G(O_2)$ of 0.0 is used throughout for suppression pool radiolysis.

2.2.2 Consideration of Other Oxygen Sources

In addition to the oxygen generated by radiolysis of water, all other oxygen sources must be considered in determining peak oxygen concentrations in the containment. Other oxygen sources could include hydriding, air inleakage to the containment from the reactor building, dissolved oxygen in the primary coolant, and air inleakage from any air supply lines within the containment. The oxygen generated from the first three sources will be negligible. Relative to the fourth source, each participating utility should show that air inleakage from such lines, if they exist, would not cause the total oxygen concentration to exceed Regulatory Guide 1.7 limits.

2.3 CONTAINMENT LEAK RATE

There is no assumed containment leakage for this analysis. This is conservative since it ignores a depletion mechanism for oxygen which will exist.

Although containment vessels are designed to provide minimum leakage to prevent radioactive fission product release in the event of an accident, it is recognized that a small amount of leakage is unavoidable particularly if the containment is pressurized by energy addition from the accident, accumulation of non-condensible gases or the use of a nitrogen CAD system. The radiological consequences of postulated accidents are evaluated assuming volumetric leak rates of approximately 1% per day to show conformance to the limits of 10CFR100. Typical plant technical specifications permit containment leakages of up to 0.5% per day by volume.

Utilities conduct leak rate tests to confirm that the actual measured leakage is less than that assumed for the radiological evaluations in accordance with 10CFR50, Appendix J. The table below shows the results of eleven of these tests for BWRs.[10]

Plant	Test Date	Leak Rate Measured [Weight % Per Day] at Room Temperature
Pilgrim 1	4/21/72	0.316 at 45 psig 0.31 at 23 psig
Peach Bottom 2	5/5/73	0.127 at 49.1 psig 0.26 at 25 psig
Duane Arnold	12/29/73	0.153 at 27 psig 0.129 at 54 psig
Peach Bottom 3	2/18/74	0.116 at 49.1 psig 0.152 at 25 psig
Hatch 1	5/6/74	0.0370 at 59 psig 0.0474 at 29.5 psig
Hatch 2	5/19/78	0.117 at 57.5 psig

Based on the results of these tests, a conservative value of containment leakage of 0.1% volume per day was selected for use in the sensitivity studies. It should be noted that containment leakage measured as volume percent is approximately equal to containment leakage measured as weight percent.

2.4 CONTAINMENT REPRESSURIZATION LIMIT

As noted in Section 1.2, the nitrogen CAD system can be used to dilute the containment atmosphere following a LOCA in order to preclude a hydrogen combustion event. Because such dilution will pressurize the containment, it is essential that the addition of nitrogen be terminated at a pressure which will not jeopardize containment integrity.

NRC regulations now in effect (10CFR50.44) specify that containment repressurization be limited to 50 percent of the containment design pressure. However, there is no apparent reason from a containment structural integrity point of view as to why repressurization should be limited to 50% of containment design pressure. The containment structure is a large, unfired pressure vessel. It is designed to withstand pressures which are at the design pressure for an indefinite period of time, or even cycled a few times between low and high (design) pressure. In this regard, it should be noted that the draft interim hydrogen rule for Mark III plants allows for pressurization of that containment under non-accident conditions to ASME Service Level A limits, which are significantly above this 50% design pressure limit.[11] In addition, due to the large inherent containment design margin, the containment could withstand a pressure of 2 to 2.5 times design pressure and still retain its integrity. This conclusion was reached in the WASH-1400 Reactor Safety Study[12] and the Limerick Probabilistic Risk Assessment (PRA).[13]

Repressurization was conservatively not included in the present analysis. However, repressurization limits were considered in the sensitivity analyses at 50% containment design pressure and at higher pressures up to the containment design pressure.

This report has been written assuming the containment design pressure is 62 psig. As a result, repressurization to 30 psig would correspond to repressurization to about 50% of containment design pressure. This is not valid for two plants listed in Appendix A which have torii design pressures of 35 psig. The cases evaluated which pressurize the containment to greater than 35 psig are not appliable to these plants.

3. DESCRIPTION AND RESULTS OF ANALYSES

3.1 ANALYTICAL MODEL

The combustible gas concentration in the containment due to radiolysis has been determined by an analytical model which has been developed to calculate oxygen transient concentrations in BWR containments. Pressure-time histories are determined by integration of the equations describing the conservation of mass and energy in the control volumes, a transient heat conduction analysis of various structures, and evaluations of equilibrium thermodynamic states.

Specific features included in the model are:

1. Mass and energy addition to the containment associated with primary system blowdown;

2. Mass and energy addition to the containment associated with hydrogen generation from metal-water reaction and core and suppression pool radiolysis;

3. Vent and vacuum breaker flows;

4. Containment atmospheric dilution (CAD) system flows;

5. Residual heat removal (RHR) system operation;

6. Leakage from the containment to the reactor building.

7. Oxygen dilution due to generation of hydrogen and steam.

The model uses four nodes for its calculations. The wetwell, drywell and reactor building gas nodes are each divided into a four-component system of nitrogen, oxygen, hydrogen and steam. Separate mass balances are maintained for each component in each node. Since thermodynamic equilibrium is assumed at each time step, only one energy equation is needed for each node. Mass and energy balances are also maintained for the suppression pool liquid node.

The basic assumptions of the model are:

1. A homogeneous mixture of nitrogen, oxygen, hydrogen and steam exists in each air node;

2. The gas mixture is in thermodynamic equilibrium;

3. All noncondensible gases are treated as perfect gases;

4. Specific heat and gas constant values for noncondensible gases and steam are assumed constant;

5. Vent clearing is calculated using a hydrostatic (manometer) model.

Figure 3-1 summarizes the elements of the model.

3.2 RESULTS OF ANALYSIS

The results of the analyses are provided in Figure 3-2 for the base analysis parameters defined in Table 3-1. The results are plotted in Figure 3-2. It is seen that the initial oxygen concentration in the drywell slightly increases as oxygen is being generated due to radiolysis in the boiling water in the core. At 12 hours when boiling in the core is conservatively assumed to stop, the oxygen concentration is seen to level off as the radiolysis source rate for oxygen goes to zero. The change in oxygen concentration from the initial value to the peak value is approximately 0.1% volume, and the peak drywell oxygen concentration of 4% is well below the Regulatory Guide 1.7 value of 5%.

The initial drywell oxygen concentration is seen to be slightly below the technical specification concentration of 4% by volume. This is a result of discharge of the drywell oxygen to the suppression chamber and addition of steam to the drywell following a LOCA event.

Since negligible oxygen is being produced by radiolysis in the non-boiling suppression pool, there is no build-up of oxygen in the wetwell, and the oxygen concentration remains at its initial value. Consequently, the results of this analysis show that the maximum oxygen concentration in the containment is well below the allowable limit in Regulatory Guide 1.7.

3.3 SENSITIVITY STUDIES

The significant input parameters which were varied are identified in Section 2.1. The cases which were run are identified in Table 3-1.

The base analysis was described in Section 1.3.4.

