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October 15, 1981

Mr. Dennis M. Crutchfield, Chief
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Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555



Subject: Dresden 2
SEP Topic: Accident & Transient topics XV-1, 3, 4, 5, 7, 8, 9, 11, 13, 14, 15, 18, 19, and 20.
NRC Docket No. 50-237

Reference:
1) Letter from R.F. Janecek to P.O'Connor dated March 30, 1980
2) Letter from T.J. Rausch to D.G. Eisenhut dated August 14, 1981.
3) Letter from H.E. Bliss to N.P. Smith dated April 7, 1981.

Dear Mr. Crutchfield:

Attached are the SEP topic assessments prepared in response to our commitments made in reference 1,2 and 3. The attached assessments for the above referenced topics were patterned by the completed topic assessments prepared by the NRC and given to us as examples to be used in preparing our assessments. While we have provided assessments of the above topics we are preparing in January, 1983 Unit 2 outage to load the reactor with Exxon fuel which will necessitate much of the analysis work to be redone. Approximately 1/3 of the core (232 out of 724) is to be replaced in a scattered loading pattern.

Please address any questions you may have concerning this matter to this office.

For the above topic assessments we are providing copies of non-GE and non-NRC originated documents which were referenced in the report that are not in your file. Topic assessment reference S and X are attached.

One (1) signed original and thirty-nine (39) copies of this transmittal have been provided for your use.

Very truly yours,

Tom R. Tramm
for
Thomas J. Rausch
Nuclear Licensing Administrator
Boiling Water Reactors



SPP:mnh
1038D
Attachment
cc: R III Resident Inspector, Dresden

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S 1/39

50-237

SAFETY ASSESSMENT
FOR THE
DRESDEN 2 NUCLEAR POWER PLANT

Received wth ltr dtd 10/15/81

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SAFETY ASSESSMENT
FOR THE
DRESDEN 2 NUCLEAR POWER PLANT
DOCKET NO. 50-237

DRESDEN 2

I. Introduction

1. Systematic Evaluation Program Description (Supplied by NRC Staff)

II. Design Basis Events

1. Accidents and Transients

A. Introduction

B. DBE Documentation History

C. Codes and Models

D. DBE Performance

1.0 Group I Events

1.1. Decrease in Feedwater Temperature (Topic XV-1)

1.2. Increase in Feedwater Flow (Topic XV-1)

1.3 Increase in Steam Flow (Topic XV-1)

1.4 Startup of Inactive Loop (Topic XV-9)

1.5 Flow Controller Malfunction Increasing (Topic XV-9)

1.6 Inadvertent Closure of Main Steam Isolation Valves (Topic XV-3)

2.0 Group II Events

2.1 Loss of External Load (Topic XV-3)

2.2 Turbine Trip (Topic XV-3)

2.3 Loss of Condenser Vacuum (Topic XV-3)

2.4 Steam Pressure Regulator Failure (Topic XV-3)

2.5 Loss of Feedwater Flow (Topic XV-5)

2.6 Feedwater Line Break

3.0 Group III Events

3.1 Steam Line Break Inside Containment (Topic XV-2)

3.2 Steam Line Break Outside Containment (Topic XV-2)

3.3 Radiological Consequences (Topic XV-18)

4.0 Group IV Events

4.1 Loss of Power to Station Auxiliaries (Topic XV-4)

5.0 Group V Events

5.1 Loss of Forced Flow (Topic XV-7)

5.2 Pump Seizure (Topic XV-7)

6.0 Group VI Events

6.1 Rod Withdrawal at Power (Topic XV-8)

6.2 Rod Withdrawal From Low Power (Topic XV-13)

6.3 Rod Drop (Topic XV-13)

6.4 Radiological Consequences of a Rod Drop (Topic XV-13)

7.0 Group VII Events

7.1 Loss of Coolant Accident (Topic XV-19)

7.2 Radiological Consequences of a LOCA (Topic XV-19)

7.3 Radiological Consequences of Failure of Small Lines (XV-16)

8.0 Group VIII Events (supplied by NRC Staff)

9.0 Group IX Events

9.1 Inadvertent Opening of a BWR Safety/Relief Valve (Topic XV-15)

10.0 Group X Events

10.1 Inadvertent Actuation of ECCS That Causes an Increase in Inventory (Topic XV-14)

11.0 Group XI Events

11.1 Fuel Loading Error (Topic XV-11)

Table 1: Reactor Protection System Scram Setpoints

Table 2: Containment Isolation Setpoints

Table 3: Engineered Safety Features Initiation Signals

Table 4: Initial Conditions and Assumptions

References

## II. Design Basis Events

### 1. Accidents and Transients

#### A. Introduction

The safety philosophy used in the design of reactor plants has traditionally been based on the concept of "defense-in-depth." Dresden 2 has four main layers of defense, to which this concept refers they are the physical barriers of the reactor fuel clad, the reactor coolant system pressure boundary, and the reactor primary containment, and the reactor building. These barriers provide defense for the general public from exposure to the radioactive products produced in the fuel by nuclear fission.

System disturbances, and malfunctions or equipment failures can occur during plant operation, and challenge the integrity of the four defense-in-depth barriers. These occurrences are analyzed to determine the capability of the plant design and installed plant systems to prevent the breeching of these barriers.

The American Nuclear Society has clasified plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. In general, this classification is also followed in the Standard Review Plan Chapter 15 review procedure for plant accidents and transients. The four categories are:

Condition I: Normal Operation and Operational Transients
Condition II: Faults of Moderate Frequency
Condition III: Infrequent Faults
Condition IV: Limiting Faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. The impact of various single failures on the course of an accident or transient is also considered.

