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DRESDEN 2 & 3 AND QUAD CITIES 1 & 2

NUCLEAR POWER PLANTS

DRYWELL TEMPERATURE ANALYSIS

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Performed by: Hoa Hoang
H. X. Hoang
Plant Performance Engineering

Approved by: RS A E Rogers
A. E. Rogers, Manager
Plant Performance Engineering

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ABSTRACT

The limiting drywell temperature envelope is calculated for Dresden 2 & 3 and Quad Cities 1 & 2 Nuclear Power Plants. This envelope is based on main steamline breaks inside the drywell ranging from 0.01 ft^2 to 0.75 ft^2. The calculated temperature envelope for one year allows drywell equipment to be qualified to a temperature lower than that required by the U.S. Nuclear Regulatory Commission (NRC). The maximum drywell temperature obtained is 334°F for less than 10 minutes, compared to the NRC bounding generic envelope of 340°F for 6 hours.

DRESDEN 2 & 3 AND QUAD CITES 1 & 2 DRYWELL TEMPERATURE ANALYSIS

1.0 INTRODUCTION

In the event of a postulated loss-of-coolant accident (LOCA), high energy coolant is released from the reactor vessel. For breaks inside the drywell, this release of steam or liquid will increase the temperature and pressure of the drywell atmosphere. Safety-related electrical equipment within the drywell must be capable of operating under accident conditions.

A plant-unique drywell temperature analysis was performed for the Dresden 2 & 3 Nuclear Power Plants. This analysis is also applicable to the Quad Cities 1 & 2 Nuclear Power Plants due to technical similarities and since the isolation condenser system is not modeled. The purpose of this analysis is to provide drywell atmosphere conditions for equipment qualification. NUREG-0588 "Environmental Qualification of Safety-Related Electical Equipment", Rev. 1, dated July 1981, specifies a generic envelope of 340^{o}F for 6 hours if a plant-unique containment temperature analysis is not performed.

This analysis considers main steamline breaks (MSLBs), inside the drywell, with break areas of 0.01 ft^2 and 0.75 ft^2. Steamline breaks are considered because they result in higher drywell temperatures than liquid breaks.

The accident scenario used in this analysis is similar to other containment pressure and temperature analysis performed by GE. The start of the accident sequence (loss of offsite power, MSIVs closure, reactor scram on MSIV position, main steamline break) is chosen so that the results will yield the most severe temperature and pressure profiles for environmental qualification. With such a consistent and conservative scenario, GE feels that the peak temperatures calculated and the temperature envelope obtained represent a reasonable upper bound for the temperatures expected after an accident.

The analysis is performed in compliance with NUREG-0588 with the following exception:

- o Heat transfer from the drywell air space to the drywell walls is considered.

  Justification: As the drywell airspace temperature is increased by the energy addition due to the break, energy will be transferred as heat to the cooler drywell walls. This approach is permitted in Appendix B of NUREG-0588 for a dry containment. The calculations have been done in accordance with the procedures specified.

2.0 SUMMARY AND CONCLUSIONS

Long-term steamline break transients have been analyzed to find the drywell temperature response. The peak drywell temperature envelope for break sizes of 0.01 ft^2 to 0.75 ft^2, is presented in Table 1.

The envelope never exceeds the generic value of 340^{o}F set by the NRC. The highest temperature, 334^{o}F, lasts for 600 seconds.

3.0 MODEL DESCRIPTION

To model each of the transients and evaluate drywell airspace temperature, over a period of one year, two calculations were made, one for short term and a second for long term response.

3.1 Coupled Reactor and Containment Model-Short Term

The first calculation uses a General Electric proprietary computer code, SHEX-03, to calculate the drywell temperature response through 1000 seconds after the accident for the IBA (0.75 ft^2) and 1800 seconds for the SBA (0.01 ft^2). This code is a coupled Reactor Pressure Vessel (RPV) and containment thermodynamics model, which is utilized to calculate the transient reponse of the containment airspaces. This model performs fluid, mass, and energy balances in the reactor primary system and drywell and wetwell airspaces, and calculates the reactor vessel water level, pressure, and the response of drywell airspace temperature. The various modes of operation of all important auxiliary systems, such as the Relief Valves (RVs), Main Steam Isolation Valves (MSIVs), Emergency Core Cooling System (ECCS), Residual Heat Removal System (RHR) for Quad Cities and Low Pressure Coolant Injection (LPCI) for Dresden, and the feedwater are modeled. Figure 1 illustrates the code model of the reactor vessel, drywell, wetwell, and the reactor auxiliary systems present at Dresden 2 & 3.

3.2 Long-Term Response Model

The calculation for the response out to 1 year uses a simpler model. Energy is added to the system by decay heat and removed by the RHR/LPCI heat exchanger. Energy is transferred within the system by ECCS flow and RHR/LPCI flow. The temperature of the drywell is found by taking mass-weighted average of the break and drywell spray enthalpies.

