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A045

CORE DAMAGE ESTIMATE I

(Primary System Breach Inside Containment)

NOTE: It is important to quickly provide a status of the present situation and a prognosis on whether the situation is expected to degrade, improve, or remain the same, (i.e., within 5 to 10 minutes of a change in plant status).

1.0 INDICATORS USED

1.1 Containment Radiation

Use Attachment 1, A, B, or C, as applicable, to determine the amount and type of fuel damage using containment radiation monitors. These figures were taken from the US NRC Response Technical Manual, RTM-96. Obtain the containment radiation levels from SPDS or the Control Room indicators.

NOTE (1): Correction for the pre-release background radiation levels may be required as listed below.

Gap or In-Vessel Melt - The background radiation monitor value is normally low (≤ 4 R/hr) relative to 1% gap or in-vessel melt release. Consequently, the monitor reading does not require correction for background level in determining the type and amount of fuel damage. If the background radiation monitor reading is > 4 R/hr, the monitor reading should be corrected for the background level in determining the type and amount of fuel damage.

Spiked or Normal Coolant - The radiation monitor value requires correction for the background level. Correct the monitor reading to account for the normal background level in determining the type and amount of fuel damage.

NOTE (2): Containment radiation will go up if there is fuel damage. The increase will depend on the type of fuel damage, and whether or not there was a LOCA, Drywell and/or Wetwell sprays were used, and the amount of blowdown from the Reactor Vessel to the Suppression Pool.

In the case of a LOCA, the fuel damage estimate depends strongly on whether or not containment sprays are being used. Special care should be taken to confirm the operation of containment sprays.

1.2 Containment Hydrogen

Use Attachment 2, taken from the US NRC Response Technical Manual RTM-96, to determine the amount and type of fuel damage using Hydrogen Concentration. Obtain the containment Hydrogen levels from SPDS or the Control Room indicators.

NOTE: Containment Hydrogen will increase if there is a LOCA inside the containment and significant fuel damage.

1.3 Coolant Fission Product Concentration vs. Core Damage

Coolant sampling will indicate the amount of fuel damage, but in most cases, will take too long for use in dose projections. If PASS sample data becomes available, the Nuclear Fuels Engineer is responsible for assuring a fuel damage calculation based on the measured fission product inventories is performed. The results of this analysis should be compared to previous calculations using other methods.