Cases 1 through 5 represent evaluations of oxygen concentration vs. time for different oxygen generation source terms, containment leakage and nitrogen pressurization. Case 6 is based on Regulatory Guide 1.7 assumptions. Case 7 examines the impact of longer assumed boiling times for reactor water in the reactor vessel.

3.4 OBSERVATIONS AND CONCLUSIONS FROM SENSITIVITY STUDIES

Figures 3-2 through 3-9 are plots of containment oxygen concentration in volume percent as a function of time for the base analysis and sensitivity Cases 1-7. Both drywell and wetwell concentrations are plotted. The results of the base analysis were previously discussed in Section 3.2.

For those cases involving nitrogen CAD system dilution and containment leakage with an assumed non-boiling source term (Cases 1, 4 and 5), the plots show an initial upward trend which corresponds to the build-up of containment oxygen due to radiolysis. Following this initial build-up, the nitrogen CAD system is turned on at about 6 hours, and the containment oxygen concentration decreases as the atmosphere is diluted with nitrogen. These two trends define the initial peak as shown in the curves.

Addition of nitrogen is assumed to be continuous until a certain containment pressure is reached. At that point the system is shut off and the oxygen concentration again begins to increase. Due to the effects of decaying radioactivity, containment leakage and nitrogen makeup to maintain containment pressure, the oxygen concentration will eventually reach a second peak value. Following this second peak, the oxygen concentration will continually decrease.

For the cases involving no containment pressurization and no containment leakage with an assumed non-boiling source term (Case 2), the oxygen concentration continually increases with time. This occurs because there is no assumed depletion mechanism for the small assumed oxygen source term.

Several important observations can be made from the sensitivity studies plotted in Figures 3-3 through 3-9. Figure 3-3 (Case 1) shows that, assuming a nominal leak rate of 0.1% volume per day and nitrogen pressurization to 30 psig, the oxygen concentration does not increase above the initial level, even using the conservative oxygen source term for non-boiling water. The oxygen concentration continually drops off from the initial value as the leak rate overshadows the source rate due to radiolysis. In fact, Figure 3-7 shows that, for plants with nitrogen CAD systems, the Regulatory Guide 1.7 acceptance criterion is met even using Regulatory Guide 1.7 assumptions on oxygen source terms with the additional considerations of a higher allowable repressurization limit. As noted in that curve, the peak oxygen concentration assuming repressurization to 62 psig (containment design pressure) is 4.9% by volume. Section 2.4 provides a technical justification for repressurization up to the containment design pressure.

Figure 3-4 (Case 2) shows two cases which take no credit for dilution from nitrogen CAD systems. For these cases, no nitrogen repressurization or containment leakage is assumed. A small oxygen radiolytic source term for non-boiling water is arbitrarily assumed for this case as described in Section 2.2.1. Figure 3-4 considers initial containment oxygen concentrations of both 4% and 3% by volume. For the case of 4% initial oxygen concentration, the limit of 5% is reached in greater than 1000 days. Additional calculations have shown that the 5% limit will not be reached until six years following the event. Consequently, even assuming a conservative non-boiling oxygen source term, the Regulatory Guide 1.7 limits are not reached for an extended period of time for plants without nitrogen CAD systems. Such an extended time period would provide the operator with the opportunity to decide on the best means of controlling oxygen concentration.

Figure 3-5 (Case 3) assumes repressurization with nitrogen to 30 psig and the arbitrary assumptions of no containment leakage and a non-boiling oxygen source term. The results show that the Regulatory Guide 1.7 limit is not reached for an extremely long time (several hundred years).

Figures 3-6 through 3-8 primarily examine the effect of arbitrarily assuming the existence of a continuous oxygen source term of the magnitude that would be appropriate for boiling conditions in the core and in the suppression pool (Cases 4-6). Cases 5 and 6 use Regulatory Guide 1.7 assumptions for oxygen generation rates in the core region. Examination of these cases shows that the drywell oxygen concentration is more limiting than the wetwell concentration.

Figure 3-6 (Case 4) applies to plants with a nitrogen CAD system and arbitrarily assumes constantly boiling water in the core. The oxygen generation rate for boiling water is chosen based on test data as described in Section 2.2.1. The results show that repressurization to 30 psig, as allowed by 10CFR50.44, will ensure oxygen concentrations which are well below Regulatory Guide 1.7 allowable limits without the need for venting.

For the cases where Regulatory Guide 1.7 oxygen generation rates are assumed for core radiolysis, it is shown (Figures 3-7 and 3-8) that repressurization to 62 psig will ensure a peak oxygen concentration below 5% by volume.

In addition to the previously described sensitivity studies, an additional case (Case 10) was run in order to examine the impact of longer assumed boiling times for reactor water in the core region. A boiling time of 2.5 days was arbitrarily assumed, and the results are plotted in Figure 3-9. Those results show that even at this extended boiling time period, the peak oxygen concentration in the containment of 4.7% is well below the allowable Regulatory Guide 1.7 limit of 5%.

In summary, these parametric studies show that, even if one arbitrarily assumed an oxygen generation rate for non-boiling water, the inerted Mark I containment is still capable of adequately controlling combustible gases, without requiring venting of the containment.

TABLE 3-1

SENSITIVITY ANALYSIS CASES

Sens. Case	Containment Leakage (% per day)	Radiolytic O_2 Yield (mol/100 eV) Core $t < 12hr$	Core $t > 12hr$	Pool	Pressurization with N_2 CAD (psig)
Base Analysis	0.0	0.10	0.0	0.0	0
1	0.10	0.10	0.003	0.003	30
2	0.0	0.10	0.003	0.003	0
3	0.0	0.10	0.003	0.003	30
4	0.10	0.10	0.10	0.003	30
5	0.10	0.25	0.25	0.003	62
6	0.0	0.25 ($t < 2.5$ days)	0.25 ($t > 2.5$ days)	0.25	30
7	0.0	0.1	0.0	0.0	0