For a new plant under review for an Operating License, the approach outlined in Regulatory Guide 1.70, Chapter 15, is used to:

1. Ensure that a sufficiently broad spectrum of initiating events has been considered,

2. Categorize the initiating events by type and expected frequency of occurence so that only the limiting cases in each group need to be quantitatively analyzed, and

3. Permit the consistent application of specific acceptance criteria for each postulated initiating event.

To accomplish these goals, a number of disturbances of process variable and malfunctions or failures of equipment should be postulated. Each postualted initiating event should be assigned to one of the following categories:

1. Increase in heat removal by the secondary system (Turbine plant),
2. Decrease in heat removal by the secondary system (Turbine Plant),
3. Decrease in reactor coolant system flow rate,
4. Reactivity and power distribution annomalies,
5. Increase in reactor coolant inventory,
6. Decrease in reactor coolant inventory,
7. Radioactive release from a subsystem or component, or
8. Anticipated transients without scram."

One of the items of information that should be discussed for each initiating event relates to its expected frequency of occurrence. Each initiating event within the eight major groups should be assigned to one of the following frequency groups:

1. Incidents of moderate frequency,
2. Infrequent incidents, or
3. Limiting faults.

The initiating events for each combination of category and frequency group should be evaluated to identify the events that would be limiting. The intent is to reduce the number of initiating events that need to be quantitatively analyzed. That is, not every postulated initiating event needs to be completely analyzed by the applicant. In some cases a qualitative comparison of similar initiating events may be sufficient to identify the specific initiating event that leads to the most limiting consequences. Only that initiating event should then be anlyzed in detail.

It should be noted, however, that different initiating events in the same category/frequency group may be limiting when the multiplicity of consequences are condsidered. For example, within a given category/frequency group combination, one initiating event might result in the highest reactor coolant pressure boundary (RCPB) pressure while another initiating event might lead to minimum core thermal-hydraulic margins or maximum offsite doses.

This approach was used in the reevaluation of accidents and transients for the SEP facilities.

The accident and transient analyses for the Dresden 2 plant are discussed and evaluated in the following subsections. In accordance with the SEP review method the evaluation includes an assessment of the ability of the plant to adequately mitigate the event. The SEP topics which affect the design basis events are identified and discussed in the evaluation of this event.

A review of the frequency of occurrence of events at the facility has shown that the current regulatory criteria used in the facility accident and transient evaluations are those found in Chapter 15 of the Standard Review Plan. In general, the acceptance criteria for moderate frequency events are (1) pressures must not exceed 110% of design pressure for the reactor coolant and steam generator systems, (2) fuel clad integrity must be maintained for essentially all fuel rods in the core, (3) a moderate frequency incident should not generate a more serious plant condition without other faults occurring independently, and (4) a moderate frequency event in combination with an assumed single active failure, or single operator error, should not cause the loss of any barrier other than the fuel cladding. A limited number of clad perforations is acceptable.

Tables 1, 2 and 3 present the setpoints for the reactor protection system, the engineered safety features initiation, and containment isolation. Table 4 summarizes the key input assumptions for each DBE.

## B. DBE Documentation History

The Dresden 2 FSAR (Reference A) was issued in November 1967. The submittal was prepared by Commonwealth Edison and General Electric. The plant operating license was issued in December 1969.

The original core loading was with 7x7 fuel. In susequent reloads, 8x8 fuel assemblies were included. General Electric has prepared generic topical reports (References B and C) for 8x8 reloads, which assess the transient analyses. The most limiting transients as identified by these topicals are reanalyzed for each reload. The topicals show that only a few of the possible plant transients result in limiting conditions so that reload submittals need only analyze these selected events, which include:

a) Rod Withdrawal Error

b) Turbine Trip w/o Bypass or Generator Trip w/o Bypass

c) Loss of FW Heating or Inadvertent HPCI Actuation

d) MSIV Closure

This approach, as well as the selection of limiting transients, has received staff acceptance, Reference D.

Reload sumittals for Dresden 2 were submitted in References E, F, G, H, and S. Staff evaluations of these submittals are included in References I, J, K, L and L-1.

The rod withdrawal error, loss of feedwater heating, load rejection w/o bypass and MISV closure have been the most severe transients in the latest cycles.

LOCA Analysis

The LOCA analysis for Dresden 2 was originally submitted in the FSAR. As technology and the acceptance criteria were refined, new analyses were provided. GE has completed "lead plant analyses" for LOCA. For each lead plant, extensive LOCA calculations are done to determine which break sizes and locations are most severe. For other plants, reference is made to the applicable "lead plant" analysis to establish limiting breaks, and only a few cases are explicitly analyzed for the non-lead plant. For Dresden 2, analysis as a lead plant is presented in reference M.

With the inclusion of some 8x8R fuel elements, a revised analysis was done to support higher MAPLHGR limits. It was based on the improved reflood characteristics of the core containing fuel with alternate flow path holes drilled in the lower tie plates. This approach was previously used for Quad Cities 1 and Pilgrim and was approved by the staff.

Reload 5 of Dresden 2 included a new prepressurized fuel design (P8x8R) analysis was done to support MAPLHGR limits for this fuel (reference P).

The analysis for Dresden 2 (Reference P) was performed as a non-lead plant, with reference made to the lead plant analysis done for Duane Arnold (Reference Q).

For small breaks, analysis presented in Reference R showed that these breaks were less severe than the large break LOCA. The codes and evaluation models have been approved for use and are in conformance with Appendix K of 10 CFR 50.

C. Codes and Models

The plant transient analyses are performed with GE codes and models such as BWR Simulator and GETAB, which have been reviewed and approved for use in transient and accident analysis.

The General Electric Thermal Analysis Basis (GETAB) utilizes the General Electric Critical Quality Boiling length correlation (GEXL), with the design limit chosen so that 99.9% of the fuel rods are expected to avoid boiling transition. The transient analysis code uses a point kinetics model with reactivity feedback from control rods, voids, and Doppler effects. The active core void fraction is calculated from a relationship between core exit quality, inlet subcooling and pressure, generated from multi-node core calculations. Principal controller functions such as feedwater flow, recirculation flow, and protection system functions are modeled.

