

OYSTER CREEK  
NUCLEAR GENERATING STATION

TOPICAL REPORT 081

May 30, 1991

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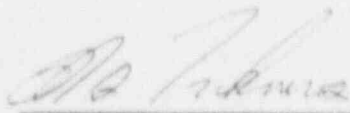
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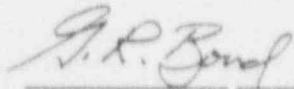
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### EXECUTIVE SUMMARY

In Enclosure 2 of their "Clarification of NRC Staff Position on Hydrogen Mitigation Requirements -10CFR50.44- Oyster Creek Nuclear Generating Station", dated November 6, 1990, the NRC staff questioned the radiolytic oxygen generation rates used in NEDO-22155. The Staff stated that the results in NEDO-22155 were applicable to pure water or water containing only minimal amounts of impurities and that including the effect of iodine could drastically change the results. The Staff also indicated that post accident hydrogen and iodine concentrations may vary during an accident and are specific for each individual plant.

In order to respond to the NRC Staff's concerns, GPUN has prepared Topical Report 081, "Oyster Creek Plant Specific Oxygen Generation Following a LOCA". This report calculates the oxygen concentration in the OC containment as a function of time following a LOCA and conservatively accounts for the hydrogen and iodine concentrations in the containment water. The methodology described in Appendix A of NUREG-0800 (USNRC SRP Section 6.2.5), "Combustible Gas Control in Containment", is utilized except that the non-boiling oxygen generation rate is calculated as a function of dissolved iodine and hydrogen.

An Oyster Creek plant specific iodine concentration was calculated for both the base case LOCA and for a more severe LOCA event in which core cooling is degraded such that a metal-water reaction 5 times that of the base case LOCA occurs. The latter case results in iodine releases that are 300 times more than the base case LOCA. A plant specific fuel heatup calculation, with and without degraded ECCS performance, was performed to determine fuel rod temperatures, metal-water reaction rates and the number of failed fuel rods. The iodine releases were determined by comparing the calculated fuel centerline temperature for the failed fuel rods against NUREG/CR-2367, "Updated Best Estimate LOCA Radiation Signature".

The results of the evaluation show that for the iodine and hydrogen concentrations that would be expected as a result of a relatively severe event, such as a LOCA with degraded ECCS performance, the oxygen concentration inside containment would remain below the 5% oxygen flammability limit. For very severe events in which 30% of the

core iodine is released and 40% of the core undergoes metal-water reaction, the flammable limit is not reached for about a year. Postulated events in which significant amounts of iodine are produced without substantial metal-water reaction are not credible.

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## 1.0 INTRODUCTION

On November 6, 1990, the NRC issued a letter to GPUN entitled, "Clarification of NRC Staff Position on Hydrogen Mitigation Requirements - 10CFR50.44 - Oyster Creek Nuclear Generating Station", (Ref. 1-1). The letter had two enclosures: Enclosure 1 stated the Staff's position on BWR Mark I compliance with the regulations in general; and Enclosure 2 was a Safety Evaluation on the BWR Owner's Group methodology for determining the oxygen generation rates by radiolytic decomposition (NEDO-22155, Ref. 1-2). The Safety Evaluation disagreed with the NEDO report on the radiolytic gas generation rate for boiling and non-boiling conditions. The data which the NRC Staff used was based on an experiment conducted by ORNL (Ref. 3-1) for pure water and a theoretical model developed by BNL for water contaminated with iodine (Ref. 3-2). Both have shown gas production rates higher than the NEDO assumed values. In this report, GPUN will use the NRC recommended model, with consideration of beyond design basis post-accident conditions for both boiling and non-boiling reactor coolant water to calculate the Oyster Creek plant specific oxygen concentration. In particular, the iodine release fraction for conditions complying with the 10CFR50.44 requirements for degraded ECCS performance and its impact on the oxygen production rate will be addressed.



## 2.0 OBJECTIVES

The primary objectives of this report are as follows:

- a) To utilize the basic methodology provided by the NRC in NUREG-0800 (SRP Section 6.2.5), Appendix A (Ref. 5-1) for calculating combustible gas concentrations in containment, with modifications to account for the effect of dissolved iodine and hydrogen on the radiolytic generation rate (G-value).
- b) To develop a value for the concentration of iodine in the containment water following a large break LOCA with a degradation of the ECC system such that the resulting metal-water reaction (and resulting hydrogen release) is 5 times that resulting from a base case LOCA (without ECCS degradation).
- c) To determine the Oyster Creek plant specific containment oxygen concentration as a function of time for the degraded ECC system performance condition evaluated and for more severe accidents as well.
- d) To show that inerting is effective in preventing a flammability condition in the containment following a relatively severe accident in which iodine is released from the core and significant metal-water reaction occurs.

### 3.0 GENERAL DISCUSSION OF POST ACCIDENT RADIOLYSIS

For post-accident radiolytic decomposition of water, Regulatory Guide 1.7 recommends that  $G(O_2) = 0.25$  be used for both boiling and non-boiling conditions. It is known that this value is overly conservative (Enclosure 2 to Reference 1-1) and that many factors will affect the G-value. For instance, temperature has an effect on the rate of decomposition. When water is non-boiling, higher temperature usually reduces the radiolytic gas concentration. However, as the water approaches boiling, higher turbulence increases the gas production rate, and the G-value increases. At the point of boiling, most dissolved gases in the water are stripped out and a higher G-value is appropriate. The presence of impurities, such as dissolved fission products, which come from post-accident fuel failure, have a strong effect with respect to increasing the magnitude of the G-value.

#### 3.1 Temperature and Turbulence Effects on Non-Boiling Water

The ORNL (Ref. 3-1) data shows the hydrogen partial pressure against integrated dose rate for a number of experimental cases. The experiments simulated various representative BWR core flow rates (100 gpm, 1000 gpm and 10,000 gpm), and used different cover gas compositions (air or 5% $O_2$ /95% $N_2$ ) and temperatures (65° C, 95° C and 125° C). The water was distilled so no impurity consideration are involved.

The test data generally concluded that:

- a. From 65° C to 95° C, the G-value decreases with temperature. It turns around when temperature is increased to 125° C (still non-boiling under pressurized condition).
- b. Initial  $G(H_2)$  varies from 0.1 to 0.3 for BWR representative core flow rates from 100 gpm to 10,000 gpm. A higher pumping rate corresponds to more turbulence and thus less recombination.

- c. Radiolytic gas pressure reaches an equilibrium ( $G=0$ ) in each case. At equilibrium, the dissolved hydrogen will recombine with any oxygen molecules produced by radiolysis. The net production is zero.

The Staff's Safety Evaluation (Enc. 2 to Ref. 1) states that for pure water (no iodine), it was determined experimentally that with no dissolved hydrogen and no boiling  $G(O_2)=0.08$ . This conclusion appears to be based on ORNL Case No. 11, which involved 95%  $N_2$  and 5%  $O_2$  gas over distilled water at 65° C and a flow rate corresponding to 10,000 gpm in a BWR (Fig. 3.1).  $G(O_2)$  becomes zero when the hydrogen's concentration reaches 2.5 cc/kg corresponding to an equilibrium partial pressure of 0.16 atm (Ref. 3-3). This was used as an argument that the G-value should be significantly greater than zero for pure water under non-boiling conditions. However, the high equilibrium pressure is mainly caused by the high pumping rate during the experiment (15 cm<sup>3</sup>/min) corresponding to 10,000 gpm in a BWR under forced flow conditions. Higher turbulence removes free radicals faster and thus reduces recombination. For post-accident BWR conditions, all pumps are tripped and a natural circulation condition is in effect in the core. The flow rate under these circumstances is closer to Case 10 of Reference 3.1 (see Fig. 3-2); from which we can derive  $G(O_2)$  becoming zero at a hydrogen partial pressure of about 0.04 atm (4% hydrogen in containment) or a concentration in water of about 0.6 cc/kg (Ref. 3-3).

Assuming a degraded core condition with 5 times the 10CFR50.46 calculated metal water reaction (2.24% MWR), the initial hydrogen concentration in the Oyster Creek containment is calculated to be about 4% (Ref. 3-3). This partial pressure of hydrogen under non-boiling conditions was shown above to result in a  $G(O_2)=0.0$  at equilibrium. The ORNL data case 10 thus supports the NEDO assumption of  $G(O_2)=0$  for non-boiling if no iodine was assumed in the post-accident water.

Since the presence of iodine in post-accident reactor coolant cannot be ignored, the pure water G-value data will not be used in the Oyster Creek plant specific calculations presented in Section 5 of this report.

### 3.2 Boiling

Boiling strips dissolved gases out of the liquid phase so that the maximum decomposition will proceed. Equilibrium between the atmosphere and the liquid will not occur during boiling. It is conservatively assumed that post-accident boiling will last 12 hours in the Oyster Creek combustible gas concentration calculation (consistent with NEDO-22155). A  $G(O_2) = 0.225$  will also be conservatively assumed for this entire duration. Enclosure 2 to Reference 1 states that the maximum values of  $G(O_2)$  for 5% MWR and 30% iodine release are between 0.19 and 0.22.

### 3.3 Impurities

The presence of impurities such as post-accident fission product iodine in the reactor water may affect the decomposition rate. A static model is used by the NRC (Ref. 3-2) as follows:

$$G(H_2) = G_{H_2} - \frac{G_{OH}}{1 + \frac{K_I [I]}{K_H [H_2]}}$$

$$G(O_2) = \frac{GH_2}{2}$$

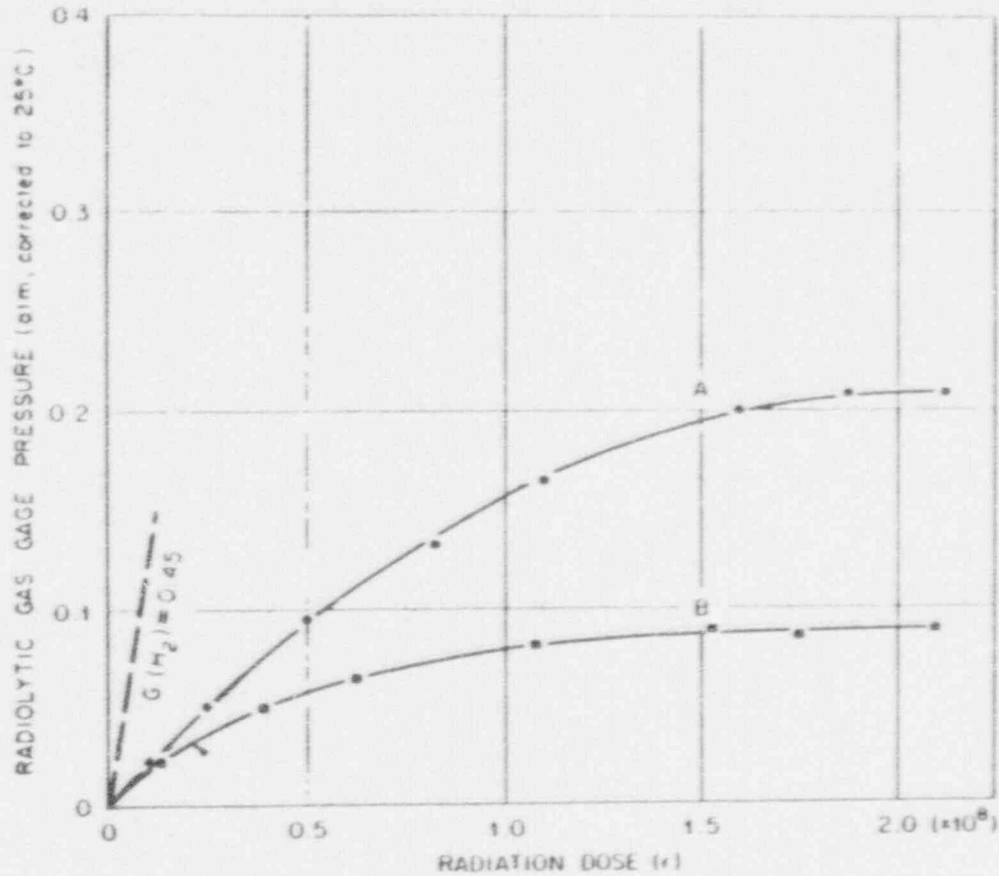
Where

- $G_{H_2}$ ,  $G_{OH}$  = initial G-value of hydrogen and OH radical, molecules/100 ev
- $[H_2]$ ,  $[I]$  = molar concentration of  $H_2$  and I
- $K_H$ ,  $K_I$  = rate constant of the reactions between OH- and hydrogen and OH- and iodine

For a small iodine concentration in the water ( $< 10^6$  gm-mole/l),  $G(H_2)$  decreases very quickly to zero as the hydrogen concentration in the water builds up. However, for a moderate iodine concentration (  $[I] > 10^6$  gm-mole/l ), which corresponds to greater than 2% of the total core iodine being released to the water, the G-value remains positive and would increase the long term oxygen build up in the containment.

Figure 3-1. ORNL Experiment No. 11

ORNL DWG. 70-7401



TEST CONDITIONS

SOLUTION, DISTILLED  $H_2O$   
 GAS-TO-LIQUID RATIO, 2.5:1  
 TEMPERATURE  
 A 65°C  
 B 95°C  
 C 125°C  
 LIQUID VOLUME, 200  $cm^3$   
 EFFECTIVE DOSE RATE,  $4 \times 10^5$  r/hr  
 FLOW RATE, 15  $cm^3/min$  ( $\sim 10,000$  gpm IN BWR)  
 COVER GAS, 95%  $N_2$  + 5%  $O_2$  (1 ATM)

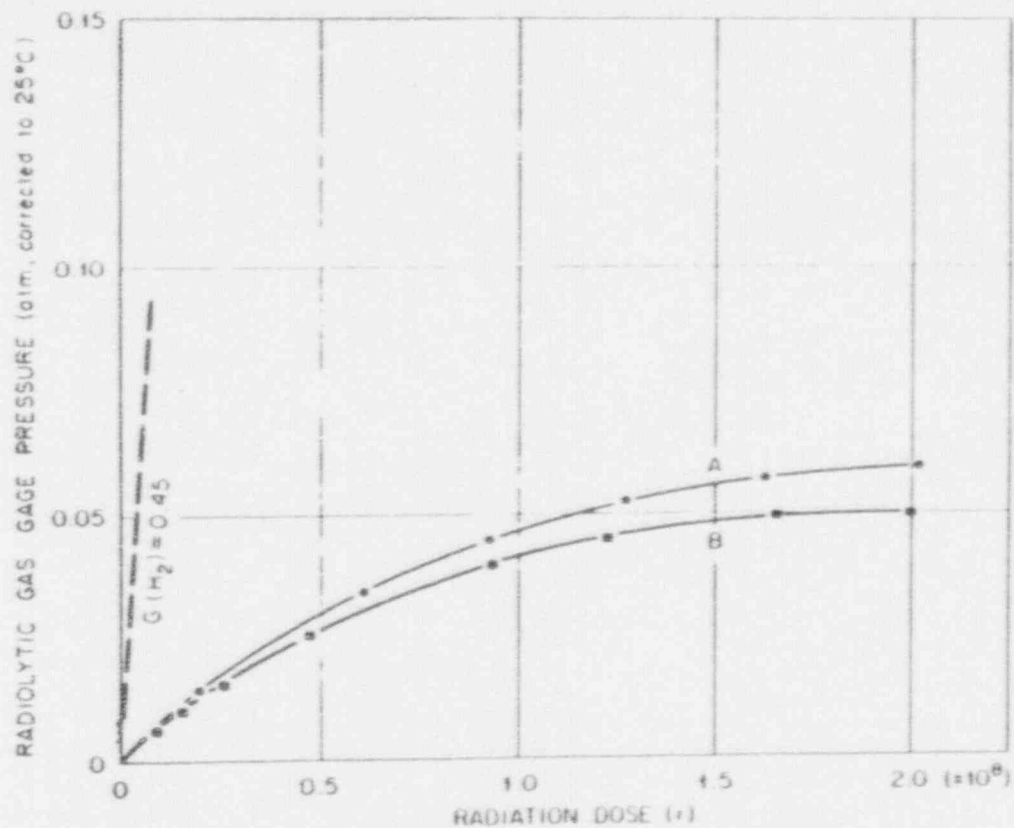
GAS ANALYSIS IN VOLUME % AT EQUILIBRIUM

	$H_2$	$O_2$	$N_2$
A	11.6	10.6	78.1
B	6.1	7.9	85.2

	BEFORE		AFTER	
	A	B	A	B
pH	6.3	6.4	5.9	5.9
L, $\text{ohm}^{-1} \text{cm}^{-1} \times 10^5$	6	7	20	15
[Fe], ppm	0	0	0	0

Figure 3-2. ORNL Experiment No. 10

ORNL-DWG 70-740Z



TEST CONDITIONS

SOLUTION, DISTILLED H<sub>2</sub>O  
 GAS-TO-LIQUID RATIO, 2.5 : 1  
 TEMPERATURE  
 A 65°C  
 B 95°C  
 C 125°C  
 LIQUID VOLUME, 200 cm<sup>3</sup>  
 EFFECTIVE DOSE RATE, 4 x 10<sup>6</sup> r/hr  
 FLOW RATE, 0.15 cm<sup>3</sup>/min (~100 gpm IN BWR)  
 COVER GAS, 95% N<sub>2</sub> + 5% O<sub>2</sub> (1 ATM)  
 GAS ANALYSIS IN VOLUME % AT EQUILIBRIUM

	H <sub>2</sub>	O <sub>2</sub>	N <sub>2</sub>
A	3.7	6.9	89.5
B	3.6	6.8	89.4

	BEFORE		AFTER	
	A	B	A	B
pH	6.2	6.3	5.8	5.9
L, ohm <sup>-1</sup> cm <sup>-1</sup> x 10 <sup>6</sup>	7	7	25	18
[Fe], ppm	0	0	0	0

#### 4.0 OYSTER CREEK SPECIFIC IODINE RELEASE AND METAL-WATER REACTION

##### 4.1 Iodine Concentration from LOCA and LOCA with Degraded Conditions

Based on the NUREG/CR-2367 fission product release data (Ref. 4-1) an Oyster Creek specific iodine release concentration was calculated for both base case LOCA and LOCA with degraded core cooling. The degraded core cooling case is defined as having a metal water reaction (and resulting hydrogen generation) that is 5 times the amount calculated for the base case LOCA used in this evaluation. The metal water reaction rate used in the base case LOCA was somewhat greater than those calculated pursuant to 50.45(b)(3) because of the ECCS code used in the evaluation. The released iodine concentrations are calculated in this section to be  $1.80\text{E-}09$  and  $5.44\text{E-}07$  gm mol/l for the base case LOCA and the LOCA with degraded conditions, respectively.

The Oyster Creek core wide metal water reaction based on compliance with 10CFR50.46 is 0.448%. The 10CFR50.44 criterion for degraded core conditions is the larger of: 1) five times the amount of hydrogen calculated in demonstrating compliance with 10CFR50.46, or 2) for the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel to a depth of 0.00023 inch. A fivefold increase in hydrogen corresponds to a fivefold increase in the metal water reaction which would be 2.24%. The metal water reaction due to the reaction of 0.00023 inches of the cladding surfaces is 0.77%. Therefore, the five time increase in MWR is the criterion used for Oyster Creek in determining the degraded core condition.

A fuel heat up calculation was performed to determine fuel rod temperatures, MWR and the number of failed fuel rods during a LOCA based on an end of cycle (EOC) core conditions. Using the NUREG/CR-2367 iodine release rate, the total iodine concentration was calculated along with a core wide metal water reaction. This served as a basis from which the degraded core cooling case could be evaluated. For the degraded core cooling case, it was assumed that the initiation of emergency core cooling was delayed and flow rates were less than Appendix K requirements. The fuel heat up



calculations were redone with reduced ECCS flow and iterating on the time for delayed core cooling until the metal water reaction increased by a factor of 5. An iodine concentration was calculated for the degraded core cooling case using the resulting fuel rod temperatures and Reference 4-1 fission product release data.

The heat up calculations were performed using the HUXY code (Ref. 4-2). The HUXY code has been approved to perform 10CFR50 Appendix K calculations for the ANF fuel loaded in Oyster Creek. The HUXY code does not calculate the mechanical response of the cladding during a LOCA. However, it does allow a temperature input which, when exceeded, fails the fuel rod and calculates a MWR based on both an inner and outer cladding surface as per Appendix K. Current licensing analyses (Ref. 4-3), show that a fuel rod will perforate at nodal exposures exceeding 19.0 GWD/MT when the peak clad temperature (PCT) exceeds 1600 F. Dividing by an approximate axial exposure peaking factor of 1.25, this translates to a bundle average exposure of 15.2 GWD/MT. For bundles having exposures less than 15.2 GWD/MT, a fuel failure temperature of 2500 F was used.

A core power distribution histogram (Figure 4.1) of number of bundles versus radial power was developed to bundle exposures below and above 15.2 GWD/MT. An EOC case was used for conservatism to maximize the number of higher exposed fuel bundles. Another conservatism was to place all fuel bundles in the peak radial power group, for the high and low exposures, at their MAPLHGR limit. The heat up calculations were repeated for the low and high exposures, for each of the radial power factors indicated, and for the base case LOCA and LOCA with degraded core cooling conditions.