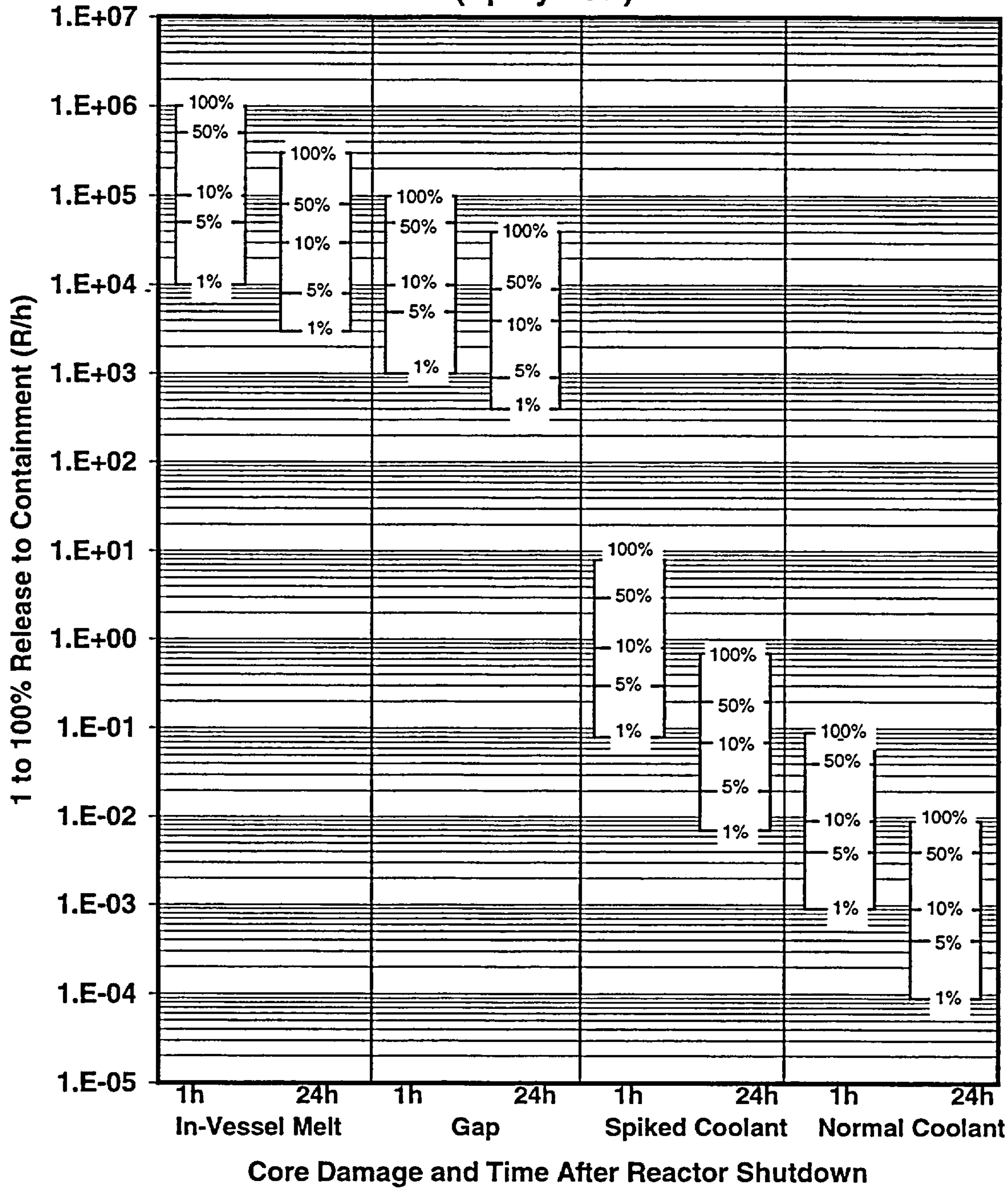
1.4 Plant Transient Precipitating Fuel Damage

If the core experienced a loss of coolant accident and is not covered within 15 minutes, refer to Attachment 3 taken from the US NRC Response Technical Manual RTM-96. The amount of time the core was uncovered can be determined using SPDS. Using the attached figures will provide an estimate of potential fuel damage. Coolant samples must be taken to accurately assess fuel damage.

The type of transient experienced by the reactor leading to fuel damage can be an indicator of the amount and type of fission products released.

- If the core experienced an overpower/pressure transient, a gap release may have occurred.
- If the core experienced a mechanical failure, which could produce flow blockage, there may be localized fuel melt.
- If the core experienced a mechanical perturbation, such as a seismic event or a large steam line break causing a large delta pressure across the core, a gap release could result.
- If the Reactor failed to shut down (ATWS) with a subsequent loss of cooling, there may be fuel melt.

