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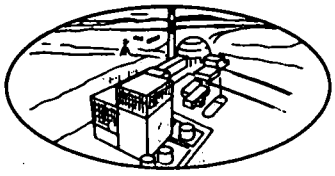
DRESDEN NUCLEAR POWER STATION

UNIT 2

PLANT DESIGN AND ANALYSIS REPORT

AMENDMENT NO. 2

ANSWERS TO AEC QUESTIONS



Commonwealth Edison Company

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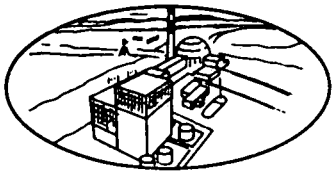


Commonwealth Edison Company

QUESTION

I. General

1. Throughout the Plant Design and Analysis Report the various sections and subsections are divided into 'Performance Objectives.' 'Bases.' and 'Description.' The manner in which each of the stated performance objectives can be met should be included in the description subsection. In this way it is possible to relate intent to practice, to consider whether or not a reasonable approach to a design problem exists, and to have knowledge of the degree of research and development that may be required to achieve some stated objective. A specific example would be Section IV-3, which relates to fuel design. The basic performance objectives were as follows: (1) no center melting of UO_2, (2) the Zircaloy-2 cladding is free standing, (3) sufficient plenum volume is available to prevent excessive internal pressures, and (4) cladding stresses shall be within design stresses. The description part of this section, however, does not support these objectives. This comment applies in various degrees throughout the Plant Design and Analysis Report, and we believe that such deficiencies should be corrected where they exist. In addition, for this example, in the "Description" part of Section IV-3, the design of the fuel assemblies, core geometry and structure are discussed. We believe that the performance objectives of the fuel assemblies and the core structure should also be stated.

ANSWER

The Plant Design and Analysis Report has been prepared in compliance with the specific requirements of Title 10, Part 50, Sections 50.34 and 50.35, Code of Federal Regulations. The report describes the proposed design of the facility, including the principal architectural and engineering criteria for design. The description is based on the design criteria for the facility as a whole and for those major component parts which are essential to the safe operation of the facility. Detailed evaluations have been made of the proposed measures and devices to prevent accidents which would create radioactive hazards or to protect against the consequences should accidents occur. The procedures for disposal of radioactive effluents has been described.

The site has been described in detail. Other information important to the operational safety of the plant, such as testing, surveillance, and procedures, has been provided where such information is currently available.

In specific instances it is recognized that information or data on certain of the new features may be lacking. The report identifies the development programs which will be undertaken with the intent of demonstrating the feasibility of additional or improved features of system design and procedures. Also, the report summarizes the major features or components for which further technical information will be provided as it becomes available.

It is the opinion of the applicant that the facility, its performance and operational characteristics, its specific safeguards features, and the site environment have been presented in sufficient detail to allow an evaluation of the adequacy of the various means to minimize the probability of danger from radioactivity to persons both on-site and off-site.

Notwithstanding the above, the following information is provided in response to the specific request for additional clarification of details with respect to the core design, fuel assemblies, and core structure.

A. Core Design

The following additional information describes the mechanical and thermal-hydraulic analysis which show that the core and fuel design will meet the stated performance objectives:

Fuel Mechanical Design Analysis

Fuel cladding stresses will be evaluated at the design overpower condition and also for overpressure or underpressure transients caused by malfunction of the reactor pressure controlling equipment to insure that the stress intensity levels given in Table IV-2 of the Plant Design and Analysis Report will not be exceeded. Stresses due to external pressure, internal pressure, bending effects at end plugs, and thermal stresses will be considered. Tensile properties used in stress analyses are based on test data of irradiated BWR fuel cladding for the applicable temperature. A fatigue life analysis will be performed based on the estimated number of temperature, pressure, and power cycles.

Preliminary stress analysis of the Dresden Unit 2 fuel rod and results of such a stress analysis for a fuel rod design essentially the same as the Dresden Unit 2 reference fuel rod design showed that the closest approach to the design performance objectives presented in Table IV-2 of the Plant Design and Analysis Report occurred at the connector location of the segmented fuel rod at the beginning of life where the calculated stress intensity was less than 90% of the design limit. During life less than 25% of the expected fatigue life would be consumed.

Core Thermal-Hydraulic Analysis

Core and fuel thermal-hydraulic design calculations involve detailed analyses of fuel rod operation and whole core performance.

The computer program used to calculate fuel rod temperatures utilizes the UO_2 thermal conductivity data as a function of temperature presented in GEAP-4624, a nominal pellet-to-clad gap conductance of 1000 Btu/hr-ft^2, and appropriate boiling heat transfer coefficients. UO_2 pellet thermal expansion characteristics and rate of UO_2 swelling due to irradiation are calculated. Thermal effects of irradiation including reduction in local power peaking factor due to U-235 depletion, buildup of Pu near the surface of the pellet, and effect of gap width and gas composition on gap conductance, are considered in assuring that the thermal-hydraulic performance objectives are met.

In the computer program used to analyze the thermal and hydraulic characteristics of the reactor core as a whole, the geometric, hydraulic, and thermal characteristics of the core design are represented, including number of fuel assemblies in each orifice zone of the core, fuel assembly dimensions, friction factors and flow restrictions, and the flow characteristics of the fuel orifices, inlet plenum region of the reactor, and leakage flow paths around the fuel channels.

Individual cases have been analyzed by providing reactor power, flow, inlet enthalpy, and appropriate power distribution factors as input to the above computer program for the reference design. The output of the program includes the calculated flow distribution among the several channel types, and a detailed analysis of the heat fluxes, steam quality, void fraction, and MCHFR at as many as 20 axial nodes for the average and peak power fuel assemblies in each orifice zone.

Comparisons of the analytical models used with fuel assembly design details such as fuel rod-to-rod and rod-to-channel clearances and spacer configurations have been made to insure that the above computer programs adequately represent the actual core and fuel design, and that design correlations, such as for CHF, are applicable.

Results of the above analyses show that with the reference core design, power peaking factors, and rated recirculation flow, the performance objectives referred to in paragraph IV-4-1 of the Plant Design and Analysis Report are met. The calculated UO_2 center temperature at the design overpower peak heat flux, 418, 800 Btu/hr-ft^2, is 4500°F, and the calculated MCHFR is greater than 1.5.

B Performance Objectives of Fuel Assemblies

The performance objectives of the fuel assemblies are:

(1) To provide proper positioning of the fuel rods, proper distribution of the coolant flow, and proper heat transfer characteristics such that the performance objectives of the fuel rods (see paragraph IV-3.1 and IV-4.1) shall be met for the design life of the fuel.

(2) Provide rigidity and protection for the fuel rods during handling and provide guide surfaces for the control rod rollers.

Performance Objectives of the Core Structure

The core structure components are designed to accommodate the loadings applied during normal operation and routine maneuvering transients considering both stress and deflection. Deflections are limited so that the normal functioning of the components under these conditions will not be impaired. In general, where deflection is not the limiting factor, the reactor internal structure design stress criteria is in accordance with the design criteria of ASME Boiler and Pressure Vessel Code, Section III.

The loading conditions which occur during an accident are also examined to determine the effect on the core structural components.

The core shroud, shroud support structure, and jet pump body which comprise the chamber within the reactor vessel around the core are designed to maintain the reflooding capability following the loss of coolant accident.

The reactor internals components are designed to preclude failure which would result in any part being discharged through the steam line in the event of a steam line break outside of the steam line isolation valve.

The structural components which guide the control rods are examined considering the loadings which would occur in either a steam line break accident or loss of coolant accident. It is the objective of the design of the core structural components that deformations produced by the accident loadings will not prevent insertion of control rods.

QUESTION

I. General (Continued)

2. The plant design and analysis report is based upon a reference design which it is recognized is subject to some change and modification as plant design progresses. It would be helpful for analysis purposes, however, to have a list of those items which are now considered firm. This would be expected to include, for example, that G. E. bottom-entry control rods will be used, that jet pumps will be used, that Zircaloy-clad UO_2 fuel will be used, that B_4C poison material in control rods will be used, etc.

ANSWER

The Plant Design and Analysis Report for Dresden Unit 2 has been prepared on the basis that the data and performance characteristics represent a "reference design", and the data are related to and have been analyzed for operation of Unit 2 for the reference design thermal output of 2255 Mwt. It is intended that as various phases of design are completed, the report will be revised by amendment to reflect design changes. To the extent practical, such changes will be made within the intent of the Principal Architectural and Engineering Criteria for Design as stated in Section II, and within the scope of the Performance Objectives as stated in other sections. In this manner, the report will reflect the current design when the project is completed.

Thus the Unit, its components, and its performance characteristics as described in the Plant Design and Analysis Report, as amended, represents the current firm design.

Many of such plant features are presented in Table I-1, Page I-4-1, and Table IV-1, Page IV-1-1 of the Plant Design and Analysis Report. Other firm features of the Unit are described in the various other sections of the report. For convenience of review, the principal of such features appropriate to the application are summarized below. Other items not listed are discussed in the various sections of the Plant Design and Analysis Report.

Principal Design Features of Dresden Unit 2

Plant

Location	Dresden Site, County of Grundy, State of Illinois
Size of Site	953 Acres
Site and Plant Ownership	Commonwealth Edison Company

Reactor	
Thermal Output	2255MW
Core Operating Pressure	1000 psig
Total Core Flow Rate	98×10^6 lb/hr
Steam Flow Rate	8.62×10^6 lb/hr
Core	
Circumscribed Core Diameter	189.7 in.
Fuel Assembly	
Number of Fuel Assemblies	724
Fuel Rod Array	7×7
Cladding Material	Zircaloy - 2
Fuel Material	UO_2
Active Fuel Length	144 in.
Cladding Outside Diameter	0.570 in.
Cladding Thickness	0.036 in.
Fuel Channel Material	Zircaloy - 2
Control System	
Number of Movable Control Rods	177
Shape of Movable Control Rods	Cruciform
Pitch of Movable Control Rods	12.0 in.
Control Material in Movable Control Rods	Boron carbide granules
Type of Control Drives	Bottom entry, hydraulic actuated
Number of Temporary Control Curtains	324
Control of Reactor Power Output	Movement of control rods and variation of coolant flow rate

Core Design Operating Conditions	
Power Density	36.7 Kw/liter
Heat Transfer Surface Area	63,527 sq. ft.
Average Heat Flux	116,000 Btu/hr - ft^2
Maximum Heat Flux	349,000 Btu/hr - ft^2
Minimum Critical Heat Flux at Overpower equal to or greater than	1.5
Core Average Voids of Coolant Within Assm.	37%
Core Average Exit Quality of Coolant Within Assemblies	9.9%
Design Power Peaking Factors	
Total Peaking Factor	3.0
Additional Allowance for Overpower	1.2
Nuclear Design Data	
Initial Average Fuel Enrichment	2.0%
Water/UO_2 Volume Ratio	2.38
Excess Reactivity of Clean Core (Uncontrolled) at 68°F	0.26 Δk
Total Worth of Control	0.30 Δk
Reactivity of Core with all Control Rods In	0.96 k_{eff}
Worth of Standby Liquid Control System	0.17 Δk
Reactor Vessel	
Inside Diameter	20 ft. - 11 in.
Overall Length	68 ft. - 0 in.
Design Pressure	1250 psig

Coolant Recirculation Loops	
Location of Recirculation Loops	Containment Drywell
Number of Recirculation Loops	2
Pipe Size	28 in.
Pump Capacity	45,000 gpm each
Number of Jet Pumps	20
Location of Jet Pumps	Inside reactor vessel
Primary Containment	
Type	Pressure Suppression
Design Pressure of Drywell Vessel	62 psig
Design Pressure of Suppression Chamber Vessel	62 psig
Design Leakage Rate	0.5% free volume per day
Secondary Containment	
Type	Reinforced concrete and steel superstructure with metal siding
Internal Design Pressure	0.25 psig
Inleakage Rate	100% free volume per day at 0.25 in. water negative pressure
Structural Design	
Seismic Resistance	0.1g
Sustained Wind Loading	110 mph
Control Room Shielding	Dose not to exceed 500 mrem in 8 hours under accident conditions assuming a full core melt
Station Electrical Systems	
Number of Incoming Power Sources	7
Separate Power Sources Provided	2 Auxiliary Transformers 1 Diesel Generator 1 Station Battery

Reactor Instrumentation System

Location of Neutron Monitor System	In-Core
Ranges of Nuclear Instrumentation	
Startup Range	Source to 0.01% rated power
Intermediate Range	0.0001% to 10% rated power
Power Range	1% to 125% rated power

Reactor Protection System

Number of Channels in Reactor Protection System	2
Number of Channels Required to Scram or Effect Other Protective Functions	2
Number of Sensors per Monitored Variable in Each Channel	2
Method to Prevent Unwarranted Withdrawal of Control Rods	Automatic Interlocks

Waste Disposal Systems

Liquid, gaseous, and solid waste disposed of in accordance with the requirements of 10CFR20

Other Engineered Safeguards - Summary of Systems and Functions

2 Core Spray Systems - To cool the core under assumed loss of coolant accident; capability to reflood the core following coolant loss.

2 Containment Cooling Systems	To remove energy associated with full core melt subsequent to assumed loss of coolant accident.
Rod Worth Minimizer	To prevent withdrawal of control rods having a worth in excess of 0.025 Δk.
Rod Velocity Limiter	To limit the free fall of a control rod from the core to approximately five feet per second.
Control Rod Drive Thimble Support	To prevent a control rod drive mechanism from falling away from the reactor pressure vessel in the unlikely event of a failure of a drive thimble.
Flow Restrictors	A construction in each main steam line to reduce rate of blow-down in event of postulated severance of the main steam line.

Isolation Condensers	To avoid overheating of the reactor fuel in the event that reactor feedwater capability is lost and other normal heat removal systems which require a-c electrical power for operation are not available.
Isolation Valves	To effect reactor containment automatically when required under accident conditions.
Core Support Structure	For Core reflooding to provide a means in conjunction with the jet pump configuration, to permit reflooding the core subsequent to a postulated loss of coolant accident.
Inerting System for Primary Containment	To provide an inert atmosphere in the primary containment system to preclude a hydrogen-oxygen reaction subsequent to a postulated coolant loss accident and full core melt-down.
Standby Gas Treatment System	To provide a means for removal of particulates and halogens from reactor building air under accident conditions prior to discharge of the filtered air through the 300 foot stack. Also provides a means for maintaining the reactor building at a negative pressure so that leakage is into the reactor building and thus prevents ground level release of building air under accident conditions.

Sizing of Systems to Accommodate Loss of Coolant Accident

The overall design of the Unit is sized to accommodate a loss of coolant accident arising from the assumed failure of any pipe inside the drywell, and the subsequent full core meltdown and associated release of hydrogen from a metal-water reaction.

QUESTION

II. Plant Design Characteristics

1. The manner in which the fuel rod internal pressure is calculated including those assumptions concerning the source of gaseous fission products should be set forth. In addition, comparable information for the control rod poison tubes should be provided.

ANSWER

Fuel rod internal pressure is due to the helium which is backfilled at one atmosphere pressure during rod fabrication, volatile content of the UO_2, and the fraction of gaseous fission products which are released from the UO_2. A quantity of 1.35×10^{-3} gram moles of fission gas are produced per MW-day of power production. It is assumed that 0.5% of the fission gas produced will be released from any UO_2 volume at a temperature less than 3000°F, 20% from any UO_2 volume between 3000°F and 3450°F, and 100% from any UO_2 volume above 3450°F. For a fuel rod similar to the reference design for Dresden Unit 2, the resultant release fraction is 27% in the hottest fuel rod and the resultant total plenum pressure is 1715 psia at end of life at operating temperature.

Design stress intensity limits for control rod poison tubes are given in the following table:

Categories	Stress Intensity Limits in Terms of Yield Strength (S_y)	Stress Intensity Limits in Terms of Ultimate Strength (S_u)
General Primary Membrane Stress Intensity	2/3 S_y	1/2 S_u
Local Primary Membrane Stress Intensity	S_y	3/4 S_u
Primary Membrane Plus Bending Stress Intensity	S_y	3/4 S_u
Primary Plus Secondary Stress Intensity	2 S_y	1.5 S_u

A stress analysis was performed of a control rod similar to that for Dresden Unit 2. It was assumed that all control rod neutron absorptions were in B-10. Based on experimental data a value of 18% was used for the fraction of He gas generated by the B-10 (n, α) Li-7 which is released from the B_4C to cause internal pressure within the poison tubes. When the nuclear life due to depletion of B-10 was reached, the internal pressure in the most highly exposed poison tube was 13,100 psi, and the resultant maximum general primary membrane stress was less than 50,000 psi compared to a design limit of 51,500 psi.

QUESTION

II Plant Design Characteristics (Continued)

2. We understand that some significant amount of analytical work is being performed to evaluate and analyze the characteristics and stability of the reactor nuclear-thermal-hydraulic system. A description of this analysis should be provided including the parameters that are considered and the manner in which the results of this analysis are related to safe operation of the facility. If the stability analysis includes a margin of stability, or threshold of instability, provide a discussion of the basis for selection of such a margin. Define "strong negative reactivity feedback," "severe transient conditions," and "requirements for over-all plant nuclear-hydrodynamic stability" as related to the above in numerical terms if possible.

ANSWER

The following summarizes the interacting parameters pertaining to plant stability, describes the analytical models for stability analysis, and delineates the stability criteria currently being used in design of the systems.

Introduction

A boiling water reactor (BWR) plant is made up of many interacting processes and their control systems. A process may be defined as "any phenomenon which shows a continuous change in time; "e.g., the boiling of water in the reactor core. The process may also be self-regulating; i.e., it exhibits a negative feedback effect. In a BWR, when a control rod is withdrawn, core power increases which causes increased boiling. The increased boiling increases the steam (void) volume in the core causing decreased neutron moderation, which is the equivalent of adding negative reactivity to counteract the positive reactivity of the withdrawn control rod. Thus, the rise in core power is limited by the negative feedback effect of the steam (void) volume.

Whenever there is negative feedback, whether it be inherent self-regulation in a process or added to the process by a control system, the potential for instability must be considered. There are many definitions of stability, but for feedback processes and control systems, the following definition may be used: A system is stable if, following a disturbance, it settles to a steady, non-cyclic state. A system may also be defined as being stable even if oscillatory, provided the oscillations are less than a prescribed magnitude. Instability, then, is a continuous departure from a final steady-state value, or it may be a continuous, greater-than-prescribed oscillation about the final steady-state value.

The mechanism for instability can be explained in terms of frequency response. Consider a sinusoidal input to a feedback control system which for the moment has the feedback disconnected. If there were no time lags or delays between input and output, the output will be in phase with the input. Connecting the output so as to subtract from the input (negative feedback or 180° out-of-phase connection) would result in stable closed loop operation. However, natural laws would cause phase shift between output and input and should the phase shift reach 180°, the feedback signal would be re-inforcing the input signal rather than

subtracting from it. If the feedback signal were equal to or larger than the input signal (loop gain equal to one or greater), the input signal could be disconnected and the system would continue to oscillate. If the feedback signal were less than one (loop gain less than one), the oscillations would die out.

This suggests that counteracting phase shift or decreasing loop gain can stabilize a feedback process or control system. In the BWR boiling process, instability can occur if the negative void reactivity feedback were sufficiently large; hence, in core design, the magnitude of the void reactivity coefficient is one of the important parameters.

It is possible for an unstable process to be stabilized by the addition of a control system. For example, it should be possible for an automatic rod control system using in-core neutron monitors to stabilize a nuclear reactor which exhibits spatial xenon instability. In general, however, it is preferable that a process with inherent feedback be designed to be stable by itself before it is combined with other processes and control systems.

Stability Considerations in BWR Design

The analysis and synthesis of large systems, such as a BWR power plant, is lengthy and time-consuming because of the large number of variables. Modeling or simulation of the plant on an analog and/or digital computer is the most effective method for evaluating performance with respect to dynamic behavior, stability, and safety. Special analytical models are also used to determine stability of important loops or processes within the system.

The block diagram of Figure II-2-1 indicates the extensiveness of the simulation of a BWR. It also indicates the parameters considered in the analysis. Many of the blocks are extensive systems by themselves. For example, the recirculation flow model is shown in expanded block diagram form in Figure II-2-2. The equations describing the processes are, in general, non-linear, coupled differential equations, approximately 100 in number. There are approximately an equal number of algebraic equations. In the complete model, the various processes and control systems are interlinked and although each process and control system may be stable when analyzed singly, it is necessary to examine the stability of the overall system. Final settings of control systems are therefore made on the plant model. With this complete "plant," all potential disturbances pertaining to safety, transient response, and stability can be investigated.

One of the special analytical models is that of the nuclear-thermal-hydraulic loop (also called power loop or power-void reactivity feedback loop) shown in Figure II-2-3. Frequency response analysis of this model on a digital code gives gain and phase margins. Comparisons with extensive experimental loop tests and operating reactor plant tests (rod oscillation) have established design criteria of 13 db gain margin and 55° phase margin. This means that a BWR will have adequate stability when it has these gain and phase margins when analyzed according to this model. A 6 db gain margin, in control systems theory, is sufficient for good dynamic performance. A 13 db criteria allows for model inaccuracies--some experiments on the test loop have shown a 6 db difference from the theoretical model.

In addition to having hydraulic-loop stability, a large core must also be strongly coupled hydrodynamically. That is, there should be a large margin against the occurrence of flow oscillations between core regions. Selective orificing of the different channels in the core (calculated by a multi-channel power-hydraulic flow

distribution digital code) results in such stability as has been shown by experience on large operating plants as Dresden Unit 1 and SENN.

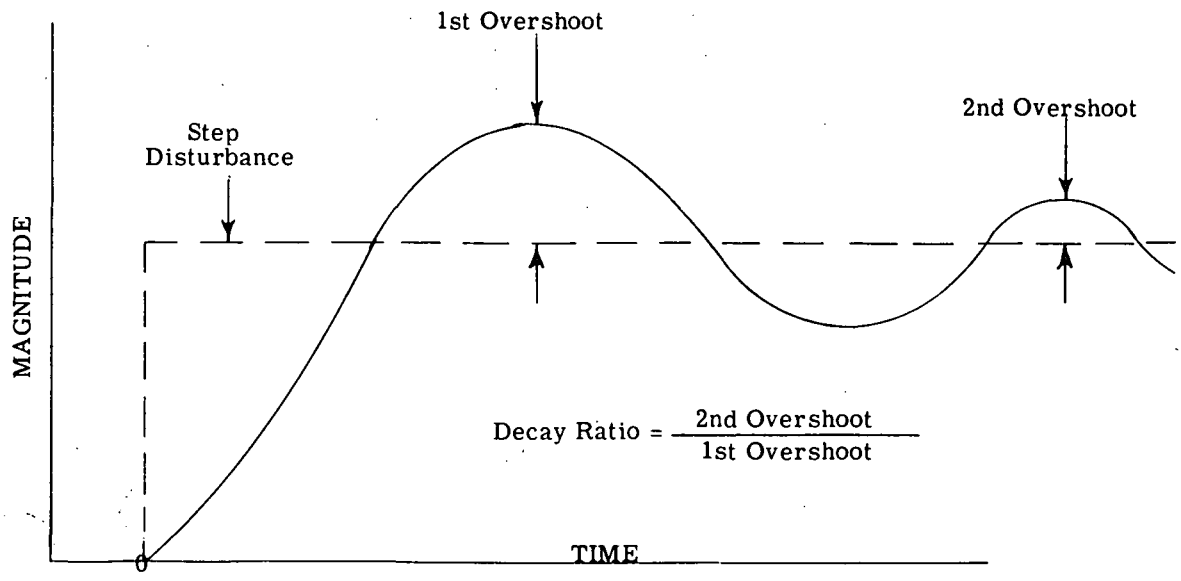
Another facet of large core stability is the response to xenon transients. A digital computer code is used to calculate xenon transient behavior in large cores. The design basis results in a power coefficient of reactivity which is more negative than $-0.01\frac{\Delta k/k}{\Delta p/p}$ which value ensures strong damping of xenon-produced power shifts. The combined reactivity effects of moderator temperature, moderator voids, and Doppler is called the power coefficient, typically, the power coefficient at beginning of core life is approximately $-0.04\frac{\Delta k/k}{\Delta p/p}$ and approximately $-0.02\frac{\Delta k/k}{\Delta p/p}$ at end of core life.

Some control loops (e.g. pressure regulating loop) are also analyzed separately, usually on an analog computer, to determine the control system characteristics prior to incorporation into the complete plant model.

It is possible to neglect a process as part of the overall system if its time response is very much slower or faster than the rest of the system. The xenon transient is so slow it has been omitted from the complete model. In contrast, boiling noise is very fast and likewise is not simulated.

Plant Stability

The most meaningful way to check plant stability is to disturb the various plant parameters and evaluate the subsequent system responses. It has been established in process control technology that response of important variables to a step disturbance should show a decay ratio of 1/4 or less for good stability; i.e., a gain margin of approximately 6 db (see the following response curve for definition of decay ratio).



A 6 db gain margin indicates, for the single loop equivalent of the system, a loop gain of 0.5 when the open loop phase shift is 180°. That is, when the feedback is in phase with the input (180° phase shift plus 180° for negative feedback connection), its magnitude is only half of its source, thus unable to amplify or diverge, which leads to a rapid die-out of any oscillation.

The perturbations which are imposed on the model are those which affect reactor power strongly and which can also be imposed on the actual operating plant so as to check the adequacy of the model. These perturbations which are imposed separately are:

1. A pressure set point change of approximately ± 5 psi.

2. A control rod position change equivalent to a local power change of five to ten percent of point.

3. A recirculation flow change equivalent to a power change of five to ten percent of point.

4. A subcooling change (by changing feedwater flow) equivalent to a power change of five to ten percent of point.

By disturbing the actual plant in this manner during the initial operation, beginning at the lowest practical power, any possible tendency to become unstable can be recognized long before actual instability occurs. The manifestation would be a more oscillatory plant response to a disturbance as an unstable condition was approached. At no time should there be a question of safety. Any significant overshoot of a responsive plant variable like neutron flux would be terminated by a safe shutdown. The most severe transient which the reactor scram system handles safely is a turbine trip at rated power. The neutron flux for this transient has a rate of rise of approximately 100% per second. In comparison, the most oscillatory condition noted so far in the models - showed a maximum neutron flux rate of rise of approximately 8% per second. Therefore, even if an oscillatory situation should occur in a large BWR, the effect it would have on neutron flux would be very mild compared to the effect of many of the normal maneuvers such as a turbine trip for which the control system has been designed. Since the scram circuit has been demonstrated as capable of quickly terminating the neutron flux transients associated with various plant maneuvers, it can certainly quickly terminate any oscillation in neutron flux since it is so much milder. Therefore, the only consequence of significance due to an oscillatory behavior would be a curtailment in the range of operation for the power reactor.

Summary

1. As complete a plant model as is practical is simulated on an analog and/or digital computer to evaluate performance with respect to stability, transient behavior, and safety.

2. Special models are used for separate analysis of important loops or processes within the system. Among these are:

   a. A linearized nuclear-thermal-hydraulic loop model which is analyzed on a frequency response basis by a digital code to give phase and gain margins.

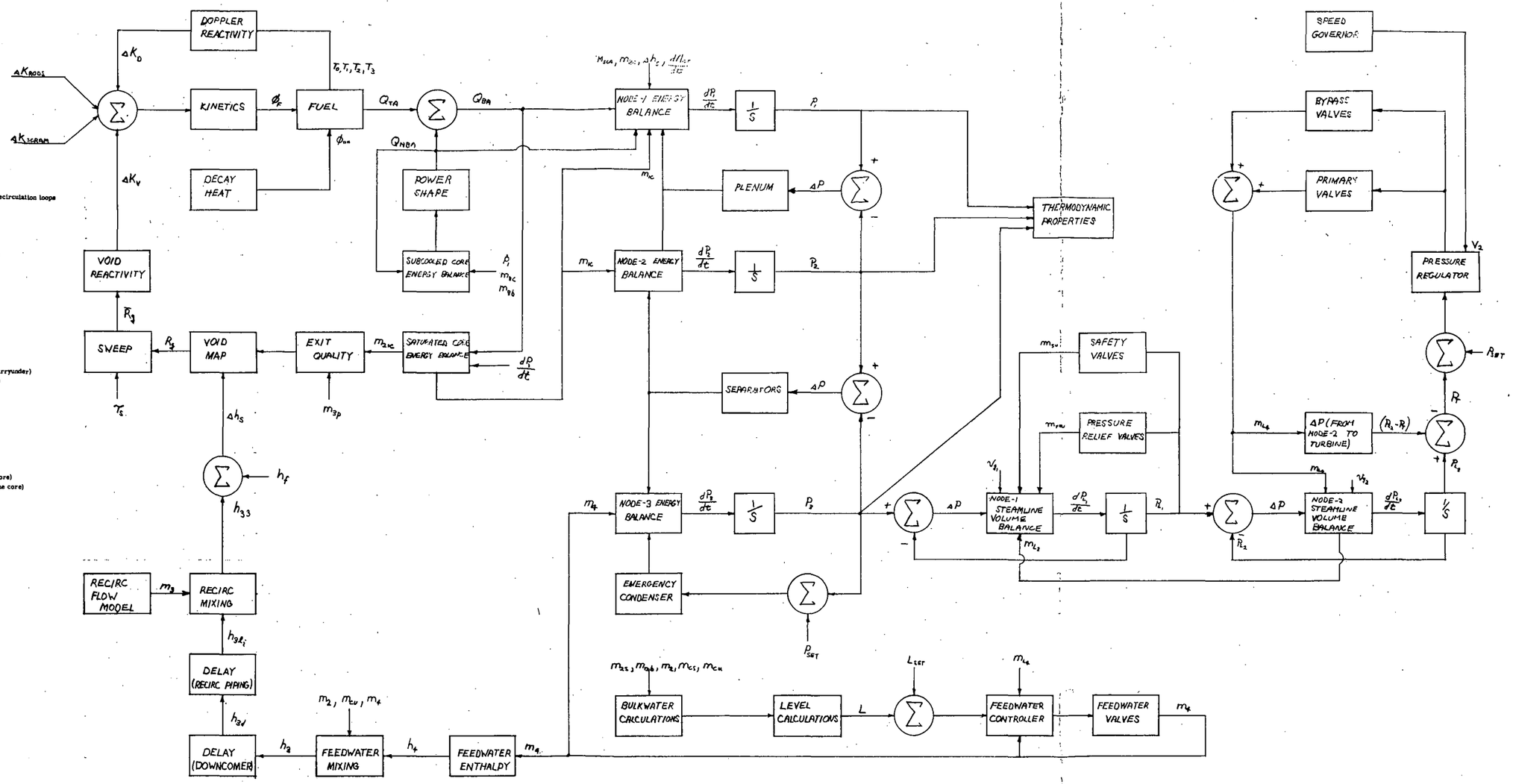
   b. A multi-channel power-hydraulic flow distribution code for specifying the orificing of the different channels in the core to achieve strong hydraulic coupling between core regions.

   c. A digital computer code to calculate xenon transient behavior in large cores.

   d. Control loop models such as the pressure regulating control system.

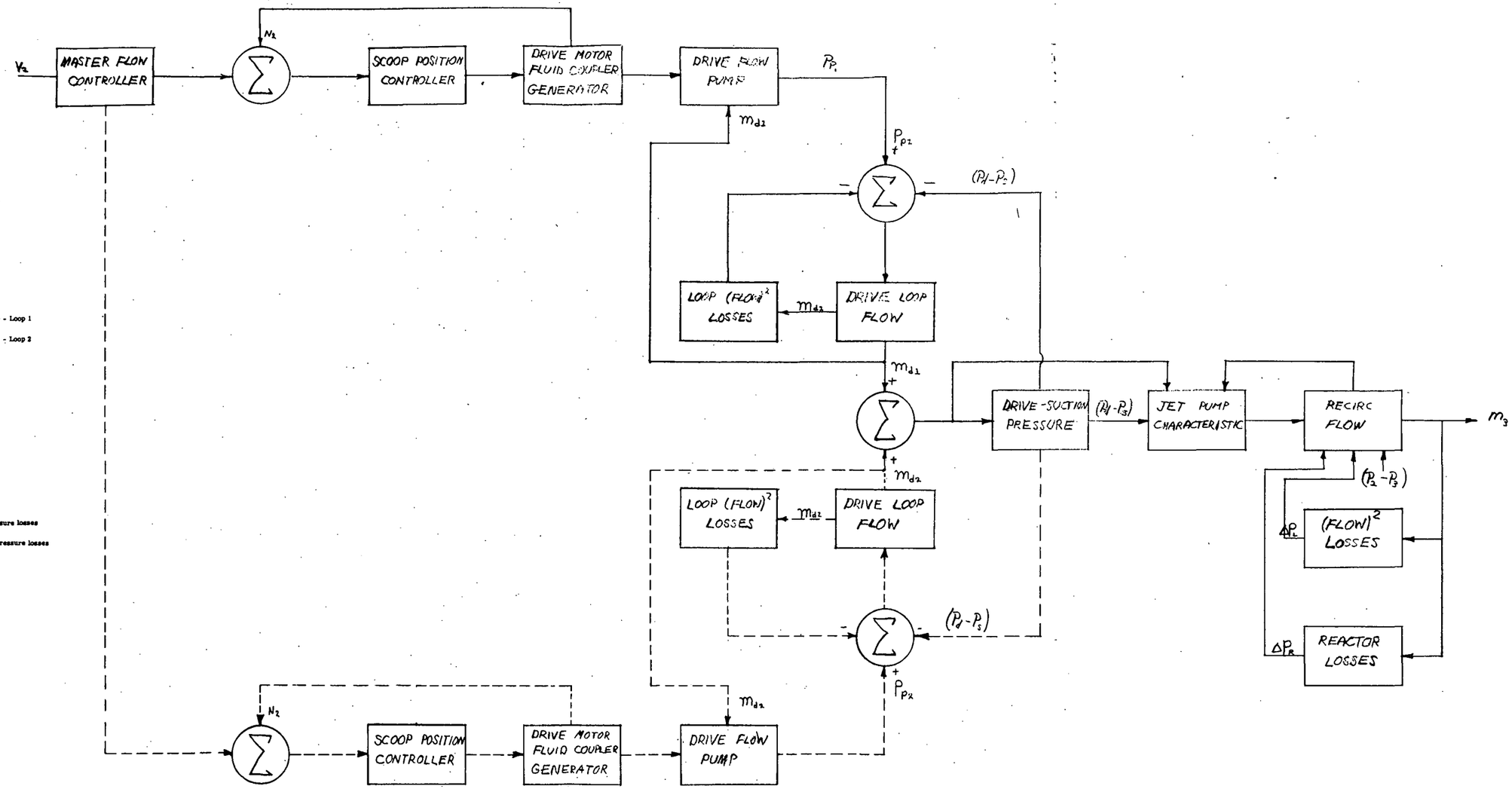
3. Definite design criteria with respect to the stable performance of the nuclear power plant exist. These criteria are based on standards accepted industry-wide and are evaluated by techniques which are standard in the control industry.

4. The securing of the power plant in the event of any possible although not expected oscillation is well within the capability of the reactor safety system.



h_3 = Enthalpy of downcomer fluid
h_4 = Feedwater enthalpy
h_{33} = Core inlet enthalpy
h_{3d} = Delayed enthalpy - enthalpy of fluid at entrance to recirculation loops
h_{3ll} = Enthalpy of fluid in i^{th} recirculation loop.
L = Reactor vessel water level
L_{set} = Level set point
M_{ct} = Total mass of liquid and steam in the core
M_{sca} = Mass of subcooled liquid in the active core
m_2 = Saturated liquid mass flow rate from Node 3 (dome)
m_3 = Recirculation mass flow rate
m_4 = Feedwater mass flow rate
m_{1c} = Core exit - steam mass flow rate
m_{2s} = Separator, saturated water mass flow rate
m_{3b} = Mass flow rate from subcooled region of the core
m_{3c} = Core, mass flow rate
m_{3p} = Core exit, mass flow rate
m_{12b} = Condensation rate - carryunder steam
m_{21c} = Steaming rate
m_{12} = Steam flow from Node 1 to Node 2 of the steam line
m_{14} = Steam flow to the turbine
m_{cs} = Saturated vapor mass flow rate into bulk - water (carryunder)
m_{cu} = Saturated vapor mass flow rate out of Node 3 (dome)
m_{prv} = Pressure relief valve mass flow rate
m_{sv} = Safety valve mass flow rate
P_1 = Pressure - Node 1 (core)
P_2 = Pressure - Node 2 (upper plenum)
P_3 = Pressure - Node 3 (dome)
P_{L1} = Pressure - steamline Node 1
P_{L2} = Pressure - steamline Node 2
P_T = Turbine pressure
P_{set} = Pressure set point
Q_{ba} = Boiling active heat (heat to safurated region of the core)
Q_{nba} = Nonboiling active heat (heat to subcooled region of the core)
Q_{ta} = Total active core heat
T_0 = Fuel temperature - center of fuel rod
T_1 = Fuel temperature - Radius r_1 of fuel rod
T_2 = Fuel temperature - Radius r_2 of fuel rod
T_3 = Fuel temperature - Radius r_3 of fuel rod
V_2 = Speed governor demand signal
v_{g1} = Specific volume - Node 1 steamline
v_{g2} = Specific volume - Node 2 steamline
Δh_s = Reactor subcooling
Δk_d = Doppler reactivity
Δk_{rods} = Control rod reactivity
Δk_{scram} = Scram reactivity
Δk_v = Void reactivity
ϕ_{dh} = Decay heat
ϕ_f = Neutron flux which produces heat in the fuel
τ_s = Sweep time constant

Figure II-2-1. Block Diagram - Single Cycle, Forced Circulation BWR



m_3 = Recirculation mass flow rate

m_{d1} = Mass flow rate - jet pump drive line - Loop 1

m_{d2} = Mass flow rate - jet pump drive line - Loop 2

N_2 = Flow control generator speed

P_2 = Pressure Node 2 - upper plenum

P_3 = Pressure Node 3 - dome

P_d = Jet pump drive pressure

P_s = Jet pump suction pressure

P_{p1} = Drive pump pressure - Loop 1

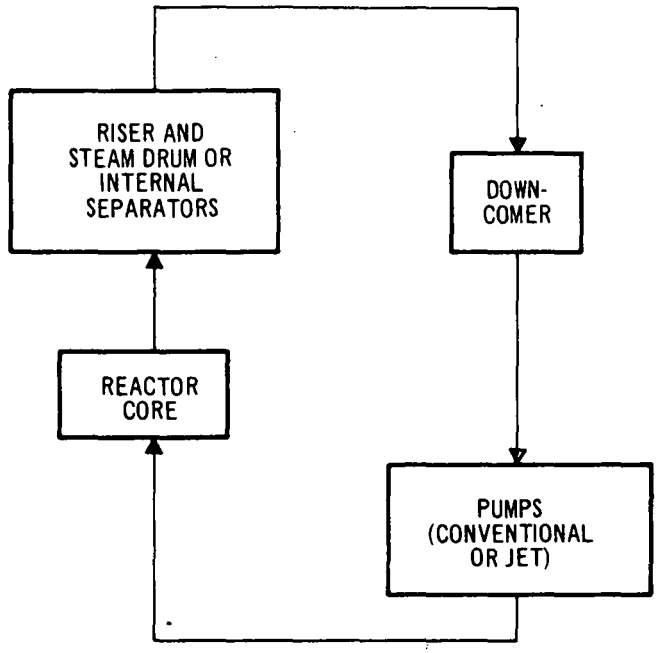
P_{p2} = Drive pump pressure - Loop 2

V_2 = Speed governor demand signal

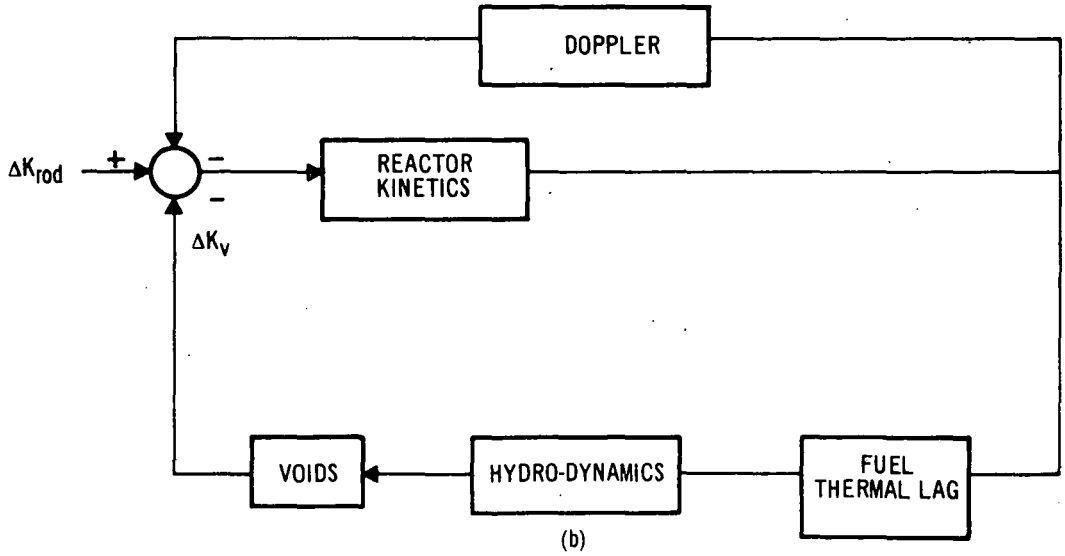
ΔP_L = External loop friction and local pressure losses

ΔP_R = Core friction, local, and elevation pressure losses

Figure II-2-2. Block Diagram - Recirculation Flow Model - Jet Pumps



(a)



(b)

Figure II-2-3. Typical Nuclear Thermal - Hydraulic Loop Models

QUESTION

II Plant Design Characteristics (Continued)

3. A discussion of the jet pumps which will be used for recirculation flow has been included in Appendix A of the Plant Design and Analysis Report. Although this is a reasonable description of jet pumps and their characteristics, per se, it does not describe the manner in which the stability of these pumps has been evaluated for use in the reactor. In order for an evaluation of this part of the application to be made, we would like to have:

   a. A description and the results of stability analyses relative to the jet pumps made to date.

   b. A discussion of test operation to be made when the pumps have been installed.

   c. The degree of pump instability which could be tolerated yet maintaining safe operation of the facility.

ANSWER

a. There have been extensive analyses conducted to predict the stability of the important nuclear-thermal-hydraulic loop of proposed product line reactors utilizing jet pumps to supply coolant flow. Tests have also been conducted with a jet pump recirculation system. These tests were carried out at pressures and temperatures duplicating reactor service conditions.

The nuclear-thermal-hydraulic analyses utilize the model discussed in the answer to the question II-2. Results showed that reactors of Dresden Unit 2 design using jet pumps meet established design stability requirements of gain and phase margins at design power. Gain and phase margins for optimum performance of jet pumps can be made equivalent to the margins obtained with non-jet pump reactors by means of proper increase in orificing of the reactor core to offset the lower resistance characteristics of the jet pump hydraulic loop.

Loop tests of a jet pump recirculation system were conducted at the General Electric Steam Separation Test Facility at the Moss Landing Power Station of the Pacific Gas and Electric Company. Simulation of reactor hydraulic conditions can be accomplished at this facility. One series of tests was made on a multiple-unit assembly of four one-third linear scale jet pumps in combination with a recirculation loop, operating as a system under nominal reactor service conditions. The major objective was to determine whether this hydraulic system would exhibit a "manifold instability" behavior within required operating ranges. The tests assessed the operational hydraulic stability of the system as constructed and provided data for inferring the behavior of reactor installed jet pump recirculation systems.

No evidence of "manifold instability" performance was ever observed, even under contitions where efforts were made to drive the system into poorly damped or unstable performance. It was also observed that a member jet pump on a common-connected system gave steadier instrument readings

than when operating alone and the efficiency of a member jet pump remained relatively constant regardless of performance distortions imposed upon companion units.

The system tested is believed to be more sensitive to instability compared to Dresden Unit 2 or other reactors of General Electric design. The design of the hydraulic loop of the system tested is such that the recirculation pump flow is artificially supplied from the jet pump discharge plenum. This results in a positive feedback influence on drive flow. In contrast, the drive flow supply of the actual reactor design is taken from the common jet pump suction flow sump which presents no such positive feedback effect.

In relation to "manifold instability", the test system has but four jet pumps operating in parallel as contrasted to 10 per manifold as proposed for the design of Dresden Unit 2. With a smaller number of jet pumps, disturbances presented to one member unit will have more effect on the manifold pressure, and thus on the total system. Also, the tested system employed jet pumps having suction flow/drive flow ratios of unity; those for Dresden Unit 2 have ratios of approximately two. Pressure differential between manifold and suction sump in the former case is considerably smaller than for the latter case. "Manifold instabilities" are expected to be sensitive to the extent of this differential, with increasing differentials lying in the direction of increased stability.

Thus, the several conditions surrounding the test situation are closer to any potential "instability threshold" than for a Dresden Unit 2 reactor system. Test demonstration of total absence of any such instability mode of operation thus provides high assurance that instability will also be absent from the reactor jet pump recirculation system.

Transient analyses have also been conducted on analog simulations of reactors comparing non-jet pump and jet pump recirculation systems. These analyses confirmed prior stability analyses of the nuclear-thermal-hydraulic loop, indicating that present design practices, coupled with proposed reactor control system designs are sufficient to assure the design stability margins.

