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SEVERE-ACCIDENT-SEQUENCE ASSESSMENT OF HYPOTHETICAL COMPLETE-STATION BLACKOUT AT THE BROWNS FERRY NUCLEAR PLANT*

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DISCLAIMER

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ABSTRACT

An investigation has been made of various accident sequences which may occur following a complete loss of offsite and onsite ac power at a Boiling Water Reactor nuclear power plant. The investigation was performed for the Browns Ferry Nuclear Power Plant, and all accident sequences resulted in a hypothetical core meltdown. Detailed calculations were performed with the MARCH computer code containing a decay power calculation which was modified to include the actinides. This change has resulted in shortening the time before core uncovery by ~18%, and reducing the time before the start of core melting by ~26%.

Following the hypothetical core meltdown accident, the drywell electric penetration assembly seals have been identified as the most likely leak pathway outside the containment. This potential mode of containment failure occurs at a pressure ~30% lower than that analyzed in the Reactor Safety Study.

SUMMARY

This paper describes the calculated responses of the Browns Ferry BWR4/Mark I Nuclear Power Plant to a hypothetical core meltdown accident initiated by a complete loss of offsite and onsite ac power. Alternate accident sequences resulting from additional failure of the high-pressure coolant injection (HPCI) and the reactor core isolation coolant (RCIC) systems and with operator's manual control of RCIC and the automatic depressurization system (ADS) have also been investigated.

Following the loss of offsite and onsite ac power, a full load rejection (i.e., fast closure of the turbine control valves) occurs. This is followed by a trip of the primary recirculation pumps and the condenser

cooling water pumps. Thereafter, the core flow is provided by natural circulation, and steam is released through the safety/relief valves into the pressure suppression pool.

When the wide range core level detector reaches the low water level setpoint (Level 2), the dc-powered HPCI and RCIC systems are automatically turned on, where the HPCI and RCIC turbine pumps are driven by steam generated by the decay heat. The RCIC system does not restart after the first cycle, while the HPCI system continues to turn on and off automatically between sensed levels 2 and 8 until the batteries run out of power, which occurs at $\sim$4 hr into the transient. Approximately 1 hr later, the core becomes uncovered, with the beginning of core melt occurring an hour later.

Five alternate accident sequences have been examined.

1. Complete Station Blackout (CSB) + No HPCI/RCIC. This sequence occurs when the HPCI and RCIC systems are not available due to closed valves or failed batteries.

2. CSB + SORV + No HPCI/RCIC. This sequence occurs when there is a stuck-open relief valve (SORV) and the injection systems are not available.

3. CSB + HPCI/RCIC. This sequence occurs when the 250 V dc-power is available and the injection systems operate as designed.

4. CSB + SORV + HPCI/RCIC. This sequence is the same as in (3) except there is a stuck-open relief valve.

5. CSB + RCIC Only + 1 ADS. This sequence occurs when the operator manually controls the RCIC system to maintain a constant water level inside the vessel until the batteries run out after 4 hr. An ADS valve is opened after 15 min to lower the reactor pressure and temperature.

Detailed calculations for the five postulated accident sequences have been performed with the MARCH computer code[1] containing a modified decay power calculation to include the actinides. In the MARCH code, the fission product decay heat source term is based on ANS Standard ANS-5.1 (1973),[2] and the decay of U-239 and Np-239 are not accounted for. The new standard ANS-5.1 (1979)[3] provides only decay parameters for U-239 and Np-239, and does not account for the decay power generated from these or the other numerous heavy nuclides (referred to as actinides) that exist in power reactors. In the modified version of the MARCH code, where the actinide decay power for the BWRs[4] is used as the source term, this change resulted in shortening of the core uncovery and core melt times by approximately 18 and 26%, respectively. These results produce better agreement with the RELAP4/MOD7 calculations.[5]