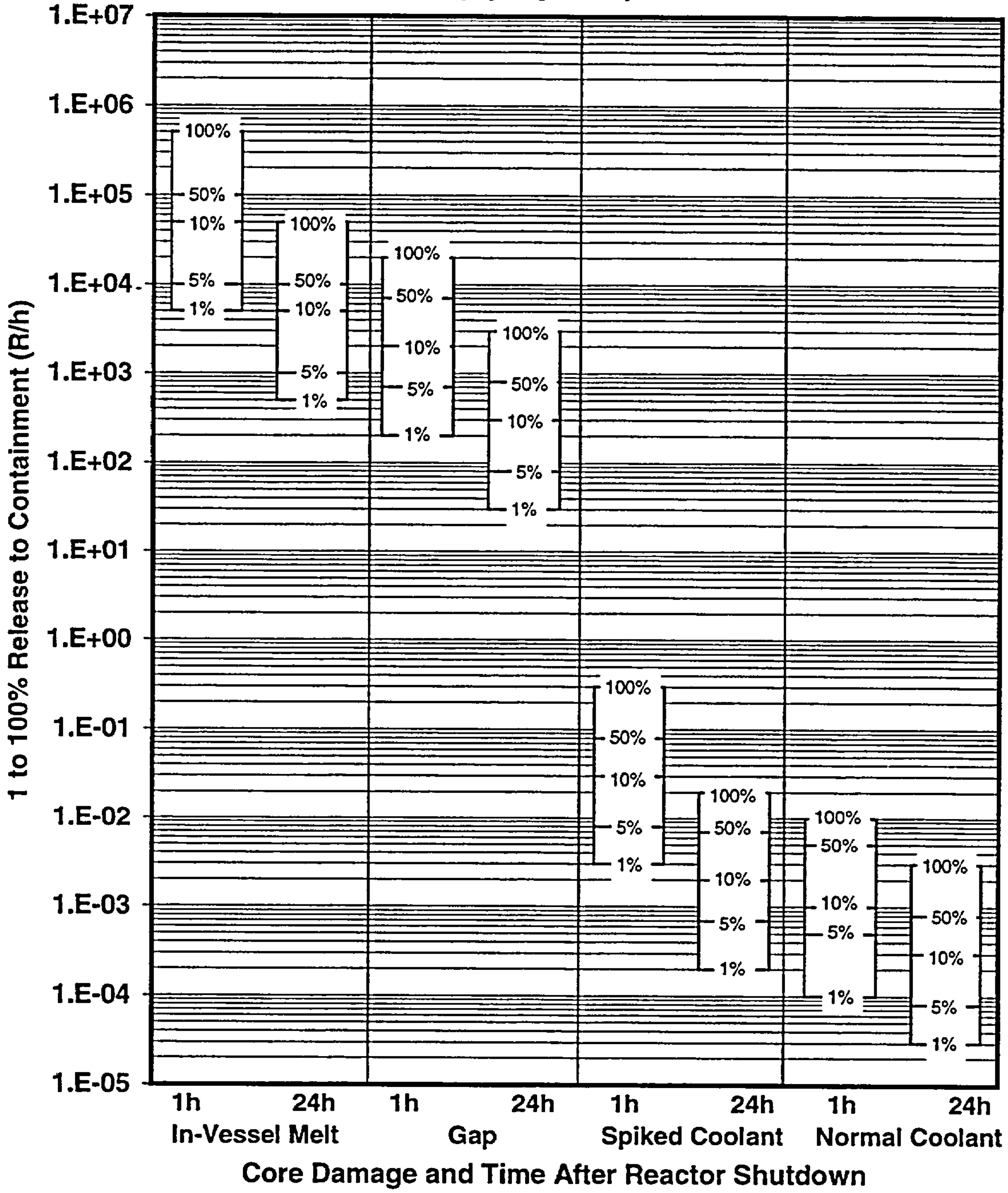
Containment Radiation Monitor Response
<u>Direct Release Path to Dry well</u>
(Sprays Off)



Note 1: This figure should be used only when there is a primary system breach inside containment and a direct release path to the Drywell.
Note 2: See Attachment 3 to determine if fuel melt occurred (core uncovered or fuel blockage).