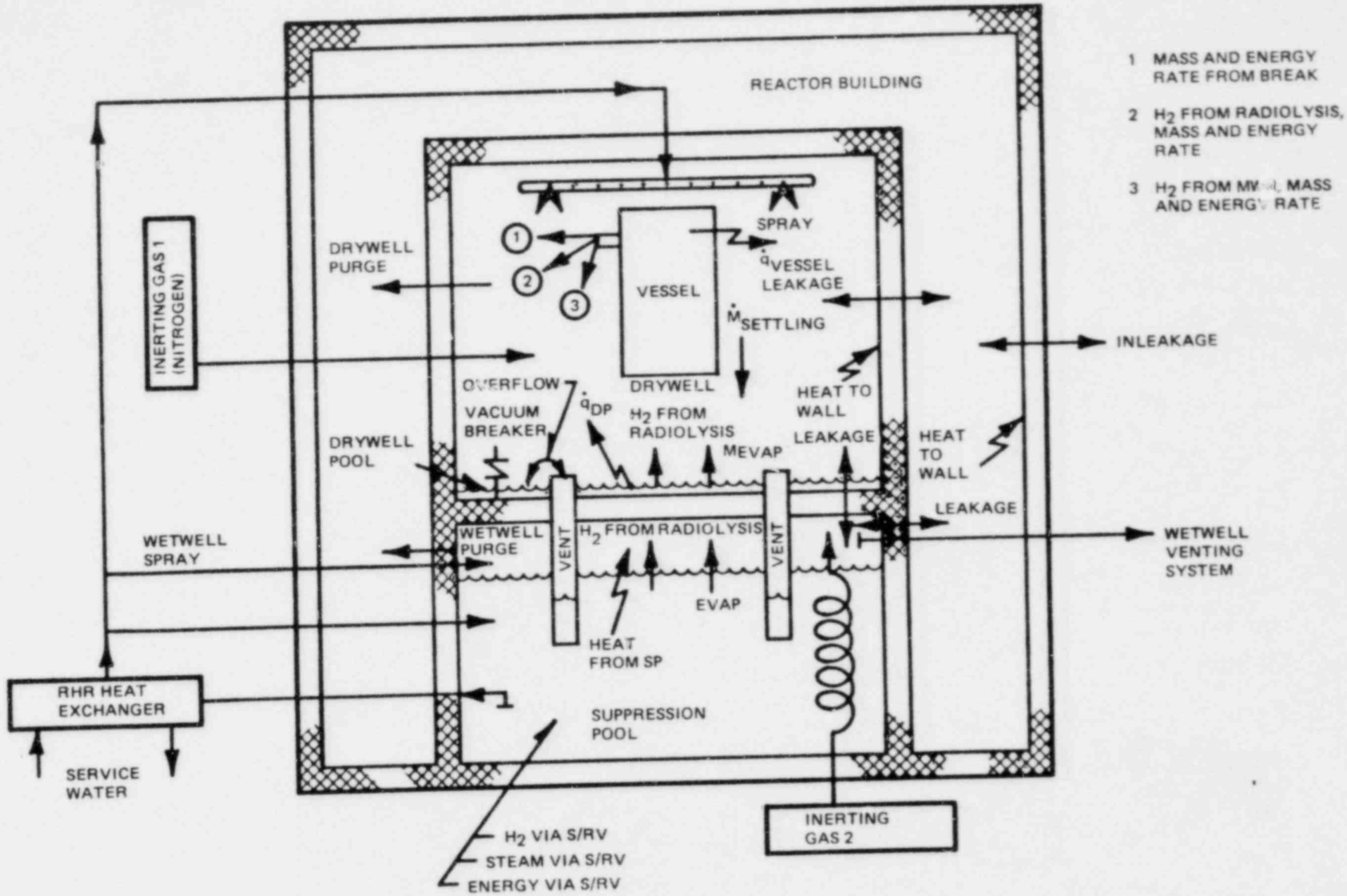

Figure 3-1. Containment Analytical Model

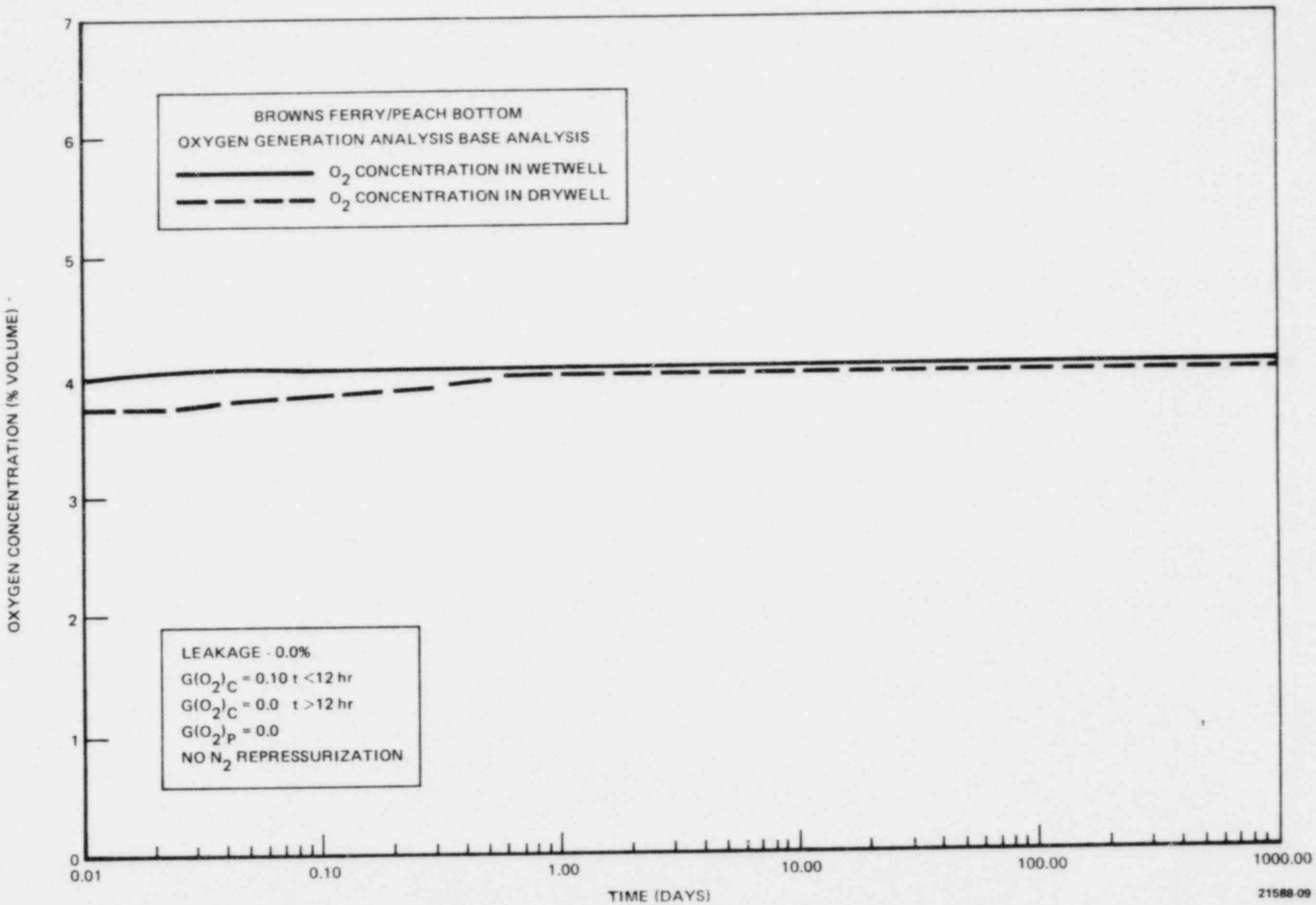

Figure 3-2. Oxygen Concentration Versus Time for Base Analysis