Currently, analyses are being conducted, utilizing data from the multiple unit tests, which will contribute to the improvement of present models of the jet pump process as used in analog simulations of the entire reactor and related control systems. Results from improved models will be factored into jet pump recirculation system designs to assure satisfactory stable reactor plant operation.

b. A test program will be performed to determine the hydraulic characteristics of the reactor recirculation system under steady state and transient conditions. The results will be based upon the test data with known performance characteristics of the recirculation pumps, jet pumps, and steam separators.

Prior to installation, the following equipment data will be obtained by performance tests:

1. Head-flow characteristics of the two recirculation pumps at hot conditions and over the full speed control range.

2. Head-flow characteristics of the four jet pumps with dual pressure taps at all flow conditions, both hot and cold.

3. The degree to which head-flow characteristics can be determined for the other 16 jet pumps with single pressure taps.

Following installation, detailed system performance tests will be made.

The operating instrumentation on the recirculation system will include:

1. Recirculation Pump ΔP and flow.

2. Four (of twenty) jet pumps with pressure taps in the exit diffuser for measuring pressure differentials.

3. Sixteen (16) jet pumps with single pressure taps located in the diffuser section, indicating a pressure differential between the diffuser and the core inlet plenum.

4. Steam separator assembly pressure inlet and outlet pressure. The ΔP vs flow characteristics of the steam separators provide a highly reliable indication of total flow over the operating range of core exit quality. At the time of Dresden Unit 2 startup, flow characteristics will have been verified by three other operating plants.

5. Core inlet and outlet pressure

   Plant tests will consist of utilizing the two independent methods (jet pump diffuser pressure and steam separator pressure differential) to accurately measure the recirculation flow rate. Tests will be performed at the following operating conditions:

   Cold
   Hot Standby
   25% Rated Power
   50% Rated Power
   75% Rated Power
   100% Rated Power

   The tests will consist of determining flow control range, coastdown characteristics following a recirculation pump trip, and single loop operation with and without the loop bypass line in operation. During all tests, the head-flow characteristics of the system will be recorded.

c. As determined from the actual tests on a multiple unit jet pump recirculation system which were discussed in II-3. a above, instabilities arising from the jet pump process itself are not expected. There is however a limit on flow oscillations which could be induced by the jet pump process itself and yet maintain safe operation of the facility.

In an analog simulation of a non-jet pump plant with flow control, recirculation flow oscillations were artificially induced by limit cycle operation of certain elements in the flow control system.

In one case analyzed, one percent of rated flow peak-to-peak oscillations of approximately 0.39 cps were induced when the reactor was at approximately 75% power and 70% flow conditions. These flow oscillations caused 5% of rated peak-to-peak neutron flux oscillations and approximately 0.5% of rated

peak-to-peak heat flux oscillations of the same frequency. No steam flow oscillations were observable, mainly because of the damping effects which result from the upper vessel and steamline steam volumes.

In the other case analyzed, 2% peak-to-peak flow oscillations of approximately 0.25 cps were induced resulting in peak-to-peak oscillation in neutron flux and heat flux of about 10% and 1.25% respectively. Steam flow oscillations again were not detected. The resolution of the steam flow traces was 0.25% of rated steam flow and thus if steam flow oscillations did exist their peak-to-peak magnitude was less than 0.25%.

It has been determined from past BWR experience that oscillations of 1% peak-to-peak in steam flow (± 0.5% of point) or 20% peak-to-peak in neutron flux (± 10% of point) could be tolerated and yet maintain safe operation of the facility.

The plant output requirements for distribution through the electrical transmission network would in most cases place lower bounds on these variables. Thus, scaling up from the two induced flow oscillation cases discussed above, it appears that peak-to-peak flow oscillations of 4% of rated (± 2% of point) can be tolerated and yet maintain safe operation of the facility with margin.

Thus if inherent instabilities exist in the jet pump process, they can be tolerated provided that they induce no more than ± 2% of point recirculation flow oscillations. As pointed out in reply to question II-3.a,even under more severe operating conditions, jet pump systems did not cause oscillations of this magnitude.

QUESTION

II Plant Design Characteristics (Continued)

4. For the type of containment proposed for this facility, the hydrogen gas from a metal-water reaction subsequent to loss-of-coolant accident could result in sufficient additional containment pressure so that containment integrity may be questioned. Assuming a loss-of-coolant accident in which the core systems malfunction, please provide an estimate of the amount of zirconium which it is believed may react. This estimate will provide a basis on which to evaluate the adequacy of the proposed containment vessel design considering the evolved and burning of the hydrogen. Any new experimental data on zirconium-water reactions and subsequent recombination which may influence your decision should be referenced or described. In addition, for the reference containment design, the maximum extent of metal-water reaction that can be tolerated without jeopardizing the integrity of the containment vessell assuming no subsequent hydrogen-oxygen reaction in one case, and then assuming a rapid hydrogen-oxygen reaction for another case, should be stated. In support of your answer to this part of the question, please state your assumptions and method of calculation used to determine your answer

ANSWER

The containment system to be provided in conjunction with Dresden Units consists of many engineered safeguard features. Each of these engineered safeguards provides a particular aspect of adequate reactor core or plant protection for a spectrum of possible system ruptures. The engineered safeguards pertaining to the containment with a description of their features are summarized in the following.

Pressure Suppression Structure

The entire reactor primary system is housed within the drywell of the pressure suppression system. This drywell is connected by sufficient venting lines to the pressure suppression chamber so that in the event of a major recirculation line rupture the drywell and suppression chamber are not over pressurized. The entire basis of the design of these structures is well established on experimental information obtained in connection with the Humboldt and Bodega Bay Projects.

Both of these systems - drywell and suppression chamber - will be designed to a pressure level of 62 psig.

Inert Atmosphere

Provisions have been made in the design of the Dresden Unit 2 containment structure so that an inert atmosphere can be provided during plant operation. This feature of inert atmosphere may be necessary to preclude any possibility for a hydrogen-oxygen burning situation should any appreciable metal water reaction occur due to core overheating. The metal water reaction between the metal of the core and any steam atmosphere would evolve hydrogen to the containment system. References indicate that an atmosphere can be rendered inert to a hydrogen burning possibility by reducing the oxygen content below 5% by volume. This can be accomplished by a nitrogen gas flushing.

Containment Cooling

The primary containment will also include a containment cooling system to provide cooling to the contents of the drywell, as well as to offer a vehicle to transfer energy to the very large heat sink offered by the water in the suppression chamber. This containment cooling system also contains the heat exchangers to provide an overall method of removing energy from the primary containment.

The containment cooling systems consist of two independent loops each provided with pumps (including a spare),heat exchanger rejecting heat to a service water system and a drywell spray system. These loops draw suction from the base of the suppression pool. These systems offer complete redundancy since either of the two systems supplies 100% cooling capability. One of the systems will start automatically from drywell high pressure signals. The other system is on standby and would start automatically if the first one failed to start. Both systems can be operated simultaneously. The electrical loads associated with these systems are considered in sizing the standby diesel generator.

Although the sizings for both the spray rates and the heat exchange capacities will not be finally specified until the completion of the system optimization study, current performance predictions are based on each cooling loop having a spray rate of 7000 gpm and 102 million btu/hr heat removal capacity for a primary side inlet temperature of 165°F.

The pressure rating of the containment system combined with the cooling capacity of either one of the two cooling loops provides a containment capable of withstanding a metal water reaction equivalent to those shown in Figure II-4-1. With just one of the two containment cooling loops in operation this containment system can stay within design pressure even with a 55% metal-water reaction over one hour or an 80% reaction over 9 hours.

Therefore, these systems offer substantial protection against a rupture due to a metal water reaction.

Reactor Core Cooling Provisions

Even though a very conservative design basis has been adopted for the containment system and its associated cooling systems, additional conservatism has also been incorporated by the provisions of reactor spray cooling systems. In the event of a loss of coolant accident the core spray cooling systems provide the dual function of removing fuel decay heat and also reflooding the inner reactor vessel.

As in other engineered safeguard features, two independent full capacity systems are provided. Both systems draw suction from the suppression pool and deliver spray to the reactor core through separate core spray spargers within the vessel. The flow capacity of each of these spray systems has been specified to be 4400 gpm. Full capacity pumping capability is provided in each of the two systems. One of the systems will start automatically from drywell high pressure signals. The other system is on standby and would start automatically if the first one failed to start. The electrical loads associated with these systems are considered in sizing the standby diesel generator.

With either one of the two core sprays in operation, the core is cooled adequately so that any hydrogen resulting from partial metal water reactions is below the flammability limit by an ample margin (a factor of 5 to 10) so that inerting would not be required. The specification of this flow rate is based on refined prototype testing of a full scale fuel bundle under actual power conditions and actual spray distribution

conditions. In order to insure that the test situations resulted in a limiting case, the test fuel rods were allowed to overheat (1250°F) prior to core spray activation and the channel boxes were allowed to stay at high temperature. The spray systems have been sized to provide the maximum required flow rate to each fuel bundle in the core. Flow distribution in the upper plenum as well as leakage flow not available to fuel rods were also taken into account in establishing the flow requirements.

The detailed behavior of these systems is discussed further in the reply to question III-7.

The capability of the primary containment system to withstand metal water reactions is shown in the curve in Figure II-4-1. Because the pressure at any time is also dependent on the temperatures which exist in the drywell and suppression chamber and the latter depend on the rate of energy removed and added, the containment pressure is a function of the metal water reaction rate.

Although the core cooling systems are assumed not to function in order to provide a basis for metal water reactions, one of the two containment cooling systems is assumed to function normally. This includes the drywell spray system with its associated cooling system.

Immediately following the major pipe rupture, the temperature of the suppression chamber water approaches 130°F and the system pressure approaches about 31 psig from the energy addition of the primary fluid blowdown. Most of the non-condensible gases (predominately air) will be in the suppression chamber but soon after initiation of the drywell spray, they will be redistributed between the drywell and suppression chamber volumes. As described above the drywell spray is sufficient to transport the decay heat, the core stored heat, and any heat resulting from the metal water reaction to the wetwell in the form of hot liquid. Steam flow will be negligible.

The energy transported to the suppression chamber water is removed from the containment system by containment spray heat exchangers. The pressure level of the system depends on both the system temperatures and the amount of non-condensible gases (air and hydrogen). Thus the capability of the system to house resulting gases from the metal water reaction varies with the rate and extent of the metal water reaction.

To evaluate this capability, various percentages of metal water reaction were assumed to take place over various durations of time. The hydrogen and energy associated with a given extent of reaction was released from the vessel uniformly for the entire duration of the event. For a given extent of metal water reaction the energy released from the vessel consisted of all the sensible heat stored in the clad and fuel of the core, all of the energy associated with the specified extent of metal water reaction, and all of the energy associated with the reactor core decay heat for the duration of the sensible and metal water heat release.

Iterative calculations were then performed to determine that percentage of metal water reaction which resulted in 62 psig (system design pressure) for various durations of energy release. These results are plotted in Figure II-4-1. It is shown in the answer to III-5 that a realistic estimate of the metal water reaction which could result from a complete core melt down is about 25%. This could be expected to occur in a minimum time of 30 minutes. As seen in the figure, this lies within the system capability. The maximum extent of metal-water reaction which can be tolerated in the Dresden Unit 2 primary containment system assuming no hydrogen-oxygen reaction is identified in Figure II-4-1.

If it is assumed that a rapid but non-detonating burn occurs subsequent to the release of hydrogen, about 4% of the total zirconium in the core can be reacted before the hydrogen concentration is such that the integrity of the containment (at a design pressure of 62 psig) is jeopardized.

Complete dispersal of the hydrogen within the drywell and suppression chamber is assumed. It is unlikely that burning of the hydrogen will occur as it is released, because of the lack of oxygen within the reactor vessel, the lack of continuous ignition source in the drywell, and the lack of sufficient hydrogen temperature level upon entering the drywell.

All the gases are assumed to follow the perfect gas laws, and 100% relative humidity is assumed to exist. Both of these assumptions are realistic and should closely represent the true case.

The entire system is assumed to be insulated from any external heat sources or sinks. Because the burning takes place rapidly, i.e., in less than 10 seconds, this is a realistic but conservative assumption.

The final pressure after combustion was calculated based on the total moles present including the combustion products and the temperature obtained from a heat balance on the energy released.

Radiant losses during the burn period are appreciable (30% because of the high temperatures in the flame front). The losses occur by radiation to the surfaces of containment. The latter represent a sizable sink even though only 2% of the thickness was calculated as being an effective heat sink. The time required to burn was based on a flame propagation velocity of 10 fps, determined by hydrogen air experiments in the literature, and the geometry of the containment. The flame was assumed to travel in two directions. The radiant heat losses were corrected for absorption by water vapor and air. The calculations are performed by an iterative procedure to yield the final temperature, immediately after combustion, of about 1200°F.

For the case of 4% metal water reaction the resulting preburn hydrogen concentration within the containment is approximately 12%. All of this hydrogen was assumed to burn. There is sufficient oxygen concentration to achieve complete burning for this amount of hydrogen. This amount of burning and associated energy release would result in a primary containment pressure level of 62 psig - the design rating of the containment.

In summary then, the proposed containment is capable of handling appreciable metal water reactions for various durations of the gas and energy release. These extents of reaction are in excess of those expected based on all calculations conducted.

Since the capability of the containment to withstand the burning hydrogen resulting from a metal water reaction is limited - approximately 4% metal water reaction - provisions have been made for an inert atmosphere in the primary containment.

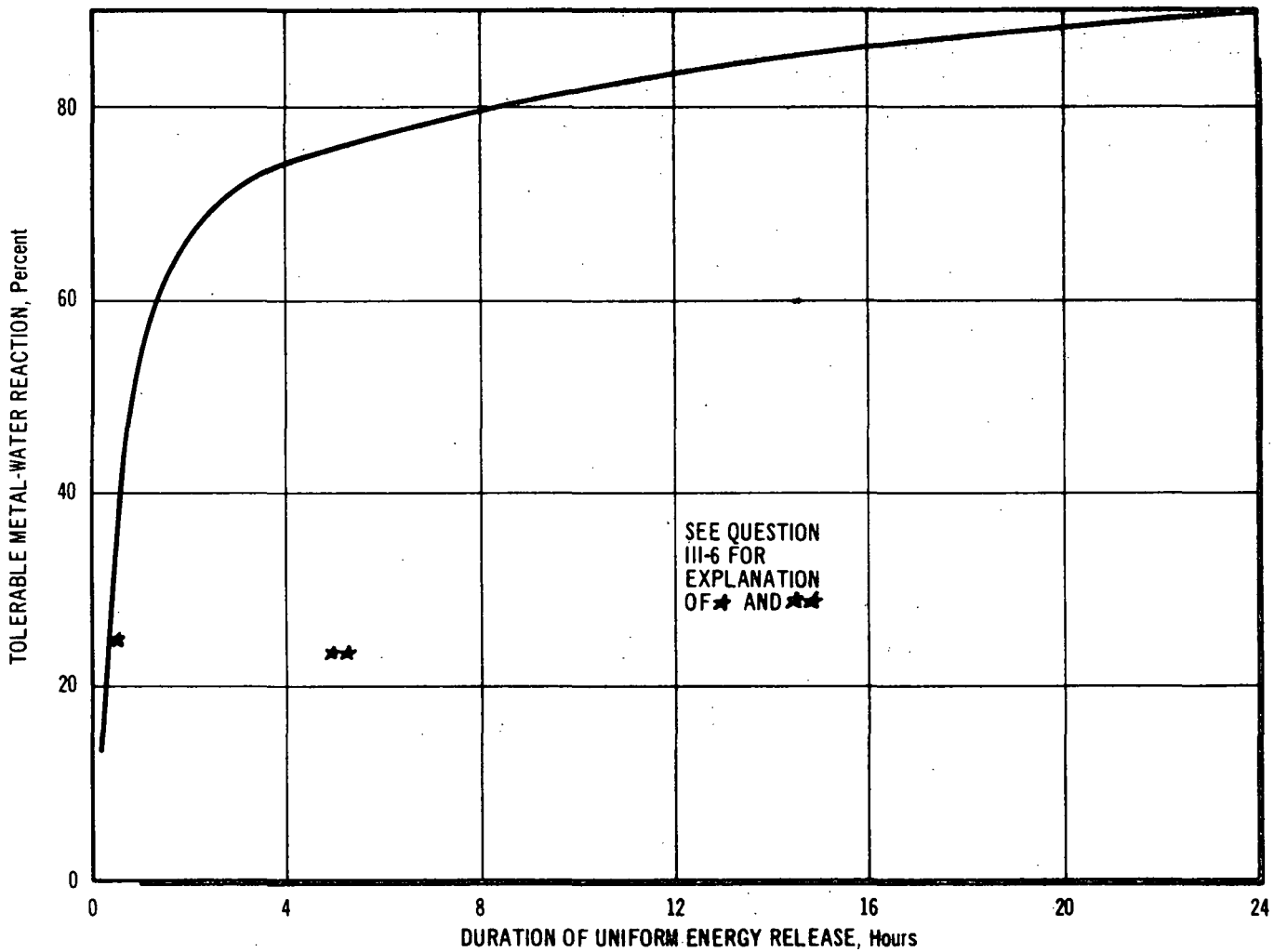


Figure II-4-1. Containment Capability - Tolerable Metal-Water Reaction as a Function of Duration of Energy and Gas Release

QUESTION

II Plant Design Characteristics (Continued)

5. There are some features in the plant design which relate directly to safety and which are new or novel in concept. These include:

   a. The core reflooding capability,

   b. The control rod velocity limiter,

   c. The steam line flow restrictor,

   d. The control rod thimble support, and

   e. The in-core nuclear instrumentation system.

   For each, a detailed description of the design objectives, design concept, and proposed testing, and/or calibration should be provided so that it may be demonstrated that each of these features satisfies its design objective and is clearly an addition to the safety of the facility. Since the in-core instrumentation appears to be quite complex in concept it would be helpful if you would include a rather complete description of the system components and a discussion of how they are coupled to form the overall system.

ANSWER

a. Core Reflooding Capability

Objective

The pressure suppression containment provides a significant advance in the evolution of engineered safeguards primarily because the residual pressure immediately following the postulated reactor coolant blowdown is minimized. As a result, the potential for leakage out of the containment is also minimized. In support of the pressure suppression containment system are several safety systems. Two redundant core cooling systems and the core flooding capability are concerned with maintaining core cooling to remove decay heat following the blowdown. The maintenance of core cooling following the postulated accident removes the potential for hazard to the public or further damage to the power plant. Containment cooling systems are provided to remove the thermal energy from the containment and further reduce the residual containment pressure.

Two separate and redundant core spray systems with 100% capacity are provided. Either system is capable of providing the necessary water to remove decay heat from the core following the loss of coolant. The addition of a second core spray system to replace the core flooding system as a source of cooling water represents a change from the core cooling systems described in the "Plant Design and Analysis Report." The reasons for this change are discussed in the following paragraphs.

The inclusion of jet pumps in the Dresden Unit 2 reactor design allows the provision of an inner vessel or container within the reactor vessel. Following a loss of coolant accident, a core spray system or the combined core spray systems will serve to restore a water level in the inner vessel for immersion of the fuel elements (as a byproduct of the cooling function). Subsequent to the restoration of a water level in the inner vessel, the continuance of core cooling is reduced to the relatively minor problem of keeping the inner vessel filled. Sufficient capacity is built into the spray systems to fill the inner vessel at a rapid rate. Consequently, the need for a special "flooding system" as such is obviated. After flooding, any system which is available to put water in the reactor vessel will provide adequate cooling, e.g. either core spray system, the reactor feed system, the control rod drive feed system, or possibly the reactor cleanup system.

Test Program

The core spray test program described in the Plant Design and Analysis Report has been completed and the amount of water required to accomplish the cooling objectives has been determined. Tests are still under way to determine how to best design the spray rings. These tests have shown the spray to be very effective as a cooling mechanism. To take advantage of this, the former core flooding system has been changed to a core spray. With both core spray systems in operation the core can be reflooded in a short time period.

In order to investigate core flooding effectiveness, tests were conducted on electrically heated "fuel" elements to determine the depth of immersion required to provide adequate cooling. The tests have been run using a twelve foot long 36 rod bundle. The power input was programmed as a function of time to correspond to the decay heat curve. The test results indicated that adequate cooling could be obtained with the "core" one third covered. The exposed portion of the rods showed peak temperatures of 1000°F to 1200°F. Exposed rod cooling is facilitated by the generation of steam in the immersed region of the bundle. In the actual reactor core the generation of steam by the lower portion of the immersed rods will be greater than the test value because of the flux skewing effect of core voids during normal operation.

In summary, the ability to fill the inner vessel in the Dresden Unit 2 reactor provides an added margin to assure the continuance of core cooling following a loss of coolant accident.

b. Rod Drop Velocity Limiter

Objective

The rod drop velocity limiter is a device which limits the free fall velocity of a control rod to five feet per second as required by the criteria of Section X-3.1.4b.

The postulated series of events leading to a control rod drop are as follows: (1) Control rod uncoupled either from failure to couple during initial installation or release of coupling due to mechanism malfunction, (2) drive withdrawal with coincident sticking of control rod and (3) sudden release of control rod. Using this postulation as a basis, the rod drop velocity limiter is designed to limit the free fall terminal velocity to the specified terminal value. The relation of the velocity limit to reactivity insertion limits is discussed on pages IV-5-2, IV-5-3, XI-5-2, XI-5-3, and in Appendix D of the "Plant Design and Analysis Report."

Design Concept

As noted in Section III of the "Plant Design and Analysis Report," the rod velocity limiter is being developed. Further information on the velocity limiter will be presented as the data accrues from the test program. A short summary description of the device and test program is included as follows:

The velocity limiter is an integral part of the bottom of the control rod. The velocity limiter is basically a loose fitting piston which travels in the control rod guide tube over the entire control rod stroke. As this piston moves along the guide tube the water which fills the tube must be displaced from one side of the piston to the other. The pressure differential which is developed across the piston results in a force which retards the piston motion. No moving parts are required to create the retarding forces or change the retarding forces developed for insertion and withdrawal. Higher pressure differentials and lower terminal velocity result in the event of a rod drop-out, yet the retarding force during scram insertion of the rod is low enough so that scram velocities can be achieved.

Test Program

A drop test facility was built to evaluate proposed velocity limiter designs. By dropping the test models first one way and then reversed, the resistance effectiveness ratios $\left(\frac{\text{velocity (scram)}}{\text{velocity (dropout)}}\right)$ were obtained. Velocity curves were recorded for the total drop in both directions. Many designs were tested. The drop test program results have established that a velocity limiter can be provided that will limit the dropout velocity of the control rod to the value established in criteria Section X-3. 1. 4b. A specific design of velocity limiter has been selected from these test results.

Another series of tests were conducted on a production control rod drive to evaluate the effect of additional loadings imposed by the velocity limiter. An increase in scram time was noted but the increase leaves the scram time well within the specification for the drive.

Tests will be made using a full scale production prototype velocity limiter and guide tube. This test will be made under cold and hot operating conditions using a production drive and simulated core. This test will confirm calculated limited dropout velocity and confirm tolerable drag forces during scram.

c. Steam Line Flow Restrictor

Objectives

In the event of a steam line rupture outside the containment the steam line flow restrictors will work in conjunction with isolation valves to:

a. Maintain adequate core coverage.

b. Eliminate the potential for core heatup.

c. Reduce total pressure loss in the vessel.

d. Reduce fission product carryover in liquid.

e. Eliminate the probability of forming water slugs of high velocity in the line.

Design Concept

Reactor steam flow rate following a steam line rupture can be limited to approximately rated steam flow by proper selection of nozzle throat area. During normal operation such nozzles would, it is estimated, result in an approximately 5 psi irreversible pressure drop, and the resulting moisture increase is negligible and can be tolerated.

In the event of a steam line break downstream of the nozzle, flow chokes in the decreased area by a two-phase mechanism similar to the critical flow phenomena in gas dynamics.

It is tentatively planned that flow nozzles will be installed in each of the four steam lines inside of the dry well, and the steam lines will be cross-tied in sets of two in front of the turbine stop valves. The flow issuing from the reactor in the event of a steam line break outside of the containment is then limited to approximately rated flow since the closing of the turbine stop valves will eliminate any steam flow from two of the four steam lines. Approximately twice normal line flow issues through the break from the reactor through the broken line flow restrictor (which chokes at double the normal flow rate or an increase of 25% to 50% of rated reactor steam flow). Because of the cross-tie, another 50% of normal steam flow issues through the break by back flow from the coupled line.

Closure of the steam line isolation valves isolates the break and prevents further steam loss.

In addition, this arrangement would limit the reactor blowdown rate to 50% of rated flow in the event of a steam line break in the drywell between a flow restrictor and the isolation valves.

Proposed Testing

Small scale tests are planned for measuring pressure loss of high-pressure steam flow through a venturi. Results will be compared to estimations already made using published recovery and expansion factors.

d. Control Rod Drive Thimble Support

Objective

As stated on page I-4-4 of Volume I of the Plant Design and Analysis Report, the control rod drive thimble support is designed to prevent significant (one to two inches) downward movement of the control rod drive with the attached control rod in the unlikely event of drive thimble structural failure. The support system will transmit to the concrete support structure loads imposed by a thimble failure. The design accounts for dynamic and static loading. The consequences of such a thimble failure, without a support device, are discussed on pages D-1 and D-2 of Volume I of the Plant Design and Analysis Report. The mechanical arrangement of the system must not create undue restrictions on operational functions such as control rod drive maintenance and/or removal, or interfere with other functional components such as incore instrumentation. There shall be provisions to determine that the system is in place.

Design Concept

The design concept shown in Figure 56 Volume II of the Plant Design and Analysis Report and discussed on Page X-3-5 of Volume I, is a modular grid supported by vertical bars fastened to cross beams that are mounted on the reinforced concrete reactor vessel support structure. The modular grid permits access to a drive for maintenance or replacement with removal of only one module. Openings for the incore instrumentation are accommodated by the open grid structure. Instrumentation is provided to assure the system is in place prior to start up of the reactor.

e. Neutron Monitoring System

Adequate neutron monitoring of the core depends on two important variables. The first is the neutron flux at the detector and the second is the neutron to gamma induced signal ratio. Calculations have shown that, because of the shielding effect of the large core and the thickness of the downcomer water gap, it is necessary to locate all neutron monitoring system detectors within the core to provide sufficient neutron flux and to meet the minimum neutron to gamma induced signal ratio.

A neutron monitoring system employing miniature chambers, all located within the core, has been designed to overcome this attenuation problem or lack of neutrons external to the reactor vessel. The complete monitoring range from source level to full power is covered by three sub-systems providing overlap coverage of: (1) source range; (2) intermediate range, and (3) power range. These sub-systems cover the following range:

Source Range	Source level to about 0.01% of rated power. These detectors can be retracted from the core to extend the range to about 10% of rated.
Intermediate Range	0.0001% to about 10% of rated power.
Power Range	1% to 125% of rated power.

All of the detectors are located in thimbles in the reactor vessel. There are a sufficient number of thimbles to provide the necessary coverage of the core to fulfill the criteria for the three ranges of instrumentation.

During the startup of the plant to rated power, the three instrumentation systems are integrated by interlocking to provide the overlap and insure proper coverage of the reactor operation. At startup, reactor mode switch is positioned in "Startup." In order to withdraw the first rod, all source range and intermediate range chambers have to be fully inserted into the core or the rods are blocked. The count level is procedurally set and must be determined if adequate by the operator.

Between approximately 5×10^3 up to 10^4 and 2×10^5 counts per second, the rod block is relieved so that the chamber can be retracted but the count rate is maintained in this range to prevent the rods from being blocked. In this way, the operator has period information up to 10% power. The operator must maintain the count rate in excess of 5×10^3 to 10^4 cps and less than 2×10^5 cps to prevent the occurrence of a rod block when in the startup mode.

The intermediate range becomes active at a minimum of one decade before the source chambers are retracted. Above the first range position of the intermediate range instrument, the downscale trip

becomes operative which gives a rod block. The high flux alarm also provides a rod block thus the operator has to maintain the intermediate range instrument reading between these two limits. Therefore, scram protection is always less than a decade from the operating level.

The transition between the intermediate and the power range requires that the power range monitor be "on scale" before the mode switch can be changed to "Run" or a rod block results. In the "Run" mode the intermediate range detector can be retracted from the core and the intermediate range downscale trip is bypassed. Also, the intermediate range high flux is bypassed if the power range monitor is "on scale" with mode switch in "Run."

1. Source Range

The source range instrumentation will be designed to provide the information needed for knowledgeable and efficient reactor startup operations.

A number of neutron sources are inserted in the reactor core to assure an adequate count rate of fission neutrons for instrumentation readout at shutdown and permit the measuring of multiplication of fission neutrons during approach to critical. The source range instrumentation covers the range from source level to about .01% of rated power. The detectors are initially located within the core just above the centerline and can be retracted to positions of lower neutron flux level to extend the usable range to about 10% of rated power. Several channels of pulse counting type instruments which have a high degree of discrimination between neutron flux and gamma flux are used to provide the operator with accurate neutron level information, particularly in the range when criticality is approached.

Each channel consists of a miniature fission counter detector, a preamplifier, and a log count rate meter with appropriate power supply. All equipment except the detector are located outside the drywell. Neutrons striking the sensitive material (U_3O_8-93% enriched) of the primary detector cause pulses of current to appear in the detector output. These pulses are amplified and shaped by the preamplifier which is just outside the drywell and transmitted to the log count rate meter in the control room for counting. Meters on the control console give continuous indication of count rate, and a strip chart recorder gives a continuous record of counting rate of the selected channels. The recorder provides an indication of count rate level as well as showing the trend of count rate with respect to time.

A downscale trip provides information as to adequacy of the count level or conditions of the instruments. An intermediate level trip blocks control rod withdrawal until the count level is above the settings if the detector is not fully inserted. A high level trip also blocks rod withdrawal. These two trips insure that the operator retracts the detector to maintain the count level between the two trips and provide period information to 10% power. Failure in the instrument is monitored by the inoperative trip which blocks rod withdrawal. One of the instruments may be bypassed if inoperative. During calibration the inoperative trip is activated.

The log count rate meter also includes circuitry to differentiate "counts" with respect to time to derive a reactor period measurement. Meters on the control console give continuous period indication. An adjustable trip circuit actuates an annunciator to warn the operator of approach to too short a period (too rapid rate of flux increase).

Each chamber has a sensitivity of approximately 1×10^{-4} counts per second per nv of neutron flux and is capable of detecting the planned source levels of the order of 6×10^{4} nv. The log count rate meter has a range of six decades with a maximum counting rate of 10^{6} counts per second.

Calibration of the source range instrumentation will be performed according to the standard procedures used for these ranges on the out-of-core instrumentation in presently operating plants. The design number and location of the detectors will be made on the basis of achieving the stated objectives during all operating conditions at any time in core life.

Motor-operated drives are provided to move the detectors from the high neutron flux region near the center of the core to a relatively low neutron flux region below the core. This serves to extend the usable range of the counting channels and prolong the life of the counters by keeping them in low flux regions except during startup. The drives are controlled from the main control console. Interlocks are provided to insure that chambers are inserted into the core before the reactor can be started up.

The detector assemblies are installed from the bottom of the reactor vessel through a flanged nozzle which penetrates the vessel and contains a dry pressure tube which extends into the core. The pressure containing tubes thus separates the assembly from the reactor water environment.

2. Intermediate Range

The intermediate range instrumentation will be designed to provide the required degree of automatic power level protection. This instrumentation will also provide the information needed for knowledgeable operation of the reactor in the intermediate power range.

Several channels of instrumentation are provided to monitor the reactor neutron level from about .0001% to 10% of rated power providing approximately a two decade overlap (a minimum of one decade) with the source range system.

Each channel consists of a miniature fission chamber detector, a preamplifier, a linear amplifier with power supply and a range selector switch. All equipment except the detector is located outside the drywell. Indicating recorders on the control panel indicate neutron level in terms of reactor power. The range selector switches for each amplifier are mounted near the recorder of the reactor control station. The amplifier is a linear-type instrument and covers a range of five decades in 10 steps.

Triggering type trip circuits are provided to serve as downscale rod block, high flux alarm-rod block and high flux scram. The alarm-rod block and scram trips are calibrated in terms of percentage of meter range rather than rated reactor power. This gives extremely fast and safe protection at all power ranges from minimum range setting to full power operation. If the operator does not keep the meters "on scale," he will initiate an alarm, a rod block, or a downscale trip. The downscale trip assures that the meters are always on the correct range before rods are withdrawn to increase power and also serves to give an early warning of amplifier or detector malfunctions.

A "channel inoperative trip" is provided that introduces a scram trip if the instrument is inoperative. During calibrating the "inoperative trip" is actuated and will cause a scram and rod block trip if the instrument is not bypassed.

The chamber arrangement in the core and in the reactor protection system is such that one intermediate range channel in each of the protection system channels may be bypassed provided the two associated chambers are not in the same region of the core.

Calibration of the intermediate range instrumentation will be performed according to the standard procedures used for these ranges in the presently operating plants. The design number and location of the detectors have been made on the basis of achieving the stated objectives during all operating conditions at any time in core life.

Motor-operated drives are provided to move the detectors from the high neutron flux region within the core to a lower neutron flux below the core. This serves to prolong the life of the detectors by storing them in low flux regions during operation above 10% rated power. The drives are controlled from the control console. Interlocks requiring the chambers to be inserted into the core are provided to insure protection when the reactor is started up.

The intermediate equipment utilizes a new method of measurement which has been under development for several years. This method uses the variation of signal that is usually considered as noise on the DC signal from a fission chamber and is known as the Campbell method. The method was developed from a statistical theory of random occuring events derived by Norman Campbell. In applicable terms, this theory is: that when random events are detected by a linear detector, they pile up on one another. Therefore, both the average and the mean square deviation about this average are a function of rate of occurrence of the events. Applying this theory the noise or AC signal is accepted and the DC signal is rejected by a coupling capacitor between the chamber and amplifier. The AC signal is amplified, squared, integrated and provides a linear indication of power level.

The Campbell approach provides excellent discrimination between neutron and gamma signals of about 1000 to 1. The equipment is AC and therefore eliminates the problems of DC leakage in cables and DC drift in the amplifier.

3. Power Range Monitoring

a. Local Power Range Monitoring

The objectives of the local power range instrumentation are to

- continuously monitor local heat flux and alarm on excessive conditions,
- permit evaluation of the critical core parameters to the accuracy required by core design and the established limits,
- and to permit demonstration or compliance with the established limits on core critical parameters with the speed and ease consistent with efficient operation of the plant.

There are approximately 164 chambers for measurement of the local neutron flux inside the reference core. Each channel consists of a chamber and separate amplifier and indicating instrument for each point. The readout instruments are located in the control rod position display panel in their general relation to the control rods.

There are approximately 41 radial locations in the reference design core which contain ion chamber assemblies. Four chambers are fabricated into one of these assemblies with the chambers spaced at 3 feet intervals, starting 1-1/2 feet from the bottom of the core. The signal from the chamber is carried in a mineral-insulated stainless steel sheathed cable. The cable is welded to the ion chambers to make a pressure-tight assembly.

The ion chamber assemblies are installed from the top of the reactor vessel when the vessel head is off. The chamber will be cooled by reactor water in direct contact with the chambers and cable. The chamber cable leads pass through a high pressure connection at the bottom of the reactor vessel and terminate in an electrical connector located outside of the vessel. The cables from the electrical connectors on each assembly are routed to amplifiers mounted in the control room. Output from the amplifiers is shown on meters in a rod-position and in-core flux level display on the main panel, with indicator lights to show high flux approach. Flux level for each in-core chamber is indicated separately.

The local power distribution monitoring instrumentation will be calibrated using data obtained from the Traveling In-Core Probe (TIP) calibration system and heat balance data such that the individual amplifier readings will be proportional to the average heat flux in the four fuel rods which surround the in-core chamber at the same elevation as the chamber.

After the calibration is performed, it is expected that the evaluation of the relevant core parameters will be performed using the amplifier readings and other data to provide

1. Three-dimensional power and heat flux distributions.

2. Quality, void, and minimum critical heat flux values throughout the core.

b. Average Power Range Monitoring

The average power range monitor will be designed to scram the reactor automatically during bulk power level transients before the power level exceeds certain specified values. The instrumentation will be used to prevent fuel damage from single operator errors or equipment malfunctions in the power range. In addition, the instrumentation will provide accurate indication of thermal power level of the reactor in the power range.

Eight channels of instrumentation are provided in the reference design core to monitor reactor neutron levels in the normal operating power range.

Each channel consists of signals from eitht or more selected in-core local power range monitor amplifiers, an average power range monitor, a recirculation flow variable trip bias unit, and meters and/or a strip chart recorder to indicate neutron flux level in terms of reactor power level. The signals from the in-core amplifiers are averaged in the average

power range monitor units. Each channel will be calibrated to read percent of rated thermal power using conventional heat balance techniques.

Each channel covers a power range from 1% to 125% of rated power. Four trip circuits are provided to serve as:

1. A downscale interlock to prevent rod withdrawal if the meter is not "on scale",
2. A "high flux warning" and rod block and flow change block to warn the operator if a preset high flux trip level is reached and,
3. A "high flux scram" to initiate reactor shut-down in the event a still higher flux level is reached,
4. An inoperative trip which provides a scram trip and rod and flow block trip if the instrument is inoperative (e.g. operate-calibrate switch is out of "operate").

The downscale interlock will serve to give an early warning of amplifier malfunction. A rod block and flow change block is associated with the "high flux warning" alarm trip to prevent the operator from increasing power by further rod withdrawals or flow increase or decrease. This alarm trip is automatically biased by the recirculation flow signal to provide core protection by preventing rod withdrawals or flow increase or decrease to power levels for which flow is insufficient.

c. Calibration System

1. The APRM system will be calibrated to read percent of rated thermal power using conventional heat balance techniques.
2. The power distribution monitoring instrumentation will be calibrated using data obtained from the Traveling In-Core Probe (TIP) calibration system and heat balance data such that the individual amplifier readings will be proportional to the average heat flux in the fuel rods which surround the in-core chamber at the same elevation as the chamber.

   After the calibration is performed, it is expected that the evaluation of the relevant core parameters will be performed using the amplifier readings and other data to provide:

   a. Three-dimensional power and heat flux distributions,
   b. Quality, void, and minimum critical heat flux values throughout the core.

3. Each local power range assembly of ion chambers contains a pressure containing tube for insertion of calibration equipment. Several complete traveling ion chamber systems are provided for calibration purposes. These are called the traveling in-core probe or "TIP" systems. Each "TIP" system consists of a miniature ion chamber, flux amplifier, readout equipment, electro-mechanical drive cable remotely operated drive unit, indexing device, valve assembly and required guide tubing. This system represents a considerable improvement over the wire insertion calibration systems used in the past. "TIP" system operation is considerably faster and permits immediate cross calibration of the in-core monitors.

The calibration tube of each in-core ion chamber assembly is connected to a guide tube which penetrates the drywell through a remotely operated valve and terminates in an indexing device. Each indexer collects eight such tubes and is directly connected to a drive assembly with a single tube. An ion chamber and cable assembly is installed in the drive unit. Any tube position can be selected for traversing with the chamber. Once the core position to be traversed is selected, the ion chamber is then moved through the tubing into the core, flux is plotted on an X-Y recorder, and the chamber returned to start position in one automatically controlled operation.

QUESTION

II. Plant Design Characteristics (Continued)

6. The standby liquid control system is designed to shut down the reactor with all control rods withdrawn within 60 minutes. Discuss those factors which governed selection of this relatively slow operation (compared to other GE boiling water reactors) and why it is considered adequate.

ANSWER

The design basis for the standby liquid control system results from the postulation that for some unexplained reason, while at full power, none of the control rods that are withdrawn can be inserted into the core. In actuality, there are numerous ways of inserting the control rods into the core, e.g., either normal drive mode or scram mode using the rod drive feed pump pressure, accumulator pressure, or reactor pressure, so that this condition has a very remote possibility of occurrence. Since all reactor control functions are provided by the control rod and drive system, the standby liquid control system is clearly redundant.

Operation of the reactor at full power with the control rods immovable could continue indefinitely until such time as the core reactivity runs out. The standby liquid control system is available under the postulated conditions to shut the reactor down as required.

The reactivity requirements for the standby liquid control system are summarized in the following table:

Parameter	ΔK	Natural Boron, ppm
Rated to Zero Power	0.040	200
Xenon	0.030	150
Hot to Cold	0.050	250
Shutdown Margin	0.050	250
Total	0.170	850
Mixing Factor	1.25	
Total	0.213	1063

The design reactivity addition rate is - 0.002 ΔK per minute (10 ppm per minute) as is the design rate for the GE boiling water reactor being supplied to the Niagara Mohawk and Jersey Central power companies.

The addition rate is an arbitrary selection made to assure that all of the solution is in the reactor primary system by the time the system is cold. Sufficient thermal energy (assuming the removal of decay heat only) is stored in the water in the vessel and in the vessel wall to allow the reactor to remain hot for an indefinite period. Even with the condenser and the shutdown cooling system available, a normal cool down period varies from 12 to 24 hours. Consequently, the selected injection rate of 0.002 Δk per minute (total injection time of 107 minutes or one hour and 47 minutes) is much faster than actually required. There is a margin of 0.050 Δk in the design of the system since 0.120 Δk is required to shut down the reactor with a cold clean (new) core from the hot operating (full power) to cold shutdown condition. On this basis, sufficient neutron absorber is in the system at the end of an hour to effect a cold shutdown. The reactor is shutdown hot in ten to twenty minutes.

The standby liquid control system is designed so that it may be completely tested to assure its operability. The pump and circulating system can be operated outside the reactor. A demineralized water purge system is provided so that the remaining portion of the system may be tested by pumping clean water through the distribution sparger and into the reactor.

QUESTION

II. Plant Design Characteristics (Continued)

7. It appears that accidental draining of the fuel storage pool when loaded with spent fuel could represent an accident of severe consequences. Please discuss the design of the fuel storage pool in terms of why accidental draining of the pool can be neglected as a potential accident or, if it cannot be neglected, evaluate the consequences of such an accident.

ANSWER

The fuel storage pool is a reinforced concrete structure, completely lined with seam welded stainless steel plates welded to reinforcing members (channels, I-beams, etc.) embedded in concrete. The pool will be adequately designed to withstand the anticipated earthquake loadings as a Class I structure. The lower liner section will be built as a free standing structure and hydrostatically tested prior to the concrete pour. The stainless steel liner will prevent leakage even in the unlikely event the concrete develops cracks. To avoid unintentional draining of the pool, there are no penetrations that would permit the pool to be drained below a safe storage level, and all lines extending below this level are equipped with suitable valving to prevent backflow. The passage between the fuel storage pool and the refueling cavity above the reactor vessel is provided with two double sealed gates with a monitored drain between the gates. This arrangement permits detection of leaks from the passage and repair of a gate in the event of such leakage. A liquid level switch monitoring pool water level is provided to detect loss of water and permit refilling of the pool from the condensate storage system. In addition, a second level switch in the pool surge tank is provided to permit almost instantaneous water loss detection.

QUESTION

II. Plant Design Characteristics (Continued)

8. Exfiltration from the reactor building could be a mechanism for uncontrolled release of fission products leaking from the reactor building portion of the containment system. To enable an evaluation of the significance of this potential source of fission products, a graph or tabulation of the exfiltration characteristics of the reactor building as a function of wind speed should be provided and the basis of this information should be stated.

ANSWER

The standby gas treatment system is designed to provide a controlled discharge to the 300-foot stack at a rate equivalent to 100% of the reactor building volume per day when the building is maintained at a negative pressure of 0.25 inch of water with respect to the outside atmosphere and with a low external wind velocity.

High winds will create pressure gradients around the building which can cause outleakage or exfiltration from the building. Calculations have been made of exfiltration in high winds based on building pressure gradients determined by Irminger and Nøkkentved[1] in wind tunnel tests of building models. Data for seven different wind directions were used in the calculations, from which it was determined that the greatest exfiltration rates arise from those associated with the wind blowing normal to the longest side of the reactor building, which measures 145' × 118'. The models used for the tests in the above reference were similar in geometry to the reactor building except that the length-to-width ratio was somewhat greater, 2:1 as compared to about 1.23:1 for the reactor building.

Leakage rate tests of buildings at low pressure differentials indicate that leakage rates may be correlated by the following equation[2]:

$$\text{Leakage rate} = a(\Delta P) + b\,(\Delta P)^{1/2}$$

where "a" and "b" are constants which are dependent upon the leakage characteristics of the building and ΔP is the pressure differential between the building atmosphere and the outside. To determine the potential range of exfiltration rates, two series of calculations were made - one assuming leakage rates to be proportional to ΔP to determine maximum exfiltration rates and the second proportional to $(\Delta P)^{1/2}$ to determine minimum rates. Other basic assumptions include the following:

1. Throughout all ranges of wind velocities, the standby gas treatment system discharges at a rate equivalent to 100 percent of the building volume per day.

2. The building inleakage rate is 100 percent of building volume per day when the building is maintained at a negative pressure of 0.25 inch of water with respect to the outside under zero wind velocity conditions.

3. Leakage sources are assumed to be distributed homogeneously on all four side walls of the building in proportion to wall length. The roof is assumed not to leak.

Maximum and minimum calculated reactor building exfiltration rates as a function of wind velocity are shown in Figure II-8-1. These data indicate that at wind velocities less than about 35 mph, there would be little, if any, exfiltration from the reactor building. These results also indicate that the maximum exfiltration rate at 40 mph is less than 20% of building volume per day as compared to the 100% per day as specified in Section V-4. 1. 1 of the Plant Design and Analysis Report. The calculations which were made also indicate that exfiltration rates are almost directly proportional to the initial inleakage rate for a given negative building pressure. That is, if the building when tested were found to have an inleakage rate of 200% per day at zero wind velocity, it might be expected to have a maximum exfiltration rate of 40% per day under 40 mph wind conditions.