4.0 INITIAL CONDITIONS

The following initail conditions were used in the analysis:

1. Reactor is operating at 102% of rated thermal power (2578 MWt).

2. Suppression pool temperature is at the normal power operation high temperature (95°F). This corresponds to the maximum service water temperatrue.

3. Suppression pool water volume is at minimum technical specification (112,000 ft^3 at Dresden and 112,200 ft^3 at Quad Cities).

4. Drywell air space is a nominal bulk temperature (150°F) and has high relative humidity (50%) for normal operation. If the drywell humidity is above 50%, the drywell temperature decreases because liquid droplets will form in the drywell atmosphere, keeping the fluid saturated.

5. Wetwell airspace is at high temperature (95°F) for normal operation and has 100% relative humidity.

5.0 ASSUMPTIONS

The following assumptions were used in the analysis:

1. Emergency power is available.

2. Normal automatic operation of the plant safety systems.

3. Control rod drive (CRD) flow is maintained constant at 5.56 lbm/sec.

4. The May-Witt decay heat curve (Ref. 2) is used for containment analysis.

5. Heat transfer to the wetwell walls from the wetwell airspace and suppression pool is not considered.

6. The main steam isolation valves (MSIVs) close in 3 seconds.

7. The control volume of the reactor includes the reactor vessel, the recirculation lines and the steam lines from the vessel to the inboard MSIVs.

8. RVs in the automatic mode have normal setpoints as specified in Reference 1: 2 at 1115 psig, 2 at 1135 psig, and 1 S/RV at 1135 psig. Flowrate is 194 lbm/sec at 1120 psia. This flow rate is 122.5% of rated which increases the long-term temperature.

9. Low values for the condensing heat transfer coefficient were used based on the NRC's formulation of the Uchida correlation in Appendix B of NUREG-0588. Based on communication with Hitachi, the driving temperature difference used was that between the bulk and wall temperatures. Experiments have shown that heat transfer can be much higher (see for example the Tagami correlation in Appendix C of NUREG-0588). J.J. Cabajo (Ref. 4) suggests heat transfer coefficients of four to five times higher than the Uchida/Tagami correlations based on the CVTR experiments.

10. The largest break size investigated was 0.75 ft^2. A steam break larger than this will result in rapid reactor vessel depressurization. This rapid depressurization will cause flashing of saturated water in the vessel, two-phase level swell and two-phase flow from the break. The drywell temperature resulting from two-phase flow will be saturation temperature at the drywell pressure which is considerably less than the temperature reached in a steam break.

11. The RHR at Quad Cities and LPCI at Dresden is used for containment spray, rather than reactor vessel level control. Reactor vessel level is maintained by other available systems.

12. The drywell airspace contains a homogenous air/steam mixture.

13. The drywell heat sink includes the drywell steel shell and the vent system. Several heat sinks were ignored, however, including the reactor building, the reator pressure vessel biological shield wall, and equipment in the drywell.

14. The break initiates simultaneously with the loss of offsite power at the start of the accident.

15. The high pressure coolant injection (HPCI) pump trip at a water level of 551.2 in. above vessel zero. The operator will throttle the low pressure core spray (LPCS) pump at high water level to avoid vessel flooding. If the operator does not run back the LPCS pump, the break will inject liquid into the drywell resulting in lower temperatures. These pumps are activated when the water level drops to 444.06 in. above vessel zero to maintain sufficient core coolant inventory. The time assumed for HPCI to reach full rated flow is 30 seconds and that for LPCS is 60 seconds. These times include diesel startup and completion of all valve motions. These values are considered conservative and are used in ECCS analysis.

16. One RHR loop at Quad Cities/LPCI at Dresden is unavailable due to diesel generator failure. This has been found to be the worst single failure. Calculations have shown that RHR/LPCI failure represents the worst single failure since the integrated steam energy from the break flow differs only by 5% with and without HPCI, thus eliminating HPCI failure from being the worst single failure.

17. In accordance with the Emergency Procedure Guidelines (Ref. 3), wetwell spray is started first and later drywell spray. For both break sizes, wetwell spray is started 600 seconds into the event. For the 0.01 ft^2 break size, the drywell spray is initiated at 1600 seconds. This is the estimated time when the drywell airspace reaches the initiation pressure of 31.9 psia (Step 3 of Primary Containment Pressure Control in Ref. 3). For the 0.75 ft^2 break size, the drywell spray is initiated at 600 seconds because the drywell pressure is higher than 31.9 psia.

18. The Automatic Depressurization System (ADS) is not used. Reference 3 states that ADS should only be used as a last contingency.

