TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

March 5, 1982

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Ms. Adensam:

In the Matter of the Application of) Docket Nos. 50-390
Tennessee Valley Authority) 50-391

Enclosed is a revision to the Watts Bar Nuclear Plant Final Safety Analysis Report section 15.4.1.2. This revision includes a change in the percent hydrogen generation to 1.5 percent.

If you have any questions concerning this matter, please get in touch with D. P. Ormsby at FTS 858-2682.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Regulation and Safety

Sworn to and subscribed before me
this 5th day of March 1982

Bryant M. Lowery
Notary Public
My Commission Expires 4/4/82

Enclosure
cc: U.S. Nuclear Regulatory Commission
Region II
Attn: Mr. James P. O'Reilly, Regional Administrator
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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15.4.1.2 Hydrogen Production and Accumulation

Hydrogen accumulation in the containment atmosphere following the Design Basis Accident can be the result of production from several sources. The potential sources of hydrogen are the zironium-water reaction, corrosion of construction materials, and radiolytic decomposition of the emergency core cooling solution. The latter source, solution radiolysis, includes both core solution radiolysis and sump solution radiolysis.

15.4.1.2.1 Method of Analysis

The quantity of zirconium which reacts with the core cooling solution depends on the performance of the Emergency Core Cooling System. The criteria for evaluation of the Emergency Core Cooling System requires that the zircaloy-water reaction be limited to one percent by weight of the total quantity of zirconium in the core. Emergency Core Cooling System calculations have shown the zircaloy-water reaction to be less than 0.1 percent, much less than required by the criteria.

The use of aluminum inside the containment is limited and is not used in safety-related components which are in contact with the recirculating core cooling fluid. Aluminum is more reactive with the containment spray alkaline borate solution than other plant materials such as galvanized steel, copper, and copper nickel alloys. By limiting the use of aluminum, the aggregate source of hydrogen over the long term is essentially restricted to that arising from radiolytic decomposition of core and sump water. The upper limit rate of such decomposition can be predicted with ample certainty to permit the design of effective countermeasures.

It should be noted that the zirconium-water reaction and aluminum corrosion with containment spray are chemical reactions and thus essentially independent of the radiation field inside the containment following a Loss-of-Coolant Accident (LOCA). Radiolytic decomposition of water is dependent on the radiation field intensity. The radiation field inside the containment is calculated for the maximum credible accident in which the fission product activities given in T1D-14844(1) are used.

The hydrogen generation calculations are performed; one using the Westinghouse model discussed below and the other using the NRC Branch Technical Position CSB6-2.

15.4.1.2.2 Typical Assumptions

The following discussion outlines the assumptions used in the calculations.

1. Zirconium-Water Reaction

The zirconium-water reaction is described by the chemical equation:

$$Zr + 2H_2O = ZrO_2 + 2H_2 + \text{Heat}$$

The hydrogen generation due to this reaction will be completed during the first day following the LOCA. The Westinghouse model assumes a 0.5- or 1.5-percent zirconium-water reaction. The NRC model assumes a 15-percent zirconium-water reaction or a core-wide average depth of reaction into the original cladding of 0.00023 inches of clad thickness. The hydrogen generated is assumed to be released to the containment atmosphere over the first two minutes following the break in both models.

2. Primary Coolant Hydrogen

The maximum equilibrium quantity of hydrogen in the primary coolant is 1120 scf. This value includes both hydrogen dissolved in the coolant water at 35 cc (STP) per kilogram of water and the corresponding equilibrium hydrogen in the pressurizer gas space. The 1120 scf of hydrogen is assumed to be released immediately and uniformly to the containment atmosphere.

3. Corrosion of Plant Materials

Oxidation of metals in aqueous solution results in the generation of hydrogen gas as one of the corrosion products. Extensive corrosion testing has been conducted to determine the behavior of the various metals used in the containment in the emergency core cooling solution at Design Basis Accident conditions. Metals tested include zircaloy, inconel, aluminum alloys, cupronickel alloys, carbon steel, galvanized carbon steel, and copper. Tests conducted at ORNL[3,4] have also verified the compatibility of the various materials (exclusive of aluminum) with alkaline borate solution. As applied to the quantitative definition of hydrogen production rates, the results of the corrosion tests have shown that only aluminum and zinc will corrode at a rate that will significantly add to the hydrogen accumulation in the containment atmosphere.