As noted in Exhibit III-6-40, the maximum wind speed at the Dresden Site is in the range of 45-50 mph, at 150 feet elevation. A high wind of this magnitude would create an exfiltration rate up to approximately 50 percent per day. The actual rate would depend upon the building leakage characteristics, i. e., whether leakage is proportional to ΔP or $\Delta P^{1/2}$ for the final design. This is indicated by the expected leakage rate range in Figure II-8-1.

The reactor building leakage rate specifications and controlled release ventilation rate will provide a more than adequate containment system for secondary containment during reactor operation and primary containment during refueling operations. Under high wind velocity exfiltration conditions, the dilution of any released material potentially reaching any off-site location is greater than that for stack release under normal wind conditions. Although a potential for increased off-site dose rates to the thyroid and lungs exists under conditions of exfiltration, because of the bypassing of the standby gas treatment system filters, the dose rates under these conditions for all of the accidents described in Section XI are far below the dose rates generally considered acceptable in unlikely accident situations such as these. Further, on the basis of wind velocity data given in Exhibit III-6-40 it is unlikely that high wind exfiltration conditions would exist for more than a few hours per year.

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1. Irminger, J. O. V. and Nφkkentved, Chr., "Wind Pressures on Building, Second Series," Copenhagen, 1936 (available from U. S. Weather Bureau, Washington, D. C.).

2. Koonty, R. L., Nelson, C. T. and Baurmash, L., "Leakage Characteristics of Conventional Building Components for Reactor Housing Construction," Transactions of the American Nuclear Society, Vol. 4, No. 2, P. 365, November 1961.

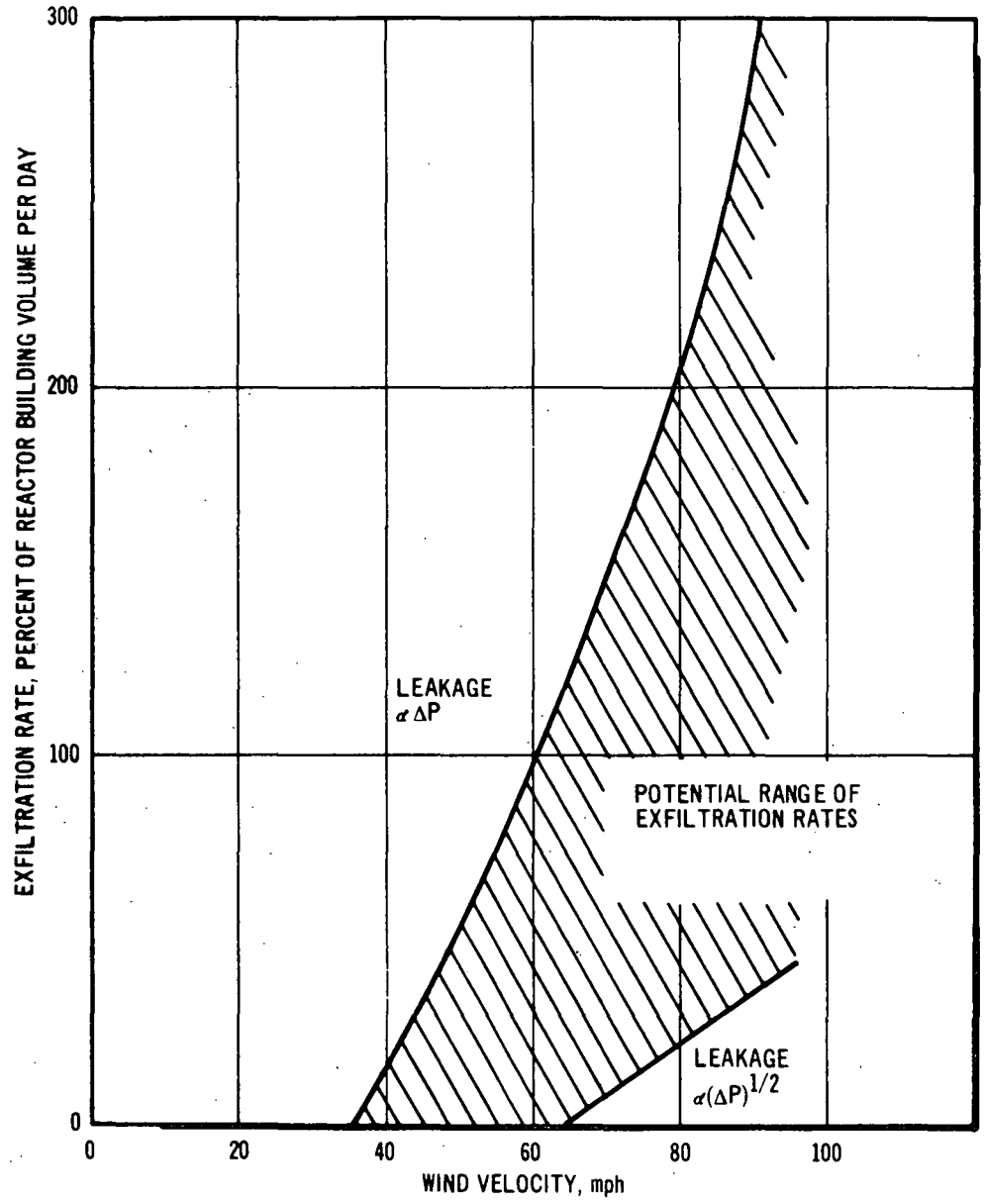


Figure II-8-1. Calculated Range of Reactor Building Exfiltration Rates as a Function of Wind Velocity

QUESTION

II. Plant Design Characteristics (Continued)

9. Please state the basis for establishing the capacity of the vacuum breakers between the drywell and the suppression pool and between the reactor building and the suppression pool.

ANSWER

The vacuum breakers, which connect the suppression chamber and drywell, are sized on the basis of the Bodega pressure suppression system tests. The vacuum breaker flow area is proportional to the flow area of the vents connecting the drywell and suppression pool. Their chief purpose is to prevent excessive water level variation in that portion of the vent discharge lines which is submerged in suppression pool water prior to a large system rupture in the drywell. The Bodega tests regarding vacuum breaker sizing were conducted by simulating a small system rupture, which tended to cause vent water level variation, as a preliminary step in the large rupture test sequence. The vacuum breaker capacity selected on this test basis is more than adequate (typically by a factor of four) to limit the pressure differential between the suppression chamber and drywell during post accident drywell cooling operations to a value which is within suppression system design values.

The vacuum breakers, which connect the suppression chamber and reactor building, provide makeup air to the containment to compensate for air density changes or to replace air which could be lost due to leakage. These vacuum breakers are sized on the basis of supplying sufficient air to limit the drywell and suppression chamber vacuum within their negative pressure design points. The air flow rate is conservatively assumed to be that necessary to compensate for steam condensation which could occur during the post accident drywell cooling operation. Drywell makeup air is supplied from the reactor building through the suppression chamber-to-reactor building breakers in series with the suppression chamber-to-drywell vacuum breakers described above.

QUESTION

II Plant Design Characteristics (Continued)

10. We believe that an important design criterion of a nuclear facility such as Dresden II is that the control room should be capable of occupancy throughout the course of any conceivable accident. To enable a more definitive evaluation of your facility in this respect, the criterion, including attenuation factors, for design of the control room with reference to occupancy during accidents should be stated. The radiation doses that will be allowed as a consequence of a maximum credible accident should be specifically identified. State also the manner in which control room ventilation will be controlled under such circumstances to assure acceptable inhalation doses.

ANSWER

The control room shielding design has not been definitely established. The control room shielding design criteria is to limit the dose in the control room to 0.5 rem in any 8 hour period following a design accident in either Unit 1 or Unit 2. The Dresden Unit 1 control room roof thickness of 3 feet allows occupancy following the "worst reasonable accident" from Unit 1 reactor. The tentative Unit 2 control room roof thickness of 2.5 feet is based on the same criteria, including some shadow shielding supplied by the Unit 1 control room. The 1.5 Mev gamma attenuation between the Unit 1 containment and the Unit 2 control room, including buildup, is 10^6, taking no credit for distance. This roof thickness is also adequate to allow occupancy of the Unit 2 control room following Unit 2 accidents that are discussed in answer III-5 of this submittal. The control room wall thickness is tentatively established as two feet. The 1.5 Mev gamma attenuation between the Unit 2 reactor building and control room, including buildup, is 10^5, taking no credit for distance. The 1.5 Mev gamma attentuation between the drywell and the control room is 10^{15}.

The radiation doses that will be allowed as a consequence of a maximum credible accident will be in accordance with 10CFR20.

The control room ventilation will consist of a system having a large percentage of recirculated air. The fresh air intake will probably be from the roof of the turbine building, above the control room, about 100 feet above grade elevation. The plant stack is 300 feet high. The dilution factor offered by 200 feet elevation difference, at about 200 feet horizontal distance, is extremely high, essentially infinite. Therefore, there will be no danger of contaminating the control room air from stack effluents. The fresh air intake can be closed to control intake of airborne activity from reactor enclosure exfiltration.

QUESTION

II Plant Design Characteristics (Continued)

11. Please state the test capability which will be incorporated in the design of the primary containment system to permit:

a. A preoperational leakage rate test of the containment vessels to verify the leak-tightness at calculated peak pressure associated with the accident analysis, after complete installation of all systems whose piping and controls penetrate the containment shells.

b. Leak detection testing of all penetrations fitted with resilient seals or gaskets, piping penetrations with expansion bellows, equipment access hatch cover seals, dry well head flange, air lock chamber (between doors), and suppression chamber manhole seals at a test pressure corresponding to the calculated peak pressure (from accident analysis).

c. Leak detection testing of containment isolation valves at calculated peak pressure (from accident analysis) and verification of functional performance of valves, associated sensors and equipment under the pressurized containment conditions selected for the conduct of the periodic integrated leakage rate test.

ANSWER

a. After complete installation of all penetrations in the drywell and suppression chamber, these vessels will be pressurized to the calculated peak accident pressure and measurements taken to verify that the integrated leakage rate from the vessel does not exceed 0.5 percent per day. Since both the drywell and suppression chamber are designed for 62 psig, it will be possible to test these vessels simultaneously at the same pressure and without the necessity of providing temporary closures to isolate the suppression chamber from the drywell. The necessary instrumentation will be installed in the vessel to provide the data required to calculate and verify the leakage rate.

b. All containment closures which are fitted with resilient seals or gaskets will be separately testable at the full design pressure of 62 psig to verify leak tightness. The covers on flanged closures, such as the equipment access hatch cover, the drywell head, access manholes and personnel lock doors will be provided with double seals and with a test tap which will allow pressurizing the space between the seals without pressurizing the entire containment system.

In addition, provision will be made so that the space between the air lock doors can be pressurized to full drywell design pressure. Since both doors on the lock will swing into the drywell vessel, it will be necessary to provide temporary structural members to brace the inner door during this test.

Electrical penetrations will also be provided with double seals and will be separately testable at 62 psig. The test taps and the seals will be so located that the tests of the electrical penetrations can be conducted without entering or pressurizing the drywell or suppression chamber.

As noted in Section V-3.3.3 of the Plant Design and Analysis Report, those pipe penetrations which must accommodate thermal movement are provided with expansion bellows. The bellows expansion joints are designed for the containment system design pressure and can be checked for leak tightness when the containment system is pressurized. In addition, these joints are provided with a second seal and test tap so that the space between the seals can be pressurized to 2 psig to permit testing the individual penetrations for leakage. A study is currently being made to establish the feasibility of providing seals which would make it possible to conduct the leakage tests of these penetrations at a pressure corresponding to the peak drywell pressure calculated in the accident analysis. Such seals will be provided if the studies demonstrate the feasibility of the design.

c. The test capabilities which will be incorporated in the primary containment system to permit leak detection testing of containment isolation valves are separated into two categories.

The first category consists of those pipe lines which open into the containment and do not terminate in closed loops outside the containment, but contain two isolation valves in series. Test taps are provided between the two valves which permit leakage monitoring of the first valve when the containment is pressurized. The test tap can also be used to pressurize between the two valves to permit leakage testing of both valves simultaneously. The valves, associated sensors and equipment which will be subjected to containment pressures during the periodic leakage tests will be designed to withstand containment design pressure without failure or loss of functional performance. The functional performance of these devices will be verified by demonstration either during the leakage tests or subsequent to the test but prior to start-up.

The second category consists of those pipe lines which connect to the reactor system, also contain two isolation valves in series. A leak-off line is provided between the two valves, and a drain line is provided downstream of the outboard valve. This arrangement permits monitoring of leakage on the inboard and outboard valves during reactor system hydrostatic tests, which can be conducted at pressures up to the reactor system operating pressure of 1000 psig.

QUESTION

II Plant Design Characteristics (Continued)

12. Please state the intent with respect to the proposed means of verifying the test leakage rate for the dry well, the pressure suppression pool, and the three (3) engineered safeguard systems directly connected to the containment vessels, and specifically whether:

a. Isolation is to be provided between dry well and pressure suppression chamber to permit independent leakage rate determinations, or

b. A composite allowable leakage rate will be specified for both dry well and suppression chamber based on testing both chambers simultaneously at the same test pressure, and

c. Any measured leakages in the dry well spray system, core flooding system, and core spray system, will be included in defining the over-all allowable leakage from the primary containment system.

ANSWER

a. As noted in answer to Question 11 a above, both the drywell and suppression chamber are designed for a pressure of 62 psig and it will not be necessary to isolate the two vessels to conduct independent leakage rate tests. Both vessels will be leakage tested simultaneously.

b. The leakage rate for the primary containment is based upon testing both vessels simultaneously at the same pressure. It is planned to initially conduct leakage rate tests at several pressures to establish a leakage rate curve. Subsequent to the integrated leakage rate test of the primary containment as discussed in Question 11a above, and prior to initial plant startup, a series of leakage rate tests can be conducted to define the leakage behavior of the containment system as a function of pressure with the capability to test the leak-tightness of containment system penetrations as described in the answer to Question 11 b above, it is anticipated that if integrated leakage rate tests during the life of the plant are required, they can be conducted at a pressure which is less than containment system design pressure. The leak rate at higher pressures may then be extrapolated from the curve. A composite allowable leakage rate and the test pressure for these tests can be specified after review of the results of the initial series of tests. The allowable integrated leakage rate will consider the permissible post-incident dose rates.

c. It will be possible to periodically pressurize and test the leak-tightness of the portions of the containment and core cooling systems which are exterior to the primary containment vessels. It will not be necessary to separately measure the leakage rate from the containment and core cooling systems since these systems will be pressurized during the leakage rate tests. Leakage from these systems will automatically be included in the overall leakage rate for the primary containment systems.

QUESTION

II Plant Design Characteristics (continued)

13. We understand that the TIP instrument calibration system will include several "Teleflex" cables penetrating the containment barrier. Further, when the TIP system is in use it will not be possible to close the associated containment isolation valves should the necessity arise. Please discuss the relationship between this and potential accident consequences, and also discuss the isolation criteria of the TIP system penetrations, and the manner in which isolation will be assured.

ANSWER

As noted in Section X-4.3.3a of the Plant Design and Analysis Report, a traveling in-core probe (TIP) system is provided for neutron flux calibration purposes. Each TIP assembly consists of a miniature fission chamber attached to an electro-mechanical drive cable, a remote operated drive unit, an indexing device, an insertion guide tube, a guide tube closure valve, and appropriate instrumentation.

A total of 50 insertion guide tubes are provided in five groups of ten tube bundles which pass from the reactor building, through the primary containment, and into the reactor vessel. That portion of the tube within the reactor vessel is closed at the end, thus forming a long thimble, and is closed in the reactor building by means of a valve. This arrangement forms two barriers between the internal environment of the reactor vessel and the reactor building.

When the TIP System cable is inserted the valve of the selected tube opens automatically and the chamber and cable are inserted. Insertion, calibration, and retraction of the chamber and cable requires approximately five minutes. Retraction requires a maximum of 1-1/2 minutes. A maximum of five valves may be opened at any one time to conduct the calibration, and any one guide tube is utilized approximately one hour per year.

If closure of the valve is required during calibration, the isolation signal causes the cable to be retracted and the valve to close automatically on completion of cable withdrawal.

If a failure were to occur to the section of the guide tube located within the reactor vessel during a time with the valve open, it has been calculated that a total of approximately 1.2 pounds of steam-water would leak through the tube to the reactor building while the cable is being withdrawn. If the tube were to fail within the drywell at a time the drywell was pressurized from a loss of coolant accident and also during a time with the valve open, it has been calculated that a total of approximately 0.3 pound of steam would leak through the tube to the reactor building while the cable is being withdrawn.

In order to assure containment integrity under normal operating conditions and under the postulated failure modes, the following isolation and containment penetration criteria are applicable:

(a) The insertion guide tube and the isolation valve are fabricated in accordance with the intent of the ASME Boiler and Pressure Vessel Code Section III.

(b) The design pressure of the insertion guide tube within the reactor vessel is 1250 psig. The guide tube between the reactor vessel and the containment penetration is designed for 75 psig external pressure.

(c) The TIP cable and fission chamber are retracted automatically within 90 seconds upon receipt of proper signal.

(d) The isolation valve closes automatically upon receipt of proper signal and after the TIP cable and fission chamber have been retracted.

(e) Penetrations of the insertion guide tubes through the primary containment are sealed by means of brazing which meets the requirements of the ASME Boiler and Pressure Vessel Code Section VIII. These seals would also meet the intent of Section III of the Code even though the Code has no provisions for qualifying the procedures or performance.

QUESTION

II Plant Design Characteristics (Continued)

14. In order to make a reasonable appraisal of the affect of the Dresden II facility on the environs, please provide an estimate of the amount of fission products per unit time which may be released via liquid and gaseous paths during routine operation. State the bases of these estimates.

ANSWER

Two general sources exist for the release of gaseous radioactive materials from the plant, i.e., activation gases and fission product gases from the fuel. The gases originate in the core and are removed from the reactor water in the air ejector, then held-up in the off-gas piping before release through the 300 foot stack.

The activation gases are released from the stack at the rate of approximately 200 μc/sec during operation at approximately 2300 MWt. as indicated on page VI-3-2 of the Plant Design and Analysis Report. The rate of release of these gases is proportional to the thermal output of the reactor and the holdup time in the system before release at the stack.

Insignificant quantities of activation gases, including Ar-41, Ar-35, and He-3 are released also.

The fission product gases may arise from tramp uranium on the surface of the fuel cladding, from imperfections, or failures which might develop in the fuel cladding. The principal gaseous isotopes from this source discharged from the stack are shown in the following tabulation.

Typical Off-Gas Release Rates

	Isotope	Half-Life	Emission Rate μc/sec.
Activation Gases	N-17	14. s	$<1 \times 10^0$
	N-16	7.35 s	$<1 \times 10^0$
	O-19	29 s	$<1 \times 10^0$
	N-13	10 m	2×10^2
	Ar-41	1.83 h	6×10^0
	Ar-37	34.3 d	2×10^{-4}
	H-3	12.36 y	1×10^{-3}
Noble Gases	Short H. L.	1-41 s	$<1 \times 10^0$
	Kr-89	3.2 m	3×10^0
	Xe-137	3.8 m	9×10^0
	Xe-135	15 m	8×10^1
	Xe-138	17 m	3×10^2
	Kr-87	1.3 h	2×10^2
	Kr-83m	1.86 h	2×10^1

Typical Off-Gas Release Rates (Continued)

	Isotope	Half-Life	Emission Rate $\mu c/sec.$
Noble Gasses (continued)	Kr-88	2.8 h	2×10^2
	Kr-85m	4.4 h	6×10^1
	Xe-135	9.2 h	1×10^2
	Xe-133m	2.3 d	2×10^0
	Xe-133	5.27 d	5×10^1
	Xe-131m	12.0 d	2×10^{-1}
	Kr-85	10.4 y	7×10^{-2}

The solid daughter products of the noble gases are removed in the filter of the off-gas system before release of gases to the stack.

If fuel failures were to occur, the quantity of noble gases discharged from the stack would be proportionately higher than shown in the tabulation, and would be dependent upon such factors as the number of failures, the size of failures, and their location in the core. In all cases, the aggregate release of radioactive materials from Dresden Units 1 and 2 shall not exceed the stack release limits prescribed in AEC Facility License DPR-2, as amended, which currently governs operation of Dresden Unit 1.

Liquid Wastes

Activity released with the liquid wastes is difficult to define since liquid wastes come from a number of sources and the quantities of activity is a strong function of plant operation including holdup time. The total amount of activity and the relative quantities of each isotope may vary significantly from day to day.

The following list of the most significant isotopes in liquid waste discharge off-site may be considered representative.

Isotope	Half-Life	Release Rate $\mu c/day$
Sr-89	54 d	1.5×10^3
Sr-90	28 y	1×10^1
Sr-91	9.7 h	7.5×10^3
Cs-137	28 y	1.5×10^1
Ba-140	12.8 d	4×10^3
1-131	8.05 d	1×10^3
1-133	20.8 h	1.5×10^3
Cu-64	12.8 h	1×10^2
Co-58	72 d	3.5×10^4
Co-60	5.27 y	4.5×10^3

The above information is based on data obtained from operation of Dresden Unit 1, adjusted wherever necessary to conform to the larger size of Dresden Unit 2.

QUESTION

II Plant Design Characteristics (Continued)

15. With reference to seismology and seismic design the following should be provided.

a. A statement of the maximum allowable stresses in terms of a percentage of working or yield stress for Class I and Class II structures and components when under combined seismic and functional load from a 0.1 g and 0.2 g earthquake.

b. A statement of the manner in which horizontal and vertical accelerations are combined.

c. A family of curves on a log-log plot of the acceleration response spectrum as a function of period from 0.01 second to 1.0 second for various values of damping. State how these curves will be used in the seismic design of facility structures and components.

d. A table of damping factors and natural periods for typical structures and components.

e. A statement of the seismic design criteria of the facility stack.

ANSWER

a. Class I structures and equipment will be designed as follows:

The stresses resulting from the earthquake accelerations shown on the family of curves given in Figure 54 of the Plant Design and Analysis Report (refer also to the log-log curves given in response to question 15.c, below) when combined with functional loading stresses, will be maintained within the established allowable working stresses for the particular materials involved.

The combined stresses resulting from functional loadings and from an earthquake having a ground acceleration of 0.20g will be such that a safe shutdown can be achieved. The combined earthquake and functional load stresses probably will not exceed yield stress. However, where calculations indicate that a structure or piece of equipment will be stressed beyond the yield point, an analysis will be made to determine its energy absorption capacity. This capacity will be such that it exceeds the energy input from the earthquake. In addition, the design will be reviewed to assure that any resulting deflections or distortions will not prevent the proper functioning of the structure or piece of equipment and will not endanger adjacent structures or components.

It is not intended that Class II structures and components be analyzed for 0.1g and 0.2g earthquakes. The seismic design of Class II structures and equipment will be in accordance with applicable codes following the normal practice for the design of power plants in the State of Illinois. As a minimum, the design will meet the requirements of the "Uniform Building Code" for Zone 1, including the

allowable stresses specified for combined functional and earthquake loadings. The usual practice of determining the stress due to earthquakes by applying a static load based on a specified seismic coefficient will be followed.

b. For the design of Class I structures and equipment, the maximum horizontal acceleration and and the maximum vertical acceleration will be considered to occur simultaneously. The resulting seismic stresses for the two motions will be combined linearly.

c. Figure II-15-1, shows a family of curves on a log-log plot of the acceleration response spectra for the design earthquake for various values of damping.

These curves are essentially the same as curves presented as Figure 54 of the Plant Design and Analysis Report except that a logrithmic scale has been used instead of an arithmetic scale.

These curves will be used for the seismic design of Class I structures and equipment. The allowable stresses resulting from combined seismic and functional loads will be as given in the answer to question 15. a, above.

d. The following table of damping factors and natural periods of vibration are based on analyses performed on as-designed structures and equipment for the Oyster Creek Unit No. 1, except as noted:

Structure or Component	Natural Period, sec	Assigned Damping Factor, %
1. Reactor Building (one mode)	0.256	10
2. Drywell		
a. Empty	0.459	2
b. Flooded	0.13	
3. Suppression Chamber		
a. Flooded	0.21	3
b. Water to normal operating level	0.142	
4. Reactor Vessel and Supporters	0.148	2
5. Recirculation Piping Loops	0.075 max	0.5
6. Stack * (one Mode)	1.87 max	5

(* Existing 300-ft-high stack at Dresden)

Since the reactor building, containment vessels, reactor vessel and recirculation piping for Dresden Unit 2 will be similar to those provided for the Oyster Creek Unit, the natural periods given in this table can be considered as being representative. Damping values will be assigned in accordance with the factors given in Section V-6 of the Plant Design and Analysis Report.

e. The seismic basis for the existing 300 foot stack is as follows: Seismic forces were analyzed in conformance with the report of the ASCE-SEAONC Joint Committee (1951), ASCE Separate No. 66, or Transactions Vol. 117, Page 716, which recommends that the total base shear be distributed over the height of the structure in accordance with the following formulae:

$$F_x = \frac{V W_x h_x}{\Sigma(Wh)}$$

and $V = CW$

where V = Total lateral base shear;

$C = 0.033$;

W = Total weight of stack;

W_x = Weight of increment under consideration (use at least 10 increments),

h_x = Height of increment above base;

$\Sigma(Wh)$ = Summation of products of weight of each increment and its height above base; and

F_x = Lateral force at section under consideration.

The stack was also designed to resist wind pressure in accordance with ACI 505-54 for a gust velocity of 110 miles per hour.

The firm of John A. Blume and Associates has recently completed an investigation of the existing 300-foot stack. The stresses and response of the structure were determined. The results of this analysis indicate that the existing 300-foot-high stack has the earthquake resistance capabilities as prescribed in the criteria for seismic design in Section V-6 of the Plant Design and Analysis Report.

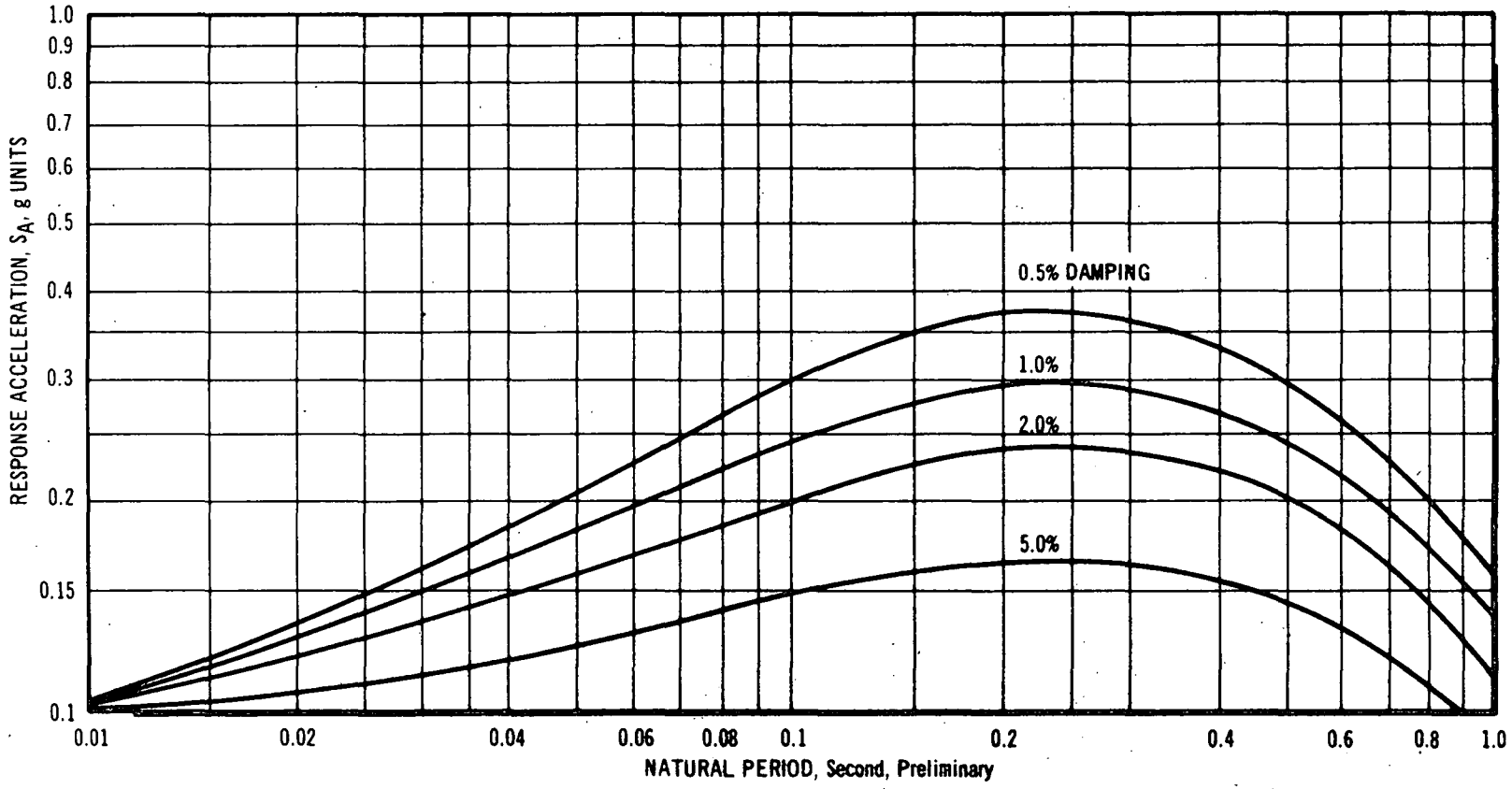


Figure II-15-1. Acceleration Response Spectrum

QUESTION

II Plant Design Characteristics (Continued)

16. We believe that the tornado design criteria should be stated in a more definitive manner. If possible, divide the structures into classes in a manner parallel to that used in the seismic design criteria so as to provide for a consistent appraisal of the relative importance of the various structures.

ANSWER

As stated in Section V-6.1.1b of the Plant Design and Analysis Report the structures are designed to assure that safe shutdown of the reactor can be achieved considering the effects of possible damage to the structures when subjected to the forces of short-term tornado loadings up to 300 mph. This design basis is interpreted to mean that the reactor primary system as described in Section V-2.0 of the Plant Design and Analysis Report, and components which directly affect the ultimate safe shutdown of the plant are located either under the protection of reinforced concrete or are located underground.

The technical literature provides little information on the wind velocities within a tornado, and the selection of 300 mph as a design value is arbitrary. The U.S. Weather Bureau has issued a publication (1) which states:

> "No structural damage is known to have resulted to a reinforced concrete building in a tornado."

A most important design and safeguards consideration is the question of what damage level to evaluate. Recognition is given to the fact that superstructure damage could be incurred to the reactor building, turbine building, storage tanks, and incoming power lines without affecting the ability to shutdown the reactor and to maintain integrity of containment and certain heat removal systems during and following a tornado which might traverse the site. Simultaneous damage to all of these items is not expected, yet as a design objective the power plant may be safely shutdown and maintained in a safe shutdown condition with the loss of all such superficial equipment.

Those items of equipment or systems located either underground or within the protection of the reinforced concrete include the following:

The Reactor Primary System
Shutdown Heat Exchangers
Control Rod Drive Hydraulic Equipment
Standby Liquid Control System
Primary Containment and Isolation Valves

Isolation Condenser System
Core Spray Systems
Containment Spray System
Service Water System
Station Battery
Diesel Generator
A Portion of the 34.5 KV Incoming Electrical Transmission Line
Electrical Controls and Instrumentation for Above Systems
Control Room

With the above equipment available, the Unit is an independent entity with safety defense capability in depth to maintain a safe shutdown condition for prolonged periods, if required.

REFERENCES

(1) Gilbertson, V. C. and Magenau, E. F., "Tornadoes," AIA Technical Reference Guide, TRG 13-2, U.S. Weather Bureau.

QUESTION

II Plant Design Characteristics (Continued)

17. The electrical power supply is a feature of the proposed facility which must be considered in detail since the power supply must function to allow operation of the engineered safeguards which protect the facility after accidents. To provide a more definitive basis for evaluation of the adequacy of the electrical power supply please provide the following information:

a. A scale drawing of the proposed site on which the major electrical equipment is located in relation to other facility structures.

b. A scale map of the proposed site and environs indicating the rights-of-way of the electrical transmission lines.

c. The manner in which the major electrical equipment, including the diesel generator and transmission lines will be protected from the effects of tornado force winds so that a continual source of electrical power can be assured.

d. A summary of experience to date on the reliability of existing 34.5 KV, 138 KV and 345 KV transmission lines.

ANSWER

a, b. The scale drawings of the site and environs which indicate the locations of major electrical equipment and the rights of way of the electrical transmission lines are shown on Figures II-17-1 and II-17-2.

c. The primary feature of the electric power system serving Dresden Station which assures a continuous source of auxiliary power to Unit 2 is the diversity of dependable power sources which are physically isolated so that any one instrument of failure affecting one source of supply will not communicate to alternate sources. Auxiliary power can be supplied from four separate and independent sources: Unit 2, the 138 KV transmission system, a 34.5 KV line and a standby diesel generator.

The auxiliary power supply from Unit 2 is provided by generator leads connected through a step-down transformer located adjacent to the turbine building. Protection of this source of power is achieved by reason of the very high reliability of the 345 KV transmission system construction.

The auxiliary power supply from the 138 KV transmission system is protected against the effect of tornadic winds by the diversity of six separate 138 KV circuits and a major generating unit feeding into the 138 KV switch yard at the Dresden site. Any one of these 138 KV circuits has sufficient capacity to furnish all of the auxiliary requirements of Unit 2.

Five of the 138 KV circuits are carried on two double-circuited tower lines and one single-circuited tower line which leave the plant in a northerly direction on a common right-of-way approximately 2-1/3 miles in length. At this point the circuits are separated on diverging rights-of-way: two circuits on a right-of-way running westward to Kewanee and Streator; two on a right-of-way leading eastward to Joliet; and the remaining circuit on a right-of-way running northward to McCook and Lombard.

The assurance of an adequate power supply from one of these five 138 KV circuits is indicated by the fact that the Edison system has never experienced the simultaneous loss due to tornado action of all transmission lines on a multiple transmission corridor. The Edison system has approximately 186 miles of multiple tower transmission corridors.

But even if it were considered credible that tornadic action would disrupt simultaneously all of the five 138 KV circuits on the common right-of-way, there remain two other sources of 138 KV power to meet the auxiliary power requirements of Unit 2. First, a sixth 138 KV transmission line is planned for installation prior to the completion of Unit 2. This line will enter the plant on a separate, independent right-of-way connecting the plant with the Bradley Transmission Substation to the south. Second, Unit 1 feeds into the 138 KV switch yard at the plant and thus is also available as a source of auxiliary power.

Based on the Edison system experience, the diversity in number of lines on diverging rights-of-way provide reasonable assurance that tornadic winds will not interrupt all sources of 138 KV supply to the 138 KV switch yard at the plant.

In the event the tornadic effects were to damage or incapacitate the 138 KV switch yard, there are still two other independent sources of auxiliary power: a 34.5 KV line and a standby diesel generator. Both sources are connected to the same auxiliary buses and each has capacity for operation of all systems required to shutdown Unit 2 and maintain it in a safe shutdown condition as described at pages VII-3-2 and VII-3-3 of Volume I of the Unit 2 Plant Design and Analysis Report.

The 34.5 KV line enters the plant from the south on a wood pole line to a point approximately 1,100 feet south of the 138 KV switch yard, at which point the line is sectionalized and a tap to serve Unit 2 will be placed underground.

The standby diesel generator will be housed in a concrete block cell at grade elevation within the turbine building. Equipment connecting the standby diesel generator and the 34.5 KV line to the auxiliary equipment needed for shutdown are protected from tornadic winds by metal enclosed switchgear located within the turbine building and underground cable with no exposed terminals.

Since the destructive path of a tornado is extremely narrow, and because of experience which is related in answer to Question 17 d, it is considered that a continued source of auxiliary power is assured to Dresden Unit 2 by the facilities described above.

d. The Commonwealth Edison system has never experienced tornado damage that has completely disabled a transmission terminal. The greatest damage that has occurred to a transmission terminal due to tornado action has been the loss of one bus-section (due to airborne debris).

Commonwealth Edison has approximately 186 miles of transmission corridors occupied by more than one line of transmission structures. The simultaneous loss of more than one double-circuited tower line due to tornado action on such corridors has never been experienced.

Nor has Commonwealth ever experienced the loss of a power production generating unit due to tornado action.

The transmission terminal at Dresden, now connected with five 138 KV supply lines, was put into service in 1960. All lines are equipped with one or two-shot automatic re-closing, which is designed to restore the line to service in approximately three and one-half seconds after trip-out due to a lightning stroke or other cause. In approximately 5-1/2 years, one line has experienced no unscheduled outages of more than 2 minutes and has been momentarily tripped by lightning twice. The second line has experienced two outages of more than 2 minutes and has been momentarily tripped by lightning three times. The third line has experienced three outages of more than 2 minutes and has been tripped by lightning four times. The fourth line has experienced four outages of more than 2 minutes and has been tripped by lightning 10 times. The fifth line has experienced 4 outages of more than 2 minutes and has been tripped by lightning once. At no time during this period have both sections of the 138 KV bus at Dresden been out of service at the same time when unscheduled. Of the outages enumerated above, the largest number of simultaneous outages has been two of the five lines.

The 34.5 KV terminal was established at Dresden in 1963 and in approximately 2 years of operation one line has experienced 9 outages of more than 2 minutes and 24 momentary interruptions. The second line has experienced one outage of more than 2 minutes and has experienced 4 momentary interruptions. The third line has experienced one outage in excess of 2 minutes and 6 momentary interruptions. Most of the momentary interruptions were caused by lightning.

As of this date, there is no 345 KV transmission constructed into the Dresden area. However, based upon system experience with 345 KV transmission since 1958, it is expected that the proposed 345 KV construction will be more trouble-free than either the 138 KV or 34.5 KV which are reported above.

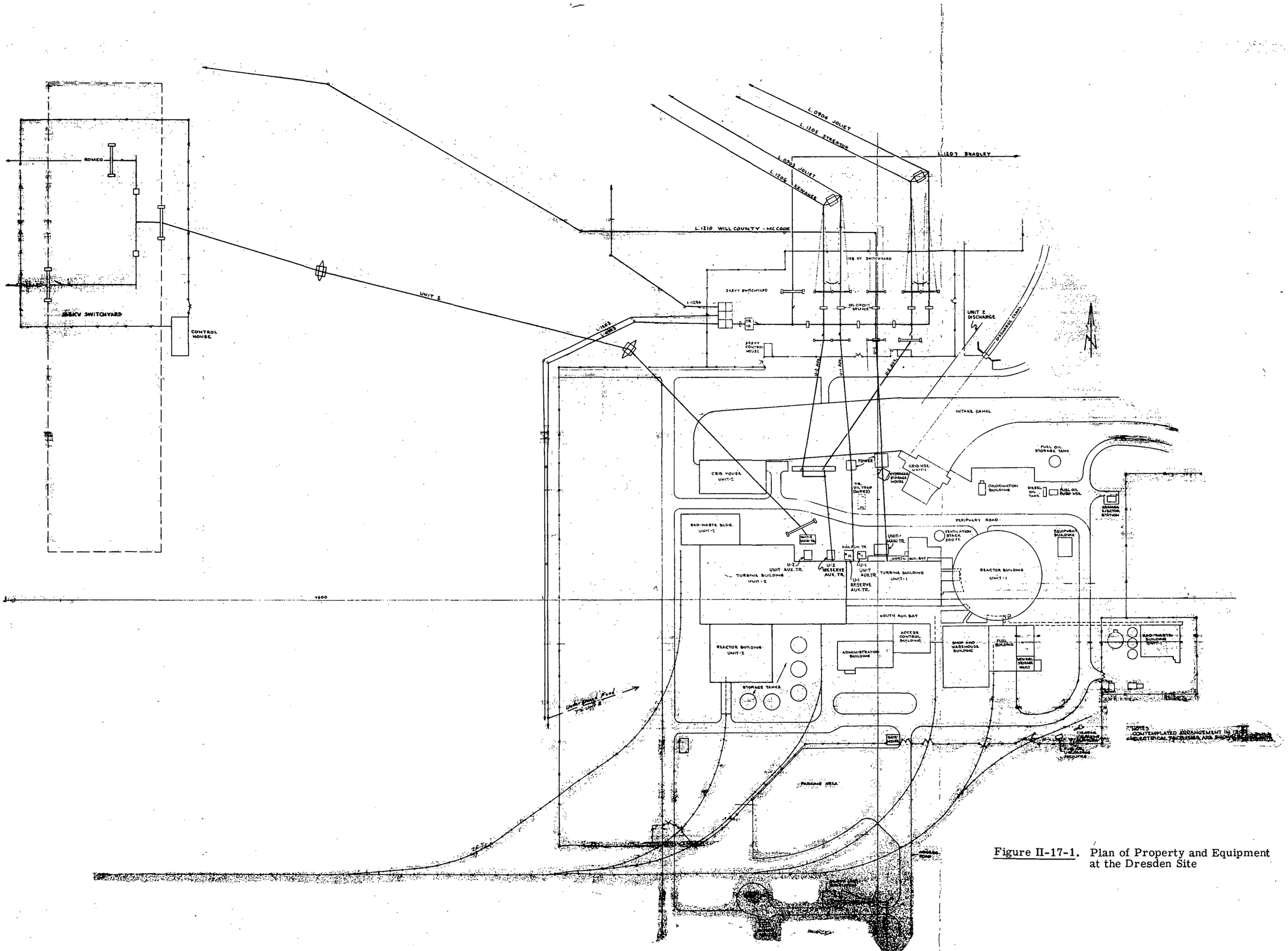
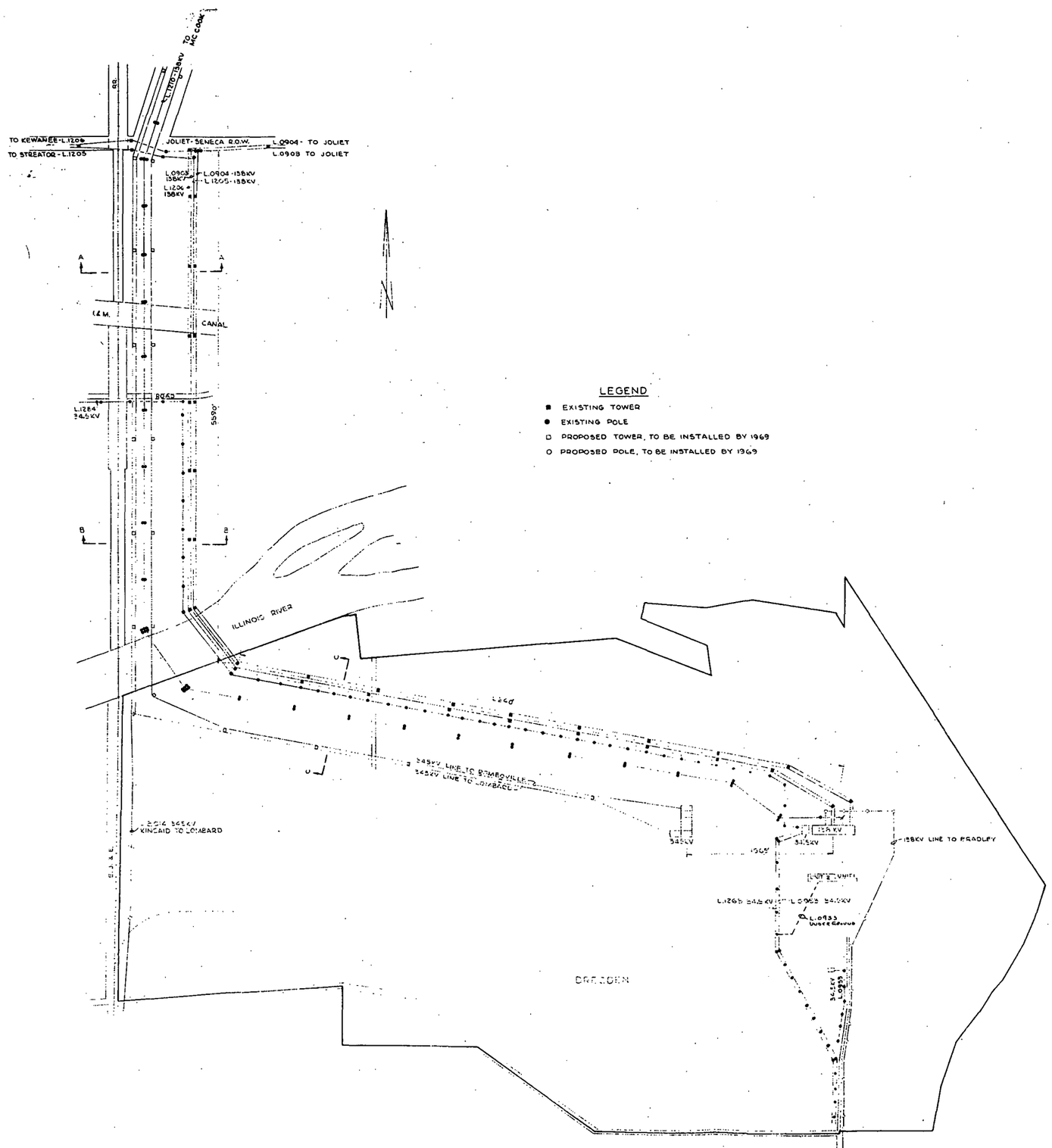
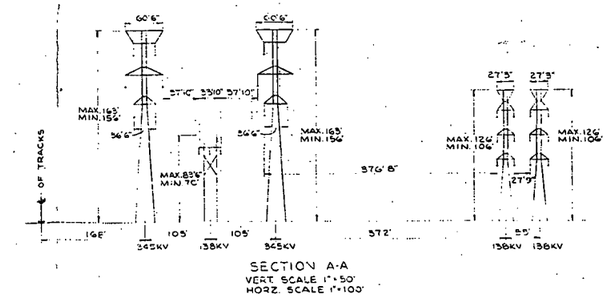


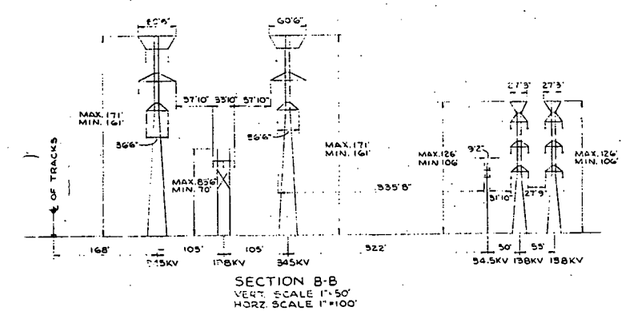
Figure II-17-1. Plan of Property and Equipment at the Dresden Site



LEGEND

- ■ EXISTING TOWER
- ● EXISTING POLE
- □ PROPOSED TOWER, TO BE INSTALLED BY 1969
- ○ PROPOSED POLE, TO BE INSTALLED BY 1969





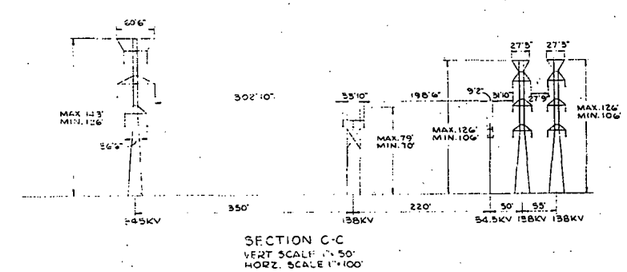


Figure II-17-2. Transmission System at the Dresden Site

QUESTION

II Plant Design Characteristics (Continued

18. Describe the experimental test program to verify that the proposed "Campbell" system truly represents reactor flux under all steady state conditions, and that the response of the system to transients is acceptable.