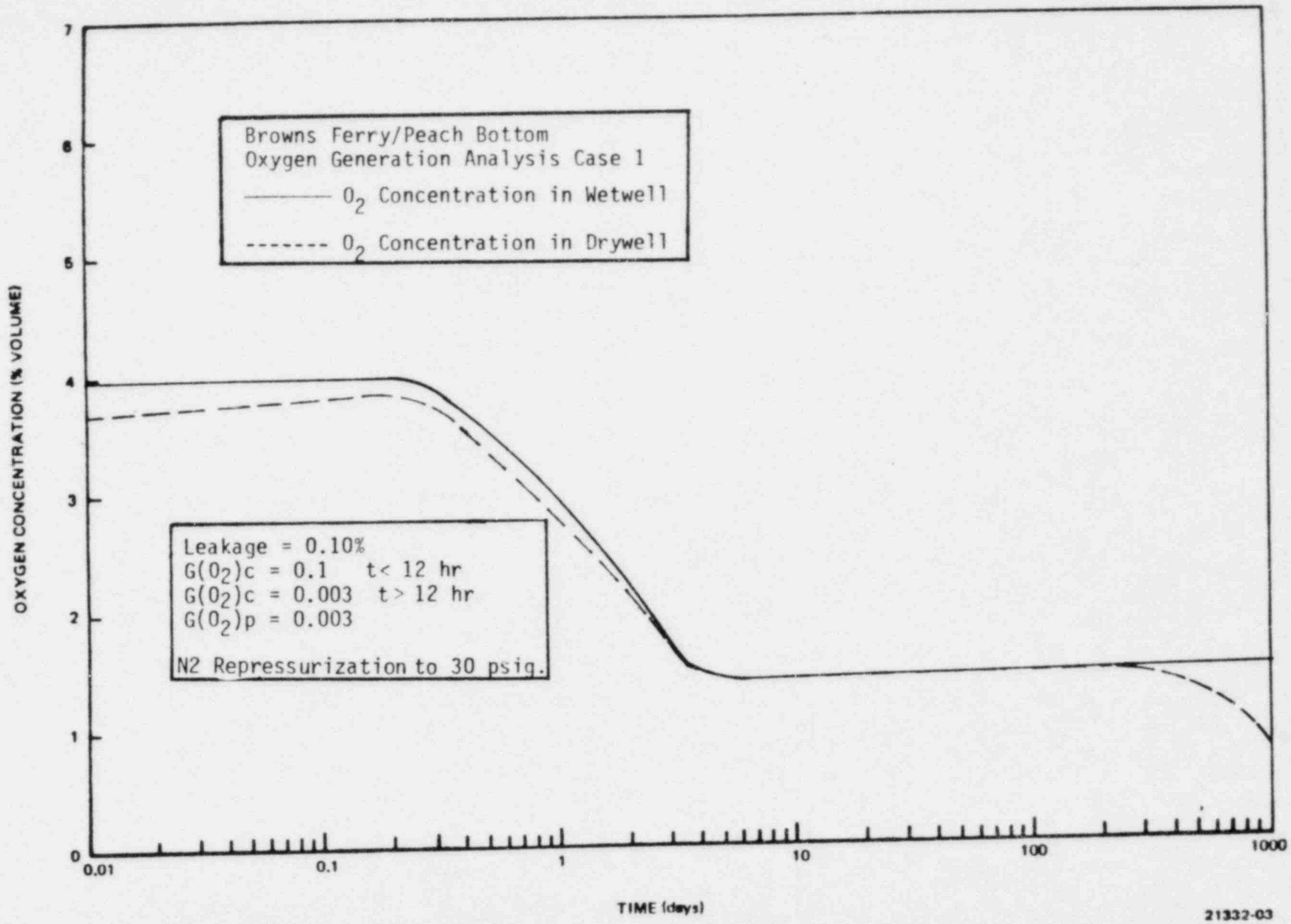

Figure 3 - 3. Oxygen Concentration Versus Time for Case 1

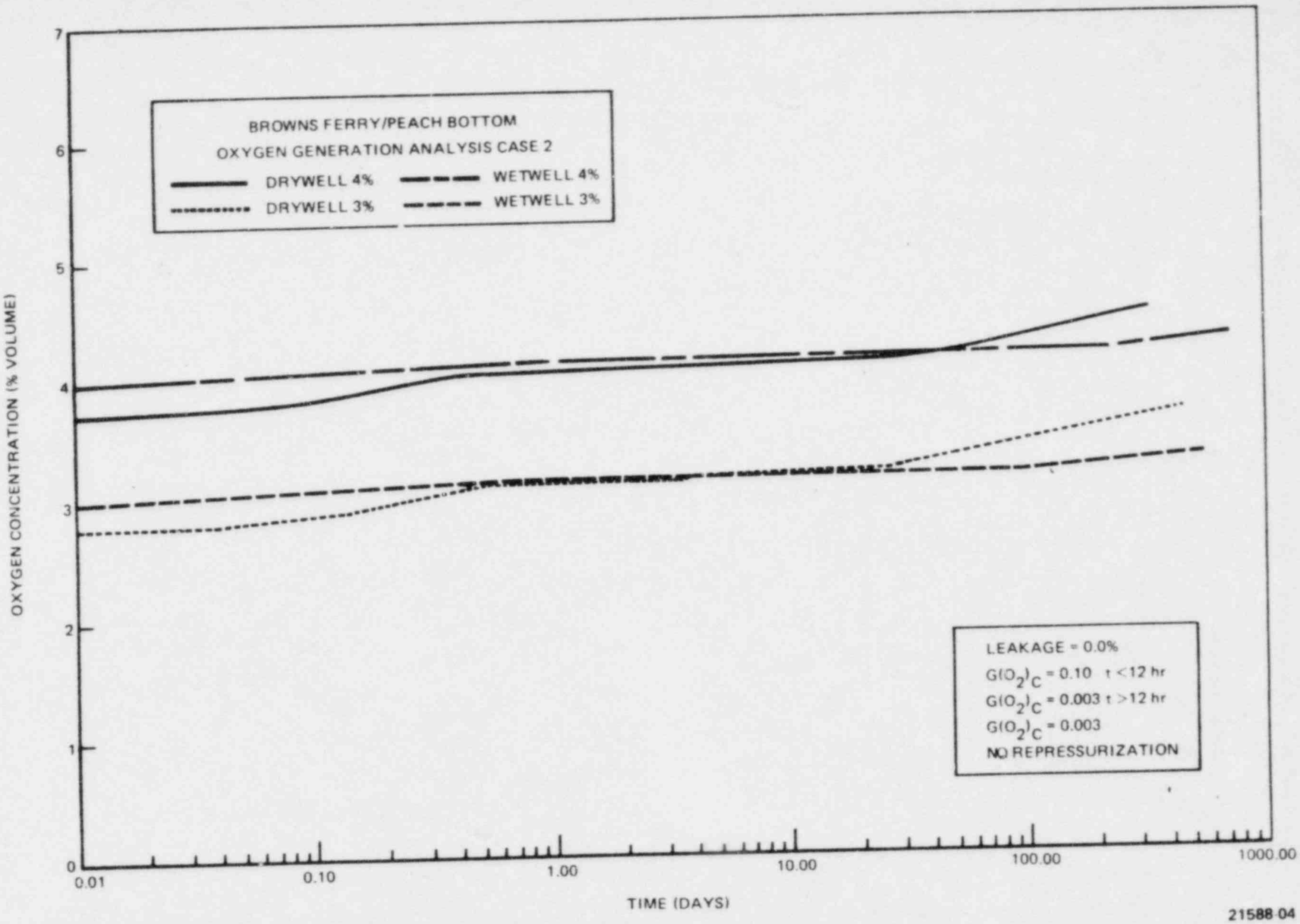

Figure 3-4. Oxygen Concentration Versus Time for Case 2