The initial conditions and assumptins used in the transient analysis were reviewed by the staff in Reference D and were determined to generally conform to the Standard Review Plan requirements. Plant system characteristics are taken at their most adverse technical specification value; end-of-cycle core nuclear parameters with conservatism factors are used and the transient initiating events are assumed to occur with a conservative magnitude and rate. A 25% penalty is applied to the void coefficient. The scram worth is assumed to be 80% of its total scram worth.

## D. DBE Performance

### 1.0 Group I Events

Group I events are moderate frequency events that involve an increase in heat removal by the secondary system or an increase in core flow.

### 1.1 Decrease in Feedwater Temperature

A decrease in FW temperature due to loss of feedwater heating can occur if the steam extraction line to a heater is shut producing a gradual cooling of the tubes, or if a feedwater heater bypass line is opened. Neutron flux increases in response to the cooler core inlet temperature which collapses voids, through response to the void coefficient.

Instantaneous loss of 145°F of heating capacity is assumed. The reactor is assumed to be in manual flow control mode, at 100% power when the heating is lost. The system establishes itself at new equilibrium with a higher heat flux. No reactor trips occur for this transient, which is the most limiting cold water event.

Analysis of this transient is provided in each reload submittal. The most recent analysis is provided in reference S. Input assumptions and evaluation models are discussed in reference C. A post TMI assessment of this transient has also been made and is presented in section 3.2.2 of Reference U.

### 1.2 Increase in Feedwater Flow

An increase in feedwater flow can occur in response to a feedwater controller failure. The FSAR assumed the most severe steam/feed flow mismatch and level transient results for an event initiated from 50% flow, and 65% power. This is the maximum power allowed for the given flow condition.

This event was analyzed in Section 11.3.3 of the FSAR.

The increase in cool feedwater flow into the core results in increasing power and vessel water level. The high reactor water level turbine trip closes the turbine stop valves, and trips the feedwater pump. Turbine stop valve closure produces shutoff of turbine steam flow, opening of the steam bypass system, and reactor scram (on 10% valve closure).

The bypass system limits the pressure rise after the turbine trip so that the relief valves do not lift.

The generic reload topical report, Reference C, identified the feedwater controller failure to maximum demand as a potentially limiting event. The transient is initiated from rated power, and is terminated by a high water level turbine trip. This event can be more severe than the similar transient at reduced power due to the higher pressure and the higher energy to be relieved after the turbine trip. The analysis for this event is included in reference S.

The sequence of events for this event is summarized below.

| Time (sec) | Event |
|---|---|
| 0 | Feedwater controller failure |
| 22 | high water level turbine trip |
| 22+ | reactor trip on stop valve closure; bypass valves open; reactor feedpumps trip. |

Following the event, the turbine and reactor have tripped, and the plant is in a stable condition. To recover, the operator must correct the feedwater controller malfunction, and initiate a normal return to power. If an extended plant shutdown is required, steam bypass to the main condenser can be used to remove decay heat. The safe shutdown discussion (Topic VII-3) details how the plant would proceed to hot, then cold shutdown.

A post TMI assessment of this event has been made and is presented in section 3.2.2 of Reference U.

## 1.3 Increase in Steam Flow

A steam flow increase can occur due to a turbine controller malfunction which results in excess opening of the turbine control valves. Maximum demand is limited to 110% by the control system, which adjusts the turbine control valves as necessary to maintain this load. As excess steam is withdrawn from the pressure vessel, reactor steam pressure decreases. This pressure reduction allows increased boiling, and thus the insertion of negative reactivity. The steam pressure decrease results in closure of the main steam isolation valves, and a subsequent reactor trip.

This transient was analyzed in the FSAR, and the transient results are shown in FSAR figures 11.2.5 (a) and(b) for a pressure regulator failure to 115%. This transient was also assessed for the generic reload topical and was found to be less severe than the worst cold water event described in section 1.1 or a steam line break (Section 3.0).

1.4 Startup of Inactive Loop

An insertion of cool water into the core can result from startup of an idle loop. Neutron Flux increases due to the void reactivity feedback as the cold recirculation flow reaches the core. This transient was anlyzed in the FSAR and the results are shown in FSAR figures 4.3.13.a and b.

For the anlysis, the inactive recirculation loop is assumed to be filled with 100°F water . The drive pump is started, and the 100°F water is discharged to the reactor vessel.

Throughout the event, diffuser flow in the loop being started remains reversed, so the cold water flows out the jet pump suctions, mixes in the downcomer region and finally reaches the lower plenum through the active jet pumps.

Normal procedures for loop startup would require warming of the loop prior to startup of an idle pump.

The highest possible initial power for one loop operating is 60% power (Reference N). However operating procedures require the operating pump to be at minimum speed during the startup of an idle loop. The power transient is controlled by the cold water induced reactivity. No automatic trips occur during this transient.

This transient was assessed in the generic reload topical (Reference C) and was found to be less severe than the worst cold water event described in Section 1.1. This is not a severe plant transient, reanalysis for each reload is not required.

1.5 Flow Controller Malfunction Causing An Increase in BWR Core Flow Rate

A malfunction in one of the motor generator set speed controllers could cause the scoop tube positioner for the fluid coupler to move at its maximum speed in the direction of increasing pump speed and flow at 10%/second. A malfunction of the master flow controller is less severe due to a slower rate of increase limit. Neutron flux increases in response to the void coefficient of reactivity, however the fuel surface heat flux increases only slightly, so no thermal margins are approached. A scram on high flux occurs at approximately 7 seconds.