The results of the calculations are summarized in Table 4.1. The base case LOCA calculations result in 6288 failed fuel rods out of the 33,600 rods in the core, and a core wide MWR of 1.16%. Both of these values are greater than the Appendix K results due to the additional conservatisms used in this

analysis. The degraded cooling case results in 17744 failed fuel rods and a core wide MWR of 5.85% (an increase of a factor of 5.04 over the base case LOCA case). The iodine release rate was calculated for each group of bundles for a given radial power factor based on the calculated fuel temperature. A fuel rod iodine concentration of 0.486 gms per fuel rod, which corresponds to a high power rod, was conservatively used for all failed fuel rods in the core. The average core iodine concentration is 0.389 g/fuel rod (Ref. 4-4). In addition, the iodine release rate for a failed fuel rod was conservatively based on the limiting (hottest) axial node in the rod.

#### 4.2 Relationship Between Metal-Water Reaction and Iodine Release

The analysis discussed in Section 4.1 provides an estimate of the core wide MWR and iodine release for degraded core conditions. The treatment of the MWR was based on the parabolic rate law of Baker and Just and the iodine release was determined using NUREG/CR-2367. Figure 4-2 is a plot of the MWR and iodine release as a function of pin centerline temperature for a failed fuel pin. As can be seen, if the pin centerline temperature increases, both percent MWR and iodine release also increase. As the temperature increases to 1600° C, the iodine release approaches 30% while the MWR approaches 70% of total.

1600° C represents a limiting fuel pin condition for the degraded core analysis reported in Section 4.1. While a few pins may approach this limiting condition, the majority of the fuel will remain well below this temperature. The inset in the upper left corner of Figure 4-2 lists the degraded core analysis results for percent of total core MWR and percent of total core iodine release. The iodine release and MWR percentages reflect the fact that for the degraded core condition, only half the pins fail and the centerline temperatures of most of the pins are well below 1600° C.

Figure 4.1  
OC Power Distribution Histogram

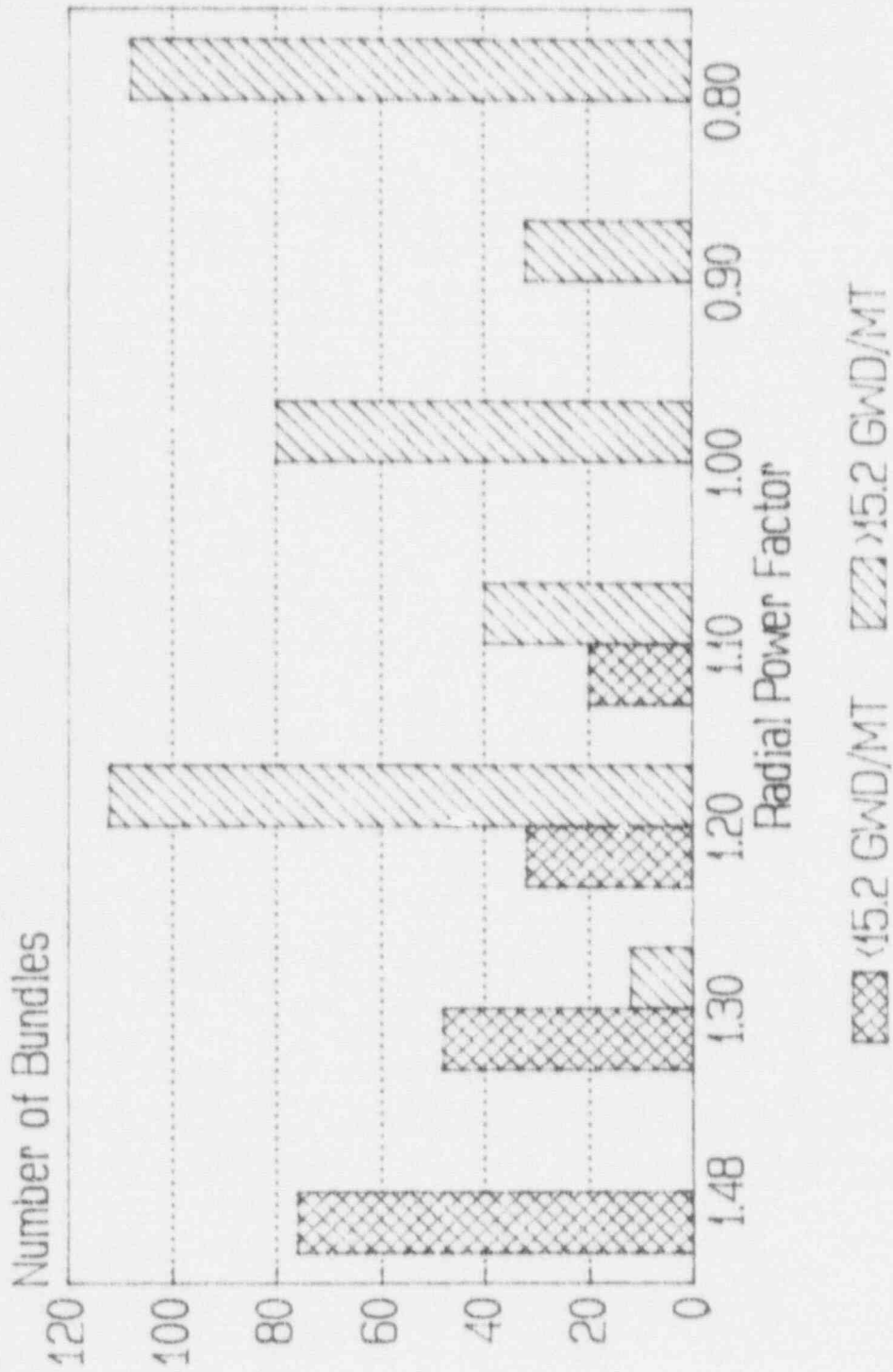


TABLE 4.1

Oyster Creek Iodine Release During LOCA with Normal Core Cooling								
EXPOSURE GWD/MT	RADIAL POWER	FAILED RODS/ BUNDLE	BUNDLES	TOTAL FAILED RODS	C/L FUEL TEMP C	MWR	IODINE RELEASE RATE	IODINE CONCENTRATION GM MOL/L
<15.2	1.48	0	76	0	1200	3.5	0.00035	0.00E+00
<15.2	1.3	0	48	0	1080	1.8	0.00024	0.00E+00
<15.2	1.2	0	32	0	1010	1.1	0.00017	0.00E+00
<15.2	1.1	0	20	0	960	0.7	0.00013	0.00E+00
>15.2	1.3	52	12	624	1160	6.1	0.00031	2.79E-10
>15.2	1.2	32	112	3584	1060	3.2	0.00022	1.14E-09
>15.2	1.1	30	40	1200	990	2.0	0.00015	2.60E-10
>15.2	1	11	80	880	930	0.9	0.0001	1.27E-10
>15.2	0.9	0	32	0	860	0.3	0.00006	0.00E+00
>15.2	0.8	0	108	0	815	0.2	0.00004	0.00E+00
TOTALS:			560	6238				1.80E-09

TOTAL CORE % IODINE RELEASED = 0.0046

TOTAL CORE % MWR = 1.16

TABLE 4.2

Oyster Creek Iodine Release During LOCA with Degraded Core Cooling

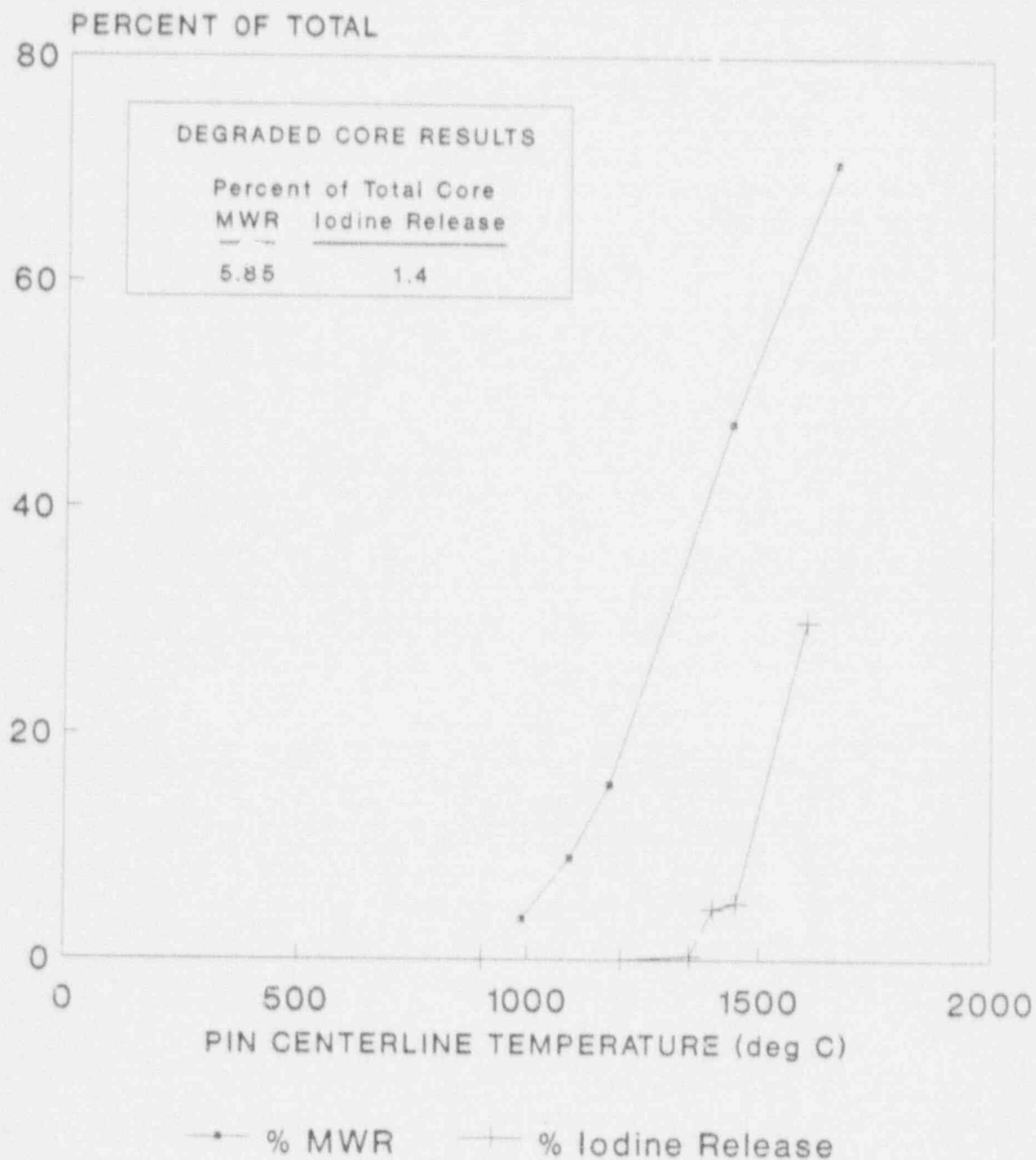
EXPOSURE GWD/HT	RADIAL POWER	FAILED RODS/ BUNDLE	BUNDLES	TOTAL FAILED RODS	C/L FUEL TEMP C	MWR	IODINE RELEASE RATE	IODINE CONCENTRATION GM MOL/L
<15.2	1.48	15	76	1140	1460	16.3	0.067	1.10E-07
<15.2	1.3	0	48	0	1390	7.8	0.037	0.00E+00
<15.2	1.2	0	32	0	1310	5.8	0.0023	0.00E+00
<15.2	1.1	0	20	0	1210	3.7	0.00053	0.00E+00
>15.2	1.3	60	12	720	1510	26.9	0.15	1.16E-07
>15.2	1.2	60	112	6720	1380	17.5	0.028	2.71E-07
>15.2	1.1	60	40	2400	1260	10.2	0.0014	4.84E-09
>15.2	1	60	80	4800	1140	5.3	0.00029	2.01E-09
>15.2	0.9	31	32	992	980	1.8	0.00014	2.00E-10
>15.2	0.8	9	108	972	910	0.7	0.00008	1.12E-10
<b>TOTALS:</b>			<b>560</b>	<b>17744</b>				<b>5.44E-07</b>