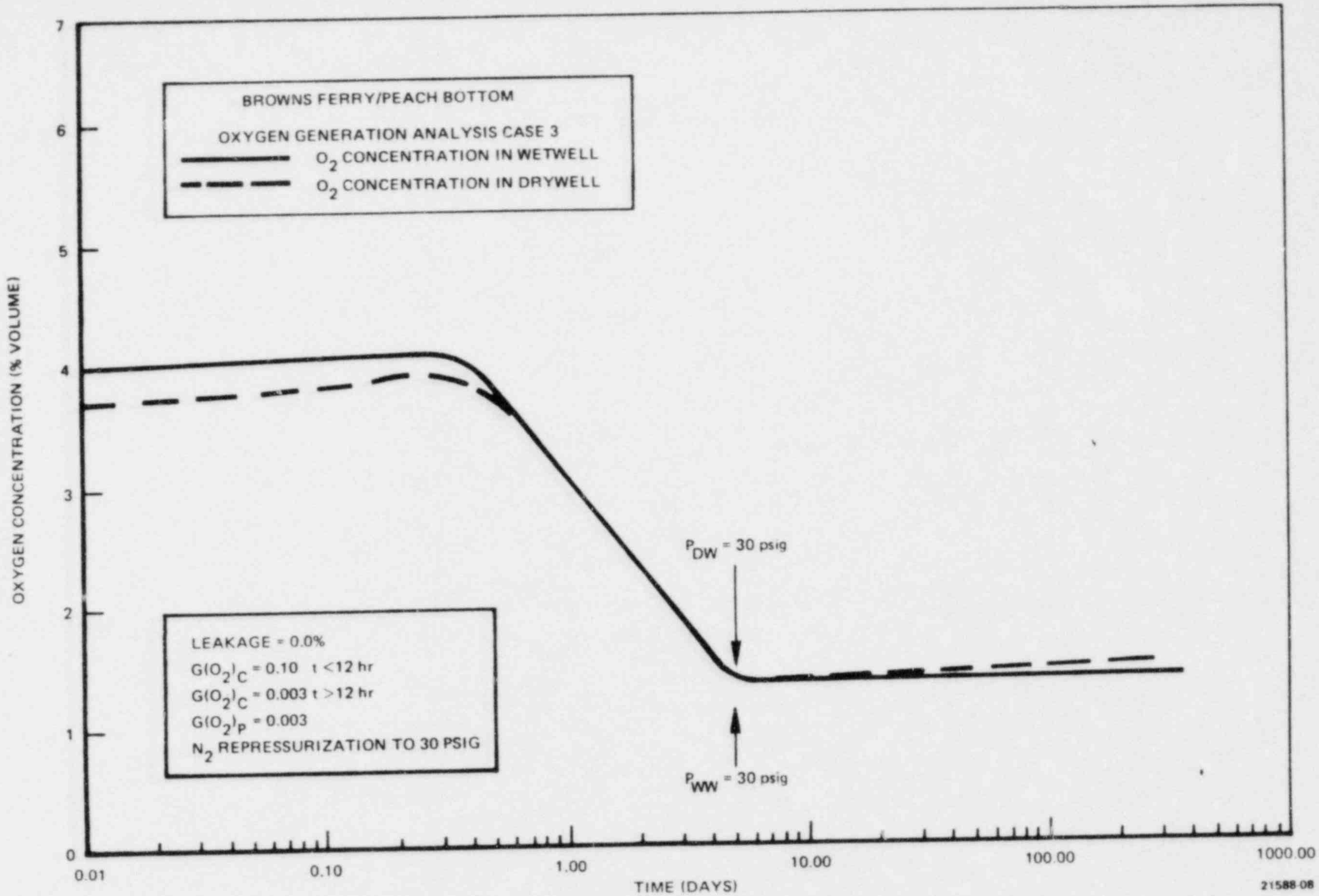

Figure 3-5. Oxygen Concentration Versus Time for Case 3

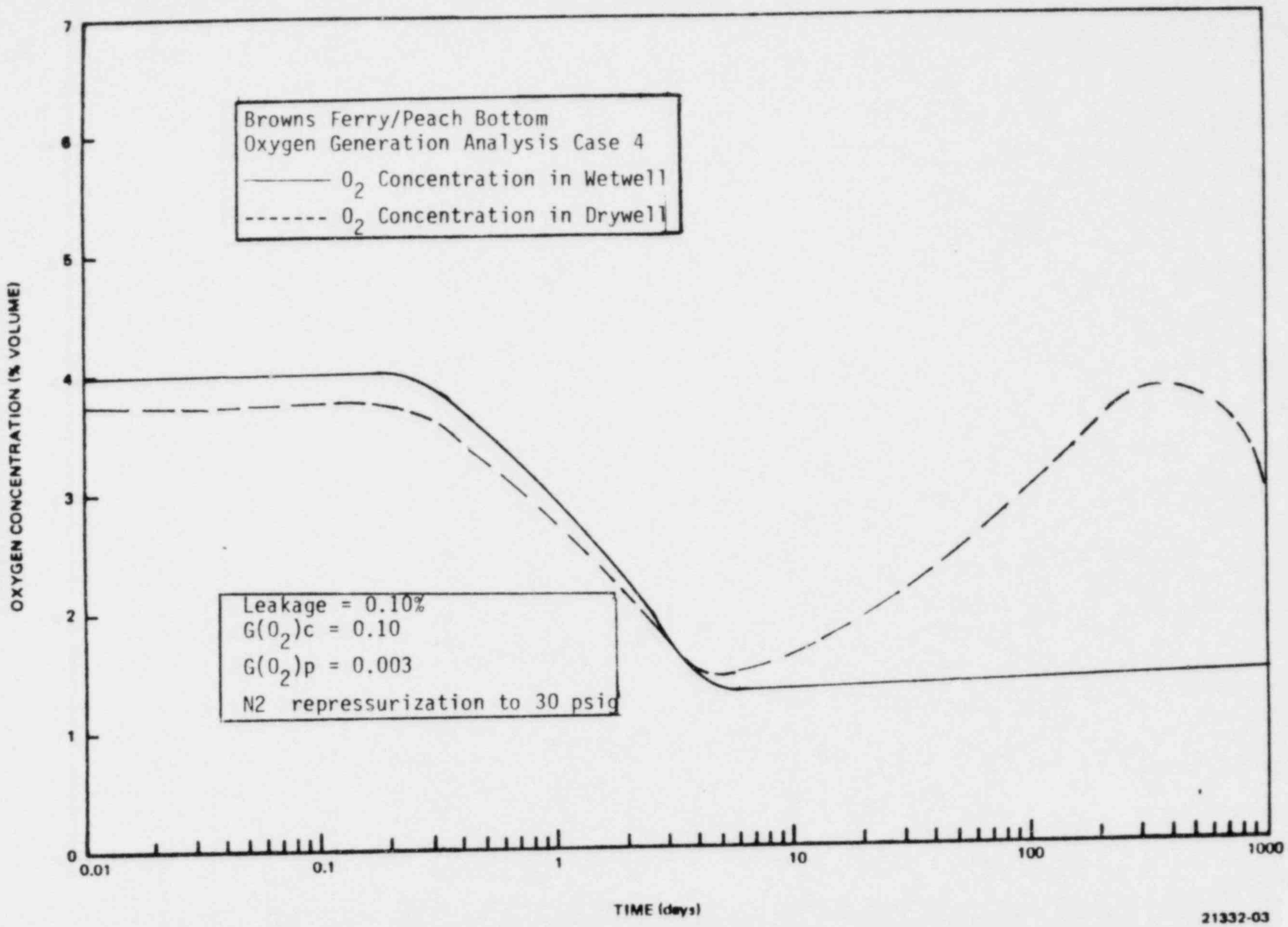

Figure 3-6. Oxygen Concentration versus Time for Case 4.