<u>ATTACHMENT 1A</u>

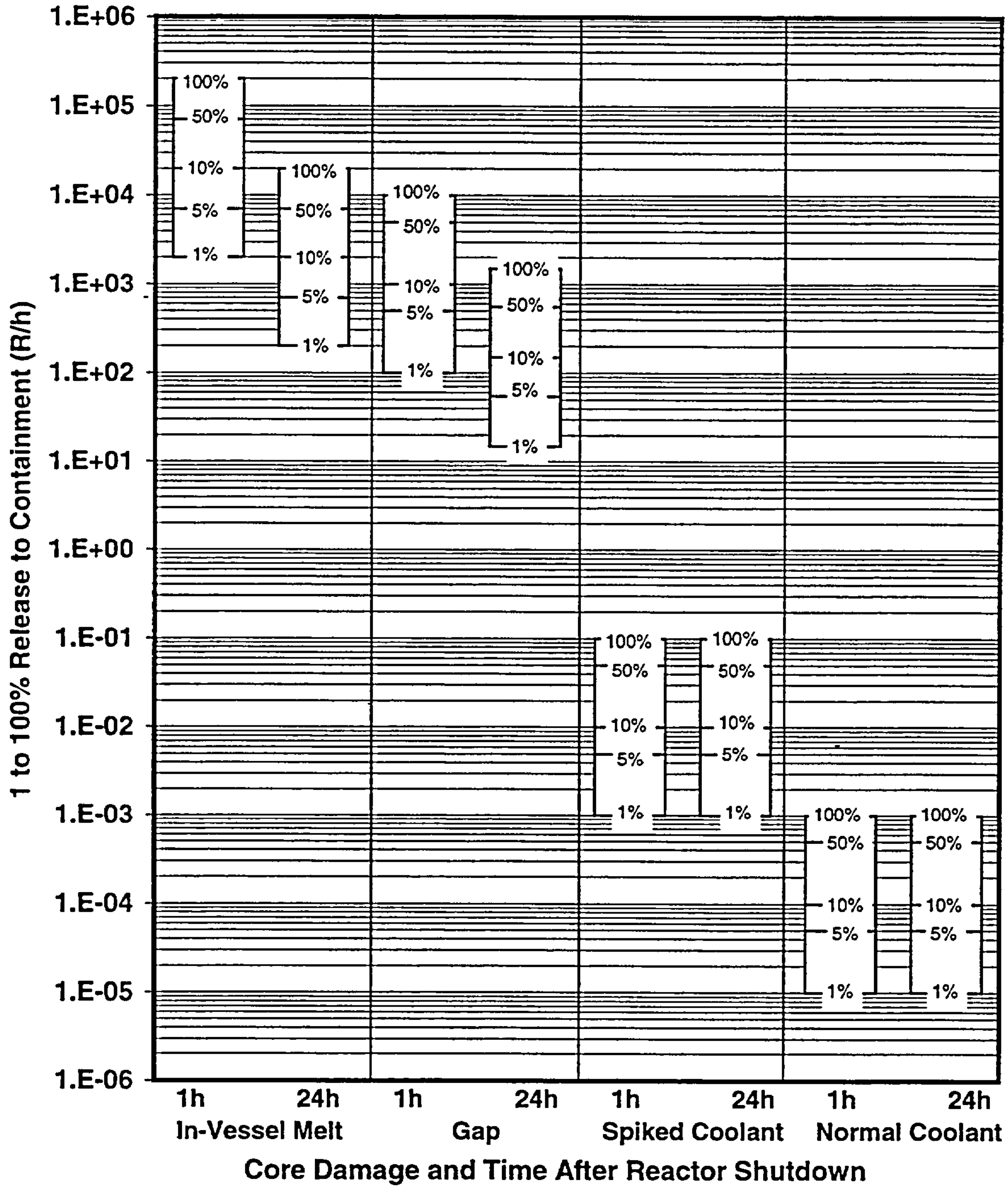
Containment Radiation Monitor Response
Direct Release Path to Dry well
(Sprays On)



Note 1: This figure should be used only when there is a primary system breach inside containment and a direct release path to the Drywell.
Note 2: See Attachment 3 to determine if fuel melt occurred (core uncovered or fuel blockage).