The assumed initial conditions are 65% power and 50% flow. The failed coupler reaches full stroke within 9 seconds. The unfailed coupler remains at its initial position for the assumed conditions. Diffuer flow in the failed loop increases, while flow in the opposite jet pumps decreases.

This event was analyzed in Chapter 4 of the FSAR, and the results are shown in Figures 4.3.9a and b. The generic reload application (Reference C) report considered this event, which is the most severe transient (in terms of MCPR) at reduced core flow. The evaluation showed that use of $K_f$ curves would ensure that the limiting MCPR is not reached. The $K_f$ curves are established for both the manual and automatic flow control mode. A post TMI assessment of this event is presented in section 3.2.2 of Reference U.

## 1.6 Inadvertent Closure of Main Steam Line Isolation Valves

Closure of the MSIV's result in overpressurization of the primary system. The valves are assumed to close in the fastest possible time. With the steam lines isolated, the reactor has lost its major heat sink, and an alternate path, such as the relief valves or isolation condenser must be used.

A direct scram occurs on 10% closure of the isolation valves. For safety valve adequacy studies, the direct scram from 10% MSIV closure is neglected and the reactor is assumed to trip on high flux. The safety valves limit the peak pressure to 1375 psig (110% of vessel design pressure), even assuming the isolation condenser and relief valves fail to operate.

The sequence of events is summarized below.

| Time (sec) | Event |
|---|---|
| 0 | MSIV's begin to fully close |
| 3 | high flux scram |
| 4 | safety valve lifts |

Single failures of the protection system and complete failure of the steam relief valves are considered in the analysis.

MSIV closure with indirect (flux) scram is a severe overpressurization transient which is reanalyzed each reload. The method used in the analysis is described in the generic reload topical report. The analysis demonstrates that for the fastest valve closing time, the system pressure remains below the limit of 110% of design pressure. The results of this analysis are shown in figure 7 of Reference S.

A post TMI assessment of this event has been made and is presented in section 3.2.2 of Reference U.

2. Group II Events

These are moderate frequency events caused by failures in the secondary system.

2.1 Loss of External Load

A loss of generator load causes a power/load unbalance which produces a turbine control valve fast closure signal. The sudden loss of heat removal causes reactor pressure, and neutron flux to increase.

A reactor scram results directly from the power/load unbalanced signal, alternate trips would result from switches that sense control valve fast closure, position switches on the turbine stop valves at 10% closure (following the turbine trip), high flux, and high reactor pressure.

The bypass valves operate to reduce pressure to prevent lifting of the relief valves. This transient is generally less severe than other overpressurization events due to the direct scram on loss of load. If, however, the bypass valves fail to open, this transient presents a challenge to the relief valves. The sequence of events is summarized below.

| Time (sec) | Event |
|---|---|
| 0.0 | loss of load |
| 1.5 | reactor trip |
| 2.0 | relief valve opens (if bypass fails) |
| 4.0 | peak vessel pressure |
| 14.0 | relief valve begins to close |

Each reload submittal considers the generator load rejection with bypass failure to assess relief valve adequacy as well as MCPR.

Reference S presents the most recent analysis for cycle 8 operation. The load rejection without bypass results in the greatest CPR for 7x7 fuel as well as for 8x8, 8x8R and P8x8R fuel (control rod withdrawal error is equally limiting for 8x8 fuels).

In addition, this event is also limiting from overpressurization considerations, and is used to evaluate the capacity of the reactor relief valves. The most recent analysis of generator trip without bypass is presented in Reference S, and shows vessel pressure reaching 1244 psig.

Loss of generator load with bypass was analyzed in Chapter 11 of the FSAR and the results are shown in Figures 11.2.2a and b. A post TMI assessment of this event has been made and is presented in section 3.2.2 of Reference U.

## 2.2 Turbine Trip

A turbine trip can occur as a result of many causes including loss of load. It produces a reactor isolation, and an increase in system pressure, which causes a void reduction, and thus positive reactivity insertion and power increase. A Scram is initiated by position swtiches on the turbine stop valves. If the turbine trip scram fails, a high flux trip would scram the reactor. Turbine stop valves closure was analyzed in Chapter 11 of the FSAR and the results are shown in Figures 11.2.3a and b.

In the FSAR, the turbine trip event was used to assess relief capacity. This transient (w/o bypass) was analyzed to evaluate relief valve capacity, assuming scram from turbine stop valve 10% closure in Chapter 4 of the FSAR and the results are presented in FSAR Figures 4.4.1a and b. The FSAR also used this transient, turbine trip without bypass, to assess safety valve capacity with the additional assumption that the turbine stop valve closure scram and reactor relief valves fail the results of this analysis is presented in FSAR Figures 4.4.2a and b.

In later analyses relief valve capacity was based on the generator load reject event described in Section 2.1 which has been found to be more limiting in terms of reactor vessel presurization, and CPR.

The turbine trip without bypass or relief has been replaced by main steam valve isoltion with failures of the valve position scram and the relief valves as the safety valve sizing event. This event is reanalyzed each fuel cycle the results for the present cycle are presented in figure 7 or Reference S. A post TMI assessment of this event has been made and is presented in Reference U section 3.2.2

## 2.3 Loss of Condenser Vacuum

An instantaneous loss of condenser vacuum event is similar to a turbine trip with bypass failure. A scram occurs when the vacuum drops to 23" Hg, the turbine stop valves close at 20" Hg, and the turbine bypass valves close at 7" Hg.

Relief valves and the isolation condenser operate to remove the stored heat. This transient was not analyzed in the FSAR since the worst case event the instantaneous loss of vacuum is identical to the turbine trip without bypass trip. It is, therefore, bounded by the load rejection transient (Section 2.1), which bounds the turbine trip, as discussed in Section 2.2.