Following the hypothetical core meltdown accident, the drywell electric penetration assembly seals have been identified as the most likely leak pathway outside of the containment. The electric penetration seals have been qualified for environmental conditions of 163°C (325°F) for 15 min followed by 138°C (281°F) for 3 hr at a pressure of 125 psig (9.64×10^5 Pa). At temperatures above 163°C (325°F), the penetration materials lose their mechanical sealing integrity, and shortly thereafter drywell venting occurs, releasing radioactive fission products outside the containment. This venting occurs at a leak rate of $\sim$250 ft^3/min (118 ℓ/hr) and a pressure of $\sim$125 psig (9.64×10^5 Pa). Meanwhile, the temperature and pressure inside the drywell continues to increase. As the drywell temperature exceeds 260°C (500°F), the penetration materials would become sufficiently decomposed such that the penetration modules are physically blown out of the containment wall. The initial leak rate through the drywell electric penetration assemblies would be in excess of $\sim 6.50 \times 10^4$ ft^3/min (3.07×10^4 ℓ/hr) at a pressure of ∿130 psig (9.98×10^5 Pa) if all of the modules failed simultaneously. The fraction of decay heat attributed to fission product which has leaked outside of the containment is ∿0.05. It is noted that this potential mode of containment failure occurs at a pressure ∿30% lower than that analyzed in the Reactor Safety Study.[6]

REFERENCES

1. R. O. Wooton and H. I. Avci, "MARCH Code Description and User's Manual," NUREG/CR-1711 (October 1980).
2. American Nuclear Society Proposed Standard, ANS 5.1, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," October (1971), Revised October (1973).
3. ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors," August (1979).
4. W. B. Wilson et al., "Actinide Decay Power," Los Alamos Scientific Laboratory LA-UR 79-283 (June 1979).
5. D. Fletcher, EG&G Idaho, Personal Communication (February 1981).
6. Reactor Safety Study, WASH-1400 (NUREG-75/014) (October 1975).

<u>N O T E</u>

The following two tables are not a part of the summary submitted for review. They should not be included in the word count. They are attached here for information only.

Brown's Ferry Nuclear Plant: Complete Station Blackout
Sequence of Events

(CSB + No HPCI/RCIC)

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 150°F Initial wetwell temperature = 95°F
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.
3.0	Turbine trips off (turbine stop valves fully closed).

Time (sec)	Event
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 100-psi pressure increase and 40-in. drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint (1060 psi) of safety/relief valves (S/RVs).
5.0	Eight (8) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 20 in. from the previous momentary 40-in. drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 8 S/RVs are completely closed.
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 235.50 in. above Level 0, or 196.44 in. above TAF.
22.0	Suppression pool water average temperature rises 0.24°F to 95.24°F in response to the first S/RV pops.
29.0	All 4 S/RVs are completely closed.
29.7	Two out of 13 S/RVs start to open.

Time (sec)	Event
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 95.54°F.
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 216.00 in. above Level 0, or 176.94 in. above TAF.
90.0	Suppression pool water average temperature is approximately 95.72°F.
101.6	The S/RV is completely closed.
107	Suppression pool water average temperature is approximately 95.81°F.
114.3	The same S/RV starts to open.
145.0	The S/RV is completely closed.
158.3	The same S/RV starts to open.
187.6	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the sequence.
625	Wide range sensed water level reaches low water level set-point (Level 2), i.e., 164.50 in. above Level 0 at 2/3 core height, or 116.50 in. above TAF.
625	HPCI and RCIC systems are not turned on because they are assumed to be unavailable.
20 min.	Suppression pool water average temperature reaches ~105°F.
33 min.	Core uncovery time. Steam-water mixture level is at 11.61 ft above bottom of the core.
40 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 2.0 psi. The HPCI/RCIC systems are not affected. Drywell and wetwell temperature are ~162°F and ~130°F, respectively. Mass and energy addition rates into the wetwell are:

Time (sec)	Event

	Mass Rate (lb/min)	Energy Rate (Btu/min)
Steam	4.36×10^3	5.26×10^6
Hydrogen	8.62×10^{-7}	1.66×10^{-3}

60 min. Mass and energy addition rates into the wetwell are:

	Mass Rate (lb/min)	Energy Rate (Btu/min)
Steam	2.07×10^3	2.88×10^6
Hydrogen	3.76×10^{-1}	1.22×10^3

70 min. Core melting starts.