ANSWER

The static sensitivity of the system has been measured for a range of neutron fluxes and for different chambers, and demonstrated to be predictable and well understood.

Verification of the acceptability of the dynamic response of the system is both experimental and analytical. Measurements have been made of the neutron flux transient during shutdown of a reactor with the "Campbell" system. The measured transient was in good agreement with the transient measured using a conventional D-C ionization system. This type of test is used to verify the acceptability of the dynamic response of the "Campbell" system during typical reactor transients.

Extensive analysis of the dynamic characteristics of the system have been performed. The validity of the analytical models has been verified experimentally. Measurements were made of the variation of output from a "Campbell" system using a scintillation crystal exposed to a rapidly varying gamma field. These experimental data were in excellent agreement with the analytically predicted output signal variation.

The results of an extensive program is reported in the series of monthly, quarterly and a pending final report under AEC Contract AT (04-3)-189, Project Agreement 22. Some of the key reports which have been prepared and submitted to the AEC include the following:

GEAP	4604	"Application of Counting Techniques to In-Core Ion Chambers"
GEAP	4747	"Theory of the Campbell System of Reactor Instrumentation"
GEAP	4797	Cable for Pulse Counting and Campbelling
GEAP	4862	Reactor Control System Based on Counting and Campbelling Techniques.

The GEAP 4862 contains a great deal of the experimental information.

QUESTION

II Plant Design Characteristics (Continued)

19. Please show that the intermediate and power range instrumentation will protect the core under all conditions of allowed bypassing of individual chambers and entire APRM units. Further, verify that every region of the core is protected, under worst case-conditions rod withdrawal, by APRM units connected to separate scram logic channels.

ANSWER

The design characteristics of the neutron monitoring system have been described in the answer to question II-5e. As noted the signal output from the intermediate and power range instrumentation is utilized in the logic circuit of the reactor protection system. By-passes and interlocks are provided to permit maintenance work on the systems, as is done on the other instrumentation systems connected into the reactor protection system, without causing unnecessary scrams and without voiding the functional performance requirements of the reactor protection system.

As required by the design criteria, the number and distribution of instruments in the core will be specified to maintain core protection from gross power disturbances with the maximum number of units bypassed. The final details with respect to number and locations of monitors, interlock requirements, bypass conditions, and response characteristics, will be based on detailed calculations which relate these requirements to the core performance parameters. Gross core power disturbances initiated, for example, by coolant flow rate changes, are sensed generally throughout the entire volume of the core. By assuring, under permitted bypass conditions, that at least one active instrument channel is still situated in each region of the core, full scram protection is maintained in the event of gross core power disturbances.

In the event of local power disturbances due to control rod withdrawal, core protection is provided by blocking rod motion rather than scramming in the event a limiting condition is approached.

Every region of the core is protected, under worst-case conditions rod withdrawal, by APRM units connected to separate safety logic channels. For the limiting local flux disturbance, the worst condition rod withdrawal, one APRM channel in each of the two reactor protection system channels will respond sufficiently to initiate a rod withdrawal block before fuel damage occurs. The rod withdrawal block is initiated by any single APRM trip, thus fuel damage is prevented even if one of the adjacent APRM units is bypassed. Analyses of the APRM system response in a large BWR using the same instrument arrangement criterion has shown that the local power disturbance from a worst case rod withdrawal is limited by the rod block device to less than fuel damage levels.

QUESTION

II Plant Design Characteristics (continued)

20. List the design criteria for the reactor instrumentation and safety systems (both nuclear and process) and the instrumentation and controls for the containment isolation and other engineered safeguards. Of especial interest are those parameters which, when their set points are exceeded, must cause a scram in order to protect the reactor. The criteria should include consideration of circuit fail-safety redundancy, invulnerability to single failure, testability, etc., relating to such parameters. Consideration relating to parameters and their associated circuits, of less safety importance should also be included.

ANSWER

Nuclear Instrument Criteria

A. Source Range

The sources and detectors will provide a minimum signal-to-noise ratio of 3/1 and a minimum count rate of 3/sec during fully controlled conditions prior to initial power operation. During subsequent operations, the above conditions will be met before the reactivity of the core exceeds the reactivity which existed during fully controlled conditions prior to initial power operation.

B. Intermediate Range

The intermediate range instrumentation will be capable of generating a trip signal to be used to prevent fuel damage due to single operation errors or equipment malfunction occurring during any low or intermediate range operation. For the worst possible combination of instrument bypass and startup rod withdrawal, the intermediate range instrument system will be capable of generating a scram signal before the bulk fission power of the reactor exceeds 0.1% of rated power and before the power density of any region in the reactor exceeds 5% of the average power density at rated power.

C. Power Range

For the worst permitted APRM channel bypass and the maximum permitted number of inoperative in-core chambers in any individual APRM channel the system will be capable of generating a scram signal during bulk neutron flux level transients before the actual bulk neutron level of the reactor exceeds 120% of the rated value or the value which results in an adequate margin for partial flow conditions.

Under the same conditions of bypass and inoperative chambers the APRM system will be capable of generating a signal to prevent rod withdrawal and to prevent flow changes in order to prevent fuel damage occurring from a local power disturbance during the worst single rod withdrawal due to operator error or equipment malfunction starting from any permitted power and flow condition.

Fuel damage is defined as perforation of the fuel cladding which would permit release of fission products. The mechanisms which cause fuel damage in reactor transients are:

a. Severe overheating of the fuel cladding caused by inadequate cooling.

b. Fracture of the fuel cladding due to strain caused by relative expansion of the UO_2 pellet.

The critical heat flux which defines the onset of the transition from nucleate boiling to film boiling is identified as the limit, below which fuel damage due to overheating will not occur.

A value of 0.7% of 1% plastic strain on the Zircaloy cladding is identified as the limit below which fuel damage due to overstraining the fuel cladding is not expected to occur. The heat flux to cause this amount of clad strain is approximately 600,000 Btu/hr-ft^2.

Neutron Monitoring Instrumentation Design Criteria

Each instrument will be designed with internal monitoring having capabilities to provide an alarm, a scram trip, and/or a rod and flow block due to an inoperative condition such as loss of power, switched out of operation when not properly bypassed, etc. Upscale and downscale trips provide alarm-rod block to insure that the instrument will be within the proper operating range. Excessively high scale will provide a scram trip in those instruments in the Reactor Protection System.

A number of channels of each type of instrument will be provided to allow a specific number to be out of service at one time for maintenance and test and still provide the required core coverage for information and/or safety. Those systems that provide protection to the reactor will always have a minimum of one channel located in each of the four Reactor Protection System logic subchannels.

Reactor Protection System Design Criteria

A. General

The reactor protection system, which has the function of scramming the reactor and providing containment isolation signals, shall be designed to provide the highest practical degree of plant safety with continuity of service as the second basic criterion. In order to provide an optimum safe compromise between reliability and freedom from spurious scrams, a two channel system has been selected as the design. Each of the two channels has at least two independent tripping sensors from each measured variable. Only one sensor must operate to trip the channel in which it is connected but both channels must trip to cause a scram. Electro-mechanical relays shall be used as the "Logic Elements" of the system.

Design objectives shall include simplicity of design and operation, ease of checking and maintenance, and arrangement with appropriate physical isolation.

The parameters which are utilized to scram the reactor, close isolation valves, and initiate other protective actions are discussed or described in Section X-5.3 and Figures 40 and 41 of the Plant Design and Analysis Report. These specific parameters have been chosen on the basis that where practical they represent direct process variables, and that either singly or in combination they will

effect the appropriate protective actions. The set points for the various sensors will be established when the final design and system performance and operating characteristics are known. These set points will take into account appropriate margins between the operating limits and the real or safety limits.

B. Reliability

The reactor protection system shall reflect an optimum safe compromise between reliability and freedom from spurious scrams, i.e., negligible probability of failure to scram on a predetermined condition and a very low probability of a spurious scram under normal operating conditions.

1. Dual Channel System

The system shall be of the dual-channel, fail-safe type requiring that both channels trip to initiate a plant shutdown as illustrated schematically on Figure 39 of the Plant Design and Analysis Report. Scram signals from duplicate sensors will feed into separate protection channels. At least one trip signal must be received by each channel in order to initiate a scram or isolation unless one of the channels has tripped because of failure, in which case a single trip signal from the other channel will initiate action.

The two channels shall be physically separated from each other and from other equipment to minimize probability of false scrams or failure to scram resulting from interaction or common causes. Each channel shall be clearly identified to reduce possibility of maintenance personnel (or others) causing inadvertent trips or mal-operations.

2. Single Channel Trip

A single channel trip shall result from any single sensor signal reaching the trip value. The logic shall be as shown on Figure 39, i.e., one-out-of-two except for the neutron flux trips which are one-out-of-four normally (one-out-of-three during maintenance).

3. Component Failure Allowance

Failure of a single sensor circuit or system component shall not prevent insertion of a sufficient number of control rods to shut down the reactor.

Redundant sensor circuits (sensors, wiring, transmitter, amplifiers, etc.) shall be electrically and mechanically independent so that they are unlikely to be disabled by a common cause.

4. Fail Safe

Each scram trip shall be designed "fail-safe" insofar as practicable, i.e., most probable component failures (including power supply failure) or open circuit in wiring shall cause a trip condition.

C. Operation and Maintenance

1. Operation Testing

Each sensor circuit shall be capable of being tripped independently by simulated signals for test purposes to verify its ability to give a single-channel-trip without causing a scram. Failure to restore normal signals to sensor circuits after test must be guarded against by making such failure conspicuous to operating personnel and adequate check off procedures followed (or locks provided).

Each pair of scram solenoid pilot valves shall be provided with a test switch for rod-scram-time test and provisions made for connecting a timing recorder to the rod position indicating circuit.

A test switch shall be provided for the scram dump volume vent and drain valves.

2. Component Maintenance

The neutron monitors can be disabled one at a time for maintenance work, but a conspicuous indication of the bypassed condition shall be given. Other sensors are not provided with maintenance bypasses.

Protection System Interlock Criteria

A. General

Interlocking mechanisms that can affect the reactor protection system are so designed and/or applied that expected failure modes of the device will render it impotent as far as the protection system is concerned. Where a conflict between plant safety and continuity of operation exists, plant safety is given first consideration and continuity of plant operation second consideration.

B. Relays - "Fail Safe" Mode

1. The deenergized condition of relays shall not cause bypass of any protective function.

2. Where contact welding could bypass a protective function, more than one contact shall be placed in series.

3. Continuous duty current ratings of relay contacts from interlocking and/or bypass function shall be at least five times their actual load.

C. Disabling of measuring channels

Disabling of a Neutron Monitor will give a trip signal except any one of 4 monitors in each of the two protection channels may be bypassed unless the two monitors are in the same quadrant of the core.

D. Monitoring of potent variables

All variables that can shutdown the plant automatically (except main steam line break) are continuously indicated and/or recorded in the main control room.

E. Independence of interlocks

No single relay shall serve interlocking functions in both protection channels, i.e., power from both protection channels shall not feed through contacts of a single relay.

F. Quality of bypass relays

Relays used for Bypassing service shall be of a quality equivalent to (or better than) the sensor relay contact being bypassed.

G. Annunciators

Annunciator circuits shall be self-monitoring, i.e., opening of a contact shall initiate an alarm.

H. Indication of interlocks

An interlock condition shall be indicated by an indicator light or an annunciator or both actuated by the bypassing device (relay or switch).

Isolation Valves Control Design Criteria

In order to meet the various operational and safety criteria associated with the various isolation valves for the reactor steam and auxiliary system, the following general rules have been followed in the design of control circuitry for the valves.

1. Failure of any single component or power supply shall not prevent isolation of any drywell effluent line.

2. Loss of all AC power shall not initiate main steam line isolation valve closure.

3. Loss of all AC power shall not prevent initiation of main steam line isolation.

4. All isolation signals are fed into the dual-redundant protection system logic relays for confirmation and trip action.

5. Manual switches on the Control Panel in the Control Room back up all trip signals for all valves.

6. Manual reset is required after trip in order to restore normal conditions.

QUESTION

III Accident Analysis

1. We believe that the goal of the accident analysis is not only to describe the probable course of an accident but also to evaluate a reasonable upper limit of potential hazard to the public from the occurrence. Assumptions made at each step of the analysis must therefore be supported by reactor experience or experiments which are clearly applicable to the conditions prevailing during the course of the accident to assure (1) an adequate appraisal of the possible consequences of the accident, and (2) that unjustified credit has not been taken for some mitigating factors. We note that the analysis of "Major Accidents" described in Section XI-5 of the report, considers that the source of radioactive material available for release to the environs would be decreased by the following effects:

   a. Only 1% of the noble gas activity and 0.5% of the halogen activity would be available from those fuel rods which experience cladding perforation,

   b. The concentration ratio of halogens in steam to water would be 10^{-5},

   c. The partition factor (water concentration/air concentration) of halogens would be 10^{2},

   d. The efficiency of the reactor building filters would be 99%, and

   e. Only 10% of the halogens would be in organic form and that such material can be filtered.

   For each of these effects some experimental data is available which tends to support its use in the calculation of the accident consequences. It has not been clearly shown, however, that the experimental data applies to the assumed accident conditions. For each of the effects listed above, please cite the applicable experimental data upon which credit for the effect is based, and compare all conditions of the experiment with those assumed for the accident thereby proving that the experimental data clearly applies to each of the accident cases. When experimental evidence has not been recorded in the published literature special care should be taken to describe the exact conditions under which the experiment was run.

ANSWER

a. Some of the fission gases, both radioactive and non-radioactive, leak from the fuel lattice into the plenum between the UO_2 fuel and the cladding. The half-lives of most fission gases are short compared to the duration of reactor operation and activity buildup in the plenum, so the gases in the plenums are predominantly non-radioactive.

   The release rates of noble gases and halogens from the fuel are dependent on the fuel density, operating temperature, fracture or crumbling status of the fuel and the gas pressure in the plenum. If the fission gas release rates from the UO_2 are known, the fission gas buildup in the plenum can be calculated.

The release rate of noble gases into the fuel plenum can be estimated from the measured release rate of noble gases from failed fuel in an operating reactor, Dresden Unit 1. A failed fuel rod is one which has experienced a break in cladding integrity, from a small hole to a circumferential severance. With variation in reactor power, water-steam can leak into fuel rods operating with failed cladding and cause leaching of fission products from UO_2 or deterioration of UO_2. This increased release of fission gases from the fuel will cause an overestimation of gas buildup in the fuel plenum and therefore will make this calculation conservative. When the fuel cladding is unbreached the gases released into the plenum cause a pressure buildup which retards release of additional gases into the plenum. This decrease in fission gas release due to back pressure does not occur when the cladding is breached, so this calculation, based on the measured release from failed fuel, is conservative. The release rate of fission gases from failed fuel rods during Dresden Unit 1 operation, which operates at about the same peak and average fuel temperatures as Unit 2 fuel, and has about the same fuel density, is measured to be below 1000 μc/sec of noble gases and below 250 μc/sec of halogens (both measured after 30 minutes decay).[1] If the number of fuel rods with cladding failures is actually greater than those verified, the release per fuel rod is even less and this calculation is that much more conservative.

The mixture of fission gases from the perforated rods is between the recoil mixture (immediate release upon formation - no holdup for decay) and the diffusion mixture (some holdup for decay). The diffusion mixture, which is weighted toward long-lived isotopes is used in the calculation for additional conservatism.

The equilibrium buildup of fission gases in the fuel plenums is calculated using the following equation:

$$A_o = \frac{R_A}{\lambda_A}$$

Where,

A_o = Equilibrium activity of isotope A in plenum (curies)

R_A = Release Rate of Isotope A from fuel (c/sec)

λ_A = Decay constant of isotope A (sec^{-1})

$$= \frac{0.693}{(T\ 1/2)_A}$$

Since Kr-85 has a 10.4 year half-life, no credit was taken for decay in the plenum. Assuming 1000 days of buildup of Kr-85 in the plenum, its radioactivity is calculated by

$$A_o = 1000 \text{ day } 8.64\ (10)^4 \frac{\text{sec}}{\text{day}} R_A$$

The total noble gas activity (Kr-85 and shorter half-life isotopes) in the plenum of one rod is calculated to be 54 curies. The total noble gas activity in one fuel rod is 8700 curies. Therefore the noble gas plenum radioactivity in a fuel rod is less than 1% at equilibrium.

The halogen activity in the plenum of one rod is calculated to be 39 curies. The total halogen activity in one fuel rod is 10,000 curies. Therefore, the halogen plenum radioactivity in a fuel rod is less than 0.5% at equilibrium.

The radiogas buildup in the fuel rod plenums occurs over a period of years. The reactivity excursions included in the accident analyses sharply increase the fuel temperature above operating conditions, followed by rapid cooling to below operating temperatures in a few seconds. The short period of increased fission gas diffusion rates at the higher fuel temperature is negligible compared to the extremely long time at operating temperatures, and results in a negligible release of radioactivity from the fuel.

We have conservatively assumed that 1% of the noble gases and 0.5% of the halogens in a fuel rod are released from perforated rods (enthalpies >170 cal/gm).

b. The partition factor (concentration of halogens in water/concentration of halogens in air) for halogens has been measured to be 6×10^6. The measurements* have been made at Dresden Unit 1 at operating power, with full steam flow and voids. The coolant pressure, temperature and chemistry are very similar to Dresden Unit 2. The measurements demonstrate that, even at high flow, saturated steam conditions with a high steam void content, the halogens are absorbed in the water and remain there.

The halogen concentrations in the reactor coolant water and in the condenser hotwell water were measured. The halogens carried away by the off-gas were measured and found to be too low to alter the calculation. Because the coolant-steam-condensate is a straight through system, the ratio of halogen concentration in the condensate to the reactor water is the decontamination ratio, 3×10^4 to 10^5. Since the steam to condensate water density ratio is greater than 20, the partition factor (concentration in water/concentration in steam space) is 6×10^5 to 2×10^6.

Sodium is absorbed in water and is non-volatile at operating conditions so its concentration ratio in condensate water to reactor coolant water is a measure of the moisture carryover with steam. Comparing the sodium carryover to the iodine carryover, 63% of the iodine in the Dresden Unit 1 condensate was carried there in the moisture and only 37% was in the steam phase. Therefore, the actual partition factor is even higher than calculated above.

The calculated water-to-steam partition factor for halogens is used in the control rod drop and steam line break accidents, at operating temperature and pressure, to demonstrate that essentially the only halogens escaping the reactor are those absorbed in the water carried down the steam line.

c, d, e. The calculation of halogens released from the plant following a loss-of-coolant accident is based on extremely conservative assumptions with respect to (a) the fraction of halogens which are of organic form, (b) fallout, (c) partition between air and water, and (d) filterability.

Since organic halogens are much less soluble in water and are more difficult to filter, a conservatively large fraction of halogens were assumed to be organic. Fuel melting experiments[2, 3, 4] have resulted in 0.1% to 3% of the release halogens being of the organic form. For the loss-of-coolant analysis, 10% of the halogens released from the fuel are assumed to be organic form. This assumption is thus conservative by a factor of 3 to 100. The pressure suppression containment contains a

*March, 1963 and January through March, 1965

tremendous volume of water for absorption of halogens. The air-to-water volume ratio is about 2.6. All the organic halogens are assumed to remain airborne, although, at an air-to-water ratio of 2.6 about half would be expected to be absorbed in water.[5] At the halogen concentration that would result from a 100% core melt, an inorganic halogen partition factor of 3 $(10)^2$ has been measured.[5] In ORR in-pile UO_2 melting experiments, the condensation of the steam in the gas stream removed essentially all halogens from the gas stream.[6] The inorganic halogen partition factor according to Allen[7] would be $(10)^2$ at 100% melt conditions. Watson[8] reports the partition factor to be greater than $(10)^4$. These experiments, including both steam condensation in vapor suppression systems and in air correspond to the conditions accompanying a loss of coolant accident. The initial blowdown through the suppression pool is mostly air, the trailing phases of blowdown is essentially all steam. Most fission product release would accompany the final steam release and would be scrubbed efficiently by the steam condensation. Airborne inorganic halogen and solid fission products in the drywell would be rapidly removed by the containment spray and steam condensation and mixed with the water in the suppression chamber. For the accident analysis, a partition factor of 10^2 for inorganic halogens is used. Inorganic halogens are assumed to be re-evolved from the water as leakage from the containment reduces the inventory of airborne halogens. Combining the assumption of a high fraction of organic halogens with no absorption in water and the conservative water to air partition factor for inorganic halogens results in a very conservative high fraction of halogens remaining airborne for leakage from the containment. (These assumptions applied to dry containment would result in approximately 27.5% of the halogens remaining airborne, compared to 25% assumed in TID 14844.)

The halogens which leak from the pressure suppression containment into the reactor building are exhausted by the standby gas treatment system through a drier, a high efficiency filter and a charcoal filter. Because the high temperature and pressure steam atmosphere is contained within the pressure suppression system, the reactor building exhaust air is at low temperature and humidity and can be treated to reduce the humidity so that the filters will be very effective for removal of organic halogens. Tests on filter efficiencies have shown that inorganic halogens are removed by charcoal filters with efficiencies greater than 99.99%,[9, 10] and tests on filter efficiencies have shown that organic halogens are removed by charcoal filters at a relative humidity below 30% with filter efficiencies from 99.9% to 99.9999%.[4, 10, 11, 12, 13] The charcoal filter on the EVESR at Vallecitos Atomic Power Laboratory has been retaining organic halogens produced at power operation with a filter efficiency from 99.8% to 99.9% at a relative humidity of 10 to 15%. The standby gas treatment system design will be based on the latest experimental results. The system will be designed to provide the necessary humidity control, residence time in filters, etc. Thus, with this design, the assumption of only a 99% filter efficiency for the removal of inorganic and organic halogens by the standby gas treatment system is conservative by orders of magnitude to 10^4. The compounding of conservative assumptions used in the loss of coolant analysis results in calculated doses from halogens that are 20 to 1000 times higher than actually expected.

REFERENCES

1. Williamson, H. E., and Rowland, T. C., "Performance of Defective Fuel in the Dresden Nuclear Power Station," APED-3894, 1962.

2. Collins, D. A., et al., "International Symposium on Fission Product Release and Transport Under Accident Conditions," Paper 59, 1965, Oak Ridge, Tennessee.

   Stainless steel and zirconium clad UO_2 heated up to above 2000°C in steam or CO_2 atmosphere, 0.1 to 1.7% of released halogen is organic. 85% of organic halogens decompose at 800°C in steam. 2000°C is above the melting point of zirconium; at the melting temperature the fuel would slump and leave the core region. Therefore 2000°C is approximately the maximum cladding and fuel surface temperature of the core, and the range of release temperatures of the test corresponds to the range which would accompany a 100% core melt. This test is very appropriate to a 100% core melt.

3. Parker, et al., SIFTOR Draft, Volume II, Chapter 18, "Fission Product Release." Stainless steel clad UO_2 melted. 80% of released halogens were I_2, 20% adsorbed on particles, no detectable organic halogens.

4. Collins, R. D., and Hillary, "International Symposium on Fission Product Release and Transport Under Accident Conditions," Paper 44, April, 1965, Oak Ridge, Tennessee.

   Zirconium clad UO_2 (unirradiated, tracer irradiated and irradiated to 100Mwd/ton) heated to between the zirconium melting temperature and the UO_2 melting temperature in steam air atmosphere. The organic halogens were a very small fraction of the halogens released, except at the highest heating temperatures, where the organic fraction reached 2 to 3%. These tests correspond to the postulated 100% melt case where the fuel heats up in a steam-air atmosphere until the zirconium melts, allowing the fuel to fall out of the core region. The methyl iodides were filtered with coconut charcoal with an efficiency from 99 to 99.9% at 100°C.

5. Diffey, et al., "International Symposium on Fission Product Release and Transport Under Accident Conditions," Paper 41, April, 1965, Oak Ridge, Tennessee.

   Tests included bubbling air and air-steam containing elemental iodine, methyl iodide, hydrogen iodide and particulates through water. Tests included both a 3 mm diameter lute (vent pipe with lower end submerged) immersed 6 cm below the water surface and a 50 mm diameter lute immersed 50 cm. The air tests correspond to the beginning of the coolant blowdown, in which air containing essentially no halogens is blown through the vent pipes. The steam-air tests correspond to the major part of the coolout blowdown, when some halogens would be contained in the air-steam mixture. The later phases of the coolout blowdown, consisting almost entirely of steam and containing a higher halogen content, would benefit from virtually complete halogen scrubbing, and therefore were not included in these experiments. The tested lutes are smaller diameter than the suppression pool vent pipes, but the maximum tested vapor velocity of 110 ft/sec is greater than the actual blowdown vapor velocity when halogens are being carried in the steam. Additional work is being conducted to determine if scaling factors are necessary. The pool temperature was 50°C in the experiments, corresponding to the pressure suppression pool temperature.

The initial absorption of halogens in a pressure suppression pool would be almost complete, followed by re-evolution until an equilibrium partition factor (water/air concentration) is achieved. Therefore, experiments were also conducted to determine the equilibrium partition factor. For the halogen water concentration corresponding to the 100% melt loss of coolant, 3 $(10)^{-6}$ g mole/l, the measured halogen partition factor was 3 $(10)^{2}$ for inorganic halogens and about 3 for organic halogens.

6. Miller, et al., "International Symposium on Fission Product Release and Transport Under Accident Conditions," Paper 12, April, 1965, Oak Ridge Tennessee.

   Stainless steel clad UO_2 was melted in-pile in the ORR in dry, moist and steamy air. The melting temperatures correspond to those accompanying the postulated 100% melt loss of coolant accident. The halogens were significantly removed from the gas stream by condensation of the moisture.

7. Allen, T. L., and Keefer, R. M., "The Formation of Hypoiodus Acid and Hydrated Iodine Cation by the Hydrolysis of Iodine," JACS 77, No. 11, June, 1955.

   The iodine partition factor was measured for pH = 5.4 to pH = 7. For the halogen concentration corresponding to the 100% melt, within the measured pH range, the halogen partition factor is $(10)^{2}$.

8. Watson, Bancroft and Hoelke, AECL-1130, "Iodine Containment by Dousing in NPD - II," 1960.

   Dousing experiments and equilibrium iodine concentration experiments were conducted in 26 gallon and 45 gallon drums and a 120 liter flask. In the dousing experiments, water was sprayed into the drums containing airborne inorganic halogens and allowed to condense. The total volumes of halogens and water do not correspond to the pressure suppression system, but the important parameters of halogen concentration in water, water chemistry, and temperature correspond to the conditions accompanying a 100% melt loss of coolout. The experiments included halogen concentrations in water from 10^{-12} to 10^{-5} gm-mole/l; the 100% melt results in 3 $(10)^{-6}$ gm-mole/l. Variation of water and quality, pH and water additives did not alter the result of the experiments. The experiments were conducted with room temperature water. Over this range of experimental conditions, including those appropriate to the postulated accident, the measured decontamination factor was greater than 10^{4}.

9. Keilholtz, G. W., and Barton, C. J., ORNL-NSIC-4, "Behavior of Iodine in Reactor Containment Systems," Page 64, February, 1965.

   Coconut and coal charcoal filters were tested. An elemental iodine removal efficiency greater than 99.99% is achievable with practical filter systems, even with extended operation and high steam content.

10. Adams and Browning, "International Symposium on Fission Product Release and Transport Under Accident Conditions," Paper 46, April, 1965, Oak Ridge, Tenn.

    Tested bituminous coke, petroleum residue and coconut shell charcoal filters. At conditions ranging from room temperature to 150°C and from low relative humidity to steam conditions, elemental iodine is removed by practical systems with efficiencies greater than 99%. At low relative humidities all charcoal filters were effective in removing methyl iodide.

11. Collins and Eggleton, ORNL-NSIC-4, "Behavior of Iodine in Reactor Containment Systems," February, 1965, page 65.

Methyl iodide was filtered with varieties of charcoal. (207B) coal charcoal demonstrated methyl iodide removal efficiencies from 99.99% to 99.9999% at room temperature and 100°C.

12. Adams and Browning, ORNL-NSIC-4, "Behavior of Iodine in Reactor Containment Systems," February, 1965, page 65.

Methyl iodide, in room temperature air, was passed over a 1.75 inch thick (207 B) coal charcoal bed at 30 to 35 ft/min for 5 hours. With moist air and high relative humidity, methyl iodide removal efficiencies of 74% to 97.3% were obtained. But, with low relative humidity the removal efficiency was found to be 99.99%.

13. Collins, D. A., et al., "International Symposium on Fission Product Release and Transport Under Accident Conditions," Paper 45, April, 1965, Oak Ridge, Tenn.

Methyl iodide in carbon dioxide or steam was passed through charcoal filters. At low humidity coal charcoal removed methyl iodide with a 99.9999% efficiency at temperatures ranging from ambient to 100°C. At low humidity coconut charcoal removed methyl iodide with a 99.9% efficiency at 100°C. Relative humidities above 30% reduce the methyl iodide removal efficiencies of all charcoals. If the charcoal is impregnated with 4 amino pyridine or morpholine, a high methyl iodide removal efficiency (99.9%) is achieved even at 100% relative humidity.

QUESTION

III Accident Analysis (continued)

2. The results of the parametric study of the control rod drop accident as a function of power level should be provided to verify that the accident occurring at hot standby is the worst case. Specify the reactivity addition rate, control rod worth, and the initial power density assumed for each power level considered. For the worst case, discuss the relative importance of assuming an overpower scram as opposed to allowing the transient to proceed uncontrolled.

ANSWER

Figure III-2-1 shows the results of a parameter study where the effects of reactor power and moderator density were investigated for a rod drop accident where the total worth of the control rod involved was the maximum (0.025 Δk). The insertion rate was a 0.025 Δk/sec equivalent ramp for the 5 ft/sec rod drop. The initial power density was $10^{-6} \times$ rated power. This power level is highly conservative since the physical case will be several decades higher in power level.

The trends shown in Figure III-2-1 clearly indicate that the hot standby condition is worse than any full power or partial power condition as the starting point for an excursion accident.

This result is due primarily to a decrease in the peaking of the flux shape at higher powers which decrease the peak enthalpy. It is also due to a substantial increase in the effective resonance integral of U-238 (the Doppler effect) which increases the negative power feedback effects.

On the other hand, these same effects operate in reverse and cause a very shallow peak to occur at about 160°C in the consequences of a given reactivity insertion rate. The difference between this peak and the consequences at hot standby are insignificant when the calculational accuracy and operating time at hot standby versus 160°C are considered.

Regarding the failure to scram, all of the consequences described above are independent of the scram system since the excursions are Doppler limited. If for some reason the reactor should fail to scram, additional power bursts may occur which are much less severe than the initial one due to the much slower rate of reactivity insertion. This latter reactivity insertion rate is controlled by the rate of heat transfer which is very slow compared to the rate at which a rod can be removed from the core under very special circumstances.

After a series of power bursts of ever decreasing magnitude the reactor power will settle out at a somewhat higher level than the initial power. That level will be determined by the initial power level, amount of partial scram, and the rod worth if it is less than 0.025 Δk.

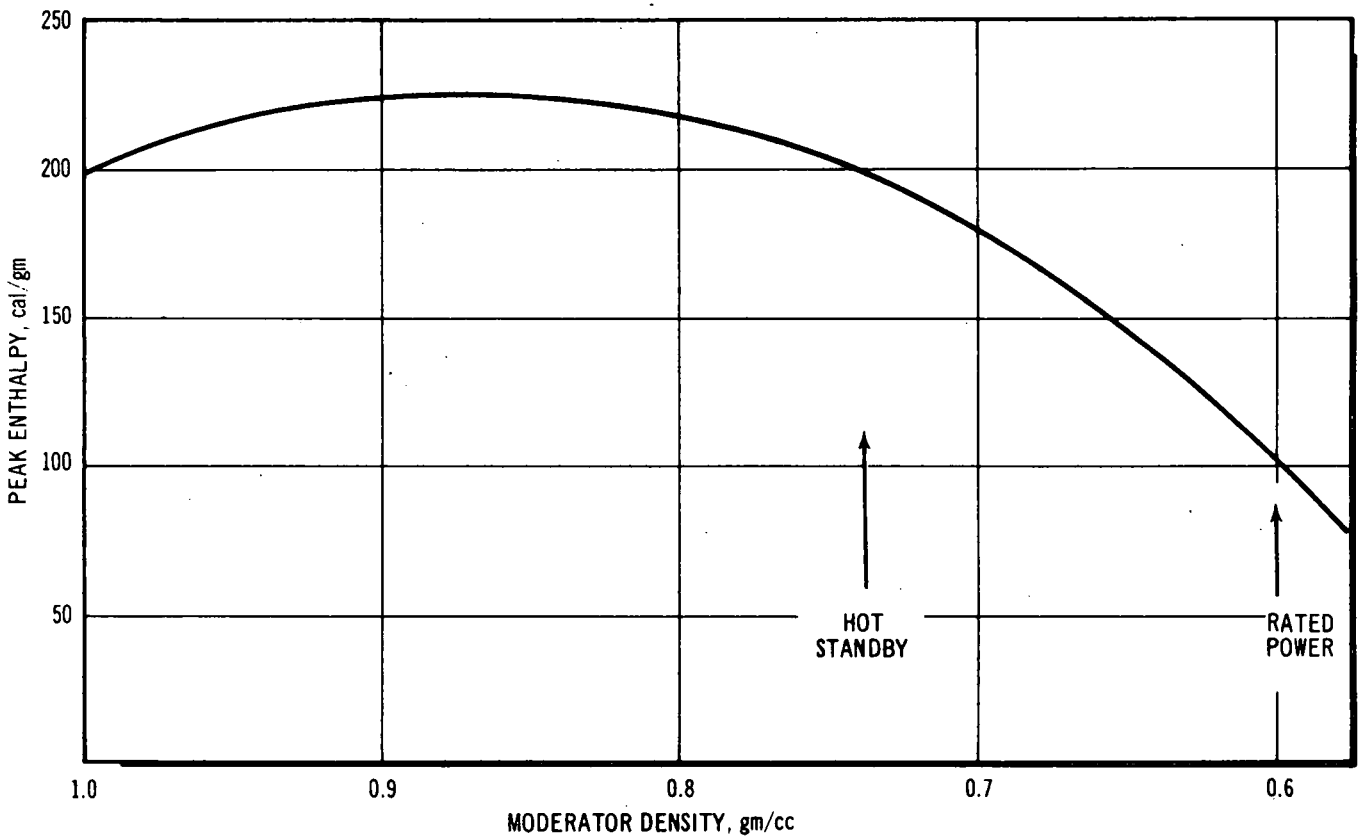


Figure III-2-1. Consequences of Rod Drop Accidents

QUESTION

III Accident Analysis (continued)

3. With reference to the refueling accident:

   a. State the energy density distribution and steam formation as a function of time,

   b. State the basis for assuming that the steam is condensed before reaching the pool surface. If the pressure-suppression data is used, discuss the differences in geometry, steam energy density, and the effect of a saturated water heat sink.

   c. What further release of fission products would result if the two control rods in the excursion zone failed to scram during the accident?

   d. Discuss possible mechanical effects on the fuel assembly and core grid plate due to the dropped fuel assembly.

   e. Discuss incorporation of Doppler coefficient feedback effect in the analysis which results in a peak temperature of 5000°F while previous analyses had indicated a peak temperature of over 10,000°F.

ANSWER

a. Figure III-3-1 shows the distribution of the 2610 MW-sec which is released during the refueling accident, in terms of the energy density histogram.

   This thermal energy which is contained within the fuel rods will then decay into the coolant at a maximum rate given by Figure III-3-2. The steam generation rate corresponding to this power is also shown on Figure III-3-2. The steam generation rate shown is extremely conservative because no credit is taken for film blanketing and the associated drastic reduction in the film side heat transfer coefficient and the subsequent reduction in the rate of energy transfer.

b. Both theoretical considerations and experimental evidence demonstrate that steam formed during the fuel loading accident will be rapidly condensed in the water and will not reach the surface of the pool. The fuel energy densities are not great enough to vaporize or melt the fuel, and thus heat transfer to the water would occur at rates associated with nucleate or film boiling heat transfer.

   The total energy transfer rate to the water is 21% of full power at time zero and 0.1% of full power at 30 seconds. The eight fuel assemblies, included in the 2 × 4 array, contain 31.4% of the excursion energy. The rate of heat transfer from the fuel assembly to the water, assuming no burnout, is about 4 times the maximum design assembly rate at full power operation. This rate decays by heat transfer to 1.7 times the maximum design assembly rate at full power operation in 5 seconds and to 0.023 in 30 seconds. At full power operation, the reactor is operating at saturated conditions with an energy of vaporization of 650 Btu/lb, and the core is submerged below 13 to 14 feet of water. During refueling

the water is at a temperature of approximately 100° to 120°F, so that it is about 100°F below the boiling point, 100 Btu/lb for a 100°F temperature rise, with an energy of vaporization of 970 Btu/lb. Additionally, the core is submerged below 50 feet of water. Some of the water in the eight hot fuel assemblies may boil but will be rapidly condensed by the water within the core region and immediately above the core, raising its average temperature less than 1°F. The halogens carried in the steam will be absorbed in the water as the steam is condensed.

The energy addition rate to water from the pressure suppression tests conducted at Moss Landing by P.G.&E. for Bodega Bay may be compared to the energy addition rate resulting from the fuel loading accident. The conditions are essentially the same. In the pressure suppression tests energy, in the form of steam-water, was added to water at around 100°F, at atmospheric pressure, heating up the water. The maximum energy addition rate to water during the tests was nearly 3×10^4 greater than the maximum energy addition rate to the water from the fuel loading accident. Additionally, in the pressure suppression tests the steam was condensed within 7 feet, whereas in the fuel loading accident 50 feet of water cover the core. The time for the steam bubble to be condensed in the Humboldt Bay pressure suppression system[(1)] was calculated to be 0.007 second. The steam from the fuel loading accident will be rapidly condensed and will not reach the pool surface.

The SPERT-1 reactor was subjected to transients[(2)] with shorter periods, 2.2 and 1.5 milliseconds, than result from the fuel loading accident, 3.9 milliseconds. Although SPERT-1 fuel temperatures did not reach the levels calculated for the fuel loading accident, the physical effects are expected to be similar because the calculated fuel energy densities for the fuel loading accident are well below levels that could cause rapid dispersal of the sintered oxide fuel and rapid heat transfer to the water. The water cover over the SPERT core was 5 to 6 feet compared to 50 feet over the Dresden Unit 2 core. In the SPERT tests no steam reached the surface of the pool and no fission products other than noble gases were detected in the atmosphere.

Even if steam condensation did not occur, the steam bubbles would be expected to be effectively scrubbed of halogens the same as they are during normal operation of the reactor. See the answer to question III 1b.

As the steam is condensed, the halogens are absorbed in the pool. They are assumed to be evolved from the pool into the air to establish an equilibrium partition factor. At the halogen concentration in the water (10^{-7} g mole/liter) from a fuel loading accident, the partition factors are 10^4 (Ref. 6 in answer to question III 1c), $2\ (10)^3$ (Ref. 7 in answer to question III 1c) and 10^4 (Ref. 9 in answer to question III 1c). In the analysis of this accident the partition factor was conservatively assumed to be 10^2. Even if the partition factor were 10^0, the dose from halogens would increase only by a factor of 2.

c. In order to evaluate the consequences of potential accidents which could occur when the primary containment (drywell-suppression chamber) is not required, a fuel loading accident has been postulated. This extremely improbable accident was analyzed because it provides the potential for a large release of fission products to the reactor building for purposes of testing the efficacy of the containment to minimize offsite doses.

In order to achieve this accident, gross compounding of operator error and interlock failures is required (see Section XI.5.2 of the Plant Design & Analysis Report).

The plant operators will be quite experienced, well trained, and more than one will be on duty during refueling operations. Procedures and interlocks will not allow maintenance of the control rod drive mechanisms during fuel handling. Further, all control rods will be inserted during fuel handling over the core.

Because of the low energy density associated with the excursion, no serious core deformation will occur and it is difficult to conceive a mechanism to prevent control rod insertion following the excursion. The scram protection system is highly reliable and no instance of failure to scram on a full scram signal has been observed on an operating reactor. Mechanically the drives are designed to apply sufficient force to insert themselves even in such cases where shearing and tearing of channel material is required due to any misalignment of fuel bundles. In any case, the reactor would be rendered subcritical upon the successful partial insertion of only one of the two withdrawn control rods in the assumed pattern, since the one stuck rod margin insures subcriticality with one rod withdrawn.

Even if it is hypothesized that no scram occurs, the power density in the small critical region of the core will equilibrate (due to the heating of the moderator and increased neutron leakage) far below rated power density. The cladding temperature will be only slightly above the water temperature, far below normal operating temperature. No fuel cladding perforations beyond that experienced during the initial power transient will occur, and no additional fission products will be released from the core.

d. Our analyses indicate that if a fuel assembly were to be dropped into the core as described in the refueling accident, damage to the core grid plate would not be incurred and the functional capability of the plate would be maintained. Also any mechanical damage to the fuel assembly, in addition to the damage to the fuel rods as calculated in the refueling accident, would not significantly change the calculated consequences or lead to other failures of more serious consequences.

e. Differences in consequences in present and prior refueling accident excursion analyses are due primarily to refinements in analytical models. In both cases, a point kinetics model is employed with Doppler feedback effects entered into the model through a single Doppler Weighting Factor (DWF) parameter. The DWF incorporates the spatial effects of the contributing fission power distribution.

In the prior analyses the power distribution characteristic of the initial prompt power burst was utilized to evaluate the DWF. A relatively crude but conservative spatial importance weighting factor equivalent to the 5/2 power of the initial neutron flux distribution was applied in determining the constant DWF used in the analysis.

In the current analysis, a marching technique with time is used. The DWF is evaluated at each time step by actual spatial calculation of the Doppler distribution. The feedback of this spatial Doppler on power shape as well as reactivity is incorporated into the model. It has been observed that the DWF calculated by this technique corresponds more nearly to a cube power importance weighting than the 5/2 power used in prior analyses. This results in about a factor of two increase in the effective DWF for the highly peaked power distributions existing in the postulated refueling accident and consequently a less severe transient.

One additional assumption contributed to the observed difference in the results of analyses. In prior calculations, energy deposition in the fuel was very conservatively assumed to correspond to the initial power burst shape. In current analyses, the energy is distributed according to the power shape

determined at each time step throughout the course of the excursion. Since the power shape flattens due to local Doppler, the effective peak-to-average energy deposition and peak temperatures are more correctly lower in current analyses than those previously reported.

REFERENCES

1. "Humboldt Bay Preliminary Hazards Summary Report, Unit 3, Amendment 6, Addendum E, Appendix I," January 29, 1960.

2. Grund, J. E., Editor, "Experimental Results of Potentially Destructive Reactivity Additions to an Oxide Core," IDO-17028, December, 1964.