19. Initial containment conditions are set to yield low pressures which result in higher drywell temperatures.

20. The breakflow fluid is assumed to be dry steam.

6.0 DRYWELL TEMPERATURE ANALYSIS

A detailed description of each event, and the corresponding results of the drywell temperature analysis are presented in this section.

6.1 0.01 ft^2 Main Steamline Break Accident Sequence

Time (Sec)	Event Description
0	Loss of off-site power. Reactor isolation followed by automatic reactor scram on MSIV position. 0.01 ft^2 break on main steamline.
3	MSIVs fully closed.
7	Feedwater flow stopped at the end of coast down period.
12	RV automatic cycling action initiated on high vessel pressure.
44	Drywell pressure reaches 16.7 psia and HPCI is initiated.
74	HPCI injection to the reactor vessel reaches rated flow.
600	Wetwell spray is initiated and operates continuously during the event. Wetwell airspace temperature reaches a maximum of 156^{o}F.
1600	Drywell spray is initiated and operates continuously during the event. Drywell temperature peaks at 262^{o}F.

6.2 Analysis Results for 0.01 ft^2 Break

The models described in Section 3.0 were used to calculate containment response. The containment temperature and pressure results are given in Figures 2 and 3, respectively. The vessel pressure is given in Figure 4. The long term drywell temperature response is given in Figure 5.

The reactor scrams automatically on MSIV closure. At 7 seconds, the feedwater flow is completely terminated due to loss of off-site power. At 44 seconds, high drywell pressure (16.7 psia) signals the startup of the HPCI system. At 74 seconds, HPCI flowrate to the vessel reaches its rated value. This coolant injection causes a drop in reactor pressure which starts to rise again as soon as HPCI is shut off. HPCI operates on and off according to the reactor water level to maintain an adequate coolant inventory. At 12 seconds, RVs begin actuating to maintain vessel pressure. This RV cycling can be seen in the vessel pressure curve. Drywell and wetwell temperature and pressure increase from the start of the accident until the initiation of wetwell spray. The operator manually actuates wetwell spray in accordance with the Emergency Procedure Guidelines, resulting in wetwell spray flow initiating at 600 seconds. This spray has only a minor effect on the drywell temperature. The drywell temperature continues to rise and reaches a maximum of 262°F at 1600 seconds when it drops sharply due to the drywell spray initiation. Wetwell and drywell sprays, once started, are left on throughout the transient. Drywell and wetwell pressure curves follow each other very closely with about a 1 psi difference until drywell spray initiation at 1600 seconds. The drywell spray condenses the steam in the drywell, causing a significant pressure drop and the wetwell pressure follows due to the vacuum breaker actuation.

A second temperature peak is reached at about 10,000 seconds, but this peak is only 162°F. This second peak is at the point where the decay heat is equal to the rate of heat removal by the RHR/LPCI heat exchanger.

6.3 $0.75\ ft^2$ Main Steamline Break Accident Sequence

Time (Sec)	Event Description
0.0	Loss of off-site power. Reactor isolation followed by automatic reactor scram on MSIV position. $0.75\ ft^2$ break on main steamline. HPCI initiated upon high drywell pressure signal.
3	MSIVs fully closed.
7	Feedwater flow completely terminated.
31	HPCI injection to the reactor vessel reaches rated flow.
125	LPCS flow to vessel begins as vessel pressure reaches LPCS pump shutoff head (318.7 psia for Dresden and 325 psia for Quad Cities).
600	Wetwell spray is initiated and operates continuously. Wetwell airspace temperature at this time is $217^{o}F$. Drywell spray is intiated and operates continuously. Drywell airspace temperature at this time is $283^{o}F$.

6.4 Analysis Results for $0.75\ ft^2$ Break

The containment response was calculated using the models described in Section 3.0. The containment temperature and pressure results are given in Figures 6 and 7, respectively. The vessel pressure is given in Figure 8. The long term drywell temperature response is given in Figure 5.

The reator scrams automatically on MSIV closure. At 1 second, high drywell pressure (16.7 psia) signals the startup of the HPCI system. At 7 seconds, feedwater flow is completely terminated due to loss of off-site power. At 31 seconds, HPCI flowrate to the vessel reaches its rated value and operates on and off depending on the reactor vessel water level. At 125 seconds, LPCS flow to the vessel begins. An early drywell temperature peak is reached at about 10 seconds due to the Uchida heat transfer coefficient formulation. The heat transfer coefficient is small for a large air/steam mass ratio and increases as this ratio decreases. Initially, the air/steam ratio is fairly high and the heat transfer is therefore small. This allows a high temperature in the drywell. As the air is purged from the drywell, the heat transfer to the walls increases and the drywell temperature turns around.

Due to the large break size, the reactor pressure drops rapidly during the first hundred seconds of the event. The wetwell and drywell sprays initiate at 600 seconds as a result of operator action. These sprays result in a significant drop in wetwell and drywell temperature, due to condensation of steam in the airspace.