80 min. Drywell and wetwell temperatures are ~167°F and ~160°F, respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate (lb/min)	Energy Rate (Btu/min)
Steam	7.51×10^2	1.26×10^6
Hydrogen	2.53×10^1	1.30×10^5

96 min. Water level in vessel drops below bottom grid elevation.

97 min. Bottom grid fails and temperature of structures in bottom head is above water temperature.

99 min. The corium slumps down to vessel bottom.

101 min. The debris is starting to melt through the bottom head. Drywell and wetwell temperatures are ~207°F and ~335°F, respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate (lb/min)	Energy Rate (Btu/min)
Steam	2.46×10^3	3.08×10^6
Hydrogen	8.93	2.04×10^4

129 min. Vessel bottom head fails, resulting in a pressure increase of ~49 psia.

129.03 min. Debris starts to melt the concrete floor of the containment building. Temperature of debris is ~2815°F initially. Internal heat generation in metals and oxides are 1.36×10^7 and 2.50×10^7 Btu/h, respectively.

Time (sec)	Event
165 min.	Drywell and wetwell temperatures are ~286°F and ~446°F, respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (lb/min)	Energy Rate (Btu/min)
Steam	722.83	9052
Hydrogen	4.38	0
CO_2	341.88	
[illegible]	[illegible]	

Time (sec)	Event
190 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 400°F and start to vent through the primary containment.
193 min.	Containment failed as the containment temperature exceeds 500°F and all electric penetration modules are blown out of the containment.
219 min.	Drywell and wetwell pressures are ~14.7 psia. Dry- [illegible] ~319°F, respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (lb/min)	Energy Rate (Btu/min)
Steam	92	9052
Hydrogen	32	0
CO_2	307	
CO	666	

The leak rate through the containment failed areas is ~6.15 x 10^5 ft^3/min. The fraction of fission product decay heat that has leaked outside of containment is ~4.49 x 10^{-2}.

Time (sec)	Event
250 min.	Drywell and wetwell temperatures are ~1247°F and ~317°F, respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (lb/min)	Energy Rate (Btu/min)
Steam	905	9052
Hydrogen	33	0
CO_2	203	
CO	695	

Time (sec)	Event
	The leak rate through the containment failed area is ~1.04 x 10^5 ft^3/min. The fraction of fission product decay heat that has leaked outside of containment is 0.224.
309 min.	Rate of concrete decomposition is ~4.65 x 10^4 gm/s. Rate of heat added to atmosphere is ~1.[illegible] x 10^4 kW.
267 min.	Drywell and wetwell pressures are ~14.7 psia and [illegible] The leak rate through the containment failed area is ~[illegible] x 10^4 ft^3/min. The fraction of fission product decay heat that has leaked outside of containment is 0.224.
733 min.	Drywell and wetwell temperatures are ~1014°F and ~369°F, respectively. The leak rate through the containment failed area is ~4.50 x 10^3 ft^3/min. The fraction of fission product decay heat that has leaked outside of containment is ~5.59 x 10^{-2}.

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0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.

Time (sec)	Event
3.0	Turbine trips off (turbine stop valves fully closed).
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 100-psi pressure increase and 40-in. drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint (1090 psi) of safety/relief valves (S/RVs).
5.0	Eight (8) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 20 in. from the previous momentary 40-in. drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 8 S/RVs are completely closed.
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 235.50 in. above Level 0, or 196.44 in. above TAF.
22.0	Suppression pool water average temperature rises 0.24°F to 95.24°F in response to the first S/RV pops.
29.0	All 4 S/RVs are completely closed.