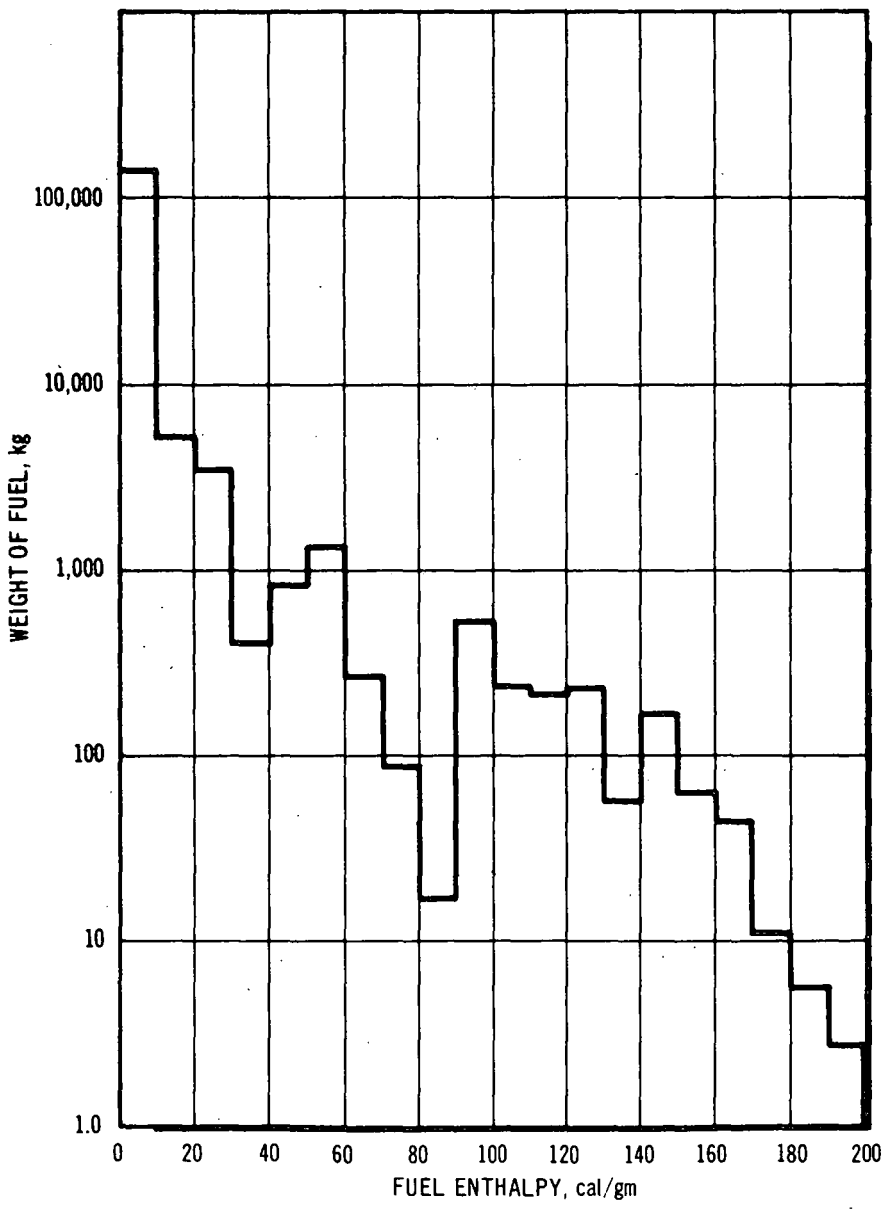


Figure III-3-1. Fuel Energy Distribution - Refueling Accident

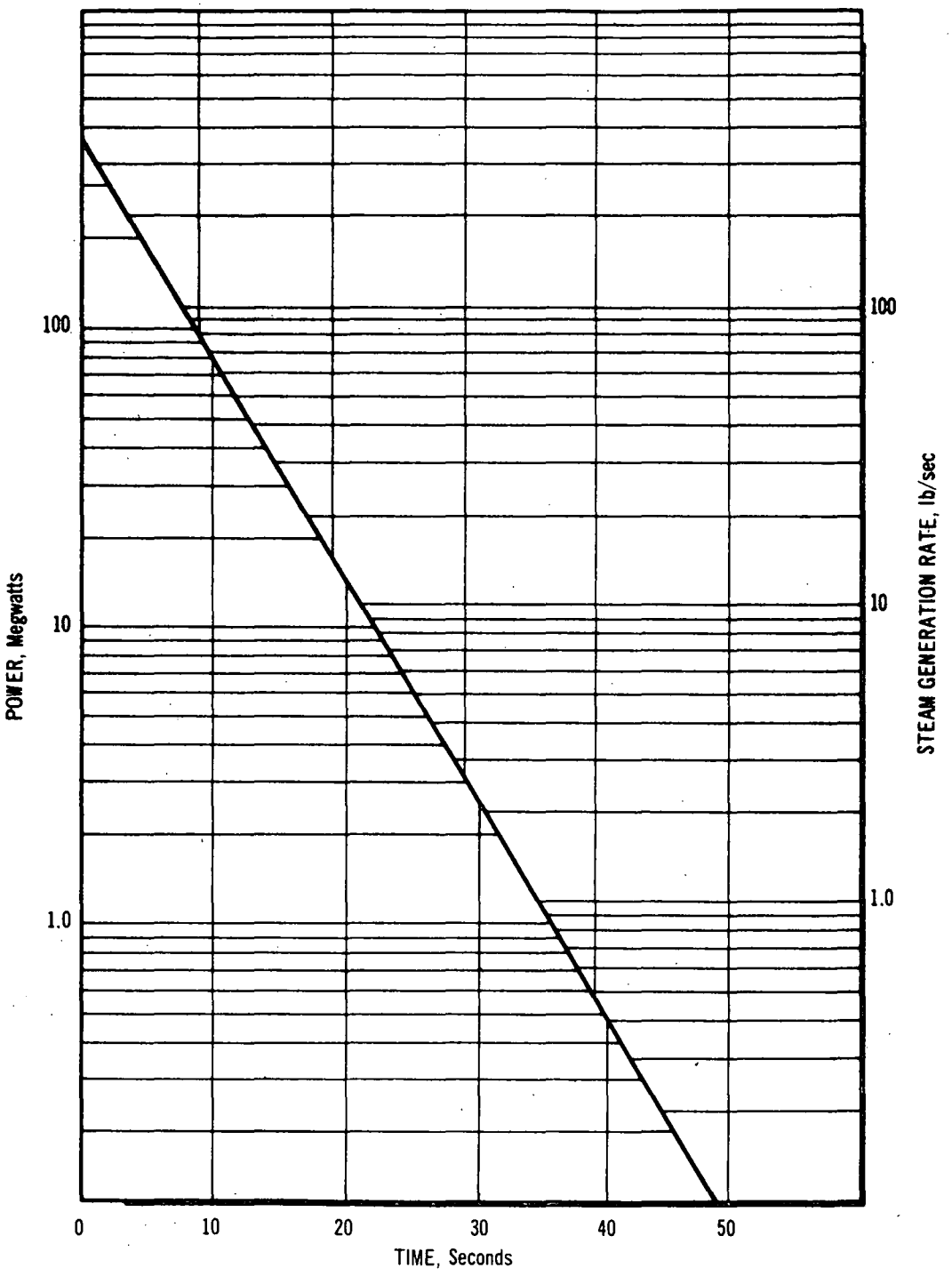


Figure III-3-2. Energy Transfer Rate to Coolant - Refueling Accident

QUESTION

III Accident Analysis (Continued)

4. With reference to the steam line break accident the following should be provided:

a. A description of the method of calculation used in estimating the amount of water which is carried with the released steam.

b. A description of the path of release of steam to the turbine building in terms of providing assurance that no mechanical damage to the pipe tunnel or the turbine building is likely to occur, and that the only path of release is via the roof ventilation system of the turbine building.

c. A calculation of the energy of the steam leaving the turbine building and the diffusion of the steam to the environs considering turbulence due to the facility buildings.

ANSWER

a. Calculation of Water Carryover in Steam Line Break

In the present design, the steam is transported from the reactor through four steam lines, each provided with flow restrictors and isolation valves. The four steam lines are cross connected in pairs as discussed in the answer to question II-5-c. Each steam line normally transports one-fourth of the total steam flow. If a steam line were to rupture, the flow in that pair of steam lines would double.

That is, 100% of rated steam would flow from the double ended rupture until the isolation valves are fully closed. The moisture carryover would be very slight and it has been calculated that the consequences would be many orders of magnitude less than that calculated and reported in Section XI-5.3 of the Plant Design and Analysis Report.

For the purposes of this calculation, however, the break has been analyzed assuming that all four steam lines are cross connected, so that the steam flow would increase above the normal rate following the rupture. Isolation valves closing in 3 to 10 seconds are now specifiedfor the steam lines. These work in conjunction with the flow limiters to limit coolant loss following a steam line rupture.

Blowdown flow rates for saturated steam/water mixtures were computed from an ideal nozzle model (1). Steam blowdown rate from 1000 psia is about 1720 lb_m/sec total from the two 0.43 ft^2 nozzles associated with the double-ended steam line break.

Decompression rate in the vessel would be less than 35 psi/sec., and the steam-water mixture level would rise at a rate less than four feet per second in the reactor vessel. Essentially no water would enter the steam line until the mixture spills over the dryers, about 12 ft. above normal water level. Steam blowdown should therefore proceed for three seconds followed by low quality mixture blowdown.

Maximum rate of mixture blowdown after 3 seconds is calculated to be about 7000 lb_m/sec. from both sides of the break. If the valve closed in about 8 seconds after mixture blow began (assuming the longest closing time specified of 10 seconds with an additional second to initiate start of closure) 56,000 lb_m of water would be blown through the steam line nozzles. Blowdown rates [1] are known to be conservative for low quality mixtures; that is, the calculated rates are slightly higher than would be expected in the actual blowdown. The estimated loss of 56,000 lb_m of saturated water during a steam line rupture outside containment is thus conservative. Further the assumption of 35,000 lb_m of water dividing from a total blowdown of 61,000 lb_m as discussed in reply to question III.4.c is even more conservative.

b. The path of release of steam to the turbine building is via the pipe tunnel between the reactor building and the turbine building, through the turbine building pipe tunnel at elevation 531'-6" and to the area under the turbine which is bounded by heavy shield walls and the turbine operating floor at elevation 551'-0". The walls and floor slabs of the pipe tunnel and the area under the turbine generator have inherent strength to assure that damage to the building will not occur.

The space under the turbine-generator and the pipe tunnels will be exhausted to the ventilation stack on the reactor building. Thus, a path for release of the steam which is not condensed on the surfaces of the turbine building and pipe tunnels will be via this elevated point. The spaces adjacent to the area under the turbine-generator and the area above the operating floor will normally be maintained at a slight negative pressure, with respect to the atmosphere, by the roof exhausters. Thus, the steam which is not released to the atmosphere through the reactor building stack will be released via the turbine building roof exhausters.

c. The water included in the steam-water mixture would rain out in the steam tunnel and in the turbine building below the turbine floor. The mixture would divide to approximately 35,000 pounds of water and 26,000 pounds of steam, assuming a water-steam enthalpy content equivalent to the average in the vessel prior to the rupture. The 35,000 pounds of water would contain most of the halogens. See references in answer to question III-1. The references include experiments and plant operations at room temperature and operating pressure, which bracket the conditions included here. The references indicate that 99 to 99.99% of the halogens would remain absorbed in the water. Nevertheless for this accident it was assumed that 100% of the halogens was airborne in the steam.

The steam pressure in the turbine building would force the steam out the vents in the roof of the turbine building or out of the sub-turbine floor ventilation system to a stack. Subsequent vaporization from the water mass is below the turbine floor and would be exhausted to the stack. The steam would contain approximately 30 $(10)^6$ Btu of energy, which for a 30 second release is 10^6 Btu/sec. The reference cited in Section XI-5.3, for the height of rise of a hot cloud in the atmosphere, indicates that the cloud would rise 4700 feet in a one mile/hour wind and 470 feet in a 10 mile/hour wind. The cloud rise experiments were conducted for energy inputs into the cloud short of that in this accident, so the equation was extrapolated to this condition. The equation was used not to predict

the absolute height to which the hot steam cloud would rise but to obtain an order of magnitude rise in order to estimate the consequent doses. Air turbulence in the building wake will cause some diffusion of the edges of the steam cloud. At low wind speeds, building wake turbulence is a minimum and the large energy of the cloud will make it relatively independent of the turbulence. At high wind speeds the building wake turbulence is greater and the cloud rises to a lower height (as predicted by the equation) but the cloud is diluted more by the wind so the lack of rise is not as important.

The reduction of halogens in the steam cloud due to absorption in the water fallen out in the turbine building will much more than off-set any variation in the actual height to which the cloud will rise or any cloud spread due to building wake turbulence.

REFERENCES

[1] Moody, F. J., "Maximum Flow Rate of a Single Component, Two Phase Mixture", Journal of Heat Transfer, Trans, ASME, Series C, Vol. 87, p134.

QUESTION

III Accident Analysis (Continued)

5. An analysis of the consequences of a loss of coolant accident in which both the core spray and core flooding systems fail to function to prevent a core meltdown should be provided. Include in the analysis consideration of the fission products which would reach the environs via all sources of leakage from the containment structures. Consider the effects of metal-water reaction including possible recombination of the hydrogen evolved. Long range effects such as slumping of the core into the bottom of the reactor vessel and subsequent termination of the accident need not be included for the purpose of this analysis. State the consequences of this accident for those meteorological conditions assumed in the report plus fumigation conditions. Consider also whether some exfiltration from the reactor building rather than release from the stack could result in more serious consequences.

ANSWER

The analysis of the loss of coolant accident provided in the Dresden Unit 2 Plant Design and Analysis Report involved a determination of the behavior of the reactor core during the period of loss of coolant as well as during that period of core heat-up prior to the core flooding system becoming effective. This analysis has been reconducted to determine the behavior of the reactor core in a loss of coolant accident if both of the core spray systems are ineffective.

The methods of analysis were the same as those employed in all reactor core heat up calculations conducted by the General Electric Company. The entire determination is conducted by digital computer programs. The reactor core is subdivided into radial and axial positions, the fuel bundles are further divided into zones of fuel rods, then the fuel reactor core subdivision allows the analysis to be conducted with a refined specification of power distribution throughout the reactor core, radially and axially, as well as throughout the fuel bundles and fuel rods. With this same degree of subdivision the temperature distributions throughout the reactor core can be quite precisely determined throughout the course of the temperature transient. Although radiation heat transfer from fuel rod to fuel rod and to the channel box is considered by the program, there is no heat loss from the overall reactor core itself allowed. Thus all decay heat and metal water reaction heat during the core heat up process is retained for the core in the form of sensible heat and latent heat of fusion. Fuel channel box material is analyzed separately from the fuel cladding and fuel pellet material. Because of the emphasis placed on the existence of any metal water reaction, the digital computer program also involves a continuous calculation of the extent and current rate of the metal water reaction for all metal surfaces within the reactor core on the above described subdivision basis. The metal water reaction is defined in the computer program by the expression of Baker, [1] which defines the rate of reaction as a function of both local metal temperature and the current extent of the reaction.

[1] Louis Baker, Jr. ANL 6548.

If it is assumed that there is no core cooling following the loss of coolant accident, the more important results are shown graphically on Figure III-5-1. This accident consists of the instantaneous severance of a main recirculation line with the subsequent complete loss of coolant from the vessel. It is assumed that neither of the core spray systems are effective and that the reactor core completely heats up to melting temperature and melts. In this analysis long range effects such as slumping of the core into the bottom of the reactor vessel and subsequent termination of the accident are not included. It was assumed that a given subdivision of the core (approximately a two foot axial section of about 150 fuel bundles) would drop from the core region upon melting of all the metal within it and further "in-core" metal water reaction would be terminated.

Figure III-5-1 demonstrates many characteristics associated with a "non-cooled core" heat up and melt.

1. The maximum extent of metal water reaction consisted of approximately 24.5% of all the available metal (channel boxes and cladding) within the core region.

   There is sufficient heat released from a reaction of this amount to heat up all the fuel cladding and channel boxes in the core from approximately 1800°F to the clad melt temperature and melt all the cladding and channel boxes. Thus 24.5% indicates a potential bound for metal water reaction for this reactor core.

2. The total duration of the reactor core meltdown is approximately one hour. However, the reactor core is effectively melted (90%) in about half an hour.

   The calcualtions make it very clear and the Figure III-5.1 indicates that once a given fuel region is heated to approximately 1800°F by means of decay heat within an insulated fuel bundle, the rate of metal water reaction begins to dominate and the remainder of the heatup is practically a straight function of the heat of metal water reaction.

Dose Analyses

The doses to off-site persons are calculated with the techniques and assumptions described in the 1% fission product release in Section XI-5.4 of the Plant Design and Analysis Report, except that 100% of the core is assumed to heat up to 3000°F and release fission products and 27.5% of metal in the core is assumed to react with water. The volume of hydrogen and the energy from the metal-water reaction are added to the containment. They cause the containment pressure to rise above the pressure calculated for the 1% melt analysis and to persist above atmospheric pressure for some time. The containment temperature and pressure transients (one containment spray system operating) are curves f of Figures III-7-2 and III-7-3. The leakage from the containment also persists at a higher rate for the duration of the accident. The assumptions for the fission product activity in the core; release of fission products from the 3000°F fuel; fallout, plateout and washout in the primary containment; leakage from the containment as a function of pressure; solids fallout in the reactor building; and discharge through filters to the stack are the same as for the 1% melt analysis.

The resulting airborne fission product activity in the reactor building is given in Table III-5-1.

TABLE III-5.1

Reactor Building Airborne Fission Product Activity
(Curies)

Time After Accident	Noble Gases	Halogens	Volatile Solids	Other Solids
30 min.	4.9×10^4	9.8×10^3	8.1×10^3	1.1×10^3
1 hour	8.6×10^4	1.6×10^4	1.2×10^4	1.9×10^3
3 hours	1.9×10^5	2.4×10^4	1.7×10^4	3.4×10^3
10 hours	4.2×10^5	3.6×10^4	2.3×10^4	4.7×10^3
1 day	5.6×10^5	3.9×10^4	7.0×10^3	1.7×10^3
3 days	5.3×10^5	3.9×10^4	1.6×10^2	2.1×10
7.5 days	4.8×10^5	3.6×10^4	0	0
12.5 days	3.5×10^5	2.0×10^4	0	0
17.5 days	2.1×10^5	1.2×10^4	0	0
22.5 days	1.3×10^5	6.8×10^3	0	0
27.5 days	7.8×10^4	4×10^3	0	0

The resulting fission product discharge rate from the stack is given in Table III-5-2.

TABLE III-5-2

Stack Discharge Rate
(Curies/sec)

Time After Accident	Noble Gases	Halogens	Volatile Solids	Other Solids
30 min.	0.57	1.1×10^{-3}	9.4×10^{-4}	1.2×10^{-4}
1 hour	1.0	1.9×10^{-3}	1.4×10^{-3}	2.1×10^{-4}
3 hours	2.2	2.8×10^{-3}	1.9×10^{-3}	3.9×10^{-4}
10 hours	4.9	4.2×10^{-3}	2.7×10^{-3}	5.4×10^{-4}
1 day	6.4	4.5×10^{-3}	8.8×10^{-4}	2.0×10^{-4}
3 days	6.2	4.5×10^{-3}	1.8×10^{-5}	2.4×10^{-6}
7.5 days	5.6	4.1×10^{-3}	0	0
12.5 days	4.1	2.4×10^{-3}	0	0
17.5 days	2.4	1.3×10^{-3}	0	0
22.5 days	1.5	7.9×10^{-4}	0	0
27.5 days	0.9	4.6×10^{-4}	0	0

The off-site radiological effects of this fission product release, calculated as described in Section XI-6.3, are shown in Tables III-5-3 and III-5-4.

Fumigation

The doses were also calculated for an assumed period of "fumigation", using the calculational methods in Meteorology and Atomic Energy AECU 3066. As indicated on page 61 of this reference, 15 minutes is the

average duration of this condition and was therefore used in this analysis. Assumption of a larger duration will give proportionally larger doses. For example, use of a 30 minute duration gives about twice as large a dose.

TABLE III-5-3

Radiological Effects of the Coolant Loss Accident*

External Passing Cloud Dose (rad)

Distance (miles)	VS-2	MS-2	N-2	N-10	U-2	U-10
	First 2-Hour Exposure					
1/2	3.2×10^{-3}	3.4×10^{-3}	3.4×10^{-3}	1.9×10^{-3}	3.4×10^{-3}	1.3×10^{-3} ‡
1	2.4×10^{-3}	3.0×10^{-3}	2.8×10^{-3}	1.1×10^{-3}	2.8×10^{-3}	7.6×10^{-4} ‡
5	-	-	-	8.0×10^{-5}	-	3.4×10^{-5} ‡
9	-	-	-	2.4×10^{-5}	-	9.2×10^{-6} ‡
12	-	-	-	1.3×10^{-5}	-	4.2×10^{-6} ‡
	Total Accident Exposure					
1/2	1.2×10^{-1}	1.2×10^{-1}	1.2×10^{-1}	7.2×10^{-2}	1.2×10^{-1}	5.2×10^{-2} ‡
1	9.0×10^{-2}	1.0×10^{-1}	4.2×10^{-2}	4.2×10^{-2}	1.1×10^{-1}	2.9×10^{-2} ‡
5	1.0×10^{-2}	1.2×10^{-2}	8.3×10^{-3}	3.0×10^{-3}	3.0×10^{-3}	1.3×10^{-3} ‡
9	3.7×10^{-3}	4.6×10^{-3}	1.7×10^{-3}	9.0×10^{-4}	7.1×10^{-4}	3.4×10^{-4} ‡
12	1.9×10^{-3}	2.4×10^{-3}	7.1×10^{-4}	4.9×10^{-4}	2.6×10^{-4}	1.6×10^{-4} ‡

Lifetime Thyroid Dose (rad)

	First 2-Hour Exposure					
1/2	a**	a	2.4×10^{-6}	5.0×10^{-5}	2.5×10^{-2}	7.3×10^{-3}
1	a	a	2.6×10^{-3}	1.4×10^{-3}	2.3×10^{-2}	5.6×10^{-3}
5	-	-	-	1.3×10^{-3}	-	5.3×10^{-4}
9	-	-	-	5.6×10^{-4}	-	2.3×10^{-4}
12	-	-	-	3.6×10^{-4}	-	1.4×10^{-4}
	Total Accident Exposure					
1/2	a	a	3.7×10^{-5}	7.5×10^{-4}	3.8×10^{-1}	1.1×10^{-1}
1	a	a	2.0×10^{-2}	2.1×10^{-2}	3.6×10^{-1}	8.5×10^{-2}
5	a	3.3×10^{-3}	1.0×10^{-1}	2.0×10^{-2}	4.0×10^{-2}	8.0×10^{-3}
9	a	1.4×10^{-2}	4.5×10^{-2}	8.5×10^{-3}	1.5×10^{-2}	5.5×10^{-3}
12	a	2.7×10^{-2}	2.8×10^{-2}	5.5×10^{-3}	1.0×10^{-2}	2.1×10^{-3}

*Calculated using meteorological methods described in Sections XI-6.3.2 to XI-6.3.2f of Volume I Plant Design and Analysis Report.

**The symbol "a" means less than 1×10^{-10}.

average duration of this condition and was therefore used in this analysis. Assumption of a larger duration will give proportionally larger doses. For example, use of a 30 minute duration gives about twice as large a dose.

TABLE III-5-3

Radiological Effects of the Coolant Loss Accident*

External Passing Cloud Dose (rad)

Distance (miles)	VS-2	MS-2	N-2	N-10	U-2	U-10
			First 2-Hour Exposure			
1/2	3.2×10^{-5}	3.4×10^{-5}	3.4×10^{-5}	1.9×10^{-5}	3.4×10^{-5}	1.3×10^{-5}
1	2.4×10^{-5}	3.0×10^{-5}	2.8×10^{-5}	1.1×10^{-5}	2.8×10^{-5}	7.6×10^{-6}
5	-	-	-	8.0×10^{-7}	-	3.4×10^{-7}
9	-	-	-	2.4×10^{-7}	-	9.2×10^{-8}
12	-	-	-	1.3×10^{-7}	-	4.2×10^{-8}
			Total Accident Exposure			
1/2	1.2×10^{0}	1.2×10^{0}	1.2×10^{0}	7.2×10^{-1}	1.2×10^{0}	5.2×10^{-1}
1	9.0×10^{-1}	1.0×10^{0}	4.2×10^{-1}	4.2×10^{-1}	1.1×10^{0}	2.9×10^{-1}
5	1.0×10^{-1}	1.2×10^{-1}	8.3×10^{-2}	3.0×10^{-2}	3.0×10^{-2}	1.3×10^{-2}
9	3.7×10^{-2}	4.6×10^{-2}	1.7×10^{-2}	9.0×10^{-3}	7.1×10^{-3}	3.4×10^{-3}
12	1.9×10^{-2}	2.4×10^{-2}	7.1×10^{-3}	4.9×10^{-3}	2.6×10^{-3}	1.6×10^{-3}

Lifetime Thyroid Dose (rad)

Distance (miles)	VS-2	MS-2	N-2	N-10	U-2	U-10
			First 2-Hour Exposure			
1/2	a**	a	2.4×10^{-6}	5.0×10^{-5}	2.5×10^{-2}	7.3×10^{-3}
1	a	a	2.6×10^{-3}	1.4×10^{-3}	2.3×10^{-2}	5.6×10^{-3}
5	-	-	-	1.3×10^{-3}	-	5.3×10^{-4}
9	-	-	-	5.6×10^{-4}	-	2.3×10^{-4}
12	-	-	-	3.6×10^{-4}	-	1.4×10^{-4}
			Total Accident Exposure			
1/2	a	a	3.7×10^{-5}	7.5×10^{-4}	3.8×10^{-1}	1.1×10^{-1}
1	a	a	2.0×10^{-2}	2.1×10^{-2}	3.6×10^{-1}	8.5×10^{-2}
5	a	3.3×10^{-3}	1.0×10^{-1}	2.0×10^{-2}	4.0×10^{-2}	8.0×10^{-3}
9	a	1.4×10^{-2}	4.5×10^{-2}	8.5×10^{-3}	1.5×10^{-2}	5.5×10^{-3}
12	a	2.7×10^{-2}	2.8×10^{-2}	5.5×10^{-3}	1.0×10^{-2}	2.1×10^{-3}

*Calculated using meteorological methods described in Sections XI-6.3.2 to XI-6.3.2f of Volume I Plant Design and Analysis Report.

**The symbol "a" means less than 1×10^{-10}.

TABLE III-5-4

Radiological Effects of the Coolant Loss Accident*

Distance (miles)	VS-2	MS-2	N-2	N-10	U-2	U-10
External Passing Cloud Dose (rad)						
First 2-Hour Exposure						
1/2	3.2×10^{-3}	3.4×10^{-3}	3.4×10^{-3}	1.9×10^{-3}	3.4×10^{-3}	1.3×10^{-3} ‡
1	2.4×10^{-3}	3.0×10^{-3}	2.8×10^{-3}	1.1×10^{-3}	2.8×10^{-3}	7.6×10^{-4} ‡
5				8.0×10^{-5}	-	3.4×10^{-5} ‡
9	-	-	-	2.4×10^{-5}	-	9.2×10^{-6} ‡
12	-	-	-	1.3×10^{-5}	-	4.2×10^{-6} ‡
Total Accident Exposure						
1/2	3.2×10^{0}	3.4×10^{0}	3.4×10^{0}	1.9×10^{0}	3.4×10^{0}	1.3×10^{0} ‡
1	2.4×10^{0}	3.0×10^{0}	2.8×10^{0}	1.1×10^{0}	2.8×10^{0}	7.6×10^{-1} ‡
5	2.8×10^{0}	3.2×10^{-1}	2.2×10^{-1}	8.0×10^{-2}	8.0×10^{-2}	3.4×10^{-2} ‡
9	1.0×10^{-1}	1.2×10^{-1}	4.6×10^{-2}	2.4×10^{-2}	1.9×10^{-2}	9.2×10^{-3} ‡
12	5.0×10^{-2}	6.4×10^{-2}	1.9×10^{-2}	1.3×10^{-2}	7.0×10^{-3}	4.2×10^{-3} ‡
Lifetime Thyroid Dose (rad)						
First 2-Hour Exposure						
1/2	a**	a	5.0×10^{-6}	2.2×10^{-4}	2.1×10^{-2}	1.3×10^{-2}
1	a	a	1.5×10^{-3}	5.9×10^{-3}	2.0×10^{-2}	7.3×10^{-3}
5	-	-	-	3.6×10^{-3}	-	5.3×10^{-4}
9	-	-	-	1.3×10^{-3}	-	1.9×10^{-4}
12	-	-	-	7.9×10^{-4}	-	9.9×10^{-5}
Total Accident Exposure						
1/2	a	a	3.3×10^{-3}	1.5×10^{-1}	1.5×10^{1}	8.8×10^{0}
1	a	a	9.7×10^{-1}	4.0×10^{0}	1.4×10^{1}	4.8×10^{0}
5	a	4.0×10^{-1}	7.5×10^{0}	2.4×10^{0}	1.2×10^{0}	3.5×10^{-1}
9	a	1.0×10^{0}	2.2×10^{0}	8.8×10^{-1}	4.2×10^{-1}	1.3×10^{-1}
12	a	2.8×10^{0}	1.2×10^{0}	4.8×10^{-1}	2.2×10^{-1}	6.6×10^{-2}

*Calculated using meteorological diffusion methods in HW-SA-2809. See Section XI-6.3.2h of Volume I Plant Design and Analysis Report.

**The symbol "a" means less than 1×10^{-10}.

An example dose was calculated for the site boundary for the 100% melt loss of coolant for comparison to first two hour and total accident doses. These calculations were performed for period of maximum stack release.

TABLE III-5-4

Radiological Effects of the Coolant Loss Accident*

Distance (miles)	VS-2	MS-2	N-2	N-10	U-2	U-10
External Passing Cloud Dose (rad)						
First 2-Hour Exposure						
1/2	3.2×10^{-5}	3.4×10^{-5}	3.4×10^{-5}	1.9×10^{-5}	3.4×10^{-5}	1.3×10^{-5}
1	2.4×10^{-5}	3.0×10^{-5}	2.8×10^{-5}	1.1×10^{-5}	2.8×10^{-5}	7.6×10^{-6}
5				8.0×10^{-7}	-	3.4×10^{-7}
9	-	-	-	2.4×10^{-7}	-	9.2×10^{-8}
12	-	-	-	1.3×10^{-7}	-	4.2×10^{-8}
Total Accident Exposure						
1/2	3.2×10^{-2}	3.4×10^{-2}	3.4×10^{-2}	1.9×10^{-2}	3.4×10^{-2}	1.3×10^{-2}
1	2.4×10^{-2}	3.0×10^{-2}	2.8×10^{-2}	1.1×10^{-2}	2.8×10^{-2}	7.6×10^{-3}
5	2.8×10^{-2}	3.2×10^{-3}	2.2×10^{-3}	8.0×10^{-4}	8.0×10^{-4}	3.4×10^{-4}
9	1.0×10^{-3}	1.2×10^{-3}	4.6×10^{-4}	2.4×10^{-4}	1.9×10^{-4}	9.2×10^{-5}
12	5.0×10^{-4}	6.4×10^{-4}	1.9×10^{-4}	1.3×10^{-4}	7.0×10^{-5}	4.2×10^{-5}
Lifetime Thyroid Dose (rad)						
First 2-Hour Exposure						
1/2	a**	a	5.0×10^{-6}	2.2×10^{-4}	2.1×10^{-2}	1.3×10^{-2}
1	a	a	1.5×10^{-3}	5.9×10^{-3}	2.0×10^{-2}	7.3×10^{-3}
5	-	-	-	3.6×10^{-3}	-	5.3×10^{-4}
9	-	-	-	1.3×10^{-3}	-	1.9×10^{-4}
12	-	-	-	7.9×10^{-4}	-	9.9×10^{-5}
Total Accident Exposure						
1/2	a	a	3.3×10^{-3}	1.5×10^{-1}	1.5×10^{1}	8.8×10^{0}
1	a	a	9.7×10^{-1}	4.0×10^{0}	1.4×10^{1}	4.8×10^{0}
5	a	4.0×10^{-1}	7.5×10^{0}	2.4×10^{0}	1.2×10^{0}	3.5×10^{-1}
9	a	1.0×10^{0}	2.2×10^{0}	8.8×10^{-1}	4.2×10^{-1}	1.3×10^{-1}
12	a	2.8×10^{0}	1.2×10^{0}	4.8×10^{-1}	2.2×10^{-1}	6.6×10^{-2}

*Calculated using meteorological diffusion methods in HW-SA-2809. See Section XI-6.3.2h of Volume I Plant Design and Analysis Report.

**The symbol "a" means less than 1×10^{-10}.

An example dose was calculated for the site boundary for the 100% melt loss of coolant for comparison to first two hour and total accident doses. These calculations were performed for period of maximum stack release.

Dose During 15 Minutes of "Fumigation" (*) for 100% Core Melt

Passing Cloud (rad)	Lifetime Thyroid (rem)
2.5×10^{-2}**	0.47
1.3×10^{-1} ‡	2.4

Exfiltration

If the wind is strong enough to cause the pressure at the downwind side of the reactor building to be more negative than the reactor building interior (standby gas treatment system operating), airborne fission products in the reactor building can leak out in the building wake (exfiltration).

From the answer to Question II-8, exfiltration will not occur below a wind speed of 35 to 65 miles/hr. From exhibit III-6-40 of the Plant Design and Analysis Report, not once during 44,000 hours of observation was the wind speed at the Dresden site recorded to be above 39 miles/hour at 15 feet above the ground. Only 10 times during 44,000 hours of observation was the wind speed reported to be above 39 miles/hour at 150 feet above the ground. Therefore, it is very improbable that a high wind speed could occur coincident with the accident. If such a wind speed did occur, it would be in gusts and of very short duration.

Nevertheless, if it is assumed that a 40 mile/hour wind occurs for 10 minutes during the period of maximum airborne fission product activity in the reactor building, the maximum whole body dose (using the maximum leakage rate of 0.16/day at 40 miles/hr) would be 0.53 mr at the nearest site boundary and the maximum integrated thyroid dose would be 160 mr.

(*)Calculated for 2 miles/hour case.

**See Section XI-6.2.2 to XI-6.3.2-f of Plant Design and Analysis Report for details of calculational methods and constants used.

‡See Section XI-6.3.2-h of Plant Design and Analysis Report for details of calculational methods and constants used.

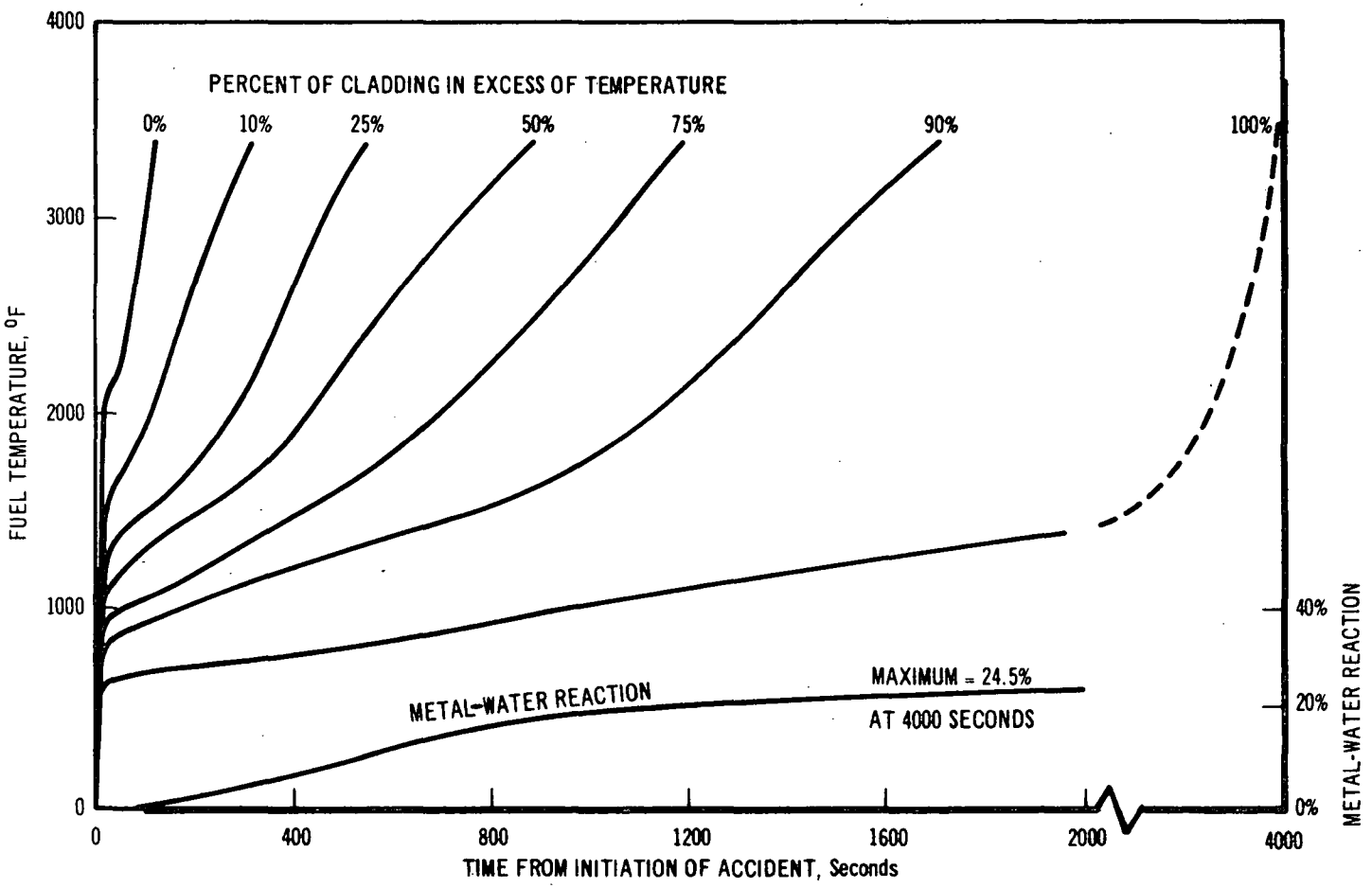


Figure III-5-1. Reactor Core Heat-up Behavior
No Engineered Safeguards

QUESTION

III Accident Analysis (Continued)

6. For the loss of coolant accident please demonstrate that the failure of the recirculation line rather than a smaller diameter line is the accident with the most serious consequences. Discuss why a larger failure, such as the pressure vessel itself, was not considered credible.

ANSWER

The failure of the main recirculation line would result in more serious consequences than the failure of a smaller diameter line inside the drywell.

If the main recirculation line is completely severed during normal operation at rated power and pressure, compounded with failure of all cooling systems, it has been calculated that approximately 25% of the zirconium in the core region would react with water, and the entire core would melt in approximately 40 minutes after the rupture (see answer III-5). For a small line break, with a three hour blowdown, it has been calculated that approximately 23% of the zirconium would react with water, and the core would melt within 2 hours after the blowdown or 5 hours after the rupture. (See Figure II-4-1 of reply to question II-4.)

During the blowdown the main parameters of interest include core pressure differentials, maximum drywell pressure, core cooling, and fuel rod cladding integrity. The pressure differentials across the vessel internals as a result of a small line break would not significently differ from those existing during normal operation. The vessel blowdown rate would be much lower for a small line break and the maximum drywell pressure during the blowdown would be only a few psig. For the small line break, the core would be cooled during depressurization to essentially the coolant temperature (300 to 500°F) at the completion of the blowdown. The fuel rod cladding temperature would be essentially that of the coolant during the blowdown and therefore cladding integrity would be maintained. Following the blowdown, the parameters of interest include cladding perforation, fission product release, core melt and metal-water reaction. For the small break investigated, the fuel rod cladding did not reach perforation temperatures for approximately 3-1/4 hours after the rupture as compared to approximately 1-1/2 minutes for the large break. The core reached redistribution temperature at approximately 3-1/3 hours for the small break as compared to minutes for the large recirculation break. Therefore, since the majority of fission products are not released until the redistribution temperature of UO_2 (3000°F) is reached, the fission products would have a longer decay time and the corresponding total curies released for a small break would be lower than for the recirculation line break. Since the time period of release of hydrogen gas from the metal water reaction for the small break is considerably longer than for the large break the containment spray system can easily quench the steam in the drywell. Therefore, the resulting pressures in the containment systems will be lower following a small break. The lower fission product leakage rates, due to both lower containment pressures and lower levels of activity being released, result in lower doses. For comparison purposes, several pertinent parameters are given in Table I. In summary, evaluations of the spectrum of small line breaks indicates that the extent of metal water reaction is very insensitive to the break size and that, in all non-core-cooled cases, the smaller break and longer depressurization results in less severe effects. Because the small

line breaks would result in longer times for blowdown and subsequent core heatup the potential for the termination of the accident by some delayed automatic or manual action is considerable.

TABLE III-6-1
Results of Loss of Coolant Accident

	Main Recirculation Line Break	Typical Small Line Break
Time for blowdown	24 sec.	3 hours
Time for 1st rod to perforate	8 sec.	3.2 hours
Time for all rods to perforate	0.8 hours	4.8 hours
Time for 1st rod to redistribute temp.	2 min.	3.3 hours
Time for all rods at redistribution temp.	1 hour	5.1 hours
Start of metal water reaction	3 minutes	3.4 hours
Extent of Metal-Water Reaction	~25%	~23%
Time to maximum metal water reaction	0.7 hours	5.1 hours
Metal-Water reaction relative to system capacity	*	**See Figure II-4-1

The evaluation of loss of coolant accidents is based upon the failure of any pipe within the drywell as noted above. A catastrophic failure mode of the reactor vessel is considered to be not credible in these analyses. This conclusion has been reached principally on the basis of the extremely stringent standards which have been adopted and utilized by the nuclear industry in the design and fabrication of reactor pressure vessels, and upon the operating conditions to which the vessels will be subjected.

The potential causes for catastrophic vessel failure have been intensively studied and the factors which cause such failures are reasonably well understood. Major factors of importance which are considered in establishing the proper design, fabrication, and operational performance of the vessel in order to preclude a brittle fracture mode include the following: (1) rigorous application of Section III of the ASME Boiler and Pressure Vessel Code, (2) assurance that all SA302B materials used in the pressure vessel have a low nil ductility temperature (NDT), (3) the materials used inherently display a high upper shelf impact energy, and (4) that the vessels be operated with appropriate margin, above the nil ductility temperature.

In the design and fabrication of the vessel and its appurtanences, detailed procedures have been devised and are followed to assure that the vessel will be of high integrity when completed. Stress analyses are made in accordance with Section III of the ASME Code so that assurance is provided that the vessel design will eliminate regions of overstress. Quality control procedures are placed into effect to assure that (1) materials are specified and utilized which have physical, mechanical, and metallurgical properties consistent with a low nil ductility temperature and an inherent high upper shelf impact energy, (2) that fabrication and inspection requirements are properly delineated so that the various processes of metal working, welding, heat treatment, radiography, etc., are properly applied, and (3) that integrity of the completed vessel is demonstrated through the application of proper testing.

During operation of the plant the vessel will be operated in such a manner that conditions required for brittle fracture will be entirely avoided. The vessel will not be pressurized at a temperature below NDT + 60°F in the core region and 40°F in other areas, and will not increase substantially during the forty-year lifetime

of the unit. The design of the reactor core and the internal vessel arrangement is such that the upper level of integrated fast neutron exposure of the vessel wall is expected to be in the range of 5×10^{17} neutrons/sq. cm., and will thus be sufficiently low that adverse radiation effects and an increase of the nil ductivity temperature will be negligible. As a further control, vessel material surveillance samples will be irradiated within the vessel so that status of the NDT may be monitored on a timely basis.

In summary, it has been concluded that by attention to the proper design, material selection, fabrication and quality control procedures for the vessel, and by prescribing operating conditions above the NDT range, failure of the vessel will not occur.

QUESTION

III Accident Analysis (Continued)

7. For the loss of coolant accident, a graph of drywell and suppression pool pressure as a function of time after the accident should be provided assuming:

   a. All engineering safeguards function

   b. One containment cooling loop does not function

   c. Both containment cooling loops do not function

   d. The core spray and core flooding systems do not function. Assume:

      1. No metal-water reaction
      2. Metal-water reaction as defined in answer to question II-4 above.

   e. Core spray, core flooding, and containment cooling loop do not function. Assume:

      1. No metal-water reaction
      2. Metal-water reaction as defined in answer to question II-4 above.

   For each of the cases considered above, discuss the drywell temperature as the accident progresses.

ANSWER

The procedure for calculating the pressure transients in the drywell and suppression pool in each of these cases will be described first and then the application to each case will be described. The initial pressure response of the system during the period when the reactor vessel is blowing down is reported in the Plant Design and Analysis Report Volume I, XI.-5-17 and 18 for the first 30 sec after the break. These pressure responses are shown in Figure III-7-1. The same response applies to all cases considered in this question. For each case, the temperature of the suppression pool was calculated as a function of time conservatively considering the pool to be the only heat absorber in the system. The effects of decay energy, stored energy in the core and possible energy from the metal-water reaction on the pool temperature were included. Also, if applicable in the particular case, the effect of heat exchangers in the drywell spray loop were included.

The drywell temperature is calculated considering an energy balance on the drywell spray and/or core spray. The drywell spray enters at the discharge temperature of the heat exchanger and the core spray enters at the suppression pool temperature. The combined flows (drywell spray and core spray) drain back to the suppression pool, having been heated by the decay energy, stored energy in the core and any possible metal-water reaction chemical energy. The drywell temperature is then taken to be 5°F hotter than the exiting flow. Where it is assumed that the drywell and core sprays are not operating, no credit for heat removal is taken.