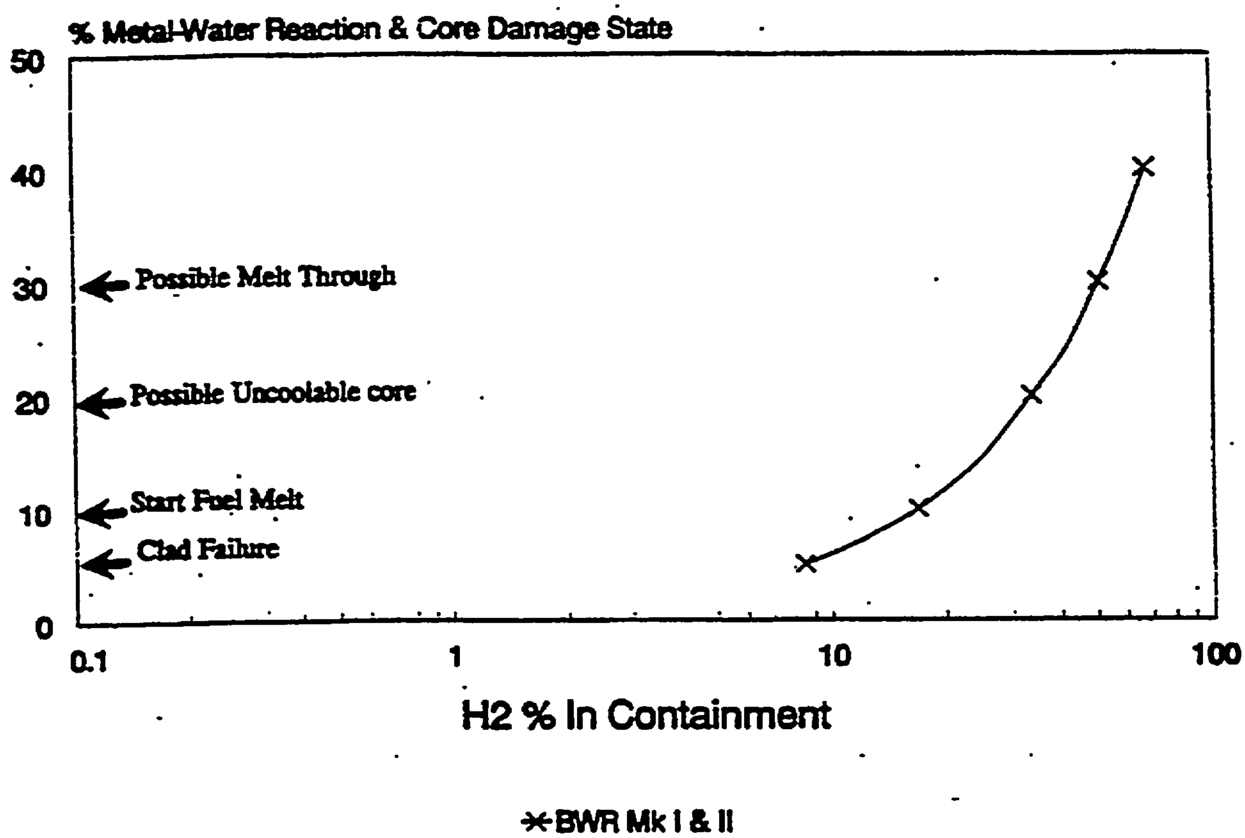
ATTACHMENT 1B

Containment Radiation Monitor Response
Direct Release to Wetwell and Not to Drywell



Note 1: This figure should be used only when there is a primary system breach inside containment and a direct release path to the Wetwell without a primary release to the Drywell.
Note 2: See Attachment 3 to determine if fuel melt occurred (core uncovered or fuel blockage).

CONTAINMENT HYDROGEN VS CORE DAMAGE



Sources: NUREG/CR-2726, p. 4-3; damage states, NUREG-4524, Vol. 5.; TMI percentage, NUREG-1370; NUREG/CR-4041; NUREG/CR-5567, Table 4.9, p. 71, confirms "dry" volume.