The corrosion of aluminum may be described by the overall reaction:

$$2\,Al + 3\,H_2O = Al_2O_3 + 3\,H_2$$

Therefore, three moles of hydrogen are produced for every two moles of aluminum that is oxidized. (Approximately 20 standard cubic feet of hydrogen for each pound of aluminum corroded.)

The corrosion of zinc may be described by the overall reaction:

$$ZN + 2H_2O = ZN(OH)_2 + H_2$$

Therefore, one mole of hydrogen is produced for each mole of zinc oxidized. This corresponds to 5.5 scf hydrogen produced for each pound of zinc corroded.

The time-temperature cycle (Table 15.4-2) considered in the calculation of aluminum and zinc corrosion is based on a conservative step-wise representation of the postulated post-accident containment transient. The corrosion rates at the various steps are determined from the aluminum and zinc corrosion rate design curves shown in Figures 15.4-1 and 15.4-1a. The corrosion data points include the effects of temperature, alloy, and spray solution conditions. Based on these corrosion rates and corrodable metal inventory given in Table 15.4-3, the contribution of aluminum and zinc corrosion to hydrogen accumulation in the containment following the Design Basis Accident has been calculated. For conservative estimation, no credit is taken for protective shield effects of insulation or enclosures from the spray and complete and continuous immersion is assumed.

Calculations based on the NRC model are performed by allowing an increased aluminum corrosion rate during the final step of the post-accident containment temperature transient (Table 15.4-2) corresponding to 200 mils (15.7 $mg/dm^2/hr$). The corrosion rates earlier in the accident sequence are the higher rates determined from Figure 15.4-1.

4. Radiolysis of Core and Sump Water

Water radiolysis is a complex process involving reactions of numerous intermediates. However, the overall radiolytic process may be described by the reaction:

$$2\ H_2O = 2H_2 + O_2$$

Of interest here is the quantitative definition of the rates and extent of radiolytic hydrogen production following the Design Basis Accident.

An extensive program has been conducted by Westinghouse to investigate the radiolytic decomposition of the core cooling solution following the Design Basis Accident. In the course of this investigation, it became apparent that two separate radiolytic environments exist in the containment at Design Basis Accident conditions. In one case, radiolysis of the core cooling solution occurs as a result of the decay energy of fission products in the fuel. In the other case, the decay of dissolved fission products, which have escaped from the core, results in the radiolysis of the sump solution. The results of these investigations are discussed in Reference (5).

15.4.1.2.3 Core Solution Radiolysis

As the emergency core cooling solution flows through the core, it is subjected to gamma radiation by decay of fission products in the fuel. This energy deposition results in solution radiolysis and the production of molecular hydrogen and oxygen. The initial production rate of these species will depend on the rate of energy absorption and the specific radiolytic yields.

The energy absorption rate in solution can be assessed from knowledge of the fission products contained in the core, and a detailed analysis of the dissipation of the decay energy between core materials and the solution. The results of Westinghouse studies show essentially all of the beta energy is absorbed within the fuel and cladding and that this represents approximately 50 percent of the total beta-gamma decay energy. This study shows further that of the gamma energy, a maximum of 7.4 percent will be absorbed by the solution incore. Thus, an overall absorption factor of 3.7 percent of the total core decay energy ($\beta + \gamma$) is used to compute solution radiation dose rates and the time-integrated dose. Table 15.4-4 presents the total decay energy ($\beta + \gamma$) of a reactor core, which assumes a full power operating time of 650 days before the accident. For the maximum credible accident case, the contained decay energy in the core accounts for the assumed T1D-14844 release of 50-percent halogens and 1-percent other fission products. To be conservative, the noble gases have been assumed to retain in the core, whereas in reality, the noble gases are assumed by the T1D-14844 model to escape to the containment vapor space where little or no water radiolysis would result from decay of these nuclides.