## 2.4 Steam Pressure Regulator Failure

A turbine pressure regulator failure can occur in either of two ways, zero output or maximum output. A failure to zero output is terminated by operation of the backup regulator. When the failed regulator attempts to close the valves, pressure rises, and the backup regulator will take over. This is a relatively minor transient and is similar to a pressure set point increase which is a normal plant operation analyzed in Chapter 7 of the FSAR the results are presented in Figures 7.2.10a and b of the FSAR.

If either regulator fails in the open direction, the maximum control plus bypass valve demand is limited by the control system to 115 percent (or less). This event was analyzed in chapter 11 of the FSAR and the results are presented in FSAR Figures 11.2.5 a and b. This transient results in a mainsteam valve isolation.

Scram occurs when the isolation valves have reached 10 per cent closed. The depressurization is stopped as soon as the isolation becomes effective and the reactor is shut down with pressure rising slowly. The isolation condenser will dissipate the decay heat for long-term shutdown.

## 2.5 Loss of Feedwater Flow

The simultaneous tripping of all feedwater pumps, or a feedwater controller failure that closes the feedwater control valves results in a complete loss of feedwater flow. The recirculation flow M-G sets run down to minimum speed when feed flow drops below 20%. This protects the recirculation pumps and jet pumps from cavitation.

The reactor water level decreases until the low water level scram trips the reactor at about 7 seconds. At about 33 seconds the low-low water level set point is reached at which point the main steam isolatin valves close, the reciruculation pumps trip and HPCI starts. Should HPCI fail the isolation condenser is independently capable of maintaining water level above the top of active fuel.

This event was analyzed in Chapter 11 of the FSAR assuming failure of the HPCI system and the results are presented in FSAR figures 11.3.3a and b.

The sequence of events are:

| Time (sec) | Event |
|---|---|
| 0.0 | loss of feedwater |
| 7.4 | low water level scram |
| 36.5 | MSIV's close (on low-low water level) |

This event was assessed in generic topical reference (C) as a potentially thermal-hydraulic event and was found not to be limiting it is therefore not reassessed for each reload.

Following the Three Mile Island Unit 2 accident loss of feedwater was reanalyzed using best estimate caculations in response to information requests by the NRC's Bulletins and Orders Task Force (B&OTF). These analyses are contained in section 3.2.1 of reference U and conclude that:

(1) The BWRs covered in this report are adequately equipped to mitigate the consequence of the LOF event as it relates to core cooling without operator assistance under all conditions within the design bais, with or without a stuck-open relief valve.

(2) Operator actions are required only under the highly improbable conditons where complete loss of high pressure injection and inventory maintenance systems occurs. In such an event, timely manual depressurization followed by injection of low-pressure systems suffices to mitigate the consequences.

(3) A stuck-open relief valve, even with a complete loss of feedwater, is a controllable event. The consequences of a SORV can be mitigated by the injection of either FWCI, RCIC, HPCI, or HPCS. If none of these injection systems are available, the consequences can be mitigated by the injection of low pressure systems following manual reactor depressurization.

Dresden 2 was covered by this analysis.

2.6 Feedwater Line Break

This topic is not applicable to BWR's

3.0 Group III Events

Group III events are accidents caused by failure of a steam line.

3.1 Steam Line Break Inside Containment

3.2 Steam Line Break Outside Containment

3.3 Radiological Consequences

The potential radiological consequences of a steam line break outside of the containment is analyzed in Chapter 14 of the FSAR. For reactor coolant activity in the normal maximum range of approximately 2.4 uc/ml, and assuming a main steam line isolation valve closure time of 10.5 seconds FSAR table 14.2.8 gives the following results.

FSAR Table 14.2.8

| Distance from Plant (miles) | Passing Cloud Dose (rem) | Thyroid Dose (rem) |
|---|---|---|
| 1/2 | $6.6 \times 10^{-7}$ | $1.3 \times 10^{-3}$ |
| 1 | $3.3 \times 10^{-7}$ | $5.7 \times 10^{-4}$ |
| 5 | $2.6 \times 10^{-8}$ | $3.5 \times 10^{-5}$ |
| 9 | $8.8 \times 10^{-9}$ | $1.2 \times 10^{-5}$ |
| 12 | $5.2 \times 10^{-7}$ | $7.0 \times 10^{-6}$ |

At the technical specification limit of 10 uc/lm the calculated doses would increase by approximately a factor of 10, still well-below the limits of 10 CFR 100.

In the safety evaluation by the Divison of Reactor licensing, U.S. Atomic Energy Commission for Dresden 2 dated October 17, 1969 the main steam line break was also analyzed. The results of the AEC evaluation are shown below to be well with in the guidelines of 10CFR100.

AEC Evaluation

| Accident | Two-Hours Dose at Site Boundary (rem) | | 30-Day Dose at the Low Population Zone (rem) | |
|---|---|---|---|---|
| | Thyroid | Whole Body | Thyroid | Whole Body |
| Steam Line Break (20 second valve closure time) | 25 | 1 | 1 | 1 |

4. Group IV Events

4.1 Loss of Power to Station Auxiliaries

Loss of AC power to station auxiliaries can occur as a result of plant electrical problems or from problems in the external power distribution system. If auxiliary power is lost, the reactor will scram due to a loss of power to the scram solenoids. In addition, within a few seconds the steam line isolation valves close.

The isolation by closure of turbine stop and bypass valves due to loss of condenser vacuum is slower than isolation valve closure, because the coastdown of the main condenser cooling water is offset by the decreasing turbine steam flow which follows the decreasing reactor power. Therefore, the transient is less severe than that discussed in FSAR Section 11.2.3. "Main Steam Isolation Valve Closure".

At no time will loss of auxiliary power prevent a reactor scram since stored pneumatic energy and reactor pressure are the driving forces of the control rods. Also, the standby diesel generator and station batteries are available for emergency operation of reactor instrumentation, isolation valves, ECCS pumps, and other critical systems.