The total number of moles of non-condensible gas in the entire system (drywell and suppression chamber) is determined from the amount of gas in the system originally plus any gas generation from a possible metal-water reaction.

With the drywell temperature, suppression pool temperature and moles of gas in the system, the system pressure is known. It was assumed conservatively that the drywell and suppression chamber gases are saturated. Also it was assumed that the drywell and suppression chamber are at equal pressure, which is reasonable since the pressure difference cannot exceed 4 ft of water (1.8 psi), the vent submergence depth, after the initial reactor blowdown.

Where it is assumed that there is no drywell spray or core spray all the non-condensible gases are assumed, conservatively, to be in the suppression chamber.

a. All engineering safeguards function.

The decay energy, stored energy in the core, and chemical energy were removed from the core by the core spray as predicted by the computer code described in the answer to question III-5. The resultant metal water reaction extent was 0.5%. The pressure response of the system is shown as curve (a) in Figure III-7-2, and the drywell temperature for this case is shown in Figure III-7-3. The drywell spray was initiated at 30 sec after the break and the system pressure rapidly decayed from the equilibrium pressure shown on Figure III-7-1 to the value shown on Figure III-7-2 curve (a). The rapid drop is due to the quenching of the steam in the drywell caused by the drywell spray.

b. One containment cooling loop does not function.

This is the same as (a) except that the containment spray is reduced. The pressure response is shown in Figure III-7-2 curve (b) and the drywell temperature response in Figure III-7-3 curve (b).

c. Both containment cooling loops do not function.

This case was handled in the same manner as (a) except no drywell spray was used. The pressure response is shown in Figure III-7-2, curve (c) and the drywell temperature response in Figure III-7-3, curve (c). Note that system pressure increases monotonically, as is to be expected, since without the cooling loop heat exchangers in operation the system's energy increases indefinitely due to core decay energy.

d-1 The core spray and core flooding systems do not function and no metal water reaction takes place.

It was assumed that the decay energy, stored core energy and chemical energy are released to the containment uniformly over a one half hour period. This duration is consistant with core thermal response reported in answer to III-5 (see Figure 5-1). A model consistant with this assumption involves a water supply in the bottom of the vessel for releasing energy from the molten material dropping from the core region.

The system pressure for this case is shown in Figure III-7-2, curve (d-1) and the temperature response of the drywell is shown in Figure III-7-3, curve d-1. The response of the drywell for case "a" and "d1" are sufficiently similar to be plotted as one characteristic. The same energy within the vessel is assumed to be transported to the drywell in both cases.

d-2 The core spray and core flooding systems do not function and a metal water reaction takes place.

This was calculated in the same manner as (d-1) except that the metal water reaction was assumed to take place in a uniform manner over a one half hour duration.

Since the energy release and hydrogen release from the reactor vessel to the containment system are based on uniform intensity for the duration of the release, a consistent model which involves a water supply in the bottom of the vessel (to allow the molten material dropping from the core region to react with water) is used. The potential metal water reaction as molten metal fell into the water and released its energy was included. The smallest potential drop sizes of molten metal which might occur from the fuel bundles were determined by a surface tension model which has been shown experimentally to give very good results. Molten zircaloy drops on the order of 0.350 to 0.400 inch were predicted. Employing the model of Louis Baker (ANL-6548) for extent of metal water reaction during droplet quenching in hot water, an additional reaction of 4% is indicated.

Therefore, the analysis reported in question III-5 indicated that on the basis of the core meltdown model, an in-core reaction of 24.5% could occur. An additional 4% metal water reaction occurs when the melted zirconium falls into the bottom of the vessel. The pressure transients reported in this investigation are based on a total reaction of 24.5%, plus 4% of the balance for a total of 27.5%.

The pressure response of the system is shown in Figure III-7-2, curve (d-2) and the temperature response of the drywell in Figure III-7-3, curve (d-2). Note that the pressure and temperature reach a peak value at 1800 sec (1/2 hr) which corresponds to the end of the uniform energy and gas release. At this time the energy release from the core is reduced to the decay power level.

e-1 Core spray, core flooding, and containment cooling loop do not function and no metal water reaction takes place.

To be conservative in the calculation of pressure, it was assumed that the energy released from the core was delivered to the suppression pool in a uniform manner over one half hour and that all non-condensible gases are in the suppression chamber. This procedure maximizes the suppression pool temperature and gas density making the calculated system pressure conservative. The pressure response of the system is shown in Figure III-7-2 curve (e-1).

This curve shows that the system pressure rises monotonically which is to be expected since there is no energy rejected from the system.

The drywell temperature can be estimated to be approximately equal to the saturation temperature of steam at the system pressure.

e-2 Core spray, core flooding and containment cooling loop do not function and a metal-water reaction takes place.

This case was calculated in the same manner as (e-1) except that metal water reaction was added which resulted in a 27.5% reaction over a one half hour period as in (d-2). The pressure response of the system is shown in Figure III-7-2, curve (e-2). This curve shows that the system pressure rises to 75.5 psig at one half hour after the accident starts and thereafter continues to rise at a

slower rate. This change in rate of pressure rise is a result of the change in the rate at which energy is added to the suppression pool. At one half hour the energy release is reduced to the decay power level. The drywell temperature can be estimated to be approximately equal to the saturation temperature of the steam at the system pressure.

An additional case was analyzed to demonstrate the capability of the system. This case is designated as case (f).

f. The core spray and core flooding system do not function; one containment cooling loop does not function and a metal water reaction takes place.

The case was calculated in the same manner as case (d-2) except only one containment cooling loop functions. The resulting pressure and temperature responses are shown in Figure III-7-2, curve (f) and Figure III-7-3, curve (f) respectively. Note the maximum pressure (27.5 psig) is less than the design pressure of 62 psig. It is this combination of effective engineering safeguards that has served as a basis for the generation of the Dresden Unit 2 Containment Capability Characteristic reported in reply to question II-4.

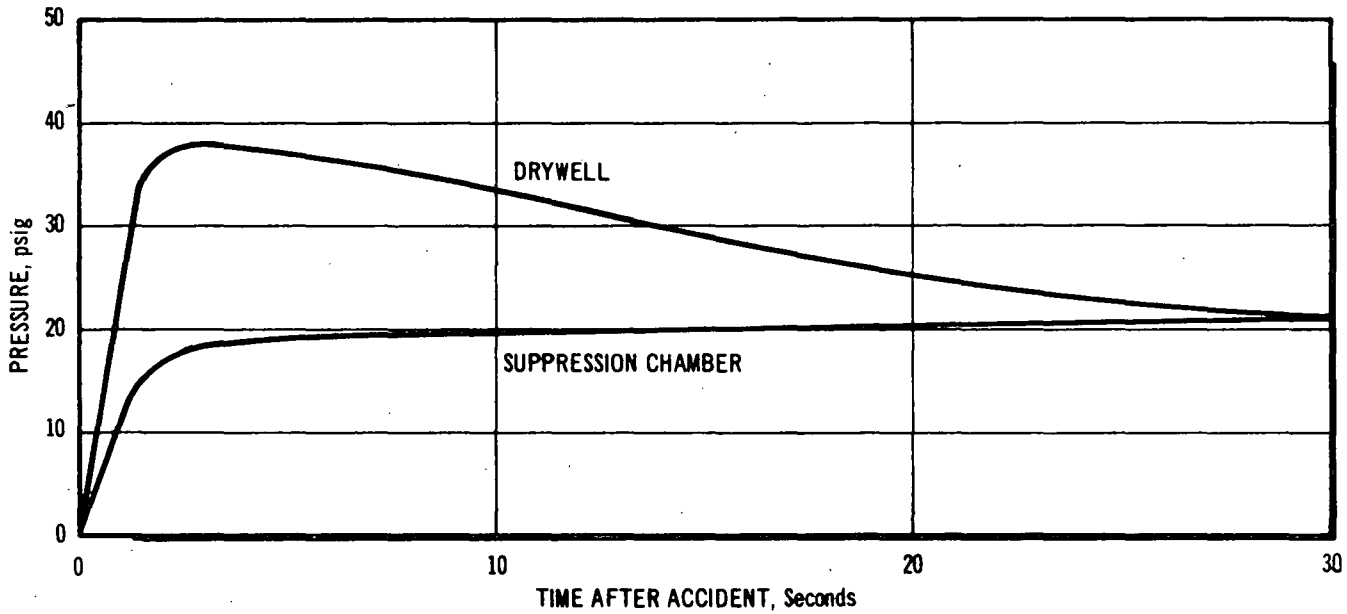


Figure III-7-1. Initial Pressure Response of Containment Following a Loss-of-Coolant Accident

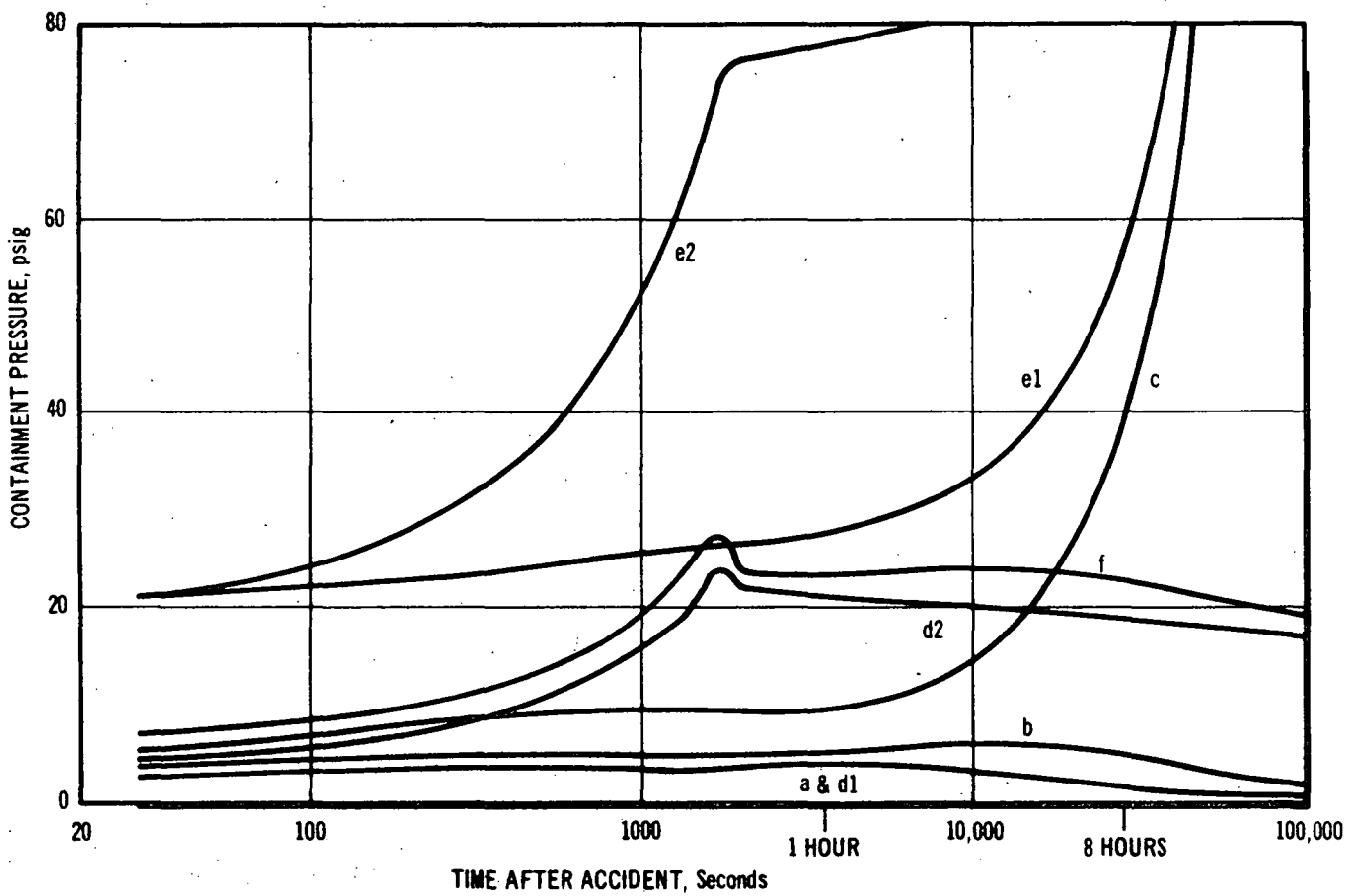


Figure III-7-2. Containment Pressure for Various Available Engineered Safeguards

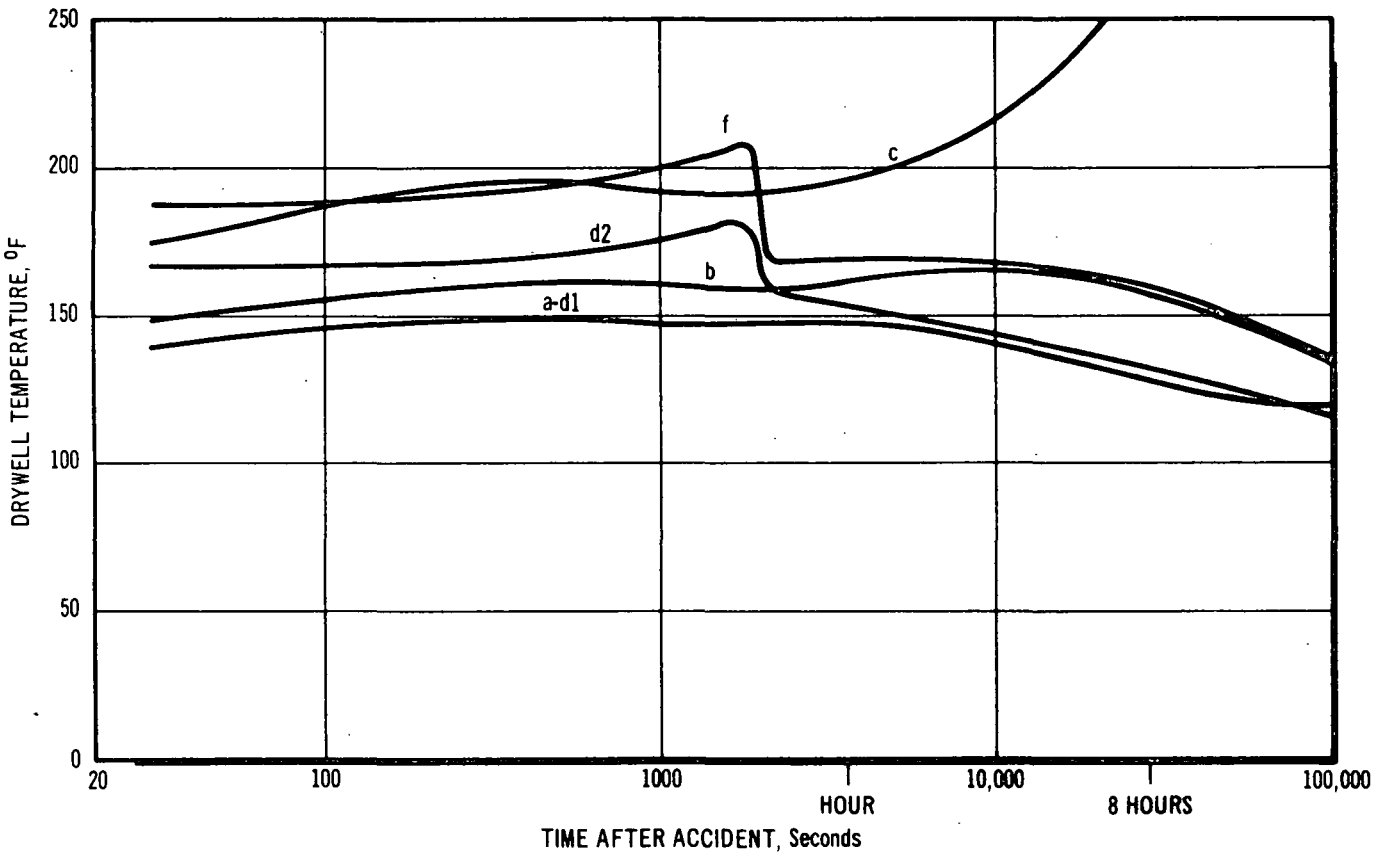


Figure III-7-3. Containment Temperature for Various Available Engineered Safeguards

QUESTION

III Accident Analysis (Continued)

8. Please state those considerations which enter into the assumption that the ventilation fans will continue to function through the course of accidents so the stated apparent stack height can be realized.

ANSWER

The ventilation fans of the Unit 1 turbine building have sufficient operational history to justify their high reliability. The turbine building ventilation system includes two fans rated at 39,000 scfm each that exhaust to the 300 foot stack. Only one fan is operated at a time.

Service on these fans, with the exception of a six day period, has been continuous since the startup of Unit 1 in 1959. During that six day outage modifications were made to the fans and duct work. During this period when the reactor was shut down, ventilation in the turbine building was not required and both fans were shut off. Normally one fan can be maintained while the other fan is in service.

The two reactor enclosure fans of Unit 1 are rated at 6,000 scfm (low speed) and 12,000 scfm (high speed) each. The operational history of these fans, although good, does not equal that of the turbine building fans. The low capacity of these fans does not make them a vital consideration in the course of accidents.

QUESTION

IV. Site and Meteorology

1. With reference to the calculation of the consequences of accidents provide the following:

   a. The criteria for determining the value of sigma theta (wind diversity) from the meteorological chart records.

   b. Justification of the application of data used to obtain values of sigma theta when measured at low wind speeds with the "Aerovane" instrument.

   c. Justification for the values of ground reflection factors and breathing rates used.

ANSWER

a. Wind charts were examined on an hourly basis. All data extracted therfore apply to a time period of one hour. That is, average wind direction is average for an hour, etc. The wind diversity was recorded as a wind direction range for each hour. Range in this case refers to the total angular direction variation within a one hour period. The range was determined from looking at the wind trace and subtracting the left-most wind direction from the right-most direction, considering any scale shift on the chart or noting where the trace traversed the 0° (also 360°) direction.

   In each case the extreme left or right direction of the chart trace was required to have a real thickness. That is, fluctuations in direction which were rapid (high frequency) such that the ink trace on the chart was either non-continuous, illegible or not at least more than the width of the recording pen trace were not counted.

   Sigma theta represents the standard deviation of the wind direction from its mean direction. Statistically this can be calculated from short duration averages from the normal root mean square relationship. However, comparison of the standard deviation calculated by the root mean square method to simply taking the range of direction and dividing by a single number has been found to be satisfactory by some workers. The number used to divide the range by to obtain sigma theta lies somewhere between about 4 and 6. The significance of this number is that, for example, using a value of 4 means the trace is such that 95% (±2 standard deviations of normally distributed data population) is included in the range reported. In the Dresden data analysis a value of 6 was used. This in effect said the trace was such that 99.7% (±3 standard deviations) was included within the range reported. In addition use of a value of 6 gives a conservative estimate of the standard deviation.

b. The purpose of extracting sigma theta from the Dresden meteorological data was to permit its use in a recently developed and more sophisticated diffusion model. In this model sigma theta multiplied by the average wind speed ($\bar{u}$) is a very important parameter. The details of this diffusion model are explained in an article published in the Journal of Applied Meteorology.[1] The extensive experimental

work on which the model is based and the meteorological instrumentation used are explained in a Geophysical Research Paper.[2]

A 400 foot meteorological tower with seven aerovanes located at seven different levels was situated near the point of release during these diffusion experiments. These instruments provided information on wind speed, wind direction and wind variability (sigma theta) for the point of release of the experimental tracer. A summary of the data taken is shown in reference (1). It can be seen in this summary that data including sigma theta were taken during low wind speed conditions. In fact, wind speed as low as 0.7 meter/sec (1.6 miles per hour) are reported. Several experiments were conducted during wind speeds ranging from 0.7-1.6 meters/sec (1.6-3.6 mph). Diffusion observed during these and other meteorological conditions was the basis for development of the diffusion model described.

The Bendix-Freiz Aerovane was originally selected for installation at Dresden in 1958 because it was comparable to the instrumentation in use at Hanford for a number of years in the various AEC R&D programs in meteorology. Thus the Dresden meteorological data were recorded on an aerovane recorder attached to a Bendix-Freiz aerovane instrument. Extraction of sigma theta from this data at any and all wind speeds was considered proper for use in a diffusion model which had been developed using the same sort of instrumentation.

c. The diffusion calculational model(s) used in the accident analysis does not account for the difference in dispersion conditions vertically in the atmosphere. Since horizontal wind speed changes vertically and since surface (ground) roughness effects vary with height above the ground, it is expected that variable dispersion will exist. The magnitude of this difference can be seen in Table 8.2, page 105 of Meteorology and Atomic Energy. The data therein show that the meteorological diffusion parameter "C" is at least twice as large at h = 0 meter compared to h = 200 meters. Thus the diffusion model(s) used overestimate ground-level air concentrations by not considering this effect.

To account for "reflection" effects, some workers multiply the diffusion equations by a factor of 2. This generally is considered to be an upper limit to the effect and a pessimistic factor in the calculation. Actually, however, the "reflection" factor of 2 increase, and the enhanced diffusion by a factor of about 2 near ground-level, (and not considered in the method) would in effect cancel. Thus a best estimate accounting for this "reflection" effect is to assign a value of one to the "reflection" factor. However, since these effects are generally ignored in diffusion calculations, a net "reflection" factor of 1.5 was used in the Dresden Unit 2 analysis.

The breathing rate used in the analysis is 230 cc/sec. This is taken from ICRP[3] where on page 16 it is stated that the "standard" man breathes 2×10^7 cc per day. From this the breathing rate is 2×10^7 cc divided by 8.64×10^4 sec or 231 cc/sec average. ICRP also points out that for purposes of calculating permissible air concentrations of radioisotopes for workers a higher rate was used. It was assumed by ICRP that due to the greater activity during an 8-hour work period, half of the intake occurred during this period. Thus a rate of 347 cc/sec is used by ICRP for calculations pertinent to workers. Continuing with the ICRP assumption, if half the intake occurs during 8 hours of work, during the 16 hours of non-work the breathing rate must be 174 cc/sec. In the accident analysis, it was assumed that a hypothetical person is located at the site boundary for the first two hours or for the total accident at another location. This hypothetical person could be in a non-work status breathing at 174 cc/sec or in a work status breathing at 347 cc/sec. For analysis purposes it was considered that this hypothetical person was not in a work status, but could be breathing at more than the non-work rate consistent with the concept of possible evacuation. Consequently, it was assumed that the breathing rate was 230 cc/sec.

REFERENCES

1. Fuquay, J. J., Simpson, C. L., and Hinds, W. T., "Prediction of Environmental Exposures from Sources Near the Ground Based on Hanford Experimental Data," Journal of Applied Meteorology, Volume 3, No. 6, December 1964.

2. "The Green Glow Diffusion Program, Volumes I and II," Geophysical Research Papers No. 73, April 1962, AFCRL-62-251 (I and II).

3. Recommendations of the International Commission on Radiological Protection, ICRP Publication 2, Report of Committee II on Permissible Dose for Internal Radiation 1959.

QUESTION

IV Site and Meteorology (continued)

2. The various characteristics of the site were described in some detail, however, the manner in which the information relative to site activities, environs occupancy, geology, hydrology, and meteorology has been applied in consideration of the design of the facility is not evident. We believe that the characteristics of the site should not only be discussed in some detail (as has been done), but also this information should be analyzed, and, based on this analysis, conclusions relative to the ability of the site to safely support the proposed activities should be drawn.

ANSWER

Section I-2 of the Plant Design and Analysis Report has been revised in its entirety to reflect additional conclusions relative to the proposed activities at the Site. The revised Section is transmitted as changes to the original report, and in addition are included here in answer to the question.

2.0 SITE AND ENVIRONS

This section summarizes the analyses and studies set forth in Volume III - Site and Environs - and sets forth the conclusions which may be drawn therefrom and which confirm the suitability of Dresden Nuclear Power Station as a site for Unit 2.

2.1 Description of Site

2.1.1 Site Description

The site for Dresden Nuclear Power Station consists of a tract of land of approximately 953 acres located in the northeast quarter of the Morris 15' quadrangle (as designated by the U. S. G. S.), Goose Lake Township, Grundy County, Illinois. The tract is situated in portions of Sections 25, 26, 27, 34, 35 and 36 of Township 34 North, Range 8 East of the Third Principal Meridian. The site boundaries generally follow the Illinois River to the north, the Kankakee River to the east, a country road from Divine extended eastward to the Kankakee River on the south and the Elgin, Joliet and Eastern Railway right-of-way on the west as shown in Figure 2A in Volume II and on Exhibit III-5-1 in Volume III. The character and contours of the site and the immediate environs are shown in Exhibit III-2-1 (aerial photograph) and Exhibit III-2-2 (vicinity map) in Volume III.

2.1.2 Site Ownership

Commonwealth Edison Company is the sole owner of the entire 953 acre tract subject only to an easement of the U. S. Government for an access road to the Dresden Island Lock and Dam maintained and operated by the U. S. Corps of Engineers. Such access road traverses the site from south to north approximately 0.8 mile west of Unit 2.

In addition to ownership of the 953 acre tract Commonwealth also leases approximately 17 acres in two narrow strips of river frontage located near the northeast corner of the site from the State of Illinois. The terms of the lease provide that these "buffer" strips shall remain idle.

2.1.3 Location of Units 1 and 2 on the Site

Unit 1 is located on the northeast corner of the site with an intake canal extending from the Kankakee River to the east and a discharge canal extending north to the Illinois River.

As shown on Figures 1 and 2 in Volume II, Unit 2 is to be constructed on the site immediately to the west of an adjacent to Unit 1. At this location Unit 2 will be situated approximately 0.5 mile from the south boundary of the site, 0.5 mile from the centerline of the Kankakee River to the east, 0.5 mile south of the center of the navigation channel in the Illinois River, and approximately one mile from the west boundary of the site.

2.1.4 Other Activities on the Site

Portions of the 953 acre tract outside the area occupied by Unit 1 and required in the construction of Unit 2 are leased to a neighboring farmer for cattle grazing and field crops. Approximately 150 acres are used for grazing with appropriate fencing provided to control the approximately 75 head of cattle that may be present during the pasturage growing season. Field crop cultivation generally occupies about 300 acres.

Some recreational activity in the form of hunting is permitted on the site outside the security fenced areas during legally prescribed seasons. Control of hunting is delegated to the lessee in recognition of his interest in preventing damage to livestock and crops.

No activities other than those enumerated are currently contemplated for the future. There are no residences on the site.

2.1.5 Access to the Site

Unit 1, including the intake and discharge canals, is completely enclosed by a security fence consisting of six-foot high chain link fencing surmounted by three stranded barbed wire. This fence also establishes the boundary of the Unit 1 site for purposes of the Price-Anderson indemnity agreement and the nuclear liability insurance policies maintained with respect to Unit 1.

Access to Unit 1 is controlled at a security gate. Unit 2 will be constructed outside the security fence surrounding Unit 1. The fence will be adjusted from time to time so that control of access to Unit 1 can be maintained without interfering with the construction work. When construction of Unit 2 is completed the security fence will be extended to control access to both Units 1 and 2.

A paved county road extends south from the gate to Dresden Nuclear Power Station and intersects several other paved county roads which connect with several state highways. Lorenzo Road which runs in an east-west direction approximately 3-1/2 miles south of the site provides access to Interstate Route 55 approximately 4 miles east of the site. Interstate Route 55 is a limited access highway between Chicago and St. Louis. Another limited access highway, Interstate Route 80, which traverses the State from east to west, lies approximately five miles north of the site and is accessible either from Interstate Route 55 or from a state highway, Illinois 47, at a point approximately two miles north of Morris, Illinois.

Railroad access to the site is provided by a siding from the Elgin, Joliet and Eastern Railway right-of-way which forms the western boundary of the site.

There are no airports within eight (8) miles of the site and the closest major airports are Chicago O'Hare International Airport and Chicago Midway Airport situated approximately 50 and 40 miles, respectively, to the north and northeast of the site.

The frontage upon the Illinois and Kankakee Rivers would permit access by water, but no facilities for boat docking or access roads to the frontage have been developed.

2.2 Description of Environs

Residential occupancy in the immediate vicinity of the Dresden site continues to remain low. Inspection of the aerial photograph (Exhibit III-2-1) shows that within a 1-mile radius there are several residences at the Dresden Dam about 0.8 mile NW of the reactor locations, a few homes at about the same distance on top of the bluffs on the opposite shore of the river to the northeast, and several farm residences at 0.8 to 1.0 mile to the south and southwest. In addition, there is a cluster of about 20 cottages on the west shore of the Kankakee River about 0.7 mile from the reactor location. Most of these are occupied only part-time for recreational purposes. Comparison of the U.S.G.S. map used in Exhibit III-2-2 which was based on aerial photographs taken in 1952, with the 1964 Aerial Photograph shows that only in this cottage area has there been any increase in the number of roofs visible.

Land usage and population in the environs is summarized in Volume III-3. It is noted that no new small village population nucleus has developed near Dresden within the past 15 years; and also that no small village of as many as 100 residents exists within 3 miles of the site.

The population within a radius of 5 miles from the site is approximately 2,600. The largest village in this zone is Channahon (pop. 1,200) which is located 3-1/2 miles to the northeast.

The total population within a 10-mile radius is about 23,000. The largest town in this area is Morris, the county seat of Grundy County, with a population of about 8,000. The increase in total population within the 10-mile zone was less than 10% in the 1950-1960 census interval. There is no reason to believe that this trend will not continue in the foreseeable future.

The population in the 10 - 25 mile zone is estimated to be 225,000. There are two population centers within this zone, the closest being Joliet, centered 14 miles northeast of the site, with a population of about 67,000. The city of Aurora, 25 miles to the north (which is situated in the southeastern corner of Kane County), comprises the other population center. By 1970 it is estimated, on the basis of the 35% growth pattern in the 1950-1960 census interval, that the population of the 10 - 25 mile zone will have increased to approximately 300,000. Present knowledge of scheduled projects and planning for this area does not indicate that the growth rate will be significantly changed in the next 5 years. Estimates beyond 5 years for a local area of this size are not considered reliable.

The land to the north and west of the site is used principally for agriculture. To the south there is about 3 miles of farm land and then a large abandoned strip mine area. The large (36,000 acres) Joliet Arsenal is located east of the site and adjacent to a recreational area of about 2500 acres owned by the State of Illinois.

2.3 Site Geology

A recent study of the geology of the Dresden site has been made by Dames and Moore, Consultants in Applied Earth Sciences, Soil Mechanics, Engineering Geology, Geophysics. Work was performed by their San Francisco and Chicago offices, and some of the core testing was done in their New York laboratory.

The previously available geological and associated data and reports for the Dresden area were reviewed, additional background data were collected, and a field reconnaissance of the area was performed by a geologist. The results of the 69 previous borings on the Dresden site were studied, and two additional test borings to the approximate 100 foot depth were made in March 1965 in the immediate area of the Unit 2 principal structures. Samples of the overburden soils and continuous cores of the underlying rock were obtained. Representative cores of rock were subjected to unconfined compression tests, density, tests, and laboratory dynamic tests to evaluate the compressional wave velocity and the shear modulus of the various rock strata encountered. Using small explosive charges, tests were performed in the test borings to measure the in-place compressional wave velocities of the various strata present.

The Dames and Moore report of the currently applicable portions of their work is presented in Section 2 of Volume III. The results of the Illinois State Geological Survey's analyses in 1957 of the previous records and cores from the previous 69 test borings and from other wells in nearby areas are also summarized.

The generalized geologic column for the site consists of an upper layer of Pennsylvanian Pottsville sandstone of variable thickness which in the two new borings showed a thickness of 40 to 50 feet. Next below is a layer of about 15 to 35 feet of Ordovician Maquoketa Divine limestone based on a 65 foot layer of Maquoketa dolomitic shale. The Ordovician system has a total thickness approaching 1000 feet, with the Cambrian system next below. Brecciated rock is found on some cross sections and is indicative of ancient faulting. The geologic evidence indicates that these faults are inactive.

Laboratory tests showed that unconfined ultimate compressive strength on boring samples ranged from 2,000 to 15,000 pounds per square inch on most samples. Laboratory wave velocity propagation tests showed 4,000 to 15,000 feet per second, and the field testing in the two borings was generally consistent with the laboratory findings.

In summary it may be said that the geological characteristics of the site, which were previously studied and determined to be suitable for Dresden Unit 1, have been confirmed by the recent studies. The load bearing capability of the rock formation is significantly in excess of that necessary for the support of the Unit 2 structures. The topographic (elevations) characteristics of the Dresden Station, particularly that proposed for location of the Unit 2, preclude possible movements (slides), either of the plant structures into the Illinois River or earth slides from adjacent higher elevations on to the Unit 2 structures.

2.4 Hydrology

The Harza Engineering Company, Chicago, Consulting Engineers - River Projects was retained to advise on the characteristics of the river systems of interest. Their report of applicable findings is given in Section 5 of Volume III.

The Dresden site at the confluence of the Des Plaines and Kankakee Rivers is at the location considered to divide the upper and lower parts of the Illinois River system. The normal pool elevation controlled at the adjacent Dresden Island Lock and Dam is 505 feet, with a maximum historical flood elevation of 506.4. Nominal ground elevation is about 516 feet at the location for the principal structures of Unit 2, which renders the probability of site flooding extremely remote. Spillway capacity at the Dresden Island Lock and Dam is well in excess of the estimated maximum instantaneous flow of the Illinois River (100, (100,000 cfs, based on the assumption that maximum flows for all contributory streams occur simultaneously). The site elevation is well above the vast valley storage area upstream from the dam.

River system flow data applicable to the Dresden site for the years 1961 - 1964 show that river flow exceeded 3,000 cubic feet per second (cfs) on 98% of the days, 3,600 cfs on 93% of the days, 4,000 cfs on 87% of the days, 5,000 cfs on 63% of the days, and 6,000 cfs on 48% of the days. Such flows are more than adequate to meet the cooling water requirements of Units 1 and 2 and assure the availability of sufficient quantities of water for dilution of radioactive liquid wastes from both units prior to discharge into the Illinois River within the limits established in 10 CFR 20 and to reduce concentrations below the point of discharge to approximately 1/1000 of the M.P.C.

The combined effects of dilution, mixing, radioactive decay and deposition on river bottom of the radioactivity will have occurred prior to the discharged water reaching the point of first domestic use at Peoria and will render the contribution of radioactivity by Dresden Units 1 and 2 negligible in relation to that present in the Illinois River from sources other than that of Dresden Station.

The principal usages of the water of the Des Plaines River below Lockport and of the Illinois River are for navigation, sewage disposal and dilution, and condenser cooling water for power plants. At and below Peoria, 110 miles downstream, the Illinois River is also used for domestic water supply. The Kankakee River is not navigable and is used for domestic supply. Corps of Engineers future planning envisions a second lock at the Dresden Dam.

River system water temperatures fluctuate principally due to the seasons. The U. S. P. H. S. in a 1963 report said that due to river usage, the net rise in temperature in the upper portion of the waterway system was about 9°C. The chemical composition of the river waters was studied in detail during 1961 - 1962, as were the biological and bacteriological conditions. The over-all effect is that the lower river system is biologically degraded, and that most sampling stations on the upper and lower system showed evidence of excessive pollution.

2.5 Regional and Site Area Meteorology

Murray and Trettel, Certified Consulting Meteorologists, Northfield, Illinois, have been retained to advise on regional meteorology characteristics, and the summary of their recent studies is given in Volume III-6. Additional studies of site atmospheric diffusion characteristics by Nuclear Safety Engineering, General Electric Company are reported in Section 7 of Volume III.

The site is located in typical "rolling prairie" Illinois terrain. The only major topographic influence, meteorologically speaking, in the area is Lake Michigan, but this is 45 miles to the northeast and is considered to have an insignificant effect on the site climatology.

Maximum temperature in the area, based on the July, 1949 - June, 1955 Argonne National Laboratory data, is 97°F, and the minimum is - 19°F.

Normal annual precipitation in the area is 33.18 inches. Within a 24-hour period a maximum of 6.24 inches has been recorded. Average yearly snowfall is 37.1 inches, with a maximum of 66.4 inches recorded in the 1929-30 winter season.

In the 50-year period, 1913-1963, four tornadoes have been reported in Grundy County. Of 140 tornadoes reported in the state as a whole, 52 are considered "destructive," i.e., caused $50,000 damage or more and/or at least one death. Average area covered by reported tornadoes is about 8 square miles. The shortest path is 1 mile, the longest 163 miles. Width of paths range from a minimum of 34 yards to 4 miles maximum. No tornado wind velocity information is available.

Annual wind frequencies show a rather uniform distribution of wind direction. The most frequent wind directions are from the west and south sectors (22-1/2 degrees). Average wind speed at the site at the 15-foot level is about 8 mph. Maximum wind velocity reported in the area of the site is 109 mph unofficially reported at Joliet on April 3, 1956, and on April 30, 1962 (the official Weather Bureau Station closed in 1952). This is a fastest gust reported during heavy thunderstorms and scattered tornadic activity. The fastest mile of wind reported at various locations in the site area is 87 mph at Chicago and 75 mph at Peoria.

Hourly wind direction variability at the site shows that an average direction range (angular change in direction) is 120 degrees in a 1-hour period, for all wind speed conditions combined. During 0 - 3 mph

wind speeds, the average range in direction is 100 degrees. Approximately 87% of the time when the wind speed is 0 - 3 mph (or 98.3% of all wind speeds) the wind direction range is 60 degrees or more, which corresponds to a value of the diffusion parameter $\sigma_\theta \bar{u}$ of 20 degree-mph or 0.16 radian-meters/second.

It is concluded that from a meterorological standpoint the site is suitable for the combined operation of Units 1 and 2. The environmental surveys of the site and surrounding areas conducted by Commonwealth, Argonne National Laboratory and the State of Illinois demonstrate that meteorologic diffusion characteristics provide a means for dispersion of gaseous wastes emitted during normal operation to a degree that they are almost undetectable in the environs of the site. There is nothing in the meteorological or topographical data which would indicate that the diffusion characteristics would not be operative during assumed hypothetical accident conditions. The "rolling prairie" terrain would not support a sustained inversion. The hourly wind direction variability of 60 degrees or more 98.3% of the time at all wind speeds provides evidence that the concentration of any accidental release of radioactive gaseous products would be rapidly diluted and dispersed.

The occurrence of tornadic activity in Illinois and throughout the midwest is recognized. Since tornado effects are generally erratic and localized to very narrow paths or areas, the probability of tornado damage at any particular point, or even an area as large as the Dresden site, is so remote as to be incalculable. Nevertheless, the possibility of tornado damage makes it necessary that Unit 2 be so designed that it can be shut down and maintained in safe, shutdown condition in the event tornadic action should cause damage at the site or interrupt any service necessary for normal operation.

The reported occurrence in the general area of the site of sustained winds in the range of 75 to 87 mph indicates that a structural design capable of withstanding wind loadings of 110 mph provides adequate assurance that Unit 2 will be available to supply power to the Commonwealth system. In the remote event of higher winds, safety of the public can be assured by providing an ability to shut down Unit 2 and maintain it in a shutdown condition.

2.6 Seismology

The Dresden site area is placed in Zone 1 (zone of minor damage) on the seismic probability map of the 1958 Uniform Building Code. The August 1958 Seismic Regionalization map by Richter gives general predictions of probable maximum intensity, and, recognizing that lines between the areas of differing intensity are approximations only, shows the Dresden region as Modified Mercalli 7 to 8.

Only several earthquakes of significant intensity in northern Illinois have been reported since 1800, and none has been accompanied by clear-cut surface faulting. A quake on May 26, 1909, caused moderate damage in Aurora, Bloomington, Chicago, and Joliet, and may have been of intensity MM7 in the Dresden area. A quake on January 2, 1912, had a reported intensity of MM6 at Aurora, Yorkville, and Morris, and probably was of similar intensity at Dresden. Consideration of an intensity of MM7 for the Dresden region appears appropriate.

The engineering consulting firm of John A. Blume and Associates, San Francisco, has been retained for advice on seismology, and they have consulted Dr. Perry Byerly, Oakland, California, on the seismicity of the site region. The consultant's findings are reported in Section 4 of Volume III.

The seismological studies indicate that the area of northern Illinois and the actual Dresden site are seismically suitable. Nevertheless, in the interest of conservatism it is appropriate to adopt a design approach which will assure the safety of Unit 2 so as to preserve the ability to maintain the plant in a safe, shutdown condition in the event of a strong earthquake having a ground acceleration of 0.20 g.

2.7 Environs Radioactivity Monitoring

The natural-and-man-made radioactivity of the environs of the Dresden site is surveyed by several monitoring programs. The long-established and continuing program of the Argonne National Laboratory monitors a radius of the order of 100 miles, thus encompassing the Dresden area, and includes one monitoring point 3 miles north of Dresden. An initial series of river samples was analyzed in 1956-57 by the National Aluminate Company under contract to the General Electric Company. The monitoring program of the State of Illinois Department of Health includes sampling of air and water near the Dresden site starting in November 1959. The continuing program sponsored by the Commonwealth Edison Company was started in September 1958, and typically includes some 3000 to 4000 radioactivity analyses and survey instrument readings each year.

Particulate radioactive material in the air is dominated by fallout from weapons testing, reaching a beta emitter peak of 1.3×10^{-11} $\mu c/cc$ in June of 1963 compared to about 10^{-12} $\mu c/cc$ in late 1964.

External gamma radiation of 2 to 3 milliroentgens per week is from natural background cosmic and ground sources, and is not significantly altered by weapons testing.

River water concentrations show a natural background of 1 to 5×10^{-8} $\mu c/cc$ due to natural radium, uranium, and radio-potassium, and have shown an order of magnitude increase during the 1963 peak weapons testing fallout.

Biological samples from the river, and vegetation and milk samples also reflect trends ascribable to weapons testing.

The over-all findings have been in general agreement with other local programs and with the national fallout surveillance network results.

2.8 Conclusions Respecting Site and Environs

The construction of Unit 2 on the Dresden site meets the reactor site criteria described by the Commission in 10 CFR 100 for the following reasons:

a. Commonwealth's ownership of the large, 953 acre, tract provides the requisite exclusion area for a power reactor such as Unit 2.

b. There are no residences on the site or within a radius of 0.5 mile of Unit 2.

c. As indicated in Sections I-7, VI-2 and VI-3, Units 1 and 2 are independent of each other to the extent that an accident in one would not initiate an accident in the other and the simultaneous operation of both units will not result in total radioactive effluent releases beyond allowable limits, but will be kept within limits now prescribed for Unit 1.

d. As indicated in Sections I-6 and IX-5, the total radiation doses under postulated hypothetical accident to an individual at the boundary of the exclusion area or at the outer boundary of the "low population zone" are within the limits prescribed by 10 CFR 100.

e. Activities which are permitted on the site but are unrelated to the operation of Unit 2 do not present any hazards to the public.

f. There are numerous access roads, including Interstate Routes 55 and 80, within the "low population zone" permitting rapid evacuation.

g. The population density and use characteristics of the site environs in the "low population zone" are compatible with the combined operation of Units 1 and 2.

h. As previously discussed, the geological, hydrological, meteorological and seismological characteristics of the Dresden site and environs are suitable for the location of Unit 2 on such site.

DRESDEN NUCLEAR POWER STATION

UNIT 2

Trans w/9-16-65 ltr.

PLANT DESIGN AND ANALYSIS REPORT

AMENDMENT NO. 4

ANSWERS TO AEC QUESTIONS

Commonwealth Edison Company

3067

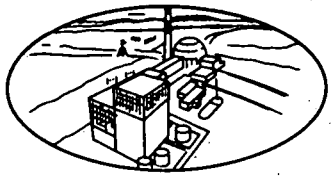
DRESDEN NUCLEAR POWER STATION

UNIT 2

PLANT DESIGN AND ANALYSIS REPORT

AMENDMENT NO. 4

ANSWERS TO AEC QUESTIONS



Commonwealth Edison Company

QUESTION

1. Present an analysis of jet pump system performance with particular regard to system stability and reactivity effects. Discuss the likelihood of injecting bubbles into the core via the jet pumps and the consequences of this on core behavior.

ANSWER

Jet Pump Reactor Coolant Recirculation System

Since 1962 General Electric has carried out development programs for the purpose of installing jet pump systems for reactor cooling recirculation in boiling water reactors. These systems consist of internally mounted static pumping devices containing no moving parts for the purpose of providing forced circulation to the reactor primary system. The use of jet pumps allows for a considerable reduction in the total volume of coolant which must be pumped external to the reactor vessel.

The jet pump design differs from the conventional forced circulation design in that only a minor fraction (approximately one-third) of the recirculation flow passes out of the reactor vessel and through a conventional recirculation pump to return to the vessel as the motive flow in jet pumps within the vessel. Figures 18 and 19 in Volume II of Plant Design and Analysis Report illustrate the use of jet pumps for reactor recirculation. Figures 12 and 30 show the jet pump assemblies and their arrangement in the downcomer of Dresden Unit 2. Figure 67 illustrates the parts of a jet pump, the pressure variation through the pump, and the pertinent nomenclature.

Operation of the jet pumps is dependent upon a driving fluid and a driven fluid. The principle of operation of the jet pumps is the conversion of momentum to pressure. A portion of the recirculation coolant flow passes externally through recirculation pumps where the pressure is increased to values higher than is required in a normal recirculation system. This high pressure fluid, the driving flow, is distributed to the driving nozzles of a number of jet pumps and emerges from the nozzles at high velocity and high momentum. By a process of momentum exchange, the driving flow entrains the remainder of the recirculation flow, the suction flow or subcooled driven fluid. The combined flow then enters the mixing sections or throats of the jet pumps where the momentum decreases with a resultant increase in pressure. The combined flow then passes through the diffuser sections of the jet pumps where additional pressure recovery is accomplished and the flow leaves the jet pumps with pressure sufficient to cause the desired circulation through the core of the reactor.