The total decay energy of the reactor core which is used to evaluate post-LOCA hydrogen production and accumulation has been compared to a decay energy curve based on ANS Standard 5.1-1979 (5A). For this comparison, the values given in ANS 5.1 for decay energy release rate at infinite irradiation time were adjusted to a 650-day irradiation time. These resultant values were then multiplied by a factor of 1.2. The results of this comparison are shown in Figure 15.4-1b. The curve presented here as the Westinghouse decay energy curve is used exclusively for post-LOCA hydrogen production calculations.

The radiolysis yield of hydrogen in solution has been studied extensively by Westinghouse and ORNL. The results of static capsule tests conducted by Westinghouse indicate that hydrogen yields much lower than the maximum of 0.44 molecules per 100 ev would be the case incore. With little gas space to which the hydrogen formed in solution can escape, the rapid back reactions of molecular radiolytic products in solution to reform water is sufficient to result in very low net hydrogen yields.

However, it is recognized that there are differences between the static capsule tests and the dynamic condition incore, where the core cooling fluid is continuously flowing. Such flow is reasoned to disturb the steady-state conditions which are observed in static capsule tests, and while the occurrence of back reactions would still be significant, the overall net yield of hydrogen would be somewhat higher in the flowing system.

The study of radiolysis in dynamic systems was initiated by Westinghouse, which formed the basis for experimental work performed at ORNL. Both studies clearly illustrate the reduced yields in hydrogen from core radiolysis; i.e., reduced from the maximum yield of 0.44 molecules per 100 ev. These results have been published. (5,6)

For the purposes of this analysis, the calculations of hydrogen yield from core radiolysis are performed with the very conservative value of 0.44 molecules per 100 ev. That this value is conservative and a maximum for this type of aqueous solution and gamma radiation is confirmed by many published works. The Westinghouse results from the dynamic studies show 0.44 to be a maximum at very high solution flow rates through the gamma radiation field. The referenced ORNL (6) work also confirms this value as a maximum at high flow rates. A. O. Allen (7) presents a very comprehensive review of work performed to confirm the primary hydrogen yield to be a maximum of 0.44 - 0.45 molecules per 100 ev.

On the foregoing basis, the production rate and total hydrogen produced from core radiolysis, as a function of time, has been conservatively estimated for the maximum credible accident case.

Calculations based on the NRC model assume a hydrogen yield value of 0.5 molecules per 100 ev, 10 percent of the gamma energy produced from fission products in the fuel rods is absorbed by the solution in the region of the core, and the noble gases escape to the containment vapor space.

15.4.1.2.4 Sump Solution Radiolysis

Another potential source of hydrogen assumed for the post-accident period arises from water contained in the reactor containment sump being subjected to radiolytic decomposition by fission products. In this consideration, an assessment must be made as to the decay energy deposited in the solution and the radiolytic hydrogen yield, much in the same manner as given above for core radiolysis.

The energy deposited in solution is computed using the following basis:

1. For the maximum credible accident, a T1D-14844 release model (1) is assumed where 50 percent of the total core halogens and 1 percent of all other fission products, excluding noble gases, are released from the core to the sump solution.

2. The quantity of fission product release is equal to that from a reactor operating at full power for 650 days before the accident.

3. The total decay energy from the released fission products, both beta and gamma, is assumed to be fully absorbed in the solution.

Within the assessment of energy release by fission products in water, account is made of the decay of halogens, and a separate accounting for the slower decay of the 1 percent other fission products. To arrive at the energy deposit rate and time-integrated energy deposited, the contribution from each individual fission product class was computed. The overall contributions from each of the two classes of fission products is shown in Table 15.4-5.

The yield of hydrogen from sump solution radiolysis is most nearly represented by the static capsule tests performed by Westinghouse and ORNL with the alkaline sodium borate solution. The differences between these tests and the actual conditions for the sump solution, however, are important and render the capsule tests conservative in their predictions of radiolytic hydrogen yields.

In this assessment, the sump solution will have considerable depth, which inhibits the ready diffusion of hydrogen from solution, as compared to the case with shallow-depth capsule tests. This retention of hydrogen in solution will have a significant effect in reducing the hydrogen yields to the containment atmosphere. The buildup of hydrogen concentration in solution will enhance the back reaction to formation of water and lower the net hydrogen yield, in the same manner as a reduction in gas to liquid volume ratio will reduce the yield.