Heat removal is accomplished by action of the relief valves and isolation condensers. A post TMI assessment of this event is contained in Section 3.2.2 of Reference U.

## 5. Group V. Events

These events involve a decrease in core flowrate degrading core heat transfer capability.

### 5.1 Loss of Forced Coolant Flow

A loss of flow results from pump malfunctions or from automatic flow controller malfunctions. Decreased flow results in core heat up, which produces negative reactivity through void formation. Power is thus reduced, but the rate of decrease is slower than that of the flow, so a power/flow mismatch results.

If both pumps trip, natural circulation provides about 30% flow at a new lower power level. No reactor scram occurs, the power is reduced to a new lower steady state condition due to the negative void reactivity feedback.

The trip of one drive pump results in a flow decrease to about 60%. Power reduces to a new lower steady-state condition due to the negative void reactivity feedback. No reactor scram occurs.

The trip of one and both recirculation pumps is analyzed in Chapter 4 of the FSAR and the results of the analyses are presented in FSAR figures 4.3.10a and b, and 4.3.11 a and b. Analysis presented in the generic reload topical Reference C have concluded all loss of recirculation flow events to be nonlimiting, therefore no reanalysis is done on a cycle-to-cycle basis. A post TMI assessment of recirculation pump trips has been made and is presented in Section 3.2.2 of Reference U.

### 5.2 Pump Seizure

The plant response to a pump seizure is similar to pump trip but is somewhat more severe since the coast down is faster (instantneous stoppage of flow). Pump shaft break has not been specifically analyzed. However, it would be bounded by the instantaneous flow stoppage of the pump seizure accident. Only one pump is assumed to be lost during two-loop operation. For single-loop operation, the plant response is analyzed in Reference (V).

The pump seizure event was analyzed in Chapter 4 of the FSAR and the results are presented in FSAR figures 4.3.12a and b. However, General Electric states (see Reference D) that the water level swell in the vessel causes a turbine trip, and thus a reactor

scram on stop valve closure. This is consistant with the post TMI assessmented of recirculation pump seizure represented in Section 3.2.2 of Reference U. No fuel perforation is predicted from this event, which has been classified on accident.

## 6.0 Group VI Events

Group VI events are the result of control rod malfunction or mispositioning causing power increases.

### 6.1 Rod Withdrawal at Power

The uncontrolled withdrawal of a control rod due an operator error can result in increased core power, temperature, and pressure. The withdrawal is terminated by the rod block monitor, which inhibits outward rod movement when its flux setpoint is reached.

This transient is analyzed for each reload to assess the adequacy of the rod block setpoint, and is described in section 5.2.1.5 of Reference (c). The most recent analysis is presented in Reference (5).

### 6.2 Inadvertent Rod Withdrawal From Low Power

An uncontrolled rod withdrawal during startup from a low power conditon could result in a rapid increase in neutron flux. This increase could result in high heat genration in the fuel and possible fuel damage. Inadvertent continuous withdrawal of a single control rod from low power conditons could result from an operator error, or an equipment malfunction.

A rod block would be initiated by the reactor protection system when a high flux signal is detected by the source range, intermediate range or power range monitoring systems. If a high-high flux signal is detected by the intermediate or power range monitors a reactor scram results.

The rod worth minimizer will initiate a rod block if an out-of sequence rod is selected, or if a rod is withdrawn one notch beyond its programmed position.

### 6.3 Rod Drop

Since BWR control rods enter from the bottom of the core, a control rod drop removes the rod from the core inducing a severe positive reactivity insertion event on the plant. The amount and rate of reactivity inserted is limited by the rod worth minimizer and the rod velocity limiter. The power excursion is terminated by Doppler feedback and the high flux reactor scram. The peak enthalopy of the fuel rods following a control rod drop is limited to 280 cal/gram.

The generic reload topical reference (c) section 5.5.1 presents a bounding analysis which shows that a rod drop excursion does not exceed the design criteria. For later reloads, key parameters (rod drop worth, scram reactivity) are compared with those of the bounding analysis to ensure its applicability. If necessary, a reanalysis would be done using the methods described in the topicals. For the most recent reload cycles, the generic analysis was determined to be applicable.

## 6.4 Radiological consequences of a Rod Drop

Based on the analysis described in section 5.5.1 of reference (c) it was conservatively determined that 850 fuel rods in a 8x8 core would reach a fuel enthalpy of 170 cal/gm which is the limit for eventual cladding perforation. The original Dresden 2 FSAR section 14.2.1 predicted 330 fuel rod failures for a 7x7 core. (had the original FSAR used the methods described in reference (c) 660 7x7 fuel rod failures would have been predicted). If the conservative assumption is made that the fractional plenum activity for the 8x8 fuel is the same as for the 7x7 fuel, the resultant dose rate increase relative to the 7x7 analysis for the failure of 330 rods is (850/330) (49/63) = 2 times the 7x7 analysis. However even if the radiological exposures are increased by a factor of two above those reported in section 14.2.1.7 of the FSAR, the effects are still orders of magnitude below those identified in 10CFR100.

# 7.0 Group VII Events

## 7.1 Loss of Coolant Accident

A loss of coolant results from a rupture of primary system piping. A spectrum of breaks from a small leak which can be controlled by makeup flow to a complete double-ended rupture of a recirculation line was considered.

The loss of inventory produces core depresssurization, and decreasing water level in the vessel.