Time (sec)	Event
29.7	Two out of 13 S/RVs start to open.
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 95.54°F.
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 216.00 in. above Level 0, or 176.94 in. above TAF.
90.0	Suppression pool water average temperature is approximately 95.72°F.
101.0	The S/RV is completely closed.
107	Suppression pool water average temperature is approximately 95.81°F.
114.3	The same S/RV starts to open.
145.0	The S/RV is completely closed.
158.3	The same S/RV starts to open.
187.6	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the subsequent cyclings of HPCI and RCIC injections.
625	Wide range sensed water level reaches low water level setpoint (Level 2), i.e., 164.50 in. above Level 0 at 2/3 core height, or 116.50 in. above TAF.
625	HPCI and RCIC systems are automatically turned on. The HPCI and RCIC turbine pumps are driven by steam generated by decay heat. System auxiliaries are powered by the 250 V dc system.
655	HPCI and RCIC flows enter the reactor pressure vessel at 5000 gpm and 600 gpm, respectively, drawing water from the condensate storage tank.
12.5 min.	Narrow range sensed water level reaches Level 8 setpoint, i.e., 270.00 in. above Level 0, or 222.00 in. above TAF.

Time (sec)	Event
12.5 min.	HPCI and RCIC trip off.
20 min.	Drywell and wetwell temperatures exceed 158°F and 102°F, respectively.
26.5 min.	Wide range sensed water level reaches Level 2 setpoint and HPCI automatically restarts. (RCIC does not automatically restart.)
27.0 min.	HPCI flow enters the RPV.
29.0 min.	Narrow range sensed water level reaches Level 8 setpoint and HPCI trips off again. The HPCI system, driven by steam generated by decay heat, turns on and off automatically between sensed Levels 2 and 8 until the batteries run out.
80 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 2.0 psi. The HPCI/RCIC systems are not isolated.
240 min.	The HPCI pump stops when the batteries run out.
260 min.	Wide range sensed water level reaches Level 2 setpoint. Mass and energy addition rates into the wetwell are:

	Mass Rate (lb/min)	Energy Rate (Btu/min)
Steam	3.68×10^3	4.37×10^6
Hydrogen	0	0

Time (sec)	Event
302 min.	Core uncovery time. Steam-water mixture level is at 11.73 ft above bottom of the core.
355 min.	Core melting starts.
389 min.	Water level in vessel drops below bottom grid elevation.
390 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
392 min.	The corium slumps down to vessel bottom.

Time (sec)	Event
394 min.	Debris starts to melt through the bottom head.
426 min.	Vessel bottom head fails, resulting in a pressure increase of 49 psia.
426.04 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is ~2611°F initially. Internal heat generation in metals and oxides are 1.05×10^7 and 1.95×10^7 Btu/h, respectively.
503.27 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 400°F and start to vent through the primary containment at a leak rate of ~250 ft^3/min (1×10^5 c.c./s).
513.59 min.	Containment failed as the containment temperature exceeds 500°F and all electric penetration modules are blown out of the containment. Mass and energy addition rates into the drywell are:

	Mass Rate (lb/min)	Energy Rate (Btu/min)
Steam	610	9052
Hydrogen	15	0
CO_2	133	
CO	311	

The leak rate through the drywell penetration seals is ~6.44×10^4 ft^3/min. The fraction of fission product decay heat that has leaked outside of containment is ~4.44×10^{-2}.

Time (sec)	Event
618 min.	Drywell and wetwell pressures are ~14.7 psia and temperatures are ~1222°F and ~271°F, respectively. The leak rate through the containment failed area is ~6.27×10^4 ft^3/min. The fraction of fission product decay heat that has leaked outside of containment is ~4.91×10^{-2}.
695 min.	Drywell and wetwell temperatures are ~1154°F and ~260°F, respectively. The leak rate through the containment failed area is ~1.37×10^5 ft^3/min. The fraction of fission product decay heat that has leaked outside of containment is ~4.91×10^{-2}.
1028 min.	Drywell and wetwell temperatures are ~1138°F and ~296°F, respectively. The leak rate through the containment failed area is ~2.83×10^3 ft^3/min. The fraction of fission product decay heat that has leaked outside of containment is ~4.91×10^{-2}.