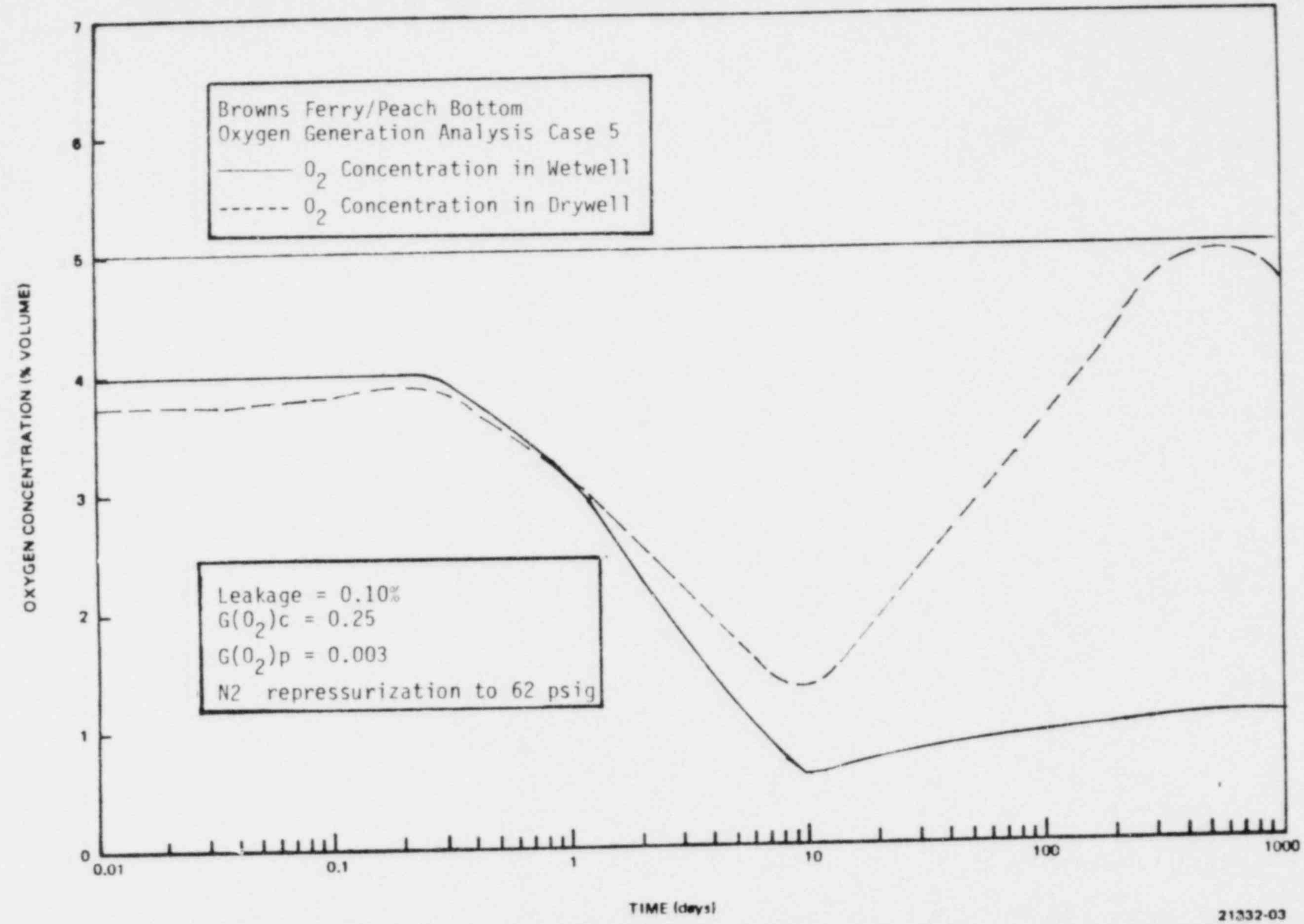

Figure 3-7. Oxygen Concentration versus Time for Case 5

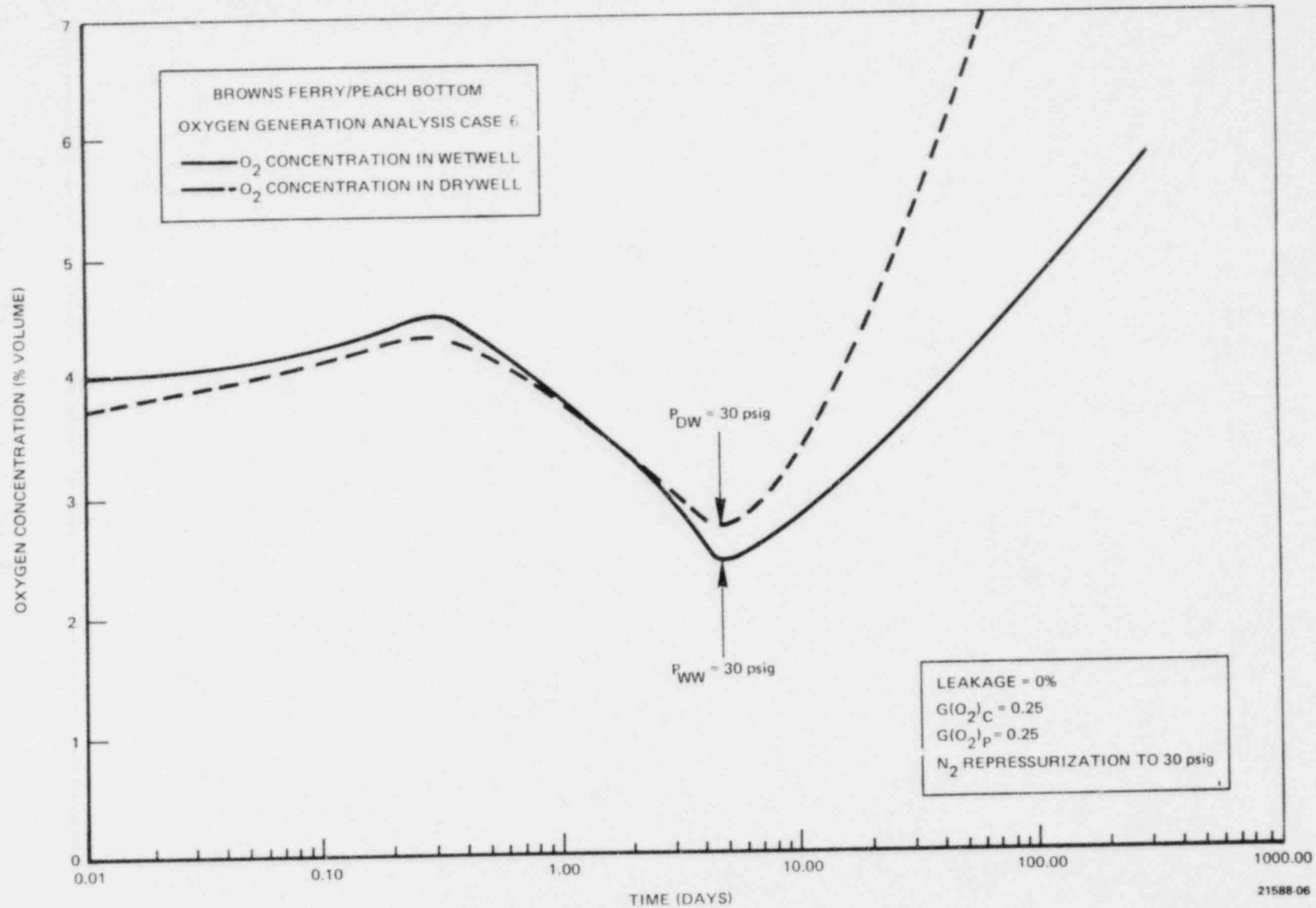

Figure 3-8 . Oxygen Concentration Versus Time for Case 6

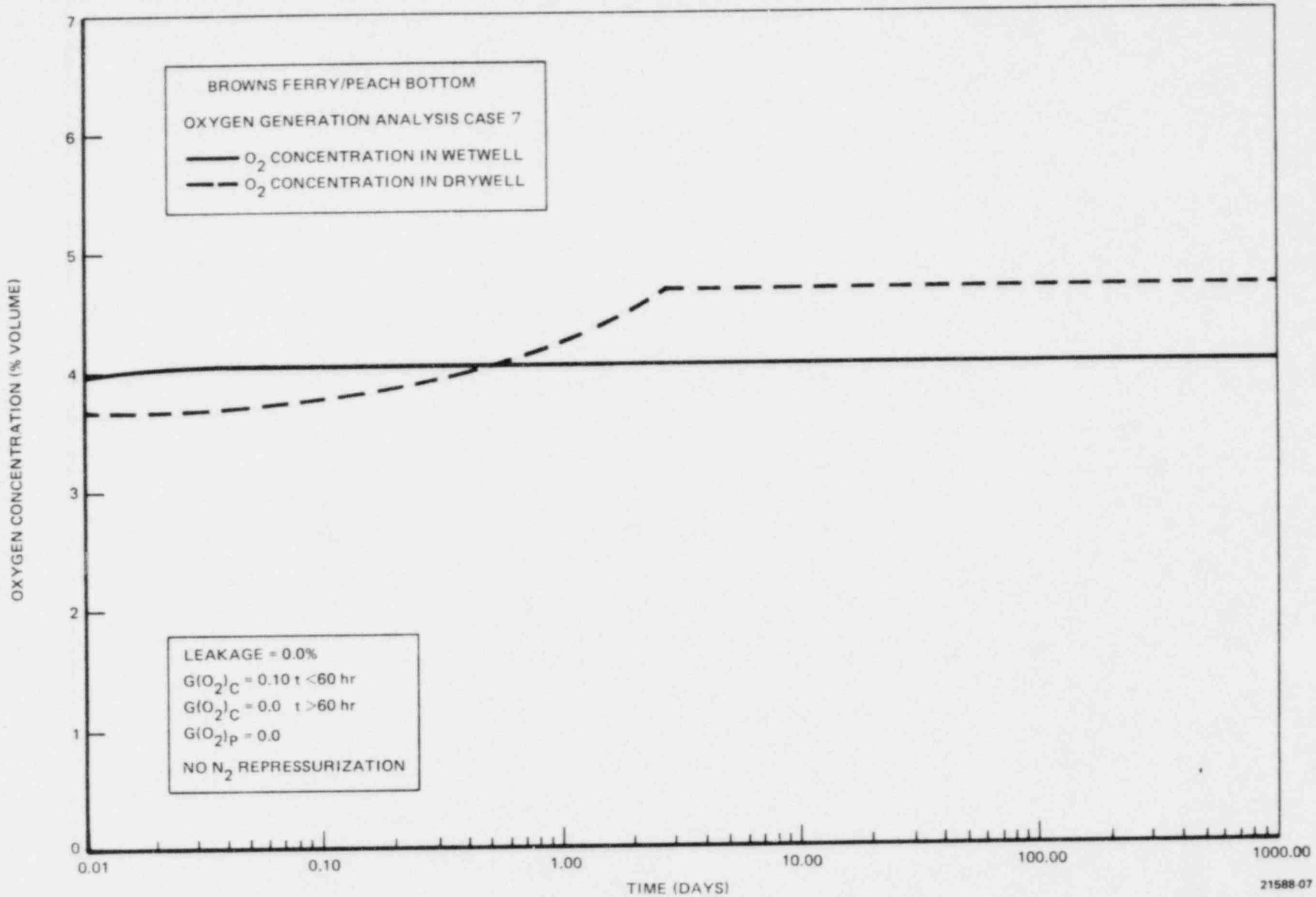

Figure 3-9 . Oxygen Concentration Versus Time for Case 7

4. CONSERVATISMS IN ANALYSIS

4.1 FRACTION OF GAMMA ENERGY ABSORBED IN COOLANT

The assumed fraction of gamma decay energy absorbed in the coolant for those cases, including the base analysis, in which core boiling will terminate at 12 hours was the Regulatory Guide 1.7 value of 0.10 for the entire analysis. This value is conservative, particularly for boiling water which contains a significant amount of voids.

Jenks and Griess have reported that only about 2% of the fast neutron and 0.5% of the gamma energy are absorbed in boiling water.[14] Results of GE analyses have indicated a value of 4.8% for the gamma absorption fraction in BWR boiling water.[15] For non-boiling water, results of calculations performed by GE indicate that the gamma absorption fraction is somewhat less than 0.10.

For those sensitivity cases in which the core water was arbitrarily assumed to continuously boil (Cases 4 and 5), the assumed gamma absorption fraction was 4.8%, which is consistent with the value given in Reference 15.