Why Use Jet Pumps?

Initial studies of jet pump recirculation systems indicated that the use of such apparatus would provide a definite increase in the inherent safety of the reactor system and at the same time provide a definite capital cost reduction for boiling water reactors.

The jet pump system has equivalent or superior safety behavior compared to a normal recirculation system for those classes of accidents associated with the recirculation system. That class of accident associated with rupture of the recirculation piping is less severe for the jet pump system in three respects: The number of major nozzle penetrations on the vessel have been reduced, the speed of the vessel blowdown in the event of a major line rupture is greatly reduced because of smaller diameter recirculation piping, and the reactor internals arrangement is such that there is a definite capability to reflood the vessel even if there is a complete severance of a recirculation line. The higher natural circulation potential of the jet pump system also tends to improve the "heat flux - coolant flow" relationship which results from a complete loss of power to the recirculation pumps.

Cost incentive included the reduction of the number of external coolant loops required with the related reduction of loop piping, and the number of valves, reactor vessel nozzles, and flow control motor-generator sets. Another major cost incentive is derived by the reductions to containment building costs offered by a two loop jet pump plant as opposed to a five or six loop normal recirculation pump plant of equivalent size.

Design Concerns Related to Jet Pump - System Usage in Product Line Plants

Before the jet pump system for reactor recirculation was chosen, it was recognized that a number of design concerns required further investigation through extensive test programs and analyses. Among these concerns were:

1. The effects of the decreased mass inertia and lower resistance characteristics of the jet pump hydraulic loop and the decreased slope of the pump head characteristics curve on reactor system stability.

2. Possible instabilities in the jet pump apparatus itself and their influence on reactor system stability.

3. The effects of carryunder on jet pump system performance.

4. Possible reactor control problems arising from usage of the jet pump system.

Thus the questions posed concerning reactor stability and the extent and effect of steam bubbles reaching the core were within the scope of design evaluation and testing.

How Design Concerns Have Been Answered

Extensive analyses have been conducted to investigate design concern areas such as those listed in the section above. In addition there have been several experimental programs of actual jet pump apparatus. These tests were made with single jet pump units and also multiple unit assemblies at both cold and reactor service conditions.

1. Analyses have been conducted to predict the stability of the nuclear-thermal-hydraulic loops of current reactors which utilize jet pump systems for recirculation coolant flow. A linearized model of this important loop is analyzed on a frequency response basis by a digital code to give

predictions of gain and phase margins. Results have shown that the proposed reactors using jet pumps meet the design stability requirements of gain and phase margins at design power as specified in Amendment No. 2. These margins for the optimum performance of jet pumps can be made equivalent to the margins obtained with non-jet pump reactors by means of proper increase in orificing of the reactor core to offset the decreased mass inertia and lower resistance characteristics of the jet pump hydraulic loop. All stability margins identified in Amendment No. 2 are met with the proper design of the jet pump system.

2. As described in Amendment 2, loop tests of a multiple unit jet pump assembly were performed at the General Electric Steam Separation Test Facility at the Moss Landing Power Station of the Pacific Gas and Electric Company. At that facility, reactor hydraulic conditions can be simulated. The major objective of these tests was to determine whether the systems as constructed would exhibit a "manifold instability" behavior. These tests assessed the operational hydraulic stability of the system as constructed and provided data for predicting the behavior of reactor installed jet pump recirculation systems.

   No evidence of "manifold instability" or other form of system instability was ever observed, even under conditions where efforts were made to drive the system into poorly damped or unstable situations. In this connection it is significant to note that the system as tested was, for reasons explained in Amendment 2, more sensitive to instability than Dresden Unit 2 design or to other designs utilizing jet pumps.

   Notwithstanding the inability to generate system instabilities in the multiloop tests, analyses were conducted on an analog simulation of a boiling water reactor to determine overall reactor effects of possible oscillations induced by the jet pump process. Such analyses demonstrated that even if flow oscillations did arise from a jet pump system, they could be tolerated provided that the recirculation flow oscillations were no more than ±2 percent of point. Flow oscillation of ±2 percent of point in flow could cause flux oscillation of ±10 percent of point, and it has been demonstrated in BWR's that oscillation of ±0.5 percent of point in steam flow or ±10 percent of point in neutron flux could be tolerated and yet maintain safe operation of the reactor facility with margin. As previously stated, flow oscillations of this magnitude resulting from jet pump system operation even under more severe and sensitive conditions than those expected to occur in normal service and maneuvering plant operations were not observed or produced.

3. The carryunder and saturated circulated flow from the separators is normally quenched and subcooled at the entrance of the downcomer by mixing with the feedwater flow. There has been considerable investigation of the effects of carryunder on jet pump system performance because of the potential for reducing the feedwater flow to the point where a jet pump system must operate with saturated coolant or possible carryunder and thus begin to cavitate. It has been determined that feedwater flow of at least 15 percent of rated, with the reactor at full recirculation flow conditions, will provide sufficient subcooling to prevent cavitation from occurring in the jet pumps. Further increases in the temperature (further decreases in subcooling) of the jet pump suction flow up to the point of cavitation produces no appreciable effect on flow. At the point of cavitation, the suction flow can no longer be increased by pushing the driving pumps harder. Driving flow itself can be increased under cavitating conditions within the limited capacity of the drive pumps. The design of the reactor internals including the downcomer

region and the feedwater sparger, are such that very good mixing is accomplished between the feedwater and the recirculation water wall before reaching the jet pump inlets. It has been demonstrated in many tests that the jet pump process is essentially a volumetric pumping process. Thus, as bubbles form, there is a reduction in the mass flow rate from the jet pumps. The effect of cavitation is a slight reduction of flow ratio as the subcooling of the suction stream is gradually decreased. There is no reduction in driving flow so the result is a gradual reduction in total flow.

In the air-water tests of the jet pump apparatus, it was determined that for a constant discharge pressure, the total flow was reduced less than 1 percent for a volumetric carryunder equivalent to 0.2 weight percent steam at reactor conditions. With a carryunder of 1-1/4 weight percent the total flow was reduced about 6 percent while discharge pressure was reduced about 14 percent. From these and other tests it has been concluded that jet pump performance would not be compromised by carryunder bubble content equivalent to 0.3 weight percent in suction flow steam which is a reasonably expected upper limit.

In the multiloop tests, which were conducted at reactor service conditions, carryunder at the entrance of the jet pumps was simulated by introducing superheated steam to the slightly subcooled suction stream. In the evaluation of these tests it was deduced that some of the superheated bubbles might possibly have passed entirely through the jet pump apparatus. Superheated bubbles, however, are an unrealistic condition and it is most probable that the bubbles formed due to cavitation will be collapsed before they pass completely through the diffuser section of the jet pumps. It is extremely unlikely for bubbles to be injected into the core via the jet pumps at reactor conditions. Loss of feedwater would cause a degradation of the flow ratio and thus a reduction of recirculation flow. This in turn would reduce the reactor power without safety implications. The loss of subcooling would further reduce the reactor power level.

4. The jet pump system has been shown to be compatible with the presently designed flow control system.

   Analyses have shown that the higher effective pump inertia associated with the higher head pumps required for drive flow of the jet pumps has an effect of causing slightly slower flow control maneuvering response. This minor operating penalty is offset however by the slower (and safer) flow decay which results from the loss of all pumping power. The range of flow control is slightly less for jet pump plants than for normal recirculation plants because of the higher natural circulation of the jet pump system. There appears to be no serious problems related to usage of the jet pump system in conjunction with the flow control system. Analyses conducted on analog simulations of jet pump systems have demonstrated that presently designed reactor control elements are sufficient to assure safe operation of jet pump plants.

Summary

The status of the resolution of the various design concerns associated with the use of jet pumps on the Dresden Unit 2 Boiling Water Reactor is:

1. Analyses conducted by digital and analog computer indicate that the jet pump design can be made equivalent to the conventional pumping designs with respect to reactor system stability.

2. Experimental programs conducted to date have indicated that there are no instabilities due to the jet pump process itself.

3. The jet pump design is such that the passage of steam through the jet pump is highly improbable based on analysis and experiment. Analysis also indicates that should steam pass through the jet pumps and into the reactor an orderly power decrease would occur as in other conventional boiling water reactors.

4. The jet pump offers no major restriction on the control of the boiling water reactor.

Therefore, sufficient investigations have been conducted to indicate that jet pumps are a definite improvement to the boiling water reactor.

QUESTION

2. Starting with adverse control rod patterns which lead to flux peaking at the bottom of the reactor and large void fraction at the top extending deep into the core, present an analysis of severe pressure transients which cause void collapse and their reactivity effect. How stable is the core to such events?

ANSWER

Introduction

As part of the overall hazard analyses associated with the design of a power reactor, considerable emphasis is placed on the investigation of effects associated with the insertion of excess reactivity into an operating reactor. These investigations are generally associated with reactivity insertions caused by inadvertent withdrawal of control rods, inadvertent addition of fuel, etc. Also, in the design of the boiling water reactor reactivity insertion associated with the collapse of steam bubbles due to pressure increases within the core and the resulting increase in moderator effectiveness is considered. Detailed investigations of the boiling water reactor system have indicated that significant pressure increases can be caused by sudden decreases in the allowable steam flow from the reactor vessel. In the course of the design of the boiling water reactor all potential sources for a decrease in the allowable steam flow from the reactor vessel are investigated and these include all normal and abnormal activation of such equipment as reactor isolation valves, turbine stop valves, turbine admission valves or bypass valves. Investigations conducted on the Unit 2 reactor system indicated that the most severe pressure transient which can be imposed on the core is that one associated with the complete closure of the turbine stop valves while at full power. This is due to the fact that this maneuver has a maximum steam flow termination (9.9 million pounds per hour) in a minimum amount of time (about 0.1 seconds) thus causing maximum rate of pressure increase on the reactor core. Further details as to the effects of this maneuver are given below.

Turbine Trip Maneuver

Should there be a need to terminate the steam flow to the turbine (e.g., excessive vibration, machine overspeed, etc.) turbine stop valves in the steam line connecting the reactor with the turbine are automatically closed. This action is undertaken principally for turbine protection. The turbine stop valves are capable of completely closing in approximately 0.1 second to adequately protect the turbine. The exact closing times are well established by actual closing time measurement on the installed system. Whenever a turbine trip-off is required for one of the reasons mentioned above, the closure of the turbine stop valves is activated and steam flow from the reactor vessel quickly terminates. The pressure level within the reactor vessel starts to increase thus causing a decrease in void content, and definite reactivity insertion and an associated increase in reactor neutron flux level. The increase in neutron flux is sufficient in both magnitude and rate to cause the reactor to

be scrammed in less than one second following the turbine stop valve closure. Detailed analyses indicate that there are negligible thermal effects from this transient. It should also be pointed out that there is sufficient concern with respect to these pressure transients that during the power operation testing program in the power plant startup, each of those transients are performed and evaluated to ascertain that adequate performance has occurred. The increase in steam inventory within the vessel due to the closure of the stop valves is terminated by the combined action of the turbine bypass system and the resulting reactor scram.

Analysis of the Turbine Trip Accident

There are three quantities which affect the void content of the core - exit quality, subcooling, and pressure. The effects of exit quality and subcooling are summarized in a void map, which relates core average voids to exit quality with subcooling as a parameter. Flux distribution also has a minor effect on the void content of the core. Void maps were obtained for three flux distributions: the design axial shape (slightly peaked toward the bottom of the core), excessive peaking to bottom of core, and excessive peaking to top of core. Table 2-1 summarizes the change in equilibrium voids and rates of change of core void content as a function of core exit quality and subcooling at the operating point $\left(\partial R_g/\partial X_e \text{ and } \partial R_g/\partial \Delta h_s\right)$ for the three cases analyzed.

TABLE 2-1

REACTOR CORE VOID CONTENT

Flux Distribution	X_e (%)	$\Delta h_s \left(\frac{Btu}{lb}\right)$	Core Average Voids (%)	$\partial R_g/\partial X_e$	$\partial R_g/\partial \Delta h_s$
Peaked to Bottom	9.6	20.4	36.5	2.12	-0.27
Peaked to Top	9.6	20.4	25.5	1.87	-0.38
Design	9.6	20.4	34.0	2.12	-0.29

For the flux distribution peaked to the bottom Table 2-1 shows negligible change in $\left(\partial R_g/\partial X_e\right)$ and a decrease in $\left(\partial R_g/\partial \Delta h_s\right)$ from the design case, i. e., for a given change in subcooling the change in voids is less than for the design distribution. For a turbine trip subcooling increases, which causes voids to decrease (slope is negative). Because the slope is less (in an absolute sense) for the flux peaked to the bottom and since $\left(\partial R_g/\partial X_e\right)$ is unchanged the turbine trip accident is less severe for this case.

From Table 2-1 it can be seen that for the axial shape peaked to the top $\left(\partial R_g/\partial X_e\right)$ is less and $\left(\partial R_g/\partial \Delta h_s\right)$ is greater than for the design case. These changes in slope are in a direction to increase the severity of the turbine trip accident.

For both of the cases mentioned above the changes are small and the change in the severity of the transient therefore should not be appreciable. To demonstrate that the above mentioned effects are small and to show that severe pressure transients do not lead to severe reactivity transients, the above effects were incorporated in the plant analysis model, an extensive digital computer program, and the turbine trip transient was run on the computer. The results for these cases are shown in Figure 2-1 and in Table 2-2.

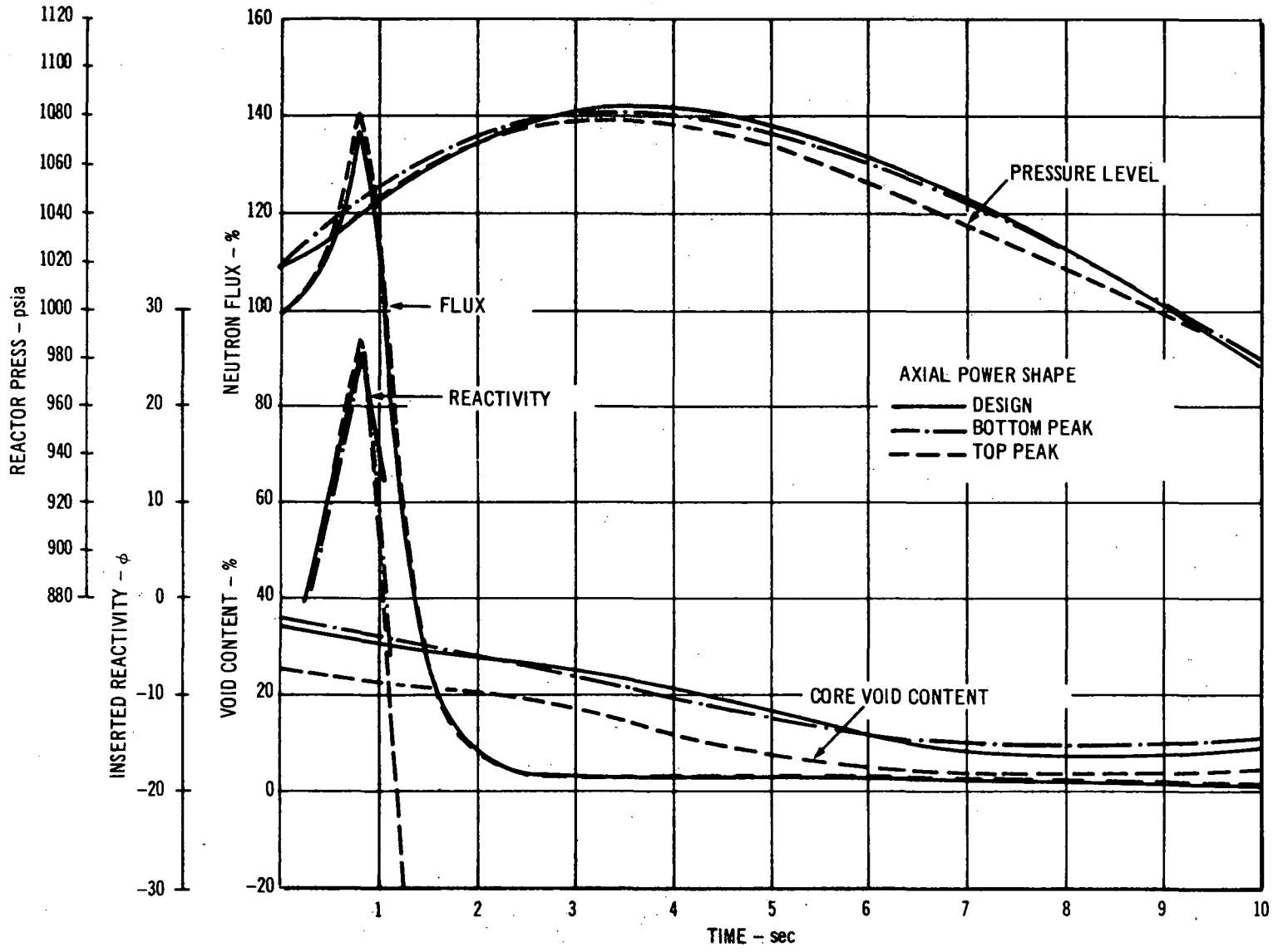


Figure 2-1. Turbine Trip Response

TABLE 2-2

RESULTS OF TURBINE TRIP RESPONSE ANALYSIS

Flux Distribution	Peak Flux (%)	Peak Core Pressure (psia)	Reactivity Inserted before Scram Takes Effect		
			Voids-%	Doppler-%	Net-%
Peaked to Bottom	139	1090	+0.19	-0.16	+0.17
Peaked to Top	140	1090	+0.21	-0.18	+0.19
Design	139	1090	+0.20	-0.18	+0.18

From Table 2-2 it can be seen that the items discussed above are confirmed by the computed results. It should also be noted that the net reactivity inserted up to the time of the flux scram, for any of the cases considered is only of the order of 0.2 percent. The surface heat flux does not exceed its rated value during the transient. When net reactivity inserted for the turbine trip accident is compared to that inserted for a rod drop accident, about 3.6 dollars, we see that for this most severe pressure transient the reactivity insertion is less by a factor of 10. Therefore, the spectrum of pressure induced transients is investigated for purposes of maneuvering capability rather than for purposes of identifying effects of reactivity accidents.

Stability

In Table 2-1 we see that the change in voids with subcooling is, in an absolute sense, less for the flux distribution peaked to the bottom of the core. This effect has a stabilizing influence on hydrodynamic stability. This is offset by the fact that for an axial shape peaked to the bottom the two phase transient time through the core is increased, which has a destabilizing effect. The net effect will be negligible since the magnitudes of the changes are approximately equal. Since the stability analyses conducted on Unit 2, as discussed in Amendment 2, Answer II-2, have indicated that the plant can be designed to meet all stability criteria, and since this analysis has indicated only negligible effects on stability, it is concluded that the effects of power shapes are not large enough to cause more than a negligible change in stability margins. For the flux peaked to the top the change in voids for a change in subcooling is greater than the design case but the two phase transient time is less. Again the net effect is negligible.

QUESTION

3. Re-examine and clearly restate the failure philosophy of the instrument system, i.e., the number of instrument and operator failures required to cause a major accident.

ANSWER

The design philosophy for the safety and control systems requires that, for those accidental power transients with the potential for endangering the health and safety of the public, the system provides, in addition to containment, at least a double level of automatic or inherent protection. For other incidents at least a single level of automatic or inherent protection is provided by these systems.

Accidental power transients may occur from either of two sources. Gross core transients may be induced by reactor primary system disturbances of core boundary conditions, i.e., coolant flow, coolant subcooling or reactor pressure. Localized power transients may develop from movement of core components, i.e., control rod motion or fuel element movement during refueling.

There are no mechanisms within the direct control of the operator or mechanisms which control primary system conditions which can cause power transients which could endanger the health and safety of the public. In the unlikely event of multiple malfunctions or failure of equipment and erroneous operation, transients due to control rod motion or fuel element drop may, in the worst cases, result in a nuclear excursion. In these cases reactivity addition rates may approach five dollars per second and reactor periods may be as short as a few milliseconds. The inherent Doppler effect terminates the excursion and engineered safeguards and system isolation limit the consequences to levels which do not endanger public health and safety. Scram, initiated by signals from safety instrumentation, is of value only in providing negative reactivity for shutdown following the power burst of the excursion. Thus as stated in paragraph IV-5.0 of the PD and AR, "Single component failure within the reactivity control system itself does not result in damage either by motion or rupture to the reactor primary coolant system".

No other transients have a potential for endangering public health and safety. The following discussion relates to the degree of protection provided for the plant.

For bulk transients inertia of the reactor system restricts rate of reactivity addition to the core from coolant or pressure transients to at most a few tens of cents per second. Resulting reactor periods are on the order of seconds or longer. Power increases generally throughout the core. These

transients are automatically terminated by the safety system by scram before core damage occurs. A dual bus safety system is used, and a trip signal from one instrument channel in each bus actuates a scram. Any of four APRM instrument channels can actuate a trip in each of the safety system buses. Bypass of one channel in each bus is permitted with certain restrictions leaving, in the worst case, at least three active channels in each of the buses.

For bulk power level transients, only one channel in each safety bus is required to initiate a scram and the location of the instrumentation within the core is not important. For the most probable failures of critical components in the safety channels, such as interruption of power or failure of crucial electronic components, the failure actuates the trips (i. e., fails in the safe direction). Failure of one of the individual in-core monitors in an instrument channel results in miscalibration, either high or low, of the channel. Recalibration is accomplished procedurally.

As noted above, at least three channels are available to actuate trips in each of the safety system buses, and only one channel is required in each bus. Therefore, in the extremely unlikely event of failure of one channel during a bulk transient, coincident with the worst bypass conditions, at least two channels are still available to provide protection in each of the safety buses.

The power range safety system is also designed to prevent fuel damage from the worst single rod motion error through its automatic rod motion blocking feature. At least eight power range instrument channels are provided, the output of four channels actuating rod motion blocking trips in each of the two safety system buses. Trips from any single instrument channel will actuate the rod motion block. One channel in each of the buses may be bypassed with the restriction that it is not possible to bypass both of the channels which monitor any one of the four reactor quadrants.

For the worst bypass conditions, two of the reactor quadrants are protected by two separate instrument channels, and the other two reactor quadrants are protected by one channel. Therefore, for fuel damage to result from the single rod motion error in the worst case, at least one instrument channel must fail to function in coincidence with the erroneous rod motion.

In the extremely unlikely event of failure of the remaining active APRM channel in a quadrant which has the other APRM channel bypassed, coincident with the worst single rod motion error in the vicinity, the ensuing local power peaking could damage up to about two hundred fuel rods (equivalent to the number of rods in about four fuel bundles). With two hundred damaged fuel rods, the fission product release would be of the same magnitude as given in Section XI-5 of the Dresden Unit 2 PDAR for the control rod drop accident in which 330 fuel rods would be damaged. System isolation capability is provided to control this release without endangering public health and safety.

QUESTION

4. Justify the philosophy of the dual channel safety system and the design to prevent scram. Justification should include the basis for the belief that the safety system will cause scram when called upon to do so. Also, state the criteria that this philosophy leads to in regard to instrument sensor location. What redundancy and backups will be supplied?

ANSWER

Conclusions

The Dresden Dual Bus Protection System has a theoretical reliability equivalent to a 1 out of 2 or a 2 out of 3 system. The dual bus system can be tested frequently to insure that no unsafe failures lie dormant which would impair its ability to function. The testing can be carried out at full power or zero power, exactly duplicating all conditions as they are in the reactor when the protection system may be called on to function.

The dual bus protection system adapts more readily to the control rod drive hydraulic scram circuitry than a 2 out of 3 system and appears to yield a higher reliability with fewer component parts. Thus the dual bus is preferred for General Electric power reactors.

Relays are used as the main building blocks because they are reliable, not hyper-sensitive to environments and excesses of stress, and adaptable to conventional contact signal inputs.

The dual bus is sub-divided into four parts to allow ample opportunity for circuit isolation. Sensors are treated as isolated inputs to the protection system. Further backup is provided so that even in the unlikely event that all rods do not scram, the unscrammed rods will also be inserted.

Theoretical Considerations

The dual bus safety system employed on Dresden 1 and planned for Dresden 2 evolved from the single bus system which had been in use on many experimental reactors prior to the advent of power reactors. The single bus system had for many years been accepted as the standard of safety and simplicity. The guiding rule of design was that a single component failure must not be allowed to prevent a scram. This led logically to redundancy, wherein two (2) or more devices were used to measure a parameter, either one of which could initiate a scram independently of the other. This is frequently called a 1 out of 2 protective system.

Safe failures, by their very nature, reveal themselves. In a 1 out of 2 system, a safe failure initiates a scram. An unsafe failure is unrevealed until a test is initiated to cause the system to operate. On the average, the unsafe failure exists for a length of time equal to half the interval of time between testings.[1]

Consider a 1 out of 2 system tested once per week. The unreliability of the system is the fraction of the time that the system is incapable of responding properly to a bonafide scram signal. This relationship is shown on Figure 4-1, the unreliability being plotted as a function of the failure rate for various systems.

The 1 out of 2 system, although considered excellent from the standpoint of safety, leaves much to be desired from the standpoint of continuity of operation. Unnecessary scrams, in addition to being costly and annoying to a utility, produce unnecessary safety hazards of their own.[2] The Dresden Dual Bus Protection System was designed to effectively solve this continuity of operation problem and at the same time make a substantial contribution to safety.

The Dresden Dual Bus Protection System is a basic 1 out of 2 system repeated twice (1 out of 2)×(2). From a theoretical standpoint (refer to Figure 4-1), the unreliability of the 1 out of 2 system is one-half the dual bus system. This difference, though insignificant in itself, is unreal. In order to test a 1 out of 2 system, the reactor must scram or be shut down. This procedure is unsatisfactory from several standpoints.

1. A test of a reactor protection system which is shut down is never quite representative of an operating reactor. A test schedule should be established and followed whether the reactor is up or down.

2. Shutting the reactor down for test of the protection system cycles the reactor through stresses and strains, which, though properly designed for, are unnecessary. The full-power steady-state mode is the safest of all reactor operating modes.

3. The utilities insist on continuity of operation, not at the expense of safety, but in addition to and enhancing safety.

Any tendency to decrease the test frequency to minimize shutdown time increases unreliability. On a 1 out of 2 system changing from once per week to once per month increases the unreliability by a factor of approximately 19 as shown by the curve on Figure 4-1.

It is of interest to compare on this same theoretical basis the unreliability of other protection systems. The 2 out of 4 system has the least unreliability, and significantly so. Furthermore, since the expression involving λ, the failure rate, is cubed, the unreliability drops more rapidly with decreasing failure rate.

The 2 out of 3 system has a slightly higher unreliability than the Dresden Dual Bus Protection System, but not significantly so. The 2 out of 2 system, long ago rejected as a reactor protection system, is shown for reference. It is worthy of note that at a failure rate of 0.1 per year, the unreliability of the 2 out of 2 system is three (3) orders of magnitude higher than the 2 out of 3 or the Dresden Dual Bus System. A failure rate of 0.1 per year is indeed a worthy goal.

Practical Considerations

The theoretical analysis of a protection system can be useful in choosing a system with good potential; but the final design of a system should be guided by many additional factors. Some of the considerations included in the design of the system are:

1) Is the protection system compatible with the types of inputs usually available?

2) Is the logic complex, difficult to trace or understand, or prone to hide unsafe failures due to redundant circuits?

3) Does the logical decision centralize onto one critical component or circuit, the failure of which could void the integrity of the protection system?

4) Can the various redundant branches of the protection system be isolated from all influences that might cause certain failure modes to be interdependent?

5) Is the protection system unduly affected by heat, moisture, or any other influence that may be impossible to isolate?

6) How much experience has been encountered with similar designs or components?

The following philosophy will give some background on the choice of the Dresden 2 Dual Bus Protection System.

Many of the inputs to a protection system originate directly from simple, reliable, and proven devices which use electrical contacts as their output. Furthermore, these signals originate from systems that may be operating at different electrical potentials. Relays were chosen as the building blocks for the protection system because they readily accept contact signals and work effectively to isolate one potential source from another.

Once relays have been selected, it behooves the designer to keep the logic simple, because complex logic can quickly cascade into a very large number of contacts and relays, all of which can contribute to unreliability. Simple logic has the added advantage that it is easier to understand by technicians charged with its care and maintenance and easier to test without fear of ambiguity. Relays are not prone to fail upon being subjected to short term excesses of temperature or voltage.

Although there is a wear-out failure mechanism built into a relay, the number of operating cycles in a protection system lifetime is so low as to be of little concern.

To the maximum extent possible, simple series circuits of contacts, any one of which tripping causes scram, should be used. Parallel circuits should be avoided because of the difficulty of testing them. For redundancy a whole redundant string of contacts is employed terminating in a separate relay rather than paralleling each individual contact with a redundant contact. These precautions tend to make most failures self-annunciating and the unsafe failures can readily be discovered by testing.

Once the logical decision has been made to scram the reactor, this decision should not be entrusted to a single circuit or component. On Dresden 2, the logical decision to scram is not made in the electrical circuits at all, but in the hydraulic system of the control rod drives. The decision is made individually and simultaneously for every control rod. Two (2) "fail safe" scram pilot valves are employed on each control rod drive. Both of them open to cause a scram of that particular rod. Failure of both of them to open does not in any way influence the ability of any other rod to scram.

The hydraulic scram circuit of the control rod drive is ideally suited as a multiple terminal point for the dual bus protection system. In 1964, a failure mode analysis was made of the scram pilot valves. The reference design two-valve system was compared with a scheme using four (4) valves suitable for a 2 out of 3 protection system. The four (4) valve 2 out of 3 scheme had one-half the unreliability of the reference design two valve system. Doubling the number of valves to cut the unreliability in half is a poor investment in redundancy. Ordinarily one should demand and expect orders of magnitude improvement by doubling the amount of equipment. As a consequence of this and other analyses, the 2 out of 3 protection system was not considered to be as suitable for application in this reactor as the dual bus protection system.

The number and location of sensors coupling into the dual bus protection system receive careful consideration. There is only one reactor pressure, and once it is measured no new information is generated by measuring it again. However, on a dual bus protection system four measurements are required to satisfy the need for four independent inputs. Every care is taken to see that the inputs are truly independent. For example, they are not connected to a common header or located in such a way that they are vulnerable to the same hazards. Although more than four inputs can be accommodated readily on a dual bus sytem, very little improvement in safety is realized from inputs in excess of four, especially if the four inputs have a high reliability to begin with.

Some parameters have a spatial distribution. Common examples are temperature and neutron flux. If only the worst case is of concern and the designer can be assured that a sensor can readily be located at the point that is known to be worst case, then four sensors are adequate. For example, it may be established that the worst case temperature is the outlet temperature in a given flow loop. If there is only one loop, four sensors should suffice. However, if there are multiple loops, the problem compounds into supplying many more sensors to assure that the worst case is discovered with adequate redundancy. If there are "n" outlets, this may mean $4 \times n$ sensors.

A more troublesome problem in sensor location is in neutron flux monitoring. A sensor located outside the core monitors only the leakage neutrons and this neutron flux is not likely to be representative of the worst case conditions. In this situation, increasing the number of chambers without limit may not appreciably improve one's knowledge of the true in-core worst case conditions. In these circumstances, a knowledge of flux distributions gained from other sources must be utilized in order to choose the minimum number of sensors. With the sensors located within the core, there must be a minimum of $4 \times n$ sensors, where "n" is the number of volumes in the core which must be treated as independent volumes. The specific design criteria for the location of sensors within the core is included as Question II-20 of Amendment 2 of the Plant Design and Analysis Report.

The Dresden Dual Bus Protection System terminates in four independent groups of pilot scram valve solenoids. The circuitry is so arranged that no single component failure can prevent all four groups from being de-energized. This scram circuit separation has the added benefit that some of the groups of rods have a high probability of scramming even if two or more failures have prevented the scram

from being unanimous. The probability that enough failures can coincidentally accumulate to prevent any rods from being scrammed is very low. In the event that only a portion of the rods scram, a back-up scram valve functions to insert all the unscrammed rods. Although this back-up scram valve does not insert the rods as rapidly as the regular scram valve, it does serve to insure that all rods eventually go in.

Any channel may be tested at any time. If the reactor is operating, the test will result in one bus being tripped. This is frequently referred to as a half scram. The half scram de-energizes half of the scram pilot valve solenoids. With a half scram effective, one rod may be scrammed completely by means of a toggle test switch which de-energizes one only scram pilot valve solenoid in the opposite bus. The reactor operating at power is tolerant of one rod scram. The Dual Bus Protection System allows a test to encompass everything from sensor input to a rod scram with an absolute minimum of artificiality. This testability precludes the possibility that a hidden unsafe fault could lie dormant and undetected for long time intervals. Testing at power also enhances the probability that the test is truly representative of the reactor's ability to scram at full power. (By the same argument, testing will also be done at zero power to assure that the reactor will scram under these conditions.)

The manual scram signal is entered into the protection system downstream and independent of all other monitored process variables. Each bus has its own scram button and independent circuit allowing the manual scram capability to be tested during full power operation. The scram buttons are located side by side and both must be depressed to effect a bona fide manual scram.

REFERENCES

1. Pearson, A., and Lennox, C. G., The Technology of Nuclear Reactor Safety, Volume 1, Thompson and Beckerley, Editors, Chapter 6, Sensing and Control Instrumentation, Page 296.

2. Epler, E. P., Reliability of Reactor Systems, Nuclear Safety, 4 (4): 72-66 (June, 1963).

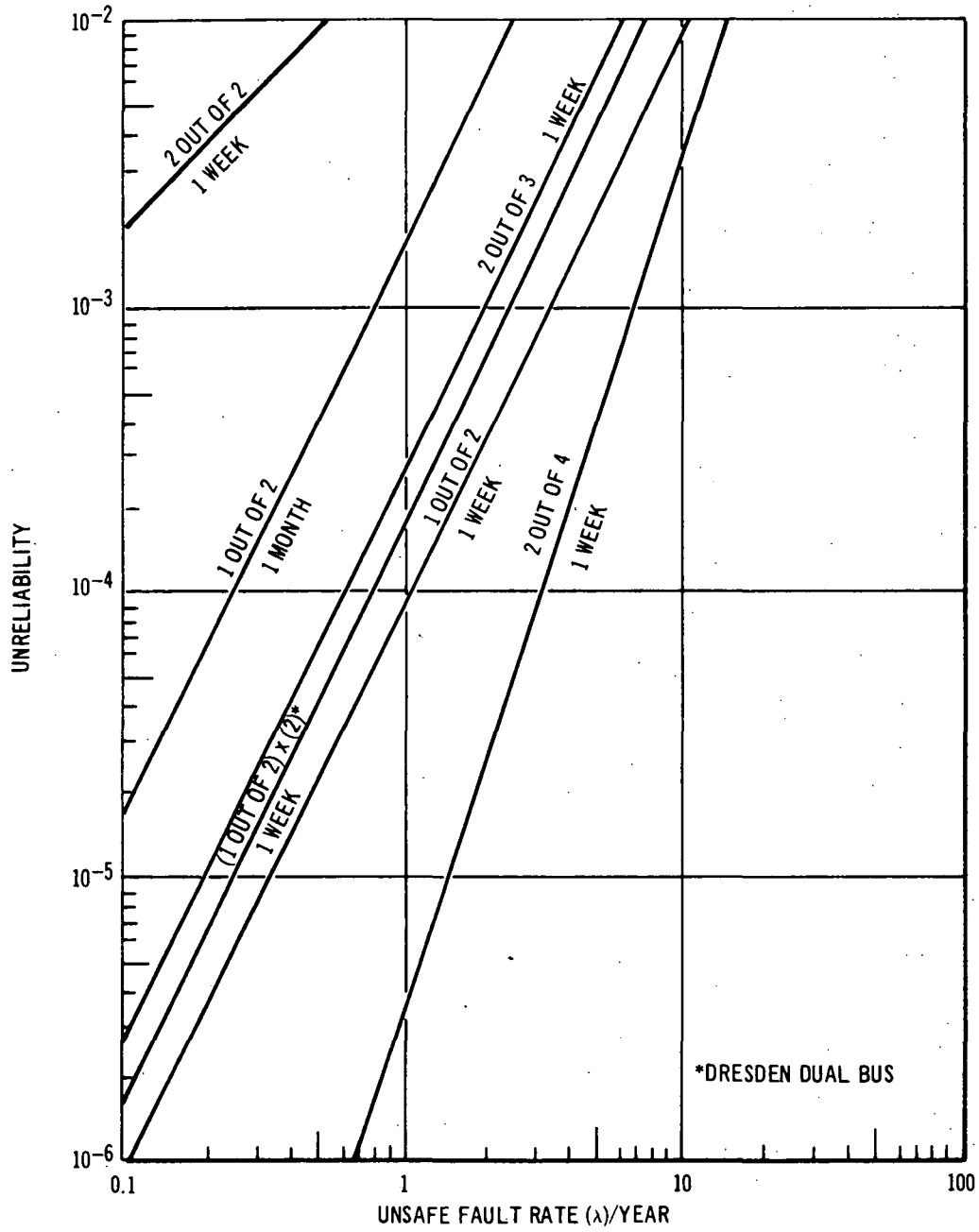


Figure 4-1. Unreliability of Coincidence Systems as a Function of Fault Rate for Various Test Intervals

QUESTION

5. In order to increase the assurance of containment integrity in the event of an accident, consider the desirability of locating the "T.I.P." System entirely within the containment - perhaps in an intermediate air lock or zone.

ANSWER

Consideration initially was given to installing the T.I.P. System completely within the primary containment. However, because accessibility to the T.I.P. equipment was considered to be of prime importance to maintain system reliability, the detector cable drive and storage mechanism were located outside the containment (See page X-4-3 of the Plant Design and Analysis Report and Page II-13-1 of Amendment 2). Each ion chamber and drive cable is housed in and guided by a stainless steel tube that penetrates the drywell and the reactor vessel. The tube is sealed and designed to withstand reactor design pressure and is located within a protective sheath which contains both the tube and the ion chambers. This sealed tube provides the first line of containment. Isolation of the guide tube is obtained with a ball valve outside the drywell.

Even though the ball valves will be opened only when the T.I.P. probe is in operation, which will occur approximately 12 times per year for a total time of approximately five minutes per operation, the remote potential exists for the containment integrity to become impaired in the unlikely sequence of events involving (1) an accident within the drywell, (2) failure of the T.I.P. tube, (3) inability to withdraw the T.I.P. probe, and (4) inability to close the ball valve. Design evaluations are currently underway to provide a redundant means of assuring containment integrity for the T.I.P. system considering malfunctions under postulated accident conditions. Although these design and analyses are not yet completed, they are being undertaken with the intent of satisfying the containment criteria as stated in the Plant Design and Analysis Report, Sections II-1.1 and V-3.1.1.

Alternate design methods are being considered. One alternative is the redesign of the enclosed room in which the T.I.P. system is located in order to make it suitable as a controlled intermediate air lock or zone. Another alternative being considered which may offer a more positive means of leakage control is the addition of shear valves, remotely operated from the control room, capable of sealing the T.I.P. tubes if the drive cable cannot be withdrawn or the ball valves become inoperable.

QUESTION

6. Submit an analysis of the reactivity behavior of the most reactive small portion of the core which behaves like a small reactor within the larger reactor, i.e., treat this as a separate small reactor with the rest of the reactor as a reflector. The manner in which local changes in k are taken into account should be clarified.

ANSWER

The maximum perturbation that can occur in the most reactive local region of the core is illustrated in Figure 6-1, which depicts a clean core geometry in which a centrally located, inserted control rod is surrounded by a minimum critical uncontrolled zone.* The remainder of the core is fully controlled. Movement of the central rod, because of the high statistical weight resulting from this geometry, produces the maximum conceivable local reactivity perturbation in the core.

Local changes in k are taken into account through the analytical treatment of this configuration by either the equivalent bare core model referred to in the question or by direct calculation of the full core. Either method leads to identical results, providing that the "reflector" properties of the controlled core are carefully and properly defined for the equivalent bare core.

Evaluation of the local reactivity effect or worth of the central rod has been made as a function of the moderator density (temperature dependent) between cold and hot-standby. The results are shown in Figure 6-2. Relative control rod worth increases up to hot-standby by approximately 46 percent of the cold value due to changes in the neutron migration properties of the surrounding fuel. Fuel reactivity, on the other hand, decreases at reduced moderator densities. Further the reflecting properties of the controlled fuel become less effective. These latter two effects require a larger region of uncontrolled fuel to reach minimum critical geometries reducing the statistical weight and worth of the central rod. The resulting maximum possible rod worth is shown in Figure 6-2.

Extension of the analysis into the power range introduces the further constraint of thermal limitations on the fuel. Attempts to produce any significant power from a minimum critical geometry would obviously exceed thermal limits and cause fuel damage. Above hot-standby, for the most reactive configuration consistent with thermal limits the maximum rod worths decrease monotonically from the hot-standby value shown to less than about 0.015 Δk at power.

It is recognized that under accident excursion conditions, as described in Section XI - 5.1 of the Plant Design and Analysis Report, rod worths associated with minimum critical geometries have potential for damaging the primary system. Operating procedures require the operator to use highly dispersed symmetrical rod withdrawal patterns in the approach to critical, heating and power ranges of operation. These procedures specifically avoid the arrangement described above. In addition, a Rod Worth Minimizer is provided to back up, by interlock, the operating procedures.

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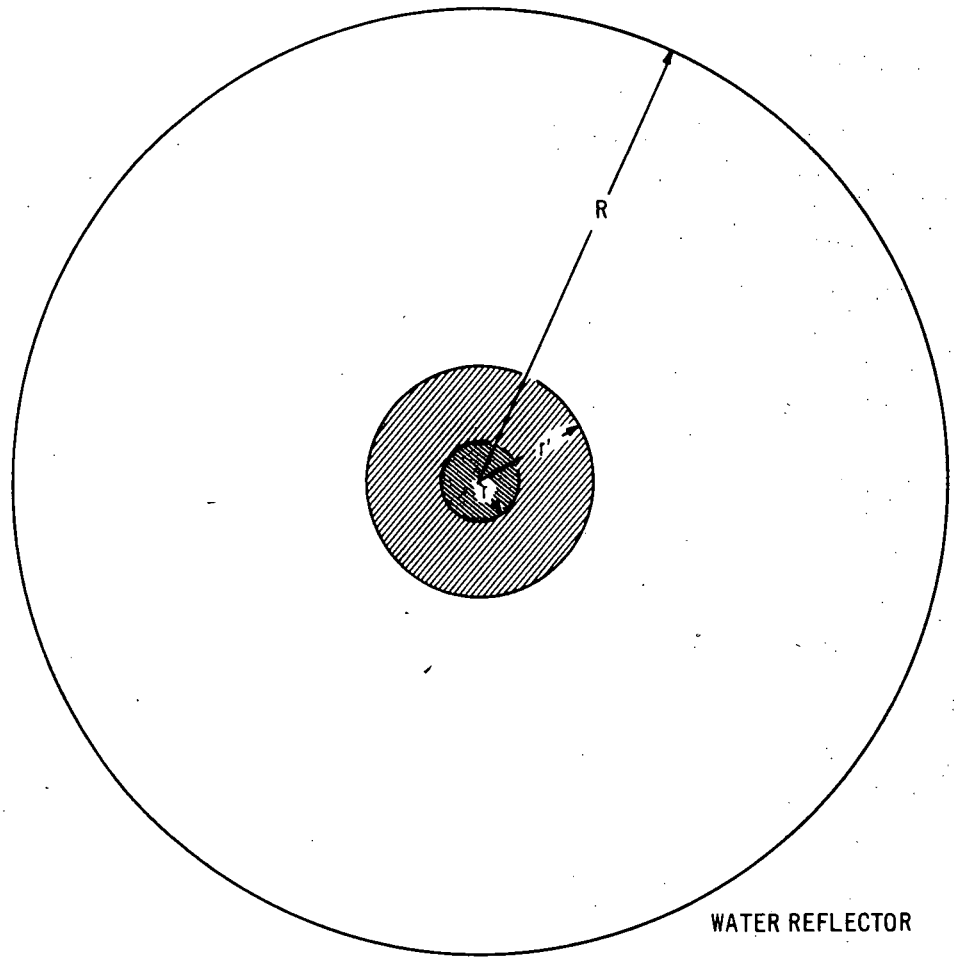
* This figure is illustrative of the configurations studies. The actual two dimensional effects are included in the analysis.



r = EQUIVALENT RADIUS OF ONE CONTROL ROD CELL
r' = EQUIVALENT RADIUS TO OUTER EDGE OF UNCONTROLLED ZONE
R = EQUIVALENT RADIUS OF FULL CORE

Figure 6-1. Core Schematic for Maximum Reactivity Perturbation Analysis

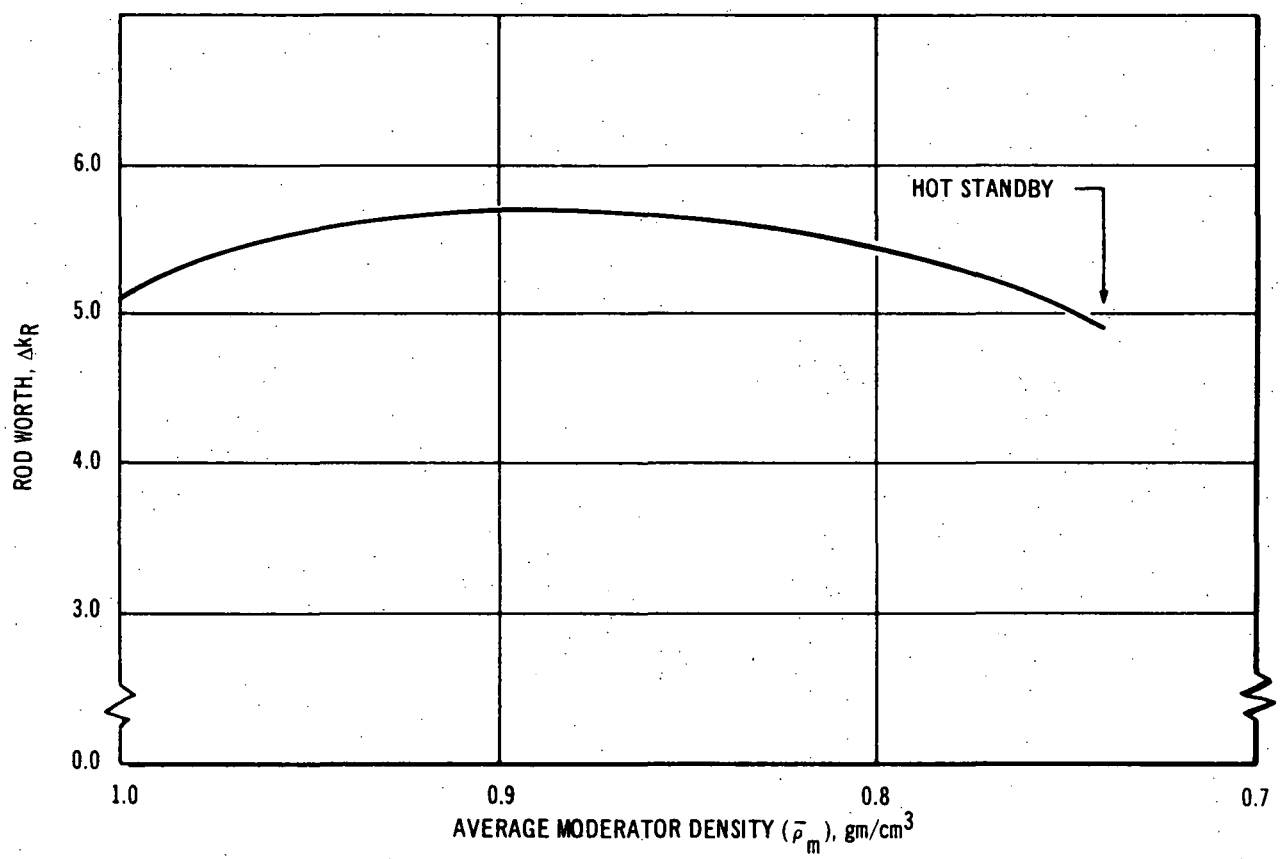


Figure 6-2. Maximum Rod Worth for Various Moderator Conditions

QUESTION

7. State the expected reactivity coefficients of the proposed core as a function of temperature, power, and core life. Include the total Doppler effect which will be available under the various possible operating conditions.