ATTACHMENT 2

WATER INJECTION REQUIRED TO COOL CORE BY BOILING

CAUTION:

These rates are those required to remove decay heat from a 3000 MW(t) plant by boiling. If there is a break requiring make up or injected water, more water than indicated will be required to both keep the core covered and cooled.

CAUTION:

If the core has been uncovered, the fuel temperature will have increased significantly. Additional flow will be required to accommodate the heat transfer necessary to return to equilibrium fuel temperature.

NOTE:

These curves are based on a 3000 MW(t) plant operated at a constant power for an infinite period and then shutdown instantaneously. The decay heat power is based on ANS-5.1/N18.6. Assuming the injected water is at 80° F, these curves are within 5% for pressures between 14 psia to 2500 psia. These curves are within 20% for injected water temperatures up to 212°F.

ATTACHMENT 3
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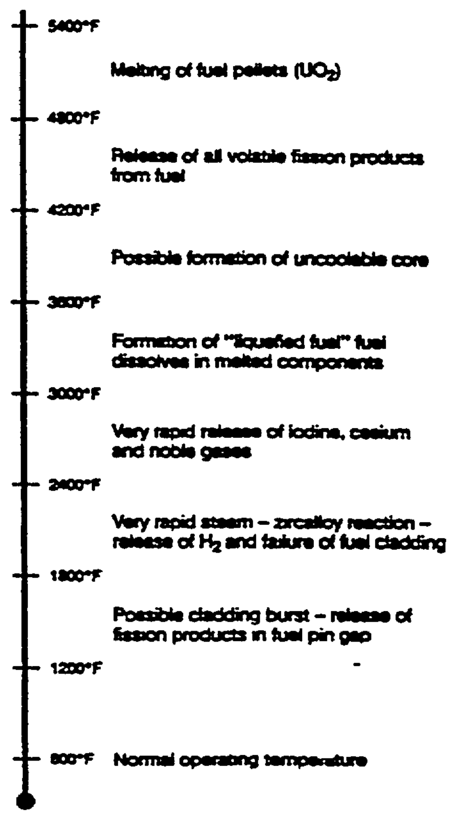
WATER INJECTION REQUIRED TO COOL CORE BY BOILING

While the top of the active core is uncovered, assume that the fuel will heat up at 1-2°F/sec. The increased core temperature will result in fuel pin damage as shown below.

Figure B-1

NOTE:

These estimates are reasonable (factor of 2) if the core is uncovered within a few hours of shutdown (including failure to scram). If there is sufficient injection, core heatup may be stopped or slowed due to steam cooling. Steam cooling may not prevent core damage under accident conditions.



Source: NUREG-0900, NUREG/CR-4524, NUREG-0956

ATTACHMENT 3
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CAUTION: **If the core is severely damaged, it may not be in a coolable state even if covered again with water.**

NOTE: If there is sufficient injection, core heatup may be stopped or slowed due to steam cooling. Steam cooling may not prevent core damage under accident conditions.

WATER INJECTION REQUIRED TO COOL CORE BY BOILING

Figure B-3

INJECTION (gpm) REQUIRED TO REPLACE WATER LOST BY BOILING DUE TO DECAY HEAT FOR A 3000 MW(t) PLANT (1/2-24 HOURS AFTER SHUTDOWN)