This is illustrated by the data presented in Figure 15.4-2 for capsule tests with various gas to liquid volume ratios. The data show a significant reduction in the apparent or net hydrogen yield from the published primary maximum yield of 0.44 molecules per 100 ev. Even at the very highest ratios, where capsule solution depths are very low, the yield is less than 0.30, with the highest scatter data point at 0.39 molecules per 100 ev.

With these considerations taken into account, a reduced hydrogen yield is a reasonable assumption to make for the case of sump radiolysis. While it can be expected that the yield will be on the order of 0.1 or less, a conservative value of 0.30 molecules per 100 ev has been used in the maximum credible accident case.

Calculations based on the NRC model do not take credit for a reduced hydrogen yield in the case of sump radiolysis and a hydrogen yield value of 0.5 molecules per 100 ev has been used.

15.4.1.2.5 Results

Figures 15.4-3 through 15.4-6 show the hydrogen production and accumulation in the containment following a LOCA Accident for both the Westinghouse and NRC models, while Figures 15.4-7 and 15.4-8 give the volume percent of hydrogen in the containment for each of the models. The figures for hydrogen accumulation and volume percent in the containment are based on the assumption that no measures are taken to remove the hydrogen (i.e., no recombination on purging of the hydrogen is taken into account). The effect of the hydrogen recombiner system on hydrogen accumulation is discussed in Chapter 6, while the effect of hydrogen purging to atmosphere is discussed in Section 15.5.

15.4.1.2 References

1. J. J. DiNunno, F. D. Anderson, R. E. Baker, and R. L. Waterfield, 'Calculation of Distance Factors for Power and Test Reactor Sites,' T1D-14844, March 1962.

2. Branch Technical Position CSB 6-2, 'Control of Combustible Gas Concentrations in Containment Following a Loss-Of-Coolant Accident.'

3. W. B. Cottrell, 'ORNL Nuclear Safety Research and Development Program Bi-Monthly Report for July - August 1968,' ORNL-TM-2368, November 1968.

4. W. B. Cottrell, 'ORNL Nuclear Safety Research and Development Program Bi-Monthly Report for September - October 1968,' ORNL-TM-2425, p. 53, January 1969.

5. W. D. Fletcher, M. J. Bell, and L. F. Picone, 'Post-LOCA Hydrogen Generation in PWR Containments,' <u>Nucl. Technol. 10,</u> 420-427, (1971).

5A. American Nuclear Society Standard ANSI/ANS-5.1-1979 'Decay Heat Power in Light-Water Reactors,' August 29, 1979.

6. H. E. Zittel and T. H. Row, 'Radiation and Thermal Stability of Spray Solutions,' <u>Nucl. Technol. 10,</u> 436-443, (1971).

7. A. O. Allen, 'The Radiation Chemistry of Water and Aqueous Solutions,' Princeton, N. J., Van Nostrand, 1961.

TABLE 15.4-3
(Sheet 1)

PARAMETERS USED TO DETERMINE HYDROGEN GENERATION

Power Level	356 Mwt
Containment Free Volume	1,230,000 ft^3
Containment Temperature at Accident	120 °F
Weight Zirconium Cladding	45,232
Hydrogen Generated Zirconium-Water Reaction	
Based on 1.5% Value	5,360 SCF
Based on 0.23 Mil	3,653 SCF
Hydrogen from Primary Coolant System	1,120 SCF
Corrodable Metal	Aluminum, Zinc

INVENTORY OF ALUMINUM IN CONTAINMENT
(NUCLEAR STEAM SUPPLY SYSTEM)

Item	Weight (lbs)	Surface Area (ft^2)
Source, Intermediate and Power Range Detectors (AL)	244	83
Control Rod Drive Mechanism Connectors (AL)	218	69
Rod Position Indicator (AL)	151	81
Flux Map Drive System (AL)	122	84
Air Handling Units (AL)	46	19
Miscellaneous Valves (AL)	230	86
Contingency (Nuclear Steam Supply System) (AL)	250	85