TOTAL CORE % IODINE RELEASED = 1.4

TOTAL CORE % MWR = 5.65

FIGURE 4-2

### FAILED PIN MWR AND IODINE RELEASE AS A FUNCTION OF TEMPERATURE



5.0 PLANT SPECIFIC OXYGEN CONCENTRATION WITH REVISED G-VALUES

5.1 Methodology

The Oyster Creek plant specific oxygen concentration was calculated for a variety of iodine and MWR assumptions. The methodology described in Appendix A of the NRC Standard Review Plan (Section 6.2.5), NUREG-0800, "Combustible Gas Control in Containment" (Ref. 5-1), was used except that G-values are calculated as a function of dissolved iodine and hydrogen (Ref. 3-2) (see Section 3.3):

$$G(H_2) = 0.45 - 2.7 / (1 + k_i [I] / k_h [H]) \quad (5.1)$$

where

- $G(H_2)$  = net hydrogen generation rate, molecules/100 ev
- $[I]$  = iodine molar concentration in the coolant
- $[H]$  = hydrogen molar concentration in the coolant
- $k_i, k_h$  = reaction rate constants for the adverse iodine reaction and the favorable hydrogen reaction in radiolysis suppression

For the first 12 hours of the LOCA,  $G(H_2)$  is given its maximum value (0.45) as in boiling.

Thereafter,  $G(H_2)$  is calculated from Equation 5.1. The dissolved hydrogen is calculated from Henry's Law;

$$P_{H_2} = K_H \cdot [H] \quad (5.2)$$

where

- $K_H$  = the Henry Law constant for hydrogen in water
- $P_{H_2}$  = the pressure of hydrogen in the gas phase

The details of the application of these equations to the specifics of the OCNCS are given in Appendix A.

## 5.2 Results

### 5.2.1 Oyster Creek Specific Oxygen Concentration

The results of the HUXY analyses discussed in Section 4.0 showed that for a base case LOCA, the iodine concentration will be  $1.80E-9$  g-moles/liter which corresponds to a total core iodine release of about 0.0046%. The total core % MWR for that case was 1.16%. The containment oxygen concentration for this event would only increase by about 0.25% as can be seen in Figure 5-1.

For the degraded ECCS case analyzed, the total core iodine release was 1.4% and the calculated % MWR was 5.85% (a factor of 5.04 increase over the base case LOCA case). The oxygen concentration in containment for this case would not increase from the initial value. This is depicted in Figure 5-2. Figure 5-3 provides the results for the same iodine release case (1.4%) with a MWR of 2.24% (5 times the 0.448% calculated for 10CFR50.46). Again, a 5% containment oxygen concentration is not reached.

### 5.2.2 Oxygen Concentration Following Severe Accidents

#### 5.2.2.1 Total Core

In this section, iodine and MWR assumptions more severe than those calculated specifically for Oyster Creek in Section 4.0 are evaluated with respect to expected oxygen concentrations. These analyses are being performed to address the release of larger amounts of iodine up to and including 30% of the total core iodine inventory. The release of such large fractions of the total core iodine inventory would require that all of the core fuel rods achieve substantial fuel centerline temperatures (Ref. 4-1). Fuel rods achieving such high centerline temperatures



would also be undergoing substantial metal water reaction. The relationship between these parameters was discussed in Section 4.2 of this report and will form the basis of the cases analyzed herein.

Figures 5.4 through 5.6 show the oxygen concentration profiles for 10%, 20% and 30% total core iodine release. The 10% iodine analysis (Fig. 5-4) shows oxygen concentration for a % MWR of 15%. The results indicate that it will take about a year to reach 5% oxygen concentration. Figure 5.5 shows the 20% iodine release results with a % MWR of 30%. Again, oxygen concentrations of 5% do not result in less than approximately one year. Similar results can be seen in Figure 5.6 for the 30% iodine release and a % MWR of 40%.

The selection of the % iodine/% MWR ratios was based upon the results depicted in Figure 4.2 which shows that for a given fuel temperature condition that results in a % iodine release, the % MWR for that condition will be substantially higher than the corresponding % iodine. A conservative ratio of 1.5 or less was used for each case analyzed.

#### 5.2.2.2 Localized Effects

This section is addressing the concern that, in the event of a LOCA, a small fraction of the core might become overheated. It is assumed that this might occur from a hot-spot resulting from local coolant flow starvation as a result of: 1) delivery of less than planned cooling to a localized area, or 2) local flow blockage. It is further assumed that the expected MWR will not occur at any time even though such an assumption is not credible.

The following conservative assumptions are being used:

- a) Fuel centerline temperatures in affected region reach 1600° C (30% iodine release).
- b) Size of affected region is 10% of core (56 bundles).
- c) % MWR for affected region is 1%\*.
- d) Remainder of core as per degraded LOCA case of Section 4.1.

\* This is conservative since higher percent MWR would produce less oxygen than the assumed case because of the suppressing effects of hydrogen on  $G(O_2)$ . The percent MWR would be about 70% for the affected region.

These assumptions result in an overall total core iodine inventory release of:

$$(1.4\%) * (0.9) + (30\%) * (0.1) = 4.26\%$$

The total core % MWR is:

$$(5.85\%) * (0.9) + (1.0\%) * (0.1) = 5.36\%$$

The oxygen profile resulting from this condition is shown in Figure 5.7. The results show that a 5% oxygen concentration in containment for this non-credible assumption will not be reached for approximately three months.

### 5.2.2.3 Iodine Release Without MWR

Iodine release without a comparable MWR is not credible. Even if a blockage of cooling water to a small region of the core is assumed as the basis for limiting the MWR, eventually either cooling will occur or fuel melt will result. Melt progression will only cease when cooling is re-established. When this occurs, MWR will also occur. The requirement of 50.44(h)(1) is that the degree of degradation is not sufficient to cause core meltdown. This implies that cooling is established, and this cooling of hot fuel must result in significant MWR.

At TMI-2 where cooling was unavailable such that significant core heatup occurred, a significant MWR resulted from the eventual re-establishment of cooling water. Even if complete core melt were to occur, significant MWR would occur when the melt material comes in contact with water inside containment.

### 5.2.3 Additional Conservatism

- a) The NRC model used in this report (Section 5.1 and Appendix A) is very conservative (overpredicts G-value) when it is applied to low impurity (low iodine and hydrogen concentrations) cases. The reason for this is the assumption of an initial  $G(H_2)$  of 0.45 which is then allowed to decrease. In the Zittel experiment for pure water (Ref. 3-1), the G-value never exceeds 0.3. This difference contributes to a larger radiolytic gas production rate and higher oxygen concentration in containment. For iodine concentrations less than approximately  $10^{-6}$  g-moles/liter, it would be more appropriate to apply the Zittel results.
  
- b) The calculation herein assumes that the precipitated  $ZrO_2$  from the metal-water reaction would occlude 10% of the water and that this water would have a  $G(H_2) = 0.45$ . The NRC in Reference 3-2 uses a value of 1% rather than 10% for this effect. The model herein would thus overpredict the radiolytic gas production slightly as a result of this.