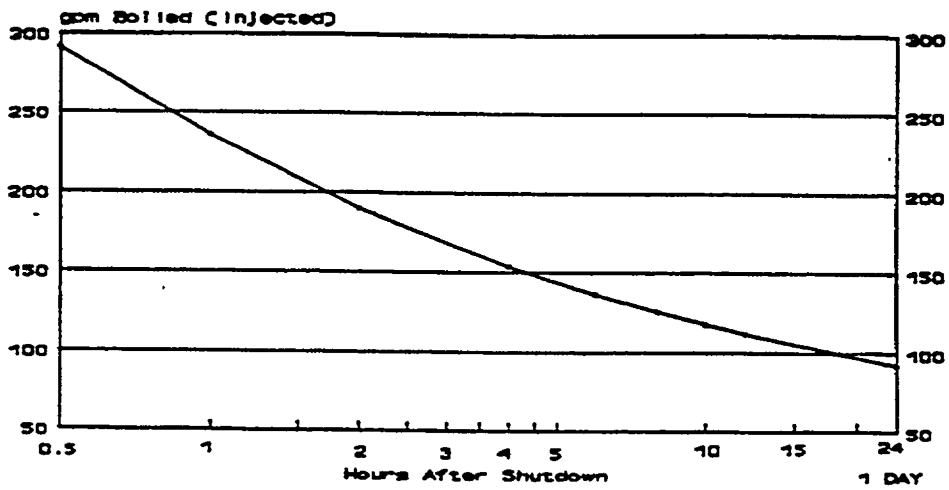
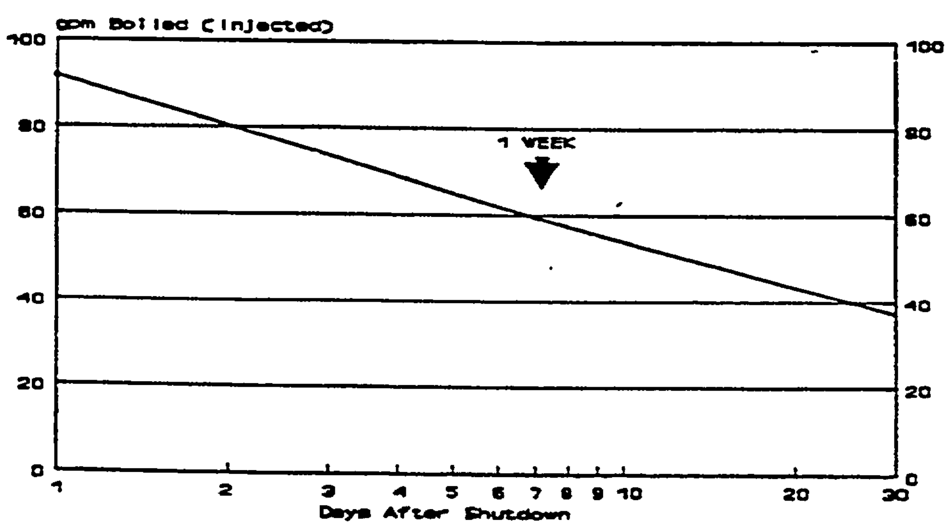


Figure B-4

INJECTION (gpm) REQUIRED TO REPLACE WATER LOST BY BOILING DUE TO DECAY HEAT FOR A 3000 MW(t) PLANT (1 to 30 DAYS AFTER SHUTDOWN)



ATTACHMENT 3
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WATER INJECTION REQUIRED TO COOL CORE BY BOILING

Core damage vs. time that reactor core is uncovered

Time PWR or 20% of BWR active core is uncovered (h)	Core temperature (°F)	(°C)	Possible core damage
0	>600	>315	• None
0.5 to 0.75	1800–2400	980–1300	• Local fuel melting • Burning of cladding with steam production (exothermic Zr-H_2O reaction with rapid H_2 generation) • Rapid fuel cladding failure (gap release from the core[a])
0.5 to 1.5	2400–4200	1300–2300	• Rapid release of volatile fission products (in-vessel severe core damage release from core[a]) • Possible relocation (slump) of molten core • Possible uncoolable core
1 to 3+	>4200	>2300	• Melt-through of vessel with possible containment failure and release of additional less-volatile fission products

Sources: NUREG/CR-4245, NUREG/CR-4624, NUREG/CR-4629, NUREG/CR-5374, NUREG-0900, NUREG-0956, NUREG-1150, and NUREG-1465.

ATTACHMENT 3
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