4.2 DECAY HEAT GENERATION RATE

The decay heat generation rate used in the anlaysis is described in SRP 6.2.5.[16] The reactor decay profiles specified by the ANS 5.1 Standard for two-year reactor operation have been fitted by several finite exponential series expressions and incorporated in the program. Between 400 and 4×10^7 seconds, the equations overpredict the standard curve by 20%. The equations underpredict the standard curve for the first 400 seconds. However, because the oxygen generation due to radiolysis is a concern only in the long term, the use of this decay heat model is considerably conservative.

4.3 ADDITIONAL CONSERVATISMS

In addition to the conservatisms previously described, other conservative assumptions were employed in the analysis. These are as follows:

- Use of maximum technical specification allowable values for the initial oxygen concentration in the containment

- Use of Regulatory Guide 1.7 assumptions for the release fractions of fission products to the reactor coolant and suppression pool

- Neglecting of other sources of oxygen depletion besides recombination and containment leakage, such as oxidation of metals

- Use of 102% core thermal power at the time of the event

- Failure of one RHR loop which reduces the amount of nitrogen which can be added before the limiting pressure is reached.

- Assumption of no containment leakage for the base case and use of a containment leakage which is below most measured data for the sensitivity studies.

- Assumption of the highest allowable suppression pool temperature which limits the amount of nitrogen addition.

5. RADIOLOGICAL IMPACT OF COMBUSTIBLE GAS CONTROL

The previous sections have indicated that for inerted Mark I BWRs as they are currently designed, venting of the containment is not required to adequately control combustible gas concentrations in the containment. Consequently, there are no offsite radiological dose consequences associated with the control of containment combustible gas concentrations.

If limited containment venting were required, the offsite dose consequences would be minimal. As noted in Section 3.4, even if a conservative oxygen generation rate were arbitrarily assumed for non-boiling water, containment venting, if it were required, would not have to occur until 6 years, assuming an initial oxygen concentration of 4% by volume. For a plant with a nitrogen CAD system pressurizing the containment to 30 psig, such venting if it were required, would not have to occur until several hundred years. Airborne iodine activity at these times would be negligible, because even if only radioactive decay were considered, the iodine activity level in containment after 6 years would be only 10^{-80} of its original value. Consequently, the offsite dose associated with containment venting for the non-boiling oxygen source term cases would be negligible.

In addition to radioactive decay, there are inherent BWR design features which act to mitigate the radiological release. Principal among these is the suppression pool, which acts as an efficient scrubber for particulate fission products (including iodine) released from the core. For fission products reaching the wetwell airspace via the suppression pool, decontamination factors of approximately 10,000 can be expected based on available test data.[17] Since the venting flows are within the Standby Gas Treatment System (SGTS) design capability, then that system will provide an additional decontamination factor of 1000 for particulates and iodine in all chemical forms. In addition to these removal mechanisms, gravitational settling, absorption and deposition will also occur within the containment to further reduce fission product releases to the containment. Consequently, the offsite doses from even limited venting of the containment would be insignificant.

6. CONCLUSIONS

The NRC, in recent clarifications of its interim rule on hydrogen control for Mark I and II plants, has indicated that, for plants with inerted containments, the capability to install hydrogen recombiners should exist. General Electric Company, on behalf of the BWR Owners Group, has performed a technical evaluation of the existing inerted Mark I containment design relative to its ability to adequately control combustible gas concentrations.

A conservative evaluation of oxygen buildup in the containment from radiolysis was performed. Values of selected key input parameters were determined based on applicable test data. The results showed that the peak oxygen concentrations for Mark I plants with inerted containments is below the Regulatory Guide 1.7 combustible gas concentration limit without the need for containment venting.

These results show that the existing inerted Mark I containment design is sufficient to assure peak combustible gas concentrations which are below allowable limits without the need to vent the containment or to install recombiner capability.

7. REFERENCES

1. Interim Requirements Related to Hydrogen Control (Final Rule), published in Federal Register on December 2, 1981, pp. 58484-58486.

2. Code of Federal Regulations, 10CFR50.44, subparagraph (g), as revised on January 1, 1981.

3. NEDO-24708A, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors, General Electric Company," Revision 1, December 1980.

4. NEDO-24934, "Emergency Procedure Guidelines - BWR 1-6," Revision 2, June 1982.

5. E. L. Burley, "Oxygen Suppression in Boiling Water Reactors, Quarterly Report 2," NEDC-23856-1, October 1978.

6. "Hydrogen Generation in a Boiling Water Reactor, Preliminary Safety Analysis Report," Amendment 23, Commonwealth Edison Company, Chicago, Illinois, Dresden Nuclear Power Station Unit 3, Docket 50249-39, August 11, 1970.

7. NRC Memorandum, W. R. Butler to R. L. Tedesco, "Three Mile Island, Unit 2: Analysis and Evaluation of Selected Containment Related Issues", April 25, 1979.

8. J. C. Conine, D. J. Krommenhoek, and D. Emanuel Logan, "KAPL Evaluation of Radiolysis Associated with the Three Mile Island Unit-2 Incident", May 1979.

9. S. Gordon, K. H. Schmidt and J. R. Honekamp, "An Analysis of the Hydrogen Bubble Concerns in the Three Mile Island Unit 2 Reactor Vessel," Argonne National Laboratory.

7. REFERENCES
(Cont'd)

10. Letter from R. E. Jagels (Bechtel) to F. R. Hayes (GE), dated May 18, 1982.

11. Interim Requirements Related to Hydrogen Control (Draft Rule), published in Federal Register on December 23, 1981, pp. 62281-62285.

12. Reactor Safety Study, "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", USNRC Report WASH-1400, October 1975.

13. "Probabilistic Risk Assessment - Limerick Generating Station - Philadelphia Electric Company," Docket Nos. 50-352 and 50-353, March 1981.

14. G. H. Jenks and J. C. Griess, "Water Chemistry in Pressurized and Boiling Water Power Reactors," Oak Ridge National Laboratory, ORNL-4173, November 1967.

15. C. L. Martin, "Nuclear Basis for ECCS (Appendix K) Calculations," NEDO-23729, November 1977.

16. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition), Section 6.2.5, September 1975.

17. Letter from G. G. Sherwood (GE) to H. R. Denton (NRC) titled "Confirmation of Suppression Pool Scrubbing Efficiency," dated April 9, 1982.

APPENDIX A

PARTICIPATING UTILITIES

This report applies to the following plants whose owners participated in the report's development.

Boston Edison	Pilgrim 1
Carolina Power & Light	Brunswick 1 & 2
Commonwealth Edison	Dresden 2 & 3, Quad Cities 1 & 2
Georgia Power	Hatch 1
Iowa Electric Light & Power	Duane Arnold
Jersey Central Power & Light	Oyster Creek 1
Niagara Mohawk Power	Nine Mile Point 1
Nebraska Public Power District	Cooper
Northeast Utilities	Millstone 1
Philadelphia Electric	Peach Bottom 2 & 3
Tennessee Valley Authority	Browns Ferry 1, 2, & 3
Vermont Yankee	Vermont Yankee

GENERAL ELECTRIC