Reactor scram occurs on low water level. ECCS is automatically actuated on either low-low water level or high drywell pressure. A loss of offsite power is assumed simultaneously with the break as well as the worst single failure in the ECCS. The assumed simultaneous loss of offsite power, causes the reciruculation pumps to coast down and also a delay in initiation of ECCS, until the diesel generators get up to speed and are loaded. The emergency power supplies are auto-started on loss of power or ECCS initiation signal. The course of the accident depends on the break size and location. For a small break, failure of HPCI is the worst single failure, and core cooling is provided by the automatic depressurization system, LPCI, and the core spray systems. For these breaks the vessel depressurizes relatively slowly due to the small break size so the ADS automaticly actuates to reduce pressure so that the low pressure cooling system can function.

For larger breaks, failure of the LPCI injection valve is the most severe single failure since the vessel depressurizes faster and the HPCI system is not available due to the low system pressure. The core uncovers as inventory is lost through the break. After several seconds, injection flow from the core spray and LPCI systems refloods the core.

For Dresden 2, the highest PCT is reached for a complete rupture of the recirculation suction line. This break results in the longest time between hot node uncovering and reflood, and an early boiling transition time. These conditions lead to minimum heat transfer, and thus peak clad temperatures.

For long-term cooling, one LPCI subsystem or one core spray pump is required to maintain core cooling. Suction taken from the suppression pool, is pumped to the vessel, and runs back to the suppression chamber through the break. The containment cooling system is operated to cool the supression chamber water.

Loss of coolant accidents have been analyzed and approved by he NRC for Dresden Unit 2 in accordance with 10 CFR 50.46 and 10 CFR 50 Appendix K. The Analytical model is described in reference (W) and the results presented in reference (P). Reference (P) has been amended to include MAPLHGR curves and LOCA results for the reload fuel. These MAPLHGR curves have been incorporated into the Dresden Unit 2 Technical Specifications.

Dresden Unit 2 Cycle 8 LOCA performance is bounded by reference (P) because of the large reload of drilled lower tie plate fuel. The Unit 2 cycle 8 reload contains more drilled lower tie plate fuel than assumed in the LOCA analysis, which is conservative. The drilled fuel allows faster reflooding and the resulting MAPLHGR curves are less restrictive than for the reference (P) analysis of a core with more non-drilled fuel.

Following the Three Mile Island 2 accident small break LOCA's have been the subject of significant review by both the NRC and by owners of boiling water reactors, the results of the owner's review are contained in Section 3.1 of Reference (U) and conclude that BWR's are designed to protect the reactor core even in the event of the extremely degraded case of high pressure system failure.

## 7.2 Radiological Consequences of a LOCA

The potential radiological consequences of a loss of coolant accident is analyzed in chapter 14 of the FSAR. The FSAR assumes cladding perforation in 45 percent of the fuel rods releasing 0.45 percent of the noble gas inventory and 0.225 percent of the core halogen inventory.

In the safety evaluation by the Division of Reactor Licensing, U.S. Atomic Energy Commission for Dresden 2 dated Ocotober 17, 1969 the loss of coolant accident was also analyzed. The results of the AEC evaluation are shown below.

AEC Evaluation

| Accident | Two-Hour Dose at Site Boundary (rem) | | 30-Day Dose at the Low Population Zone (rem) | |
|---|---|---|---|---|
| | Thyroid | Whole Body | Thyroid | Whole Body |
| Loss of Coolant | 185 | 8 | 90 | 2 |

In both the FSAR and AEC analyses the dose resulting from a LOCA were shown to be well within the limits of 10CFR 100.

7.3 Radiological Consequeces of Failure of Small Lines

(Analysis performed by NRC - See Letter dated 6-30-81; D.M. Crutchfield to J.S. Abel, Director of Licensing)

8.0 Group VIII Events (Supplied by NRC Staff)

9.0 Group IX Events

These moderate frequency events resemble a small break LOCA. However, the consequences should not violate the fuel clad integrity limit.

9.1 Inadvertent Opening of a Safety/Relief Valve

The inadvertent opening of a safety or relief valve results in a reactor coolant inventory decrease and a decrease in reactor coolant system pressure. Neutron flux decreases due to additional void formation.

A failed open relief valve blows down to the suppression pool, below the water level. Safety valves discharge directly to drywell atmosphere.

On relief valve opening the pressure regulator senses the pressure decrease and partially closes the turbine control valves. No trip occurs, conditions stabilize at a power level near the initial power, and the feedwater system makes up the continuing loss of inventory.

Should the feedwater system become unavailable due to a single failure or loss of offsite power, the HPCI system could provide water. HPCI would be automatically actuated on low-low-water level.

If the pressure regulator fails to respond, the increased steam flow would cause a decrease in steam pressure, and close the MSIV's, as discussed in Section 1.3.

In the long-term, if the safety/relief valve fails to reclose, the plant continues to depressurize, and would be taken to cold shutdown so that repairs could be made.

Operator action following a failure to close of a relief valve is directed toward closing the valve, maintaining core cooling and controlling torus temperture.

The operator tries to close the relief valve by cycling the control switch from AUTO or (MANUAL) to off. If this is unsucessful, the operator initiates a manual scram. After the scram, water level and cooldown rate are monitored. As suppression pool temperature rises, the operator initiates torus cooling.

Following the TMI-2 accident analysis of the effect of stuck open relief valves as an inventory threatening event has been performed. These analyses are presented in Section 3.5.2.1.7 of Reference U and demonstrate the capability of BWRs to maintain adequate core cooling, even under severely degraded conditions.

## 10.0 Group X Events

### 10.1 Inadvertent Actuation of ECCS Resulting In Increase In Coolant Inventory

Actuation of an ECCS (HPCI) pump causes an increase in the flow of cool water to the reactor, this produces an increase in neutron flux due to void collapse. Since the flow rate from the HPCI pump is only a fraction of that of the feed pumps, this transient does not result in a reactor trip or a thermally limiting condition. The consequences of this event are bounded by the analysis for startup of an idle loop (Section 1.5) and loss of feedwater heating (Section 1.1). This is shown by analysis results presented in Reference N.