FIGURE 5-1  
BASE CASE LOCA

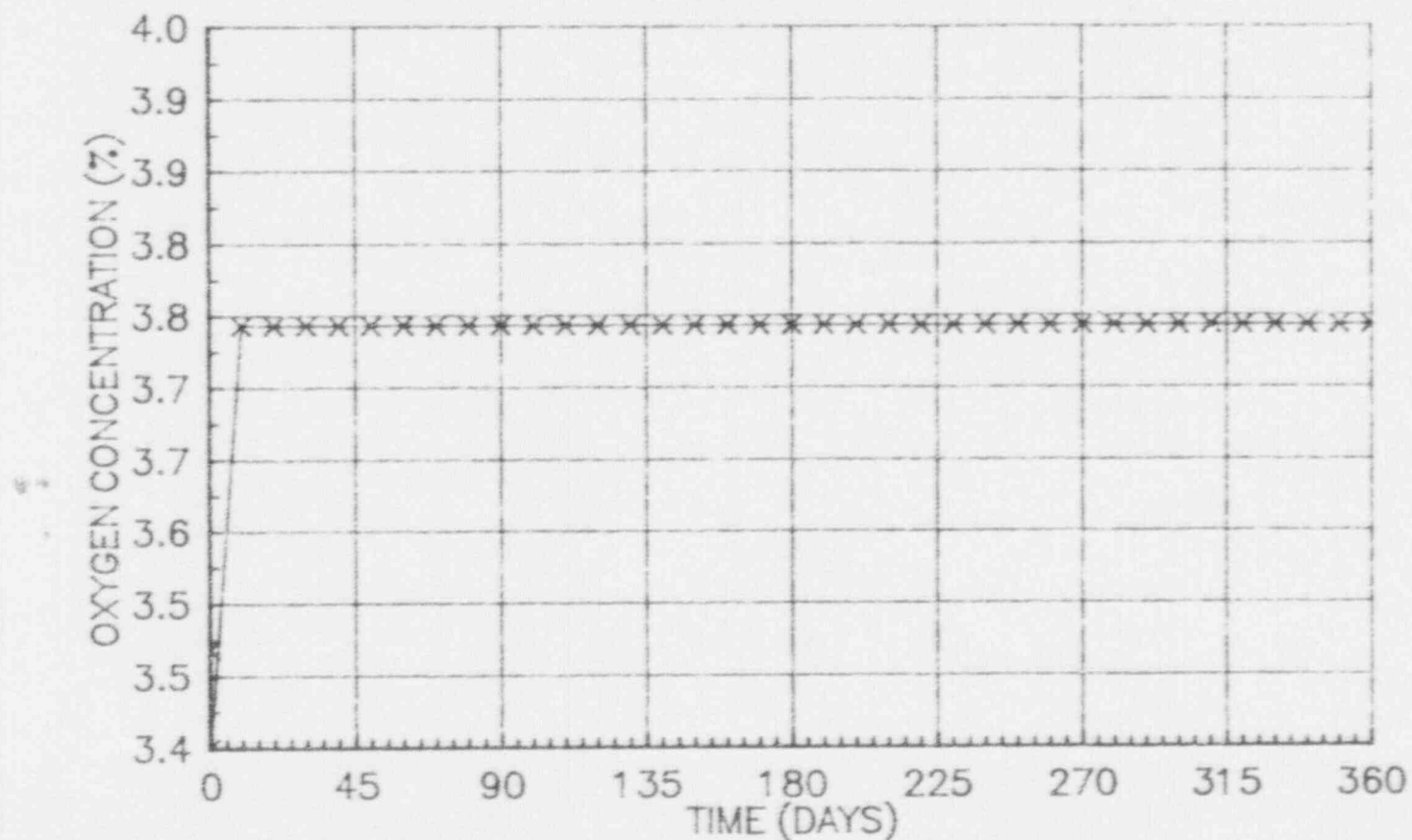


FIGURE 5-2  
IODINE=1.4%; MWR=5.85%

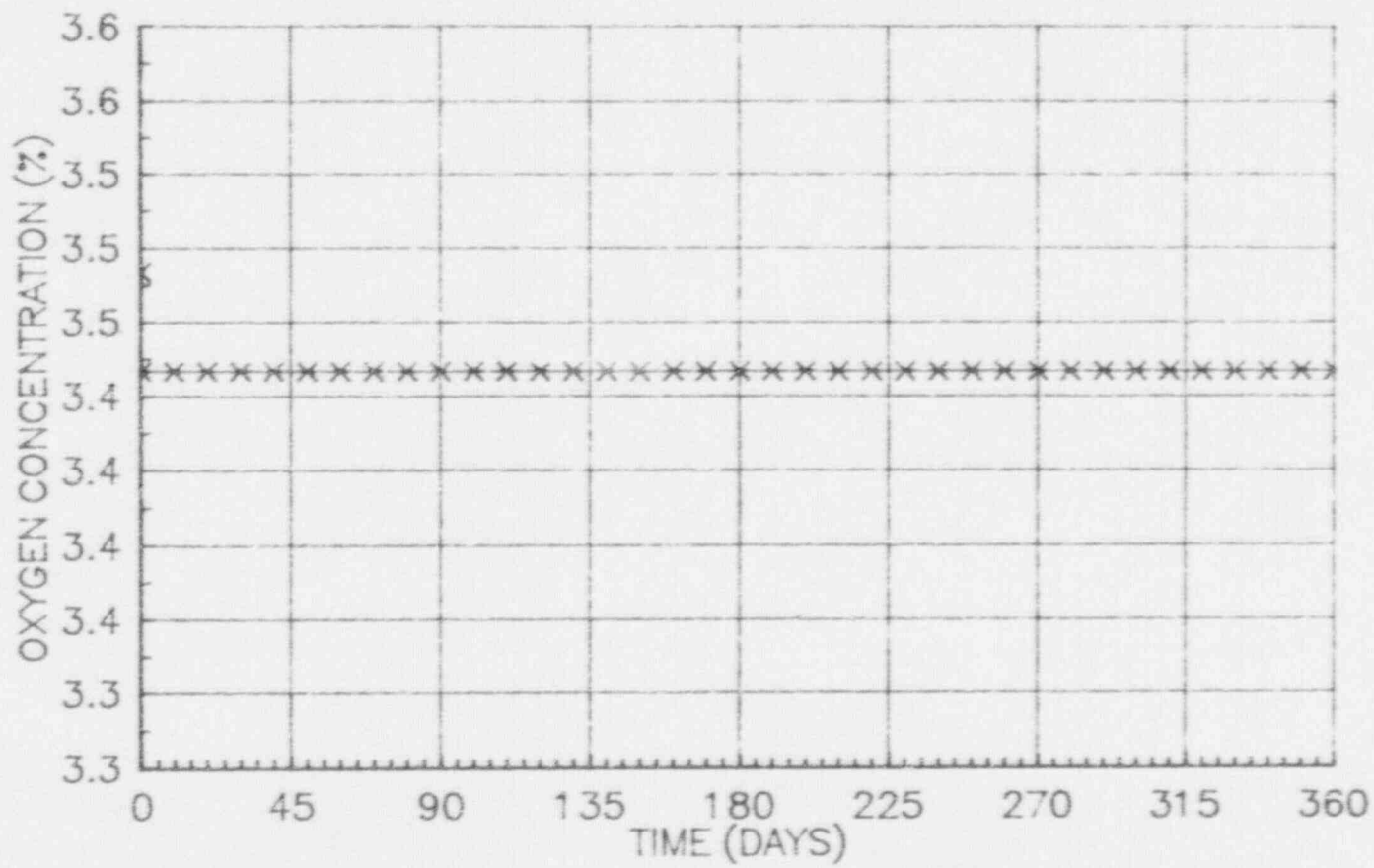


FIGURE 5-3  
IODINE=1.4%; MWR=2.24%

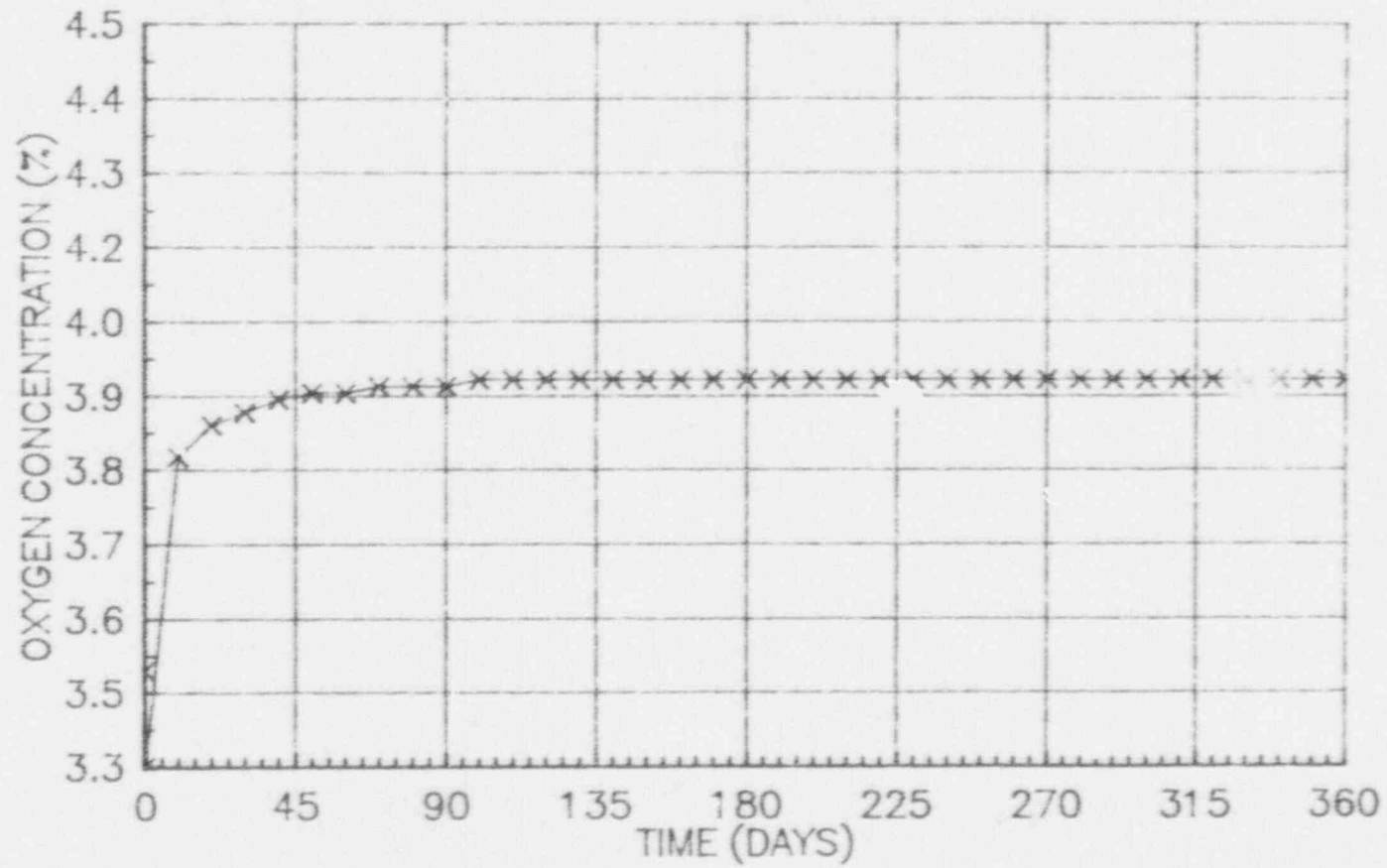


FIGURE 5-4  
IODINE=1.0%; MWR=1.5%

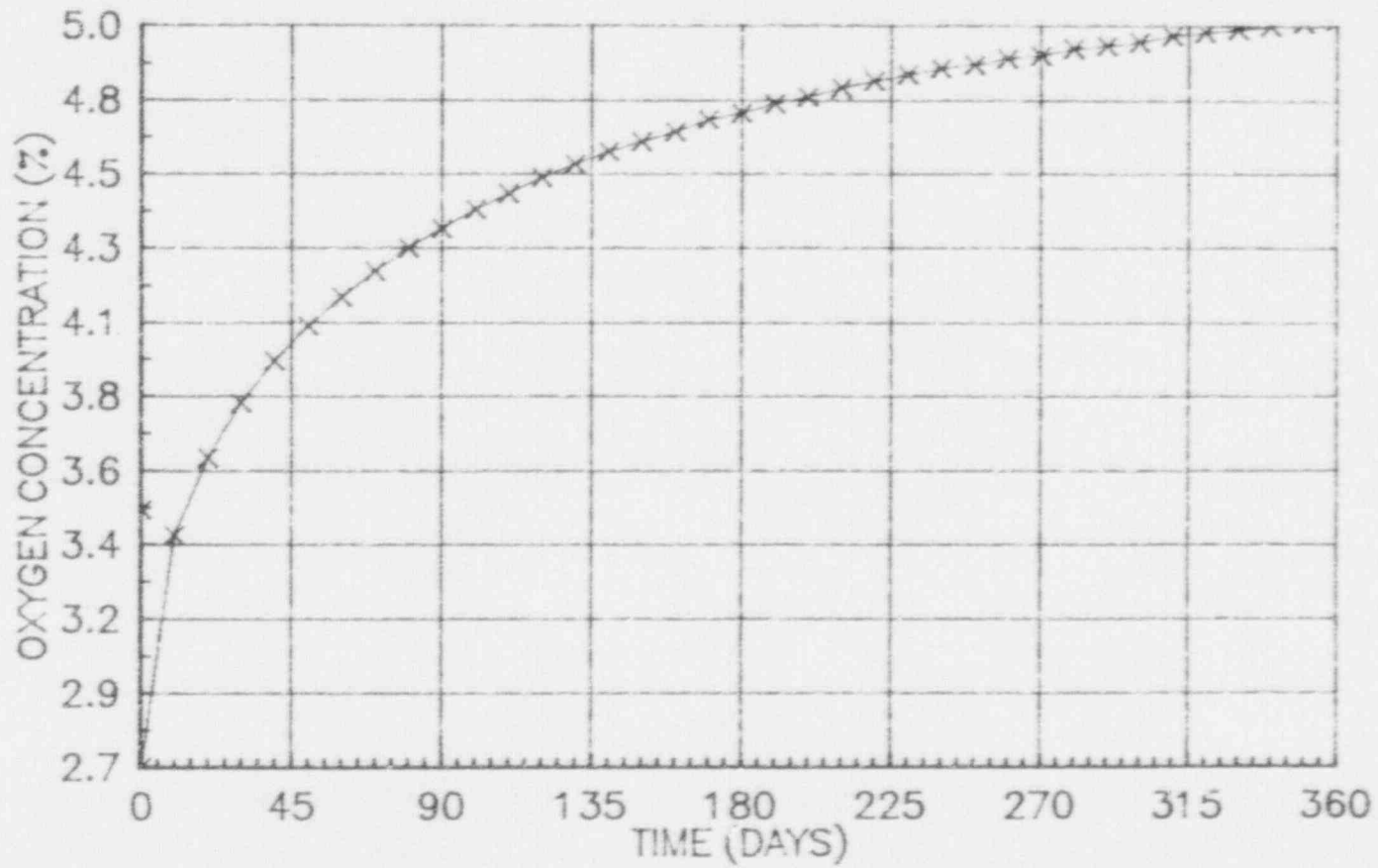


FIGURE 5-5  
IODINE=20%; MWR=30%

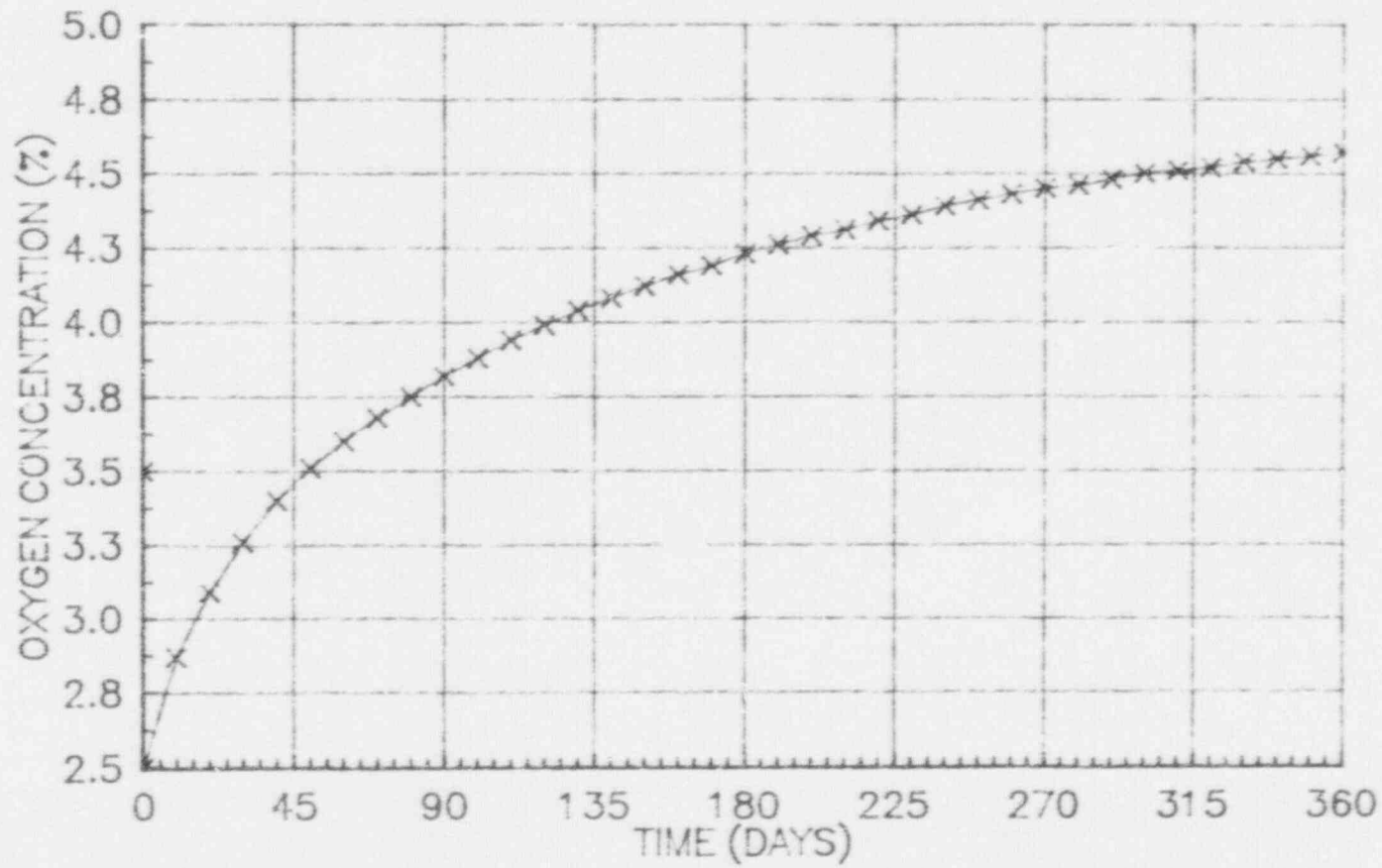




FIGURE 5-6  
IODINE=30%; MWR=40%

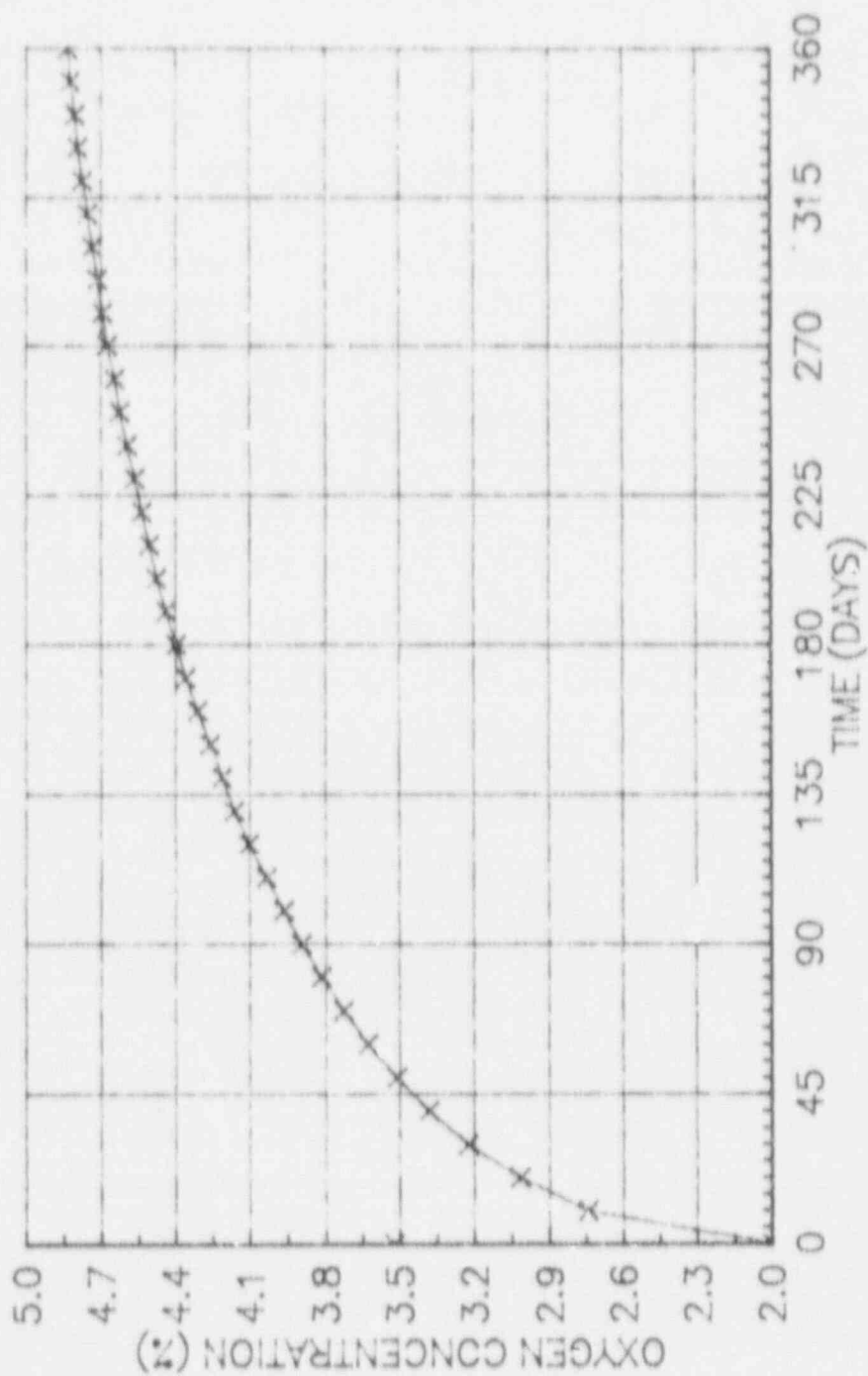


FIGURE 5-7  
IODINE=4.26% : MWR=5.36%

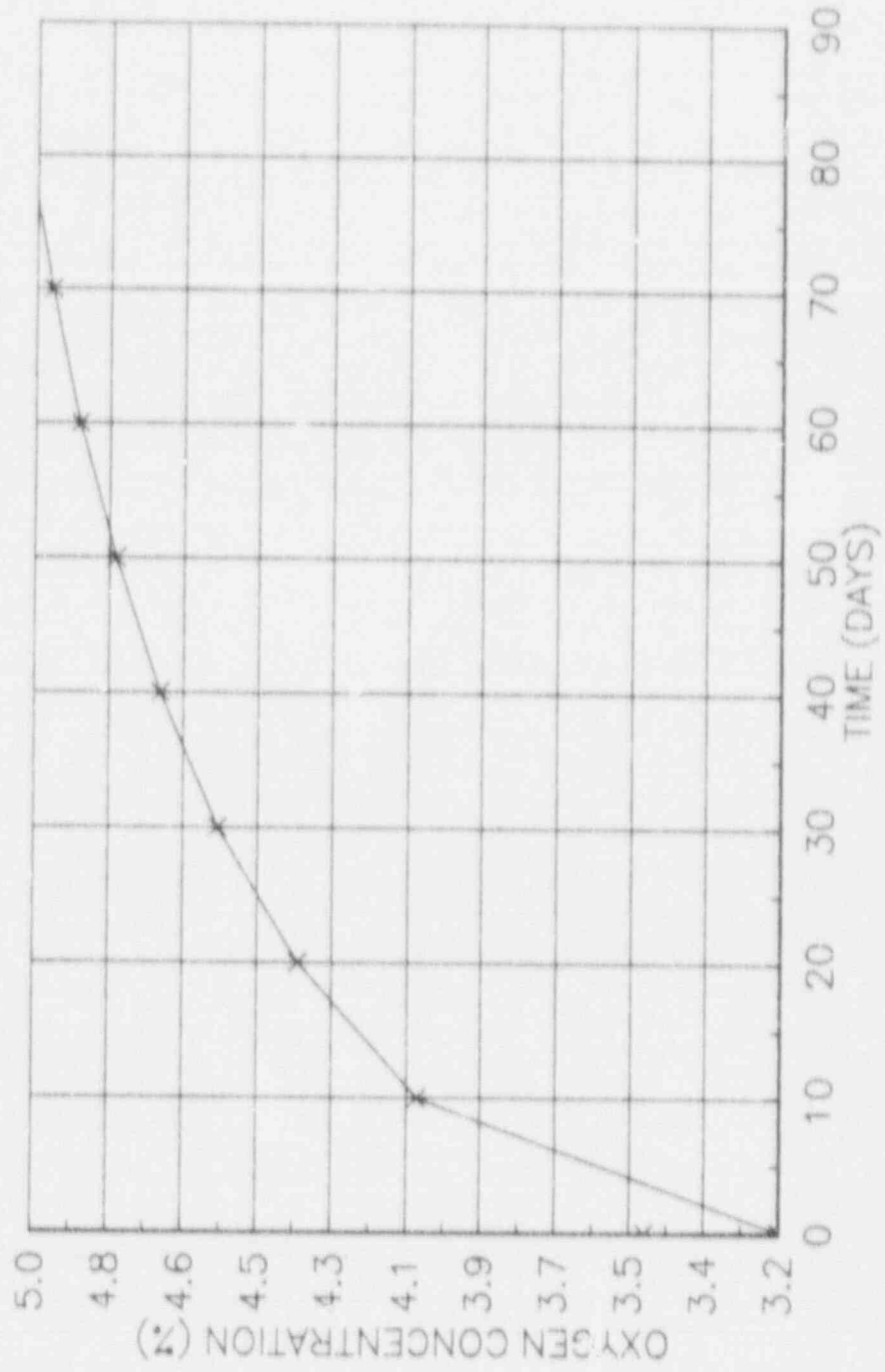


TABLE 5-1  
SUMMARY OF RESULTS

<u>CASE DESCRIPTION</u>	<u>% IODINE</u>	<u>% MWR</u>	<u>TIME TO 5% OXYGEN (DAYS)</u>
Base case LOCA	0.0046	1.16	> 1000
LOCA with Degraded ECCS	1.4	5.85	> 1000
LOCA with Degraded ECCS	1.4	2.24	> 1000
Severe Accident	10	15	360
Severe Accident	20	30	> 1000
Severe Accident	30	40	> 1000
Localized Effect	4.26	5.36	90

## 6.0 CONCLUSIONS

The following conclusions can be reached as a result of the analyses discussed in the report:

- a) The non-boiling  $G(O_2)$  is not zero with the presence of dissolved iodine, but the effect of the dissolved iodine in the containment water on the  $G(O_2)$  is offset by the effect of the dissolved hydrogen resulting from the initial metal-water reaction and from radiolysis.
- b) For both a LOCA and a LOCA with severely degraded ECCS performance, the oxygen concentration in the Oyster Creek containment would not reach 5%.
- c) For severe accidents in which 30% of the total core iodine is released, i.e., NRC assumption of fuel rod centerline temperatures of 1600° C over the entire core, the oxygen concentration would not reach 5% in less than approximately one year for a conservatively low zirconium water reaction rate of 40%. For zirconium water reaction rates greater than 40%, it would take even longer to reach 5%.
- d) There is no credible mechanism by which substantial amounts of core iodine can be released without a substantial amount of metal-water reaction.
- e) Even in the event that flow is blocked to a small fraction of the core following a LOCA, oxygen concentration in containment would not reach 5% for several months.

## 7.0 REFERENCES

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- 3-1 ORNL-TM-2412 Park VIII, "Design Considerations of Reactor Containment Spray Systems - Part VIII Boiling Water Reactor Radiolysis Studies", H. E. Zittel, October 1970.
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- 3-3 GPUN Calculation C1302-243-5450-060, "OC Post LOCA Hydrogen and Iodine Concentration in the Containment and Gas Production Rate by Radiolysis", June 1991.