A second temperature peak is reached at about 10,000 seconds, but this peak is only 162°F. This second peak occurs at the point where the decay heat is equal to the rate of heat removal by the RHR/LPCI heat exchanger.

TABLE 1

DRESDEN 2 & 3 AND QUAD CITIES 1 & 2 DRYWELL TEMPERATURE ANALYSIS
TEMPERTURE ENVELOPE

Time		TEMPERATURE (^oF)
0 - 600 sec.		334
600 - 850 sec.		287
850 - 1600 sec.		282
1600 - 1680 sec.	(≈0.02 day)	282 - 165 *
0.02 - 0.25 day		165
0.25 - 2.5 days		165 - 128
2.5 - 25 days		128 - 112
25 - 400 days		112 - 104

*Corrected entry 1/10/83

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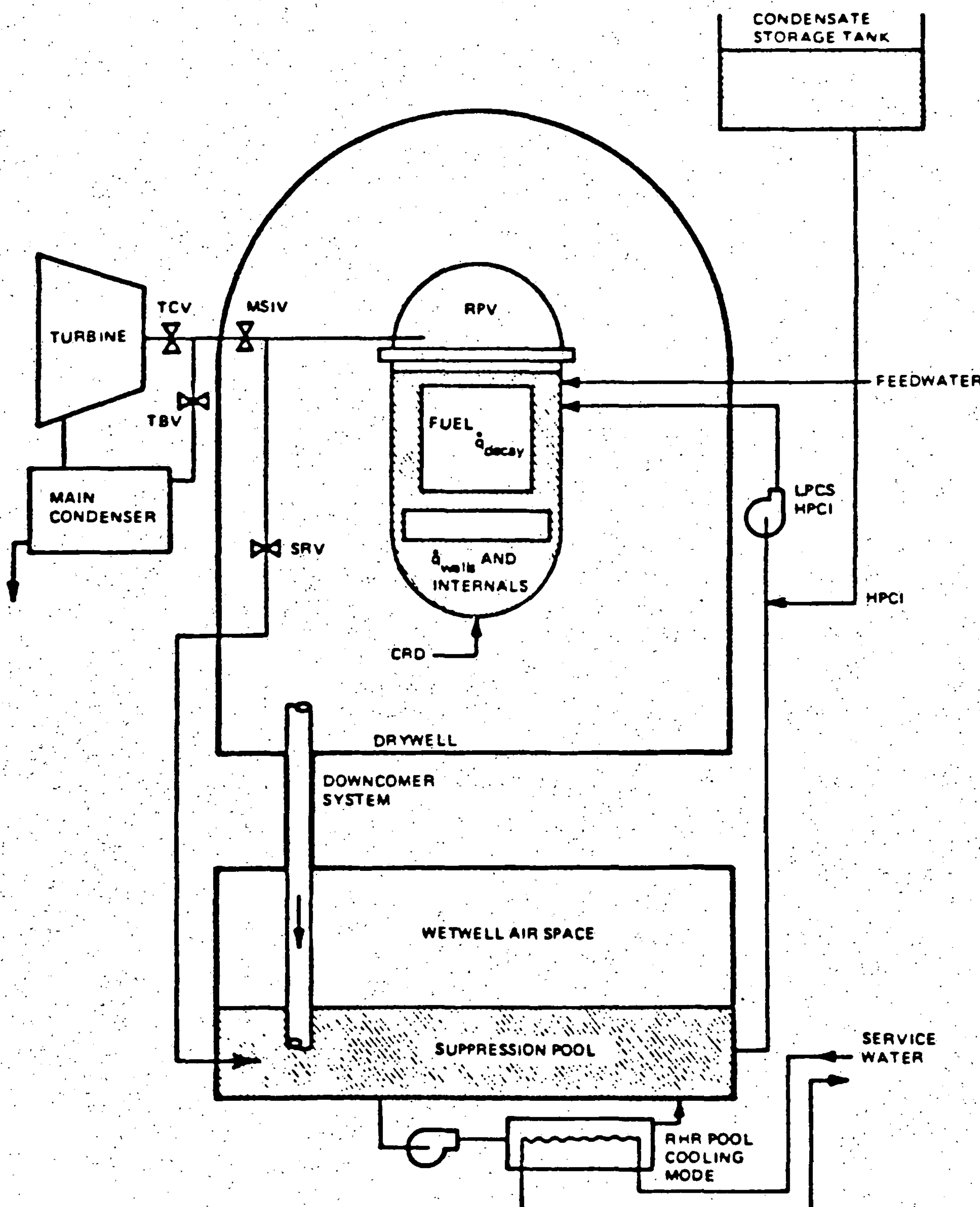

Figure 1. Coupled Reactor and Suppression Pool Model