## 11.0 Group XI Events

Group XI events result from fuel assemblies being improperly loaded causing power anomalies.

### 11.1 Fuel Loading Error

A loading error for each reload core configuration is defined as:

1. A reload bundle is rotated or a bundle is inserted in an improper location; and

2. the error is not discovered in the subsequent core verification and the reactor is operated.

Historically, this event has been treated as an accident, and therefore did not impact plant operating limits. Since 1978, NRC has required that this event not violate the safety limit MCPR. To acommodate this more restrictive criterion, GE has developed new procedures which were reviewed and approved in 1978. Reference (X) discusses these developments and provides a statistical basis for no longer analyzing the mislocated bundle error. As a result plant-cycle specific misloaction errors have not been analyzed since January 1, 1981.

The results of the worst fuel loading error event was reanalyzed for the present Dresden 2 reload. This anlysis is contained in Reference (S), and demostrates compliance with the MCPR safety limit.

TABLE 1: REACTOR PROTECTION SYSTEM SCRAM SETPOINTS

| Function | Setpoint |
|---|---|
| High Neutron Flux | 120% Power |
| High Reactor Pressure | 1060 psig |
| High Drywell Pressure | 2 psig |
| Reactor Low Water Level | 143 inches above top of active fuel |
| Turbine Condenser Low Vacuum | 23 inches Hg vacuum |
| Scram Discharge Volume High Level | 50 gallons |
| Main Steamline High Radiation | 3 times noraml background |
| Main Steam Isolation Valve Closure | 10% closure |
| Generator Load Rejection | --- |
| Turbine Stop Valve Closure | 10% closure |
| Turbine Control Valve fast Closure | --- |
| Loss of oil pressure to turbine hydraulic control system. | 900 psig |
| Manual | -- |

TABLE 2 CONTAINMENT ISOLATION SET POINTS

| Function | Setpoint | Valves Affected |
|---|---|---|
| Low-low water level | 83 inches above top of active fuel | MSIV<br>Drain/Sample Lines |
| Main steam line high radiation | 3 times normal background | Isolation Condenser vent |
| Main steam line high flow | 120% of rated | |
| Main steam line tunnel high Temperature | 200°F | |
| Main steam line low pressure | 850 psig. | |
| Reactor low water level | 143 inches avove top of active fuel | Drywell vent/drains<br>Drywell $N_2$ makeup<br>Suppression chamber vent |
| High drywell pressure | 2 psig | Drywell vent to SBGTS |
| Reactor low water level | 143 inches above top of active fuel | shutdown cooling system<br>Cleanup demineralizer<br>Reactor head cooling |
| HPCI steam line high flow | 150 inches water | HPCI turbine steam supply |
| High temperture around HPCI steam line | 200°F | |
| Low reactor pressure | 350 psig | |
| High isolation condenser $\Delta p$ | 20 psi difference | Isolaton condenser steam supply and condensate return |
| High isolation condenser condensate flow | 32 inches water difference | |

TABLE 3. ENGINEERED SAFETY FEATURES INITIATION SIGNALS

| Function | Setpoint | Systems Started |
|---|---|---|
| Reactor low-low water level | 83" above top of acive fuel (+4";-0") | LPCI, Core Spray (in inconjunction low reactor pressure)<br><br>HPCI<br>Standby Gas Treatment<br>Diesel Generators |
| Reactor Low Pressure Permissive | 350 psig | Open Core spray and LPCI admission valves |
| Timer Auto Blowdown | 120 seconds | Initiates auto blow-down with coincident low-low water level and high drywell pressure |
| Low Pressure Core Cooling Pump Discharge Pressure Permissive | 100 psig | Permissive for initiating auto blowdown |
| Under voltage on EMERGENCY buses | | Diesel Generator |
| High Containment Pressure<br>Core Water Level | 1.5 psig<br>2/3 core height | Containment Spray Interlock |
| High Reactor Pressure | 1060 psig for 15 seconds | Isolation Condenser |

TABLE 4. INITIAL CONDITIONS AND ASSUMPTIONS

| | |
|---|---|
| Decrease in Feedwater Temperature | 100% power 145°F decrease |
| Increase in Feedwater Flow | 65% power 50% Flow |
| Increase in Steam Flow | 100% power 115% Steam demand |
| Startup of Inactive Loop | 60% power 50% Flow<br>Manual flow control |
| Inadvertent Closure of MSIV's | 100% power<br>Failure of relief valves<br>Failure of direct scram on vlave closure |
| Loss of External Load | Failure of load rejection scram<br>Failure of bypass<br>100% power |
| Turbine Trip | 100% power<br>Failure of bypass<br>Failure of scram on stop valve closure |
| Loss of Condenser Vacuum | 100% power<br>Instantaneous loss of vacuum |
| Steam Pressure Regulator Failure (Closed) | 100% power |
| Loss of Feedwater Flow | 100% power<br>No inventory makeup from HPCI |
| Steam Line Rupture Inside Containment | See LOCA |
| Steam Line Rupture Outside Containment | 100% power<br>Loss of offsite power |
| Loss of Power to Station Auxiliary | 100% power |
| Loss of Forced Flow/Pump Seizure | 100% power |
| Rod Withdrawal | Range of power levels<br>Maximum possible rod worth |
| Rod Drop | See Reference 0 for reactivity Parameters |
| Inadvertent Opening of Relief Valve | |
| Inadvertent Actuation of HPCI | See Increase in Feedwater Flow and Decrease in Feedwater temperature |

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F. NEDO-21145 Reload No. 2, March 1976.

G. NEDO-24034 Reload No. 3, October 1977.

H. NEDO-24160 Reload No. 4, November 1978.

I. Dresden Station Special Report No. 29, original and supplements 7/73-3/74.

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