- 4-1 "Updated Best-Estimate LOCA Radiation Signature," NUREG CR 2367, D. D. Thayer.
- 4-2 HUXY: A Generalized Multipod Heatup Code with Appendix K Heatup Option XN-CC-33(A) July 28, 1975.
- 4-3 "Oyster Creek NGS SAFER/CORECOOL/GESTR-LOCA Loss of Coolant Accident Analysis," August 1987.
- 4-4 GPUN Calculation C1302-226-5411-236, "Iodine Release During LOCA with Degraded Core Cooling", May 1991.
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- A-1 NUREG-0800, USNRC Standard Review Plan: Section 6.2.5 Combustible Gas Control.
- A-2 US NRC Memorandum, K.I. Parczewski to Victor Benaroya, Chief, Chem. Engrg Branch, Div. of Energy, "Radiolysis of Coolant Water in Millstone 1.
- A-3 GPUN Calculation: C1302-240-6340-005, "Radiolysis After LOCA at OCNCS", May 1991.

## APPENDIX A

### OXYGEN VS. TIME CALCULATION METHODOLOGY

#### INTRODUCTION

The ERP (Ref. 1) Section 6.2.5 calculations are used with modifications as noted.

#### DISCUSSION

In the event of a loss-of-coolant accident (LOCA), hydrogen and oxygen gases will be generated within the Oyster Creek reactor containment by:

1. Metal-water reaction involving the zirconium fuel cladding and the reactor coolant, producing free hydrogen.
2. Radiolytic decomposition of the post-accident emergency cooling solutions, producing both oxygen and hydrogen.

If a sufficient amount of hydrogen is generated, it may react with the O<sub>2</sub> present in the containment atmosphere or, in the case of inerted containments, with the oxygen generated following a LOCA.

The extent of zirconium-water reaction and associated hydrogen production depends strongly upon the course of events assumed for the accident. Analytically the reaction can be described by:



$$1 \text{ lb Zr} \rightarrow 0.043956 \text{ lb H}_2$$

$$1 \text{ lb Zr} \rightarrow 0.021978 \text{ lb-mole H}_2$$

Therefore, one pound of reacted zirconium will produce 0.021978 pound-moles of free hydrogen. Assuming the perfect gas relationship, this is equivalent to 8.4866 scf/lb Zr.

$$V = \frac{MRT}{P}$$

$$V = 0.021978(10.71)(530) / 14.7 \text{ (Standard conditions taken as 14.7 psia, 530° R)(70° F)}$$

$$V = 8.4866 \text{ scf/lb Zr}$$

The total amount of hydrogen produced is based on the amount of reacted zirconium. The computer program, to maintain a degree of generality, allows the reaction percentage to be specified as an input quantity. The expression used is:

$$WG = (.022)(WZr)(f_{ZW})$$

where

WG = pound moles of hydrogen generated

WZr = weight of zirconium fuel element clad

$f_{ZW}$  = zirconium-water reaction fraction