TABLE 15.4-3
(Sheet 2)

INVENTORY OF ZINC IN CONTAINMENT

Item	Weight (lbs)	Surface Area (ft^2)
Ice Condenser Components (Galvanized) (ZN)	51,600	206,400
Ice Condenser Wall Panels (ZN)	737	29,500
Cable Trays, Conduit (Galvanized) (ZN)	4110	55,000
Painted Surfaces (ZN)	2750	45,000

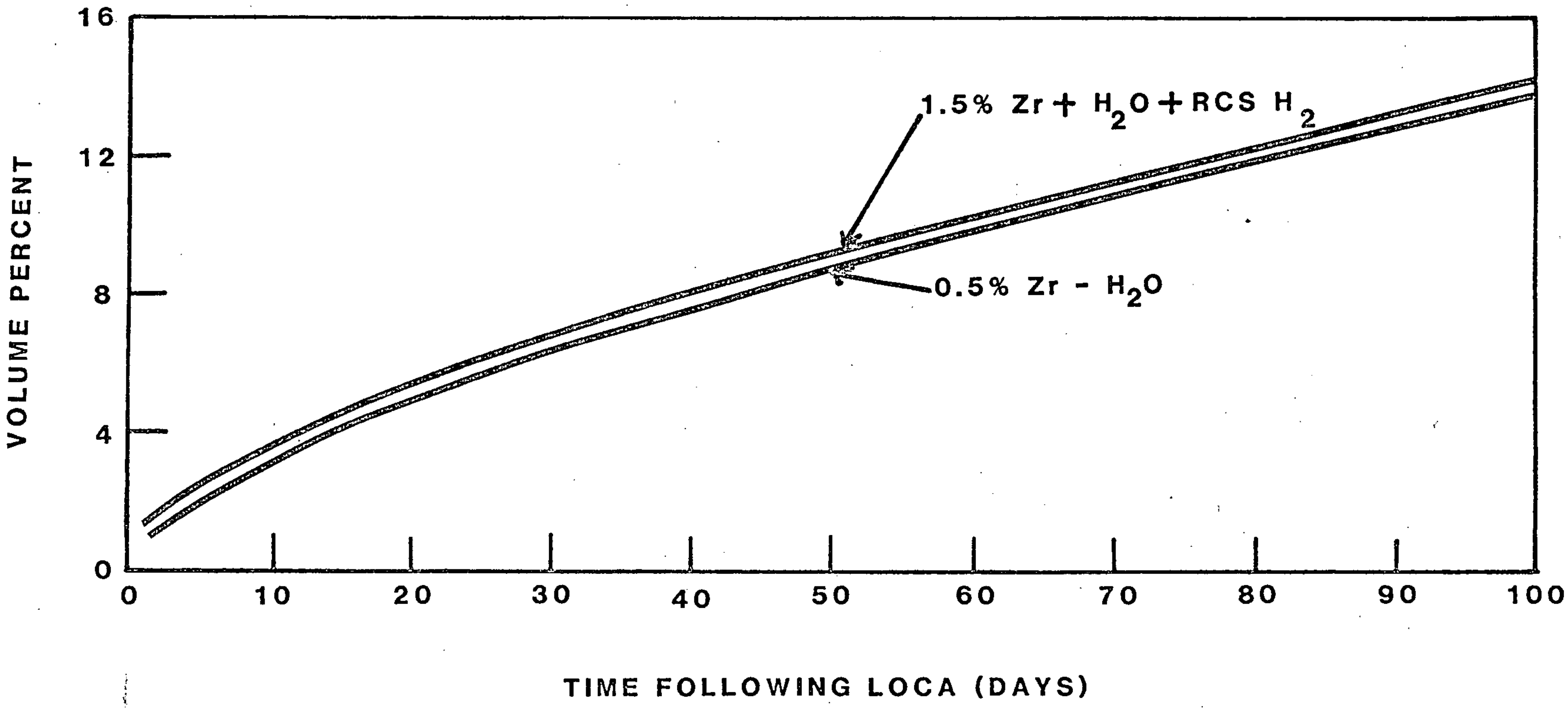

FIGURE 15.4-7 Volume Percent of Hydrogen in Containment - Westinghouse Model

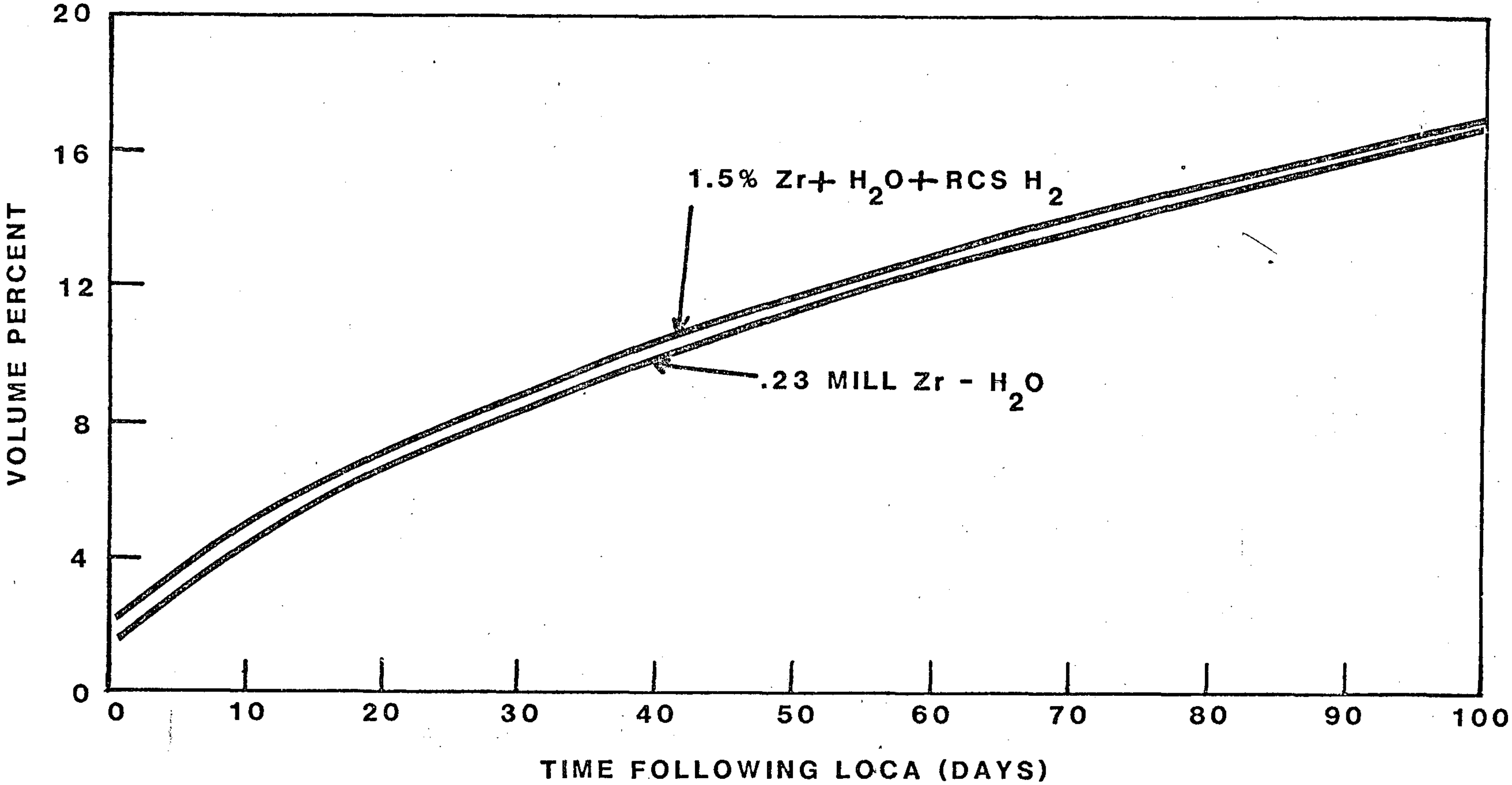

FIGURE 15.4-8 VOLUME PERCENT OF HYDROGEN IN CONTAINMENT - NRC MODEL