The rate of gas production from radiolysis depends upon the power decay profile and the amount of fission products released to the coolant. The radiolytic hydrogen production rate at time (t) is given by:

$$S_H(t) = \frac{P}{(B)(N)} \frac{G_c E_c(t) + G_s E_s(t)}{100} = \frac{P}{(B)(N)} \frac{G(H_2)}{100} (E_c(t) + E_s(t))$$

where

$S_H(t)$  = hydrogen production rate, lb-mole/sec

P = operating reactor power level, MWt

B = conversion factor, 454 gm-mole/lb-mole

N = Avogadro's number,  $6.023 \times 10^{23}$  molecules/gm-mole

$G_c$  = radiolytic hydrogen yield in core, molecules/100 ev

$E_c(t)$  = gamma ray fission product energy absorbed by core coolant, ev/sec-MWt

$G_s$  = radiolytic hydrogen yield in solution, molecules/100 ev

See below for definition of  $G(H_2)$

$E_s(t)$  = energy absorbed in coolant outside core due to fission products dissolved in coolant, ev/sec-MWt

The quantity  $E_s(t)$  is defined by:

$$E_s(t) = (f_k) H_k(t)$$

where

$$(\xi_c) = \text{fraction of fission product gamma energy absorbed by coolant in core region} \\ = 0.1$$

$$H_\gamma(t) = \text{gamma energy production rate, ev/(sec-MWt)}$$

Similarly,  $E_c(t)$  is defined by:

$$E_c(t) = (\xi_{sp})_c H_{\gamma,sp}(t) + f H_I(t)$$

where

$$(\xi_{sp})_c = \text{fraction of total solid fission product energy absorbed in coolant outside core} \\ = 0.01$$

$$H_{\gamma,sp}(t) = \text{total solid fission product energy production rate, ev/sec-MWt}$$

$$f = \text{fraction of iodine isotope energy absorbed in coolant outside core} \\ = 100\% \text{ of the fraction of iodine energy released to the coolant}$$

$$H_I(t) = \text{iodine isotope energy production rate, ev/sec-MWt}$$

The equations for oxygen generation by radiolysis are identical to those above describing hydrogen evolution except that the yield is one-half that of hydrogen. For calculational purposes, the reactor decay profiles ( $H_\gamma(t)$ ,  $H_{\gamma,sp}(t)$ , and  $H_I(t)$ ) specified by the ANS-5.1 standard for two-year reactor operation have been fitted by several finite exponential series expressions and also incorporated into the program. The resulting equations are:

$$H_\gamma(t) = 10^{22} (5.1912e^{-9.8 \times 10^{-5}t} + 0.8743e^{-6.5 \times 10^{-4}t} + 0.6557e^{-5.7 \times 10^{-3}t} + 0.4098e^{-7.4 \times 10^{-2}t} + 0.01850e^{-8.0 \times 10^{-10}t})$$

$$H_{\gamma,sp}(t) = 2.0 H_\gamma(t)$$

$$H_I(t) = 10^{22} (0.8197e^{-6.1 \times 10^{-5}t} + 0.3279e^{-1.21 \times 10^{-3}t} + 0.0574e^{-1.0 \times 10^{-2}t})$$



where

$t$  = time after reactor shutdown, sec.

Between 400 and  $4 \times 10^7$  sec, the equations overpredict the standard curve by 20%. The equations underpredict the standard curve soon after shutdown. However, this does not seriously affect the results due to the short time period involved. The equations are equivalent to the afterheat decay curve in BTP ASB 9-2 over the times of interest for post-accident hydrogen generation. It should also be noted that this formulation overpredicts the radiolytic hydrogen generation by a small amount due to a "double-counting" of the gamma energy of those fission products assumed to be released from the fuel rods.

$G(H_2)$  is taken as

$$\begin{aligned} &= 0.45 \text{ during boiling} \\ &= 0.45 - 2.7 / (1 + k_i * [I] / k_{h2} * [H]) \text{ in non-boiling water} \end{aligned} \quad (A-1)$$

where

$$\begin{aligned} k_i &= 1e10 \text{ l}/(\text{m}^3 \cdot \text{s}) \text{ (liter/mol sec)} \\ k_{h2} &= 3e7 \text{ l}/(\text{m}^3 \cdot \text{s}) \\ [I] &= \text{dissolved iodine, mol/liter} \\ [H] &= \text{dissolved hydrogen, mol/liter} \end{aligned}$$

We assume (as per A. O. Allen, Ref. A-2)

1. Water system consists of suspended  $ZrO_2$  and dissolved iodine in water.
2. Water included in the porous  $ZrO_2$  particles continues to boil.
3. The fraction of water (and decay energy) absorbed in the  $ZrO_2$  is  $f_z$ , and this is the same in both the core and torus ( $f_z = 0.10$ ).

The dissolved hydrogen at any time is calculated from

$$[H] = P_{H_2}/KH \quad (A-2)$$

where

$P_{H_2}$  = partial pressure of hydrogen in the total gas volume, (wetwell + drywell), psia

$KH$  = Henry's Law constant, psia/(mol/l)

A basic computer program performs these calculations. The hydrogen inventory is calculated by step-wise integration of the hydrogen production rate, the  $P_{H_2}$  is calculated assuming a perfect gas in the drywell, the  $[H]$  calculated from Equation A.2 and the G-value for the next time increment is calculated from this  $[H]$  value and A.2 to repeat the cycle.

The percent oxygen at each step is calculated from

$$\% \text{ oxy} = 100\% \text{ mox}/(\text{mox} + \text{mh} + \text{mn})$$

where

$\text{mox}$  = total moles oxygen (original inventory plus 0.5 times the moles hydrogen produced radiolytically)

$\text{mh}$  = total moles hydrogen (radiolytic plus Zr/water reaction moles)

$\text{mn}$  = total moles nitrogen originally present

The radiolytic hydrogen formed during boiling (the first 12 hours LOCA) can be calculated analytically since the G is independent of time, and the decay energy expression integrates to a sum of terms in the form  $B*(1-EXP(-C*t))$  with b and c constant and t=12 hours.

APPENDIX B

SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

GENERAL ELECTRIC COMPANY'S METHODOLOGY FOR DETERMINING RATES

OF GENERATION OF OXYGEN BY

RADIOLYTIC DECOMPOSITION (NEDO-22155)



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

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ENCLOSURE 2

SAFETY EVALUATION BY THE  
OFFICE OF NUCLEAR REACTOR REGULATION  
GENERAL ELECTRIC COMPANY'S METHODOLOGY FOR DETERMINING RATES  
OF GENERATIONS OF OXYGEN BY  
RADIOLYTIC DECOMPOSITION (NEDO-22155)

In June 1982 General Electric (GE) issued the subject report containing a description of the methodology for determining rates of generation of oxygen by radiolytic decomposition of water in the inerted Mark I containments. In this report, GE assumes that after an accident water in the containment will boil for 12 hours only. During this time it will undergo radiolytic decomposition with oxygen generated at the rates corresponding to  $G(O_2)=0.1$ . Where  $G(O_2)$  is a number of molecules of oxygen generated by 100 ev of radiant energy absorbed. This value was based on the results from the measurements of the hydrogen evolution rate in the offgas systems during normal (boiling) operation and during refueling shutdowns and confirmed by the experiments performed in the KRB Nuclear Power Plant.

For radiolysis of water beyond 12 hours, when boiling ceases,  $G(O_2)=0$  was assumed and consequently there was no net generator or radiolytic oxygen. This last assumption was based on the analytical results obtained by Knolls Atomic Power Laboratory (Reference 1) and by Argonne National Laboratory (Reference 2) in connection with the Three Mile Island accident. The values of  $G(O_2)$  in the GE report differ considerably from the value of  $G(O_2)$  in Regulatory Guide 1.7 which for both boiling and non-boiling cases recommends  $G(O_2)=0.25$ . However, this value is not based on any specific mechanism of radiolysis but is chosen to bound all possible cases and consequently it tends to overpredict the rates of generation of radiolytic oxygen. In 1982 an extensive effort was undertaken by the Northeast Utilities and by the NRC in connection with the Millstone 1 licensing action to determine a more realistic method for calculating rates of radiolytic oxygen generation. In performing this task the staff was assisted by a consultant from BNL. The results of this effort: have indicated that  $G(O_2)$  is not a constant parameter but varies with the amount of hydrogen dissolved in water and with the concentrations of certain impurities, most notable among them iodine. Since concentrations of these substances may vary with time and may be different for different accidents, the true value  $G(O_2)$  should be expressed as a function of these variables.

In general, an increase of concentration of hydrogen in water results in a decrease of radiolysis due to promotion of recombination reactions. On the other hand an increase of iodine concentration tends to promote radiolysis by destroying free radicals which are required for the recombination reactions to proceed. The highest rate of oxygen generation is achieved when  $G(O_2)=0.22$ ,

which is the highest theoretical limit for gamma radiation. This occurs when water is completely free of dissolved hydrogen, or when the concentrations of dissolved iodine are extremely high. However, in most cases  $G(O_2)$  will be lower and at certain concentrations of hydrogen and iodine the rates of radiolytic dissociation and recombinations reactions may become equal resulting in  $G(O_2)=0$  and no net generation of radiolytic oxygen. During the boiling regime hydrogen will be stripped by vapor bubbles and it is expected that  $G(O_2)$  will be higher than in non-boiling water.

Quantitative evaluation performed by the staff was based on the model developed by the BNL consultant (Reference 3) and on the experimental data from ORNL (Reference 4). For pure water (no iodine) it was determined experimentally that with no dissolved hydrogen and no boiling  $G(O_2)=0.08$ . However, when under non-boiling conditions the concentration of dissolved hydrogen reached 2.5 cc/kg of water, corresponding to equilibrium hydrogen pressure of 0.16 atm.,  $G(O_2)$  became zero and generation of radiolytic oxygen stops. This finding contradicts the information in the GE report where  $G(O_2)=0$  was assumed for all non-boiling cases.

For water containing dissolved iodine no applicable experimental data were available and the staff calculated  $G(O_2)$  corresponding to the maximum credible iodine concentration in water using the BNL model. Since all iodine in the containment water comes from failed fuel, an accident had to be postulated which would result in a release of this amount of iodine. In such an accident fuel was assumed to fail by oxidation of Zirconium cladding and hence, in addition to released iodine, additional hydrogen was produced. Concentrations of both these substances had to be considered in calculating  $G(O_2)$ .

The accident considered consisted of a LOCA in which 5 percent of fuel cladding was oxidized by reaction with steam producing failure of all fuel rods and overheating of the core, but without initiation of fuel melting. This case represented maximum degradation of core allowed by 10 CFR 50.44(d)(1) and 10 CFR 50.46(b)(3). The analyses performed by Sandia (Reference 5), based on the experimental work on fuel rods from the H. B. Robinson plant, have indicated that for this type of accident 30 percent of total fuel iodine inventory was released. The released iodine consisted of the initial gap inventory and of the iodine diffused from the overheated fuel. Assuming that all the released iodine was dissolved in water and using plant parameters corresponding to a typical BWR with Mark I containment, the iodine concentration in water was determined to be  $1.11 \text{ E-5}$  moles/liter and the partial pressure of hydrogen in the containment 0.12 atm. This partial pressure corresponds to an equilibrium concentration of 1.9 cc hydrogen/kg of water. Inserting this value of iodine concentration into the BNL mathematical model a relationship between  $G(O_2)$  and partial pressure of hydrogen in the containment was developed. From this relationship it was determined that for a non-boiling case, when partial pressure of hydrogen was 0.12 atm.,  $G(O_2)=0.19$ . It also found that  $G(O_2)$  would not reach zero value until partial pressure of hydrogen in the containment reaches 1 atm. For boiling case, when hydrogen is stripped from the solution,  $G(O_2)$  would be slightly higher, somewhere between 0.19 and 0.22.

These values differed considerably from those in the NEDO-22155 report. The main difference was probably due to the GE results being applicable to pure water or to water containing only minimal amount of impurities. Including the effect of iodine, which would be released during certain types of LOCA, could drastically change the results.

#### CONCLUSIONS AND RECOMMENDATIONS

1. The NEDO-22155 report underpredicts generation of radiolytic hydrogen for both boiling and non-boiling cases. This is due to the use of too low values for  $G(O_2)$ .  $G(O_2)=0.1$  for boiling case was based on the measurements made in an environment of zero or low iodine concentrations.  $G(O_2)=0$  for non-boiling case was derived from the data calculated by the codes which did not consider effects of dissolved iodine. The results were also in disagreement with the experimental data from ORNL.
2. Since  $G(O_2)$  is a function of hydrogen and iodine concentrations in the containment water, it may vary during an accident and is specific for each individual plant.
3. The maximum values of  $G(O_2)$ , calculated with the NRC radiolysis model for LOCA (5% metal-water reaction and 30% iodine release) in a BWR with Mark I containment, are  $G(O_2)=0.19$  for non-boiling and between 0.19 and 0.22 for boiling cases. They are considerably higher than the values presented in the General Electric's NEDO-22155 report.
4. The value of  $G(O_2)=.25$  in Regulatory Guide 1.7 is overly conservative. However, it is not very much different from the maximum values calculated for a LOCA using the BNL model. It is recommended, therefore that until a better understanding of post accident radiolytic decomposition of water is developed, this value should be used for predicting generation rates of radiolytic oxygen in the containment.

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