ANSWER

Figures 7-1 and 7-2 show the moderator temperature and void coefficients of reactivity for beginning of life and at 10,000 MWD/T fuel exposure. Because refuelling will be done on a batch basis, utilizing symmetrical loadings which avoid concentrations of the most exposed fuel, the 10,000 MWD/T point is the highest effective exposure the initial core will experience in the Dresden Unit 2 reactor and thus represents a limiting case on both of these coefficients. As shown on Figure 7-1, the temperature coefficient at low temperatures is only slightly positive at 10,000 MWD/T thus satisfying the basis outlined in the Plant Design and Analysis Report. Similarly the void coefficient satisfies the basis in that it remains negative throughout core life.

Figure 7-3 shows the Doppler coefficient as a function of fuel temperature and steam voids for unirradiated fuel. The behavior with exposure is assumed constant for accident analysis although in fact contributions from plutonium, particularly Pu-240, will increase the magnitude of the coefficient by 10 percent to 15 percent at 10,000 MWD/T.

Figure 7-4 illustrates the total Doppler reactivity defect (the negative integrated Doppler reactivity coefficient) existing in the core under normal steady-state operating conditions up to average fuel temperatures of 1000°F. This curve includes the effects on the Doppler reactivity defect of both fuel temperature and steam voids characteristic of normal operation.

Figure 7-5 includes total Doppler availability under abnormal conditions. Doppler defects are shown for adiabatic fuel heating transients starting from cold, hot-standby and rated power fuel temperatures. Fuel temperatures on the abscissa represent effective average fuel temperatures in the core. In Figure 7-6 is shown the integrated Doppler reactivity as a function of time that is actually involved during a hot-standby 0.025 Δk rod drop accident power excursion. Comparison of Figures 7-5 and 7-6 indicates substantially more Doppler is available than is required to terminate the excursion.

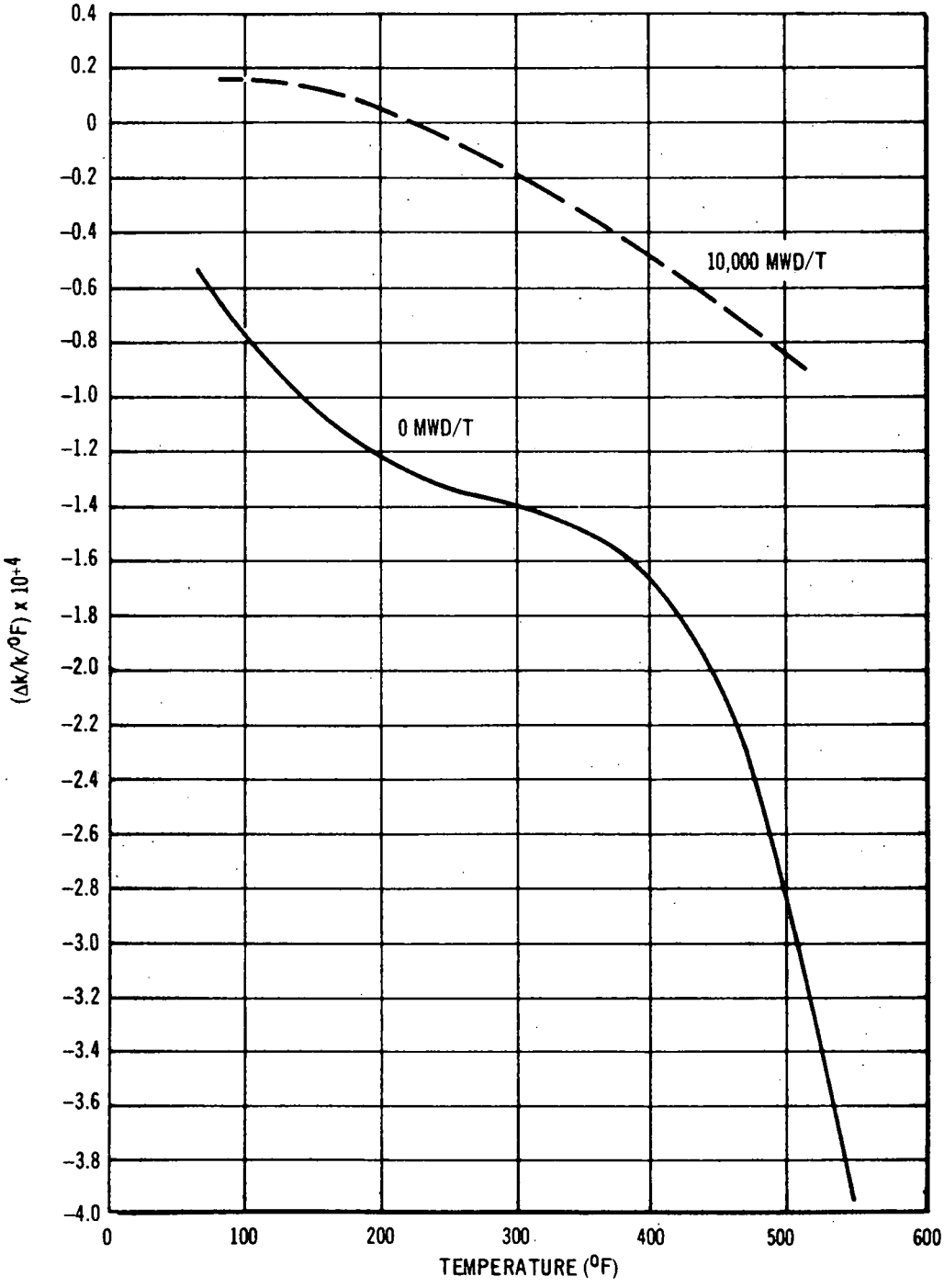


Figure 7-1. Preliminary Temperature Coefficients

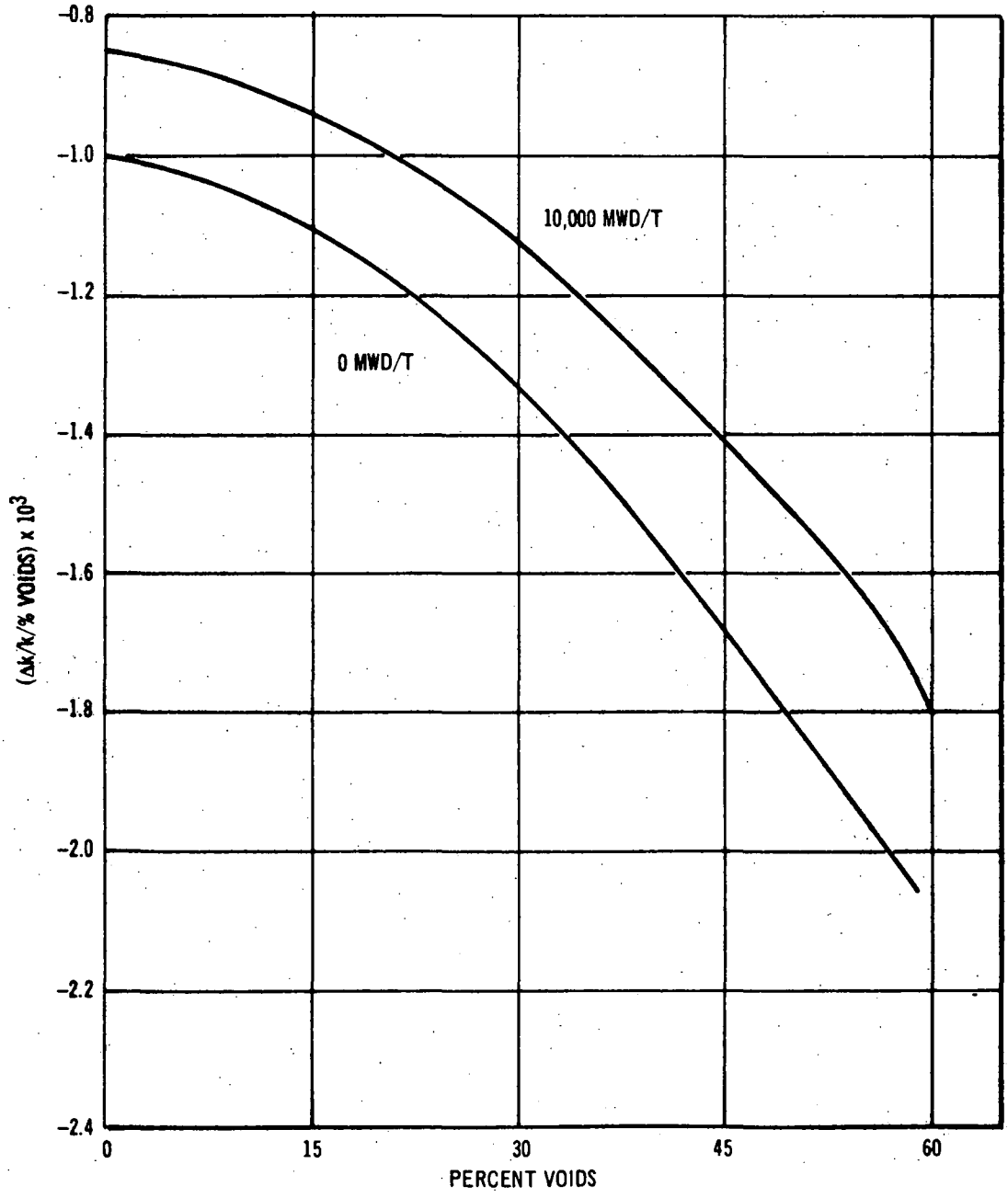


Figure 7-2. Preliminary Void Coefficients

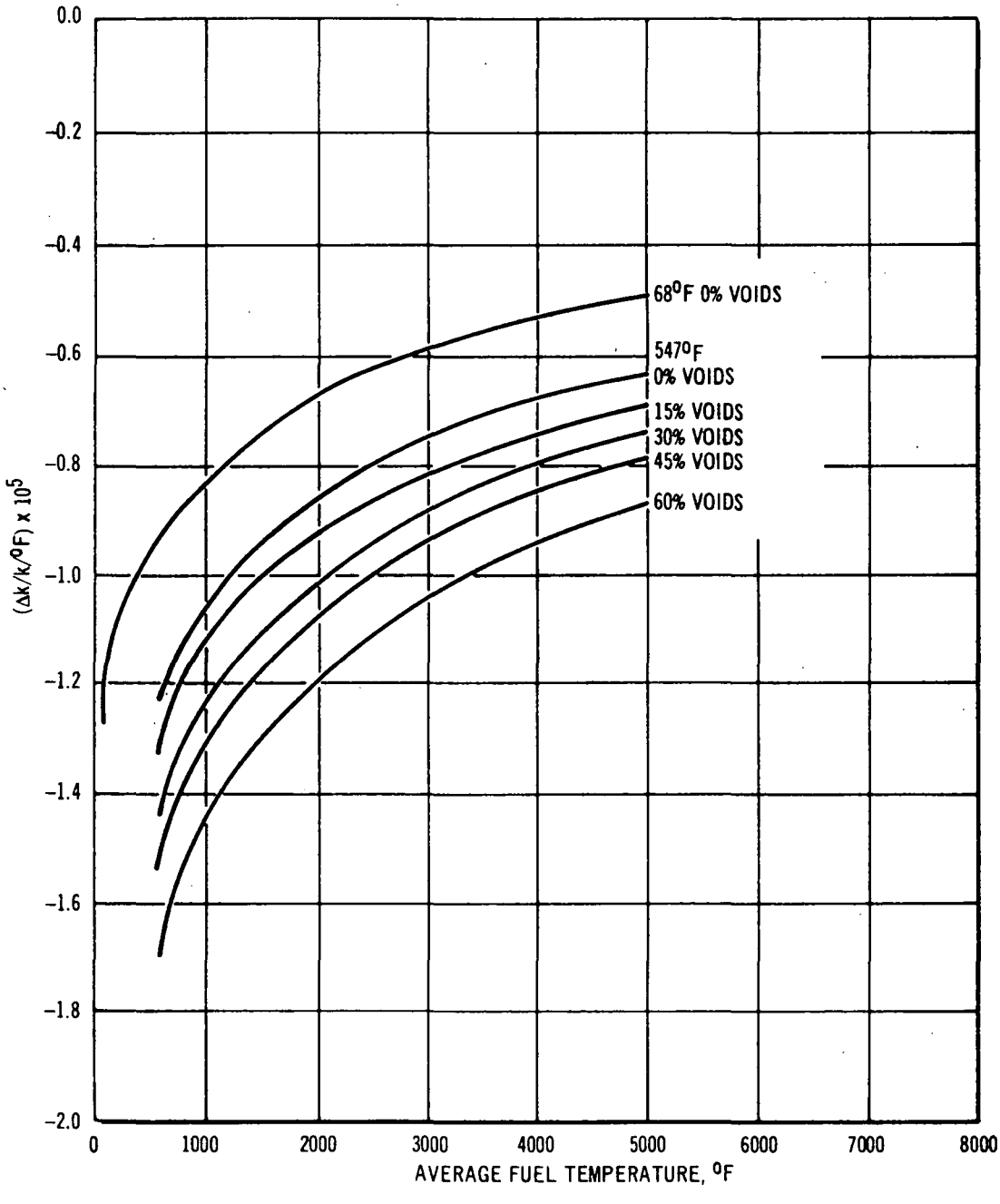


Figure 7-3. Preliminary Doppler Coefficient

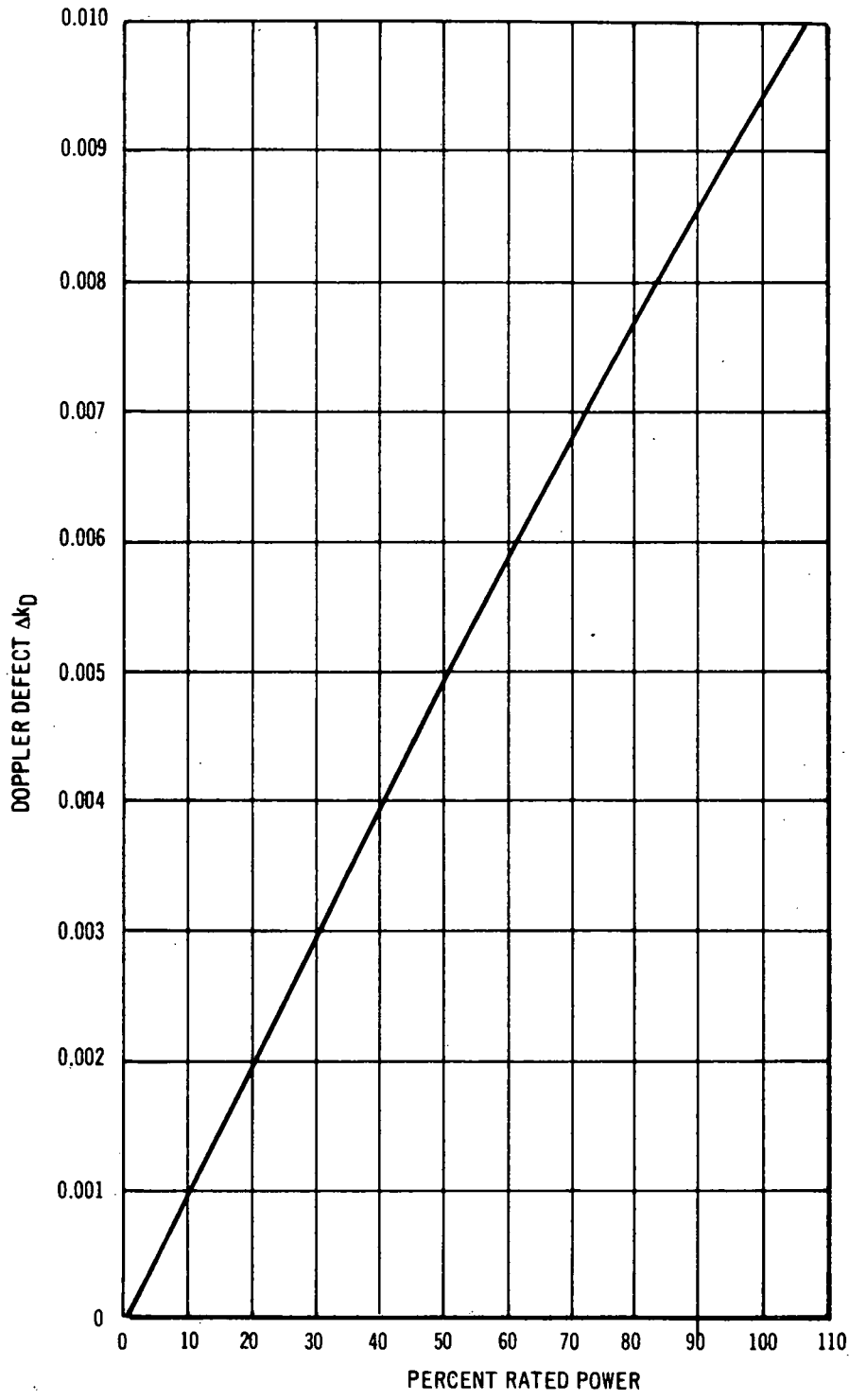


Figure 7-4. Core Average Doppler Defect Versus Core Power Level

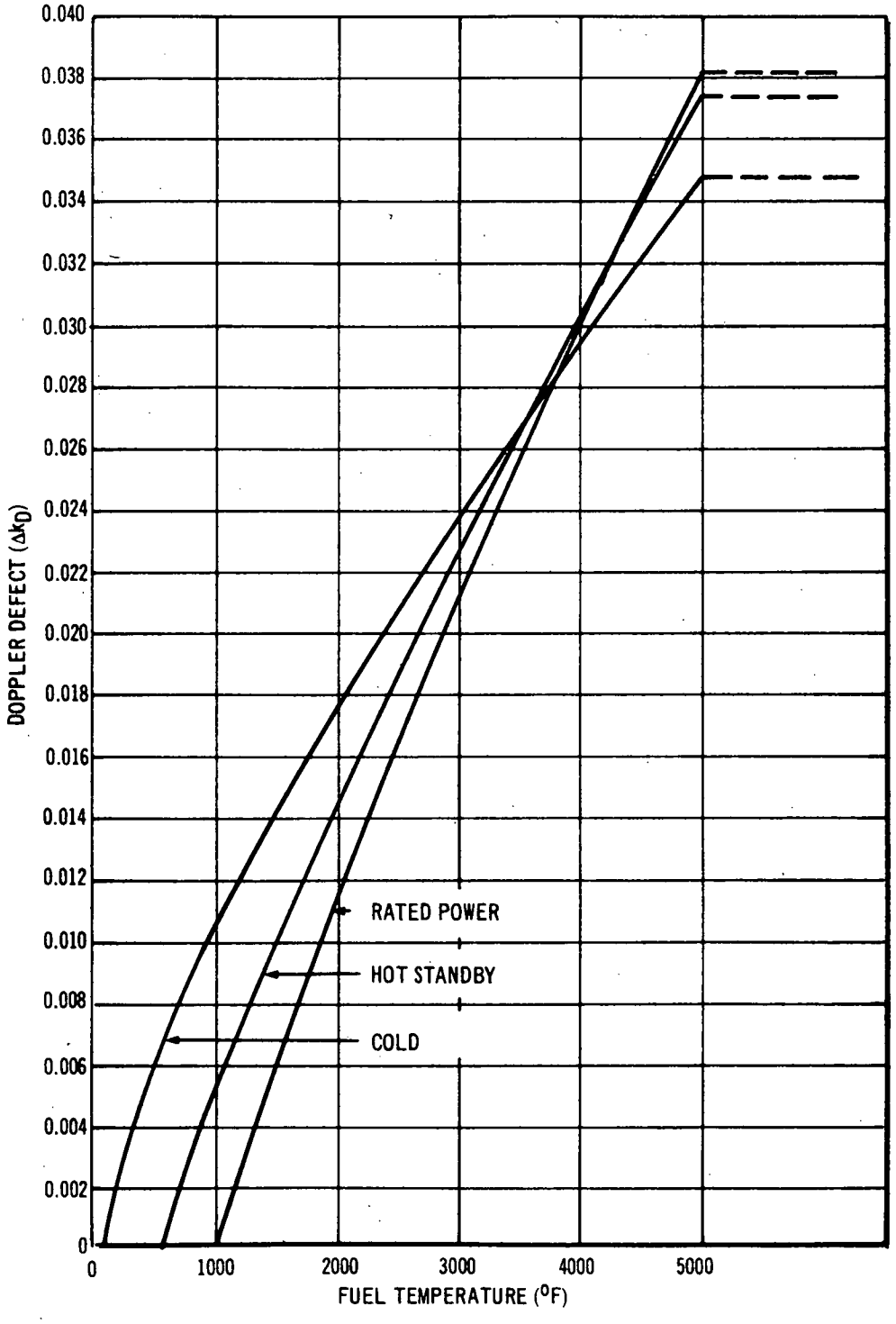


Figure 7-5. Doppler Reactivity

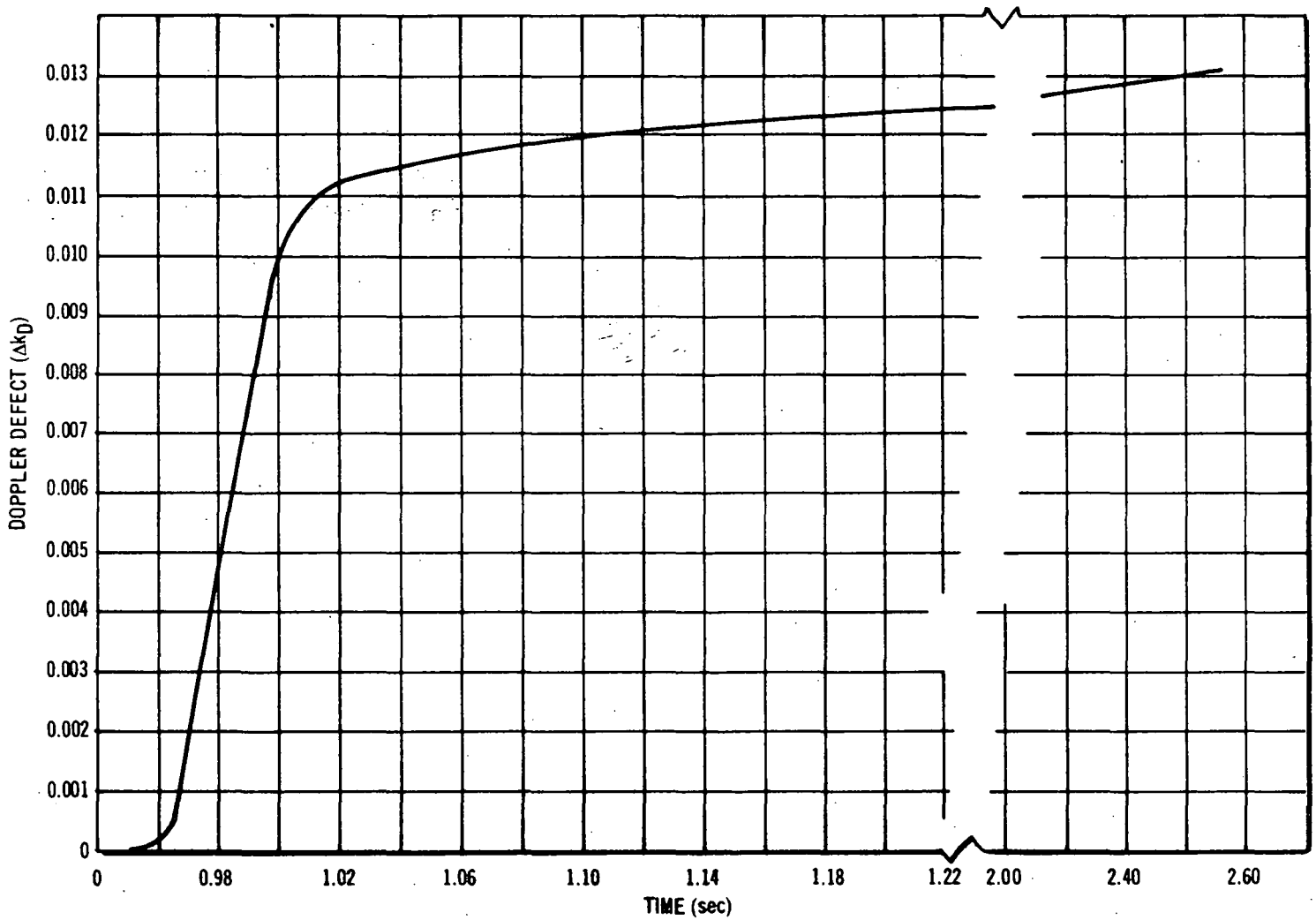


Figure 7-6. Core Average Doppler Reactivity Versus Time after Start of Rod Motion for Rod Drop Accident at Hot Standby

QUESTION

8. How will the piping inside the drywell be protected from whipping around in the event of a pipe break?

ANSWER

All large pipes which penetrate the containment are designed so that they have anchors or limit stops located outside of the containment to limit the movement of the pipe. These stops are designed to withstand the jet forces associated with the clean break of the pipe and thus maintain the integrity of the containment.

The space between the containment vessel and the concrete is controlled so that in areas which are backed up by concrete and are subjected to jet forces the integrity of the containment will not be violated. Where concrete is not available, such as at the vent openings, barriers are put across these openings for jet protection. No additional special devices are provided to prevent piping from whipping within the containment.

The quality control of the fabrication of the pipe and the inspection of the pipe and the conservative design of the pipe is given the same degree of attention as the reactor vessel for all pipe associated with primary water. This approach to prevent pipe failure is substantiated by the long history in the utility industry during which time no such circumferential pipe failures have been recorded for the piping materials to be used for this plant.

If a pipe leak should occur, means for detecting even small leaks are available in the present design so that proper action could be taken before they could develop into an appreciable break.

Therefore, based upon the conservative piping design utilizing proven engineering design practice, the proper choice of piping materials, the use of conservative quality control standards and procedures for piping fabrication and installation, and extensive studies of modes of pipe failure, it is concluded that pipes will not break in such a manner as to bring about movement of the pipes sufficient to damage the primary containment vessel.

QUESTION

9. Provide a design description of the inner vessel within the pressure vessel which provides core reflooding capability. Describe the testing of this inner vessel after installation and during operation.

ANSWER

Those components (refer to Figure 12 of the Plant Design and Analysis Report) within the reactor pressure vessel essential in providing a core reflooding capability are:

1) The shroud
2) Reactor vessel bottom head
3) The jet pump diffuser and throat (refer to Figure 30 of the Plant Design and Analysis Report).
4) The core spray spargers and piping

The shroud, reactor vessel bottom head and jet pump diffuser and throat sections form the vessel within the reactor vessel which may be reflooded. Either of the two spray spargers mounted on the inside of the shroud and the separate piping manifolds which deliver water to the spray spargers constitute the means for introducing water inside this vessel for reflooding.

The lower portion of the shroud is furnished integral with the reactor vessel and has a diameter of approximately 196 inches. This portion of the shroud consists of the bottom course of the cylindrical shroud with the bottom end at approximately the 9 foot vessel elevation, a horizontal baffle plate between the shroud bottom course and the top of the vessel bottom head at approximately the 10 foot vessel elevation, and several vertical support legs. The support legs are approximately 4-1/2 feet in length and extend from the bottom shroud course to the vessel bottom head. The baffle plate and legs are full penetration welded to both the reactor pressure vessel bottom head and the bottom course of the shroud.

The lower portion of the shroud is designed to accommodate the differential expansion of the ferritic reactor vessel and the austenitic stainless steel upper portion of the shroud and the jet pump diffuser. This lower part of the shroud, including the baffle plate and the support legs is fabricated either of carbon steel clad on both sides with stainless steel or of solid Inconel, probably the latter. The support legs carry almost all of the vertical loads carried through the shroud including the weight of the shroud and the other structural components which mount on it, the pressure loading on the shroud both under normal operating conditions and accident conditions, and the vertical loads developed to resist the vertical component of earthquake induced loads and the overturning moment of the horizontal component of earthquake induced loads.

The upper portion of the shroud is cylindrical and is joined to the bottom course of the shroud with a full penetration weld. All of these elements are fabricated of austenitic stainless steel. The principal stresses produced in the shroud are due to the pressure loading and the differential expansion of the shroud upper part and the shroud lower part. Loading due to weight of components supported on the shroud and earthquake loading are also taken into account.

The jet pump diffuser is a gradual conical section terminating in a straight cylindrical section at the lower end which is welded into the baffle plate. The throat section is a tube with integral contoured inlet section and the upper part of the conical diffuser. The joint between the throat and the diffuser is a slip fit at approximately the 18-foot elevation of the vessel. The upper end of the diffuser section and the throat are braced from the reactor vessel through the riser pipe between each pair of jet pumps. The throat is held down in the slip fit socket in the top of the diffuser by a bolted connection at the top of the riser pipe.

The design of the jet pump parts take into account the pressure loading both in normal and accident conditions and the reactions at the supporting brackets due to differential thermal expansion of the pump and reactor vessel.

The design criteria for these and other parts of the core structure components are as given in Amendment No. 2, to the Plant Design and Analysis Report, pp I-1-3 and 4.

The shroud, vessel bottom head and jet pump diffusers and throats form a vessel within the reactor vessel having a volume of approximately 6000 cu ft with 4000 cu ft below the core which can be reflooded to the height of the top of the jet pump throats, which are located at approximately the 26-foot elevation of the vessel. Water admitted through either of the two core spray spargers will fill this vessel to the point of overflowing the jet pump throats. The top of the jet pump throats will be at approximately two-thirds of the core height (about one-third of the active fuel zone will be above the tops of the jet pump throats). A finite leakage will occur through the slip fit of the jet pump throat in the top of the diffuser. The additional core spray flow will flow over the tops of the jet pumps.

During plant pre-operational testing the core spray system is operated. This test will demonstrate the ability of the system to fill the shroud up to the height of the top of the jet pumps. Timing the rate at which the level drops after the core spray flow is shut off will verify the leakage through the slip joint between the jet pump throat and diffuser. Subsequent to the pre-operational tests it is anticipated that no special testing of this feature will be performed. Analyses indicate that any potential leakage past the slip fit joint of the jet pump diffuser section would be a very small fraction of the total capacity of the core spray system, and that such leakage would have no effect on reflooding capability.

It is estimated that the pressure differentials across the reactor internals essential to core reflooding are not substantially different than those which will exist during normal operation. Therefore, the pre-operational test of the recirculation system will virtually test the components to their design loading.

QUESTION

10. Provide a description of the control rod velocity limiter in sufficient detail to show the location of this component in an individual control rod drive system and in a control rod guide tube. State the effect of the limiter on control rod scram time and the reason for selecting a drop velocity of 5 feet per second.

ANSWER

As noted in Section III-1.4 of the Plant Design and Analysis Report the rod velocity limiter is being developed. Further information on the rod velocity limiter will be presented as the data accrue from the test program. A short description of the device and test program is summarized below.

The rod velocity limiter is an integral part of the bottom of the control rod. The rod velocity limiter is basically a loose fitting piston which travels in the control rod guide tube over the entire control rod stroke as shown in Figure 10-1. As this piston moves along the guide tube, the water which fills the tube must be displaced from one side of the piston to the other. The rod velocity limiter is shaped to provide a streamlined profile in the scram (upward) direction and a non-streamlined profile in the drop-out (downward) direction. In addition, added resistance is obtained in the drop-out direction by directing water flow from the center of the limiter to the outside annulus. The pressure differential which is developed across the piston results in a force which retards the piston motion. No moving parts are required to create the retarding forces or change the retarding forces developed for insertion and withdrawal. The high pressure differential results in a low terminal velocity in the event of a rod drop-out, yet the retarding force during scram insertion of the rod is low enough so that adequate scram velocities are achieved. The rod velocity limiter is arranged in the reactor vessel as indicated in Figure 12 of the Plant Design and Analysis Report. Note that it always remains in the guide tube except when the control rod is removed. The fuel support must be removed before the control rod can be removed because of the shape of the velocity limiter.

Various tests have been performed to evaluate the effect of the rod velocity limiter on control rod scram time. These tests include drop tests of full-sized rod velocity limiter models in the scram direction, and full-scale scram tests of control rod drive systems using prototype rod velocity limiters which approximate the drag characteristics of the final limiter design. Further scram tests will be performed using the final design velocity limiter. Based on testing thus far, the limiter will cause an increase in scram time of approximately 0.2 second (measured from start-of-motion to 90 percent of stroke, at reactor pressures of zero and 1000 psig).

Design specification for scram time at these conditions is 3.4 seconds, and actual times range from 1.6 to 2.6 seconds. Therefore, the increase in time due to the rod velocity limiter can be accommodated within the margin between actual and required scram times.

The rod velocity limiter is designed to limit the free fall control rod drop velocity (5 feet per second) sufficiently to adequately limit the consequences of the drop of a maximum worth control rod but not hinder control rod insertion. Section XI-5.1 of the Plant Design and Analysis Report shows that the combination of a 5 foot per second limiter with the maximum potential rod worth allowed by the Rod Worth Minimizer yields control rod drop accident consequences that have a sufficient safety margin below the estimated threshold of potential damage to the primary system. A 5 foot per second rod velocity limits the potential consequences of dropping the maximum worth control rod allowed by the rod worth minimizer to:

a. no vaporized fuel

b. no fully molten fuel

c. minimal cladding damage

Adequate clearances have been left to prevent an interference of rod insertion by the rod velocity limiter and to prevent excessive forces on the control rod drive mechanism.

The (~0.75 inch radial) spacing between the rod velocity limiter and the guide tube provides adequate margin against binding or interference and does not appreciably increase the control rod scram time. A lower velocity would require smaller clearances and would inpose additional column loads on the index tube during scram. Therefore, the 5 foot per second rod velocity limiter is the best compromise between added safety during an improbable accident and minimum effects on normal operation.

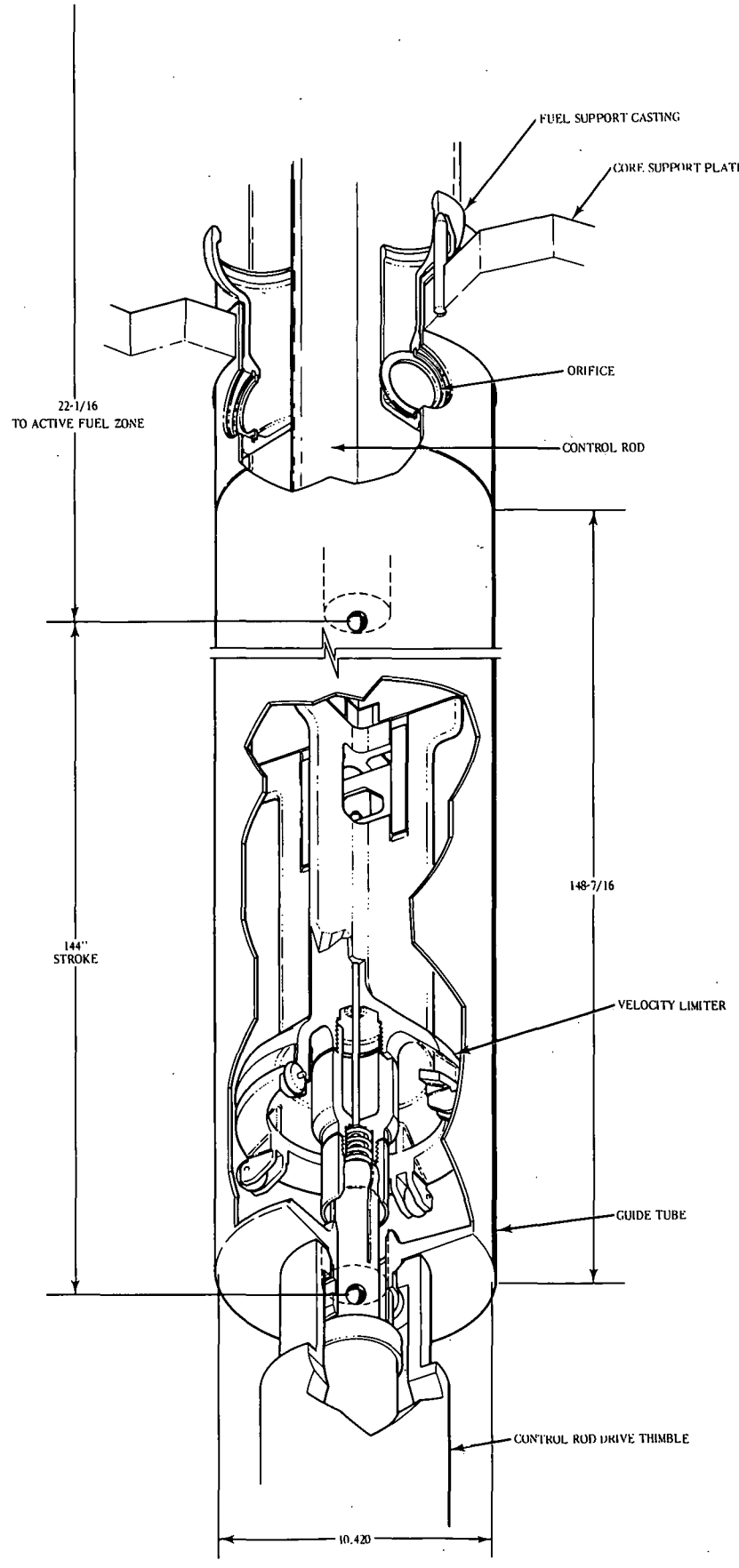


Figure 10-1. Control Rod Velocity Limiter

QUESTION

11. Provide the design basis for the control rod thimble support and discuss why the proposed design will be adequate to satisfy this basis.

ANSWER

Design Basis

A detailed discussion of the control rod drive thimble design, and the margins provided to preclude failure of the thimbles is contained in Appendix B.3 of Volume I, Plant Design and Analysis Report. The design basis for the thimble support system is based on the assumption that, despite the conservatism employed in the thimble design, any one drive thimble could experience a complete instantaneous circumferential failure with the reactor vessel at design pressure of 1250 psig. For the dynamic loading associated with this unlikely failure, the support system is designed to achieve the following objectives:

1. Limit the total downward travel of any control rod and its drive to approximately 3 inches and prevent the drive and thimble from falling away from the reactor vessel.

2. Provide clearance between the thimble support plates and the thimble to prevent contact due to thermal expansion which will occur during normal operation.

3. Provide a system which is easily removable in sections to permit access to the control rod drive mechanisms, position indicators, and in-core thimbles for maintenance and inspection.

4. Provide support for the control rod drive in the unlikely event of instantaneous failure of the flange bolts on any one drive. (This is a less severe accident condition with respect to the total force on the support system. In this case the force is a function of reactor pressure acting on the inside rather than the outside area of the thimble.)

5. Provide instrumentation to assure that the supporting system is in place prior to startup of the reactor.

Description of Support System

The support system consists of structural members, rods, support plates and disc springs (see Figure 56 of the Plant Design and Analysis Report). The structural members are placed between the rows of thimbles and are located in the spaces between the thimbles below the bottom head of the reactor vessel. These members are supported on the reinforced concrete pedestal which supports the reactor vessel.

The support plates are located under the drive flanges. These plates are attached to the rods which are supported from the structural beams. A stack of disc springs is provided on each rod. The support system therefore is an elastic structure which is capable of absorbing the energy resulting from the assumed failure. This system also limits the magnitude of the resulting dynamics forces on the supports.

When the supporting system is installed a gap of about 1 inch will be provided between the lower support plates and the contact surface on the control rod drive flanges. During system heatup this gap will be reduced due to a net downward expansion of the thimbles with respect to the support plates. In the hot, operating condition, the gap will be approximately 1/4 inch.

The downward travel of the thimble following the assumed thimble failure will be the sum of the initial gap, plus the elastic deflection of the supporting structure under dynamic loading. The support system to be provided will limit the total downward movement of the drive and thimble to 3 inches under the worst case, i.e., with a gap of approximately 1 inch. The total deflection would normally be substantially less than 3 inches because an operating gap of 1/4 inch exists between the lower support plates and the contact surface on the control drive flange. Thus the drive movement following a thimble failure, will always be less than one normal drive "notch" position.

The equivalent static force considered in the design of the supporting system is equal to the gross cross-sectional area of the 6-inch-o.d. drive thimble times the design pressure of the reactor vessel multiplied by the impact factor resulting from the acceleration of the thimble through a distance equal to the one-inch gap. The total design load (pressure × area × impact factor) is, therefore, a function of the elasticity of the supporting structure.

A number of different arrangements regarding the size of the rods, the number of springs and the stiffness of the support beams and plates are being investigated to determine the optimum design for this system. The stresses in the various components of the supporting structure will be limited to 90 percent or less of the yield strength of the materials. The stress criteria was selected to provide a system that is as elastic as possible and which can be considered adequate for a "one-time" loading condition.

The design of the instrumentation for the support system has not been completed and will require further study. The intent is to provide a system which will remotely indicate that the supporting system is in place.

QUESTION

12. Provide a parametric study of the amount of fission products which could leak from the reactor building by exfiltration and yet net exceed 10 CFR 100 guideline exposures. Perform this study for the 100 percent core meltdown case.

ANSWER

The standby gas treatment system is designed to provide a controlled discharge to the 300-foot stack at a rate equivalent to 100 percent of the reactor building volume per day by maintaining the building at a negative pressure of 0.25 inch of water with respect to the outside atmosphere. High winds will create pressure gradients around the building which can cause outleakage or exfiltration from the building.

An analysis of potential exfiltrations from the reactor building is described in Question II-8, Amendment 2 to the Dresden 2 Plant Design and Analysis Report. Such analysis is based upon a model developed from wind tunnel and actual field experiments. The model was developed for a rectangular building, such as the Dresden 2 reactor building, and is based upon a uniform leakage through all walls, varying as a function of ΔP. The pressure profiles from the tests indicate that the maximum external negative pressures occur near the center of the wall surfaces, with smaller negative pressures near the edges of the building. The most probable locations for leakage are the roof-to-the-wall joints, wall-to-wall joints, and doors. These locations are near the edges of the wall surfaces and the negative pressures at these locations are below the wall average, so that exfiltration from the reactor building should be less than that calculated by the model.

The analyses show that exfiltration will begin to occur at wind speeds in the range of 35 to 65 miles per hour and will increase at higher wind speeds. If a wind high enough to cause exfiltration from the reactor building occurs during a loss of coolant accident with 100% core meltdown, the doses at the nearest site boundary would be negligible.

An evaluation of the exfiltration rates greatly in excess of the referenced calculation has been performed. This evaluation shows that with a 35-mile per hour wind (the minimum wind speed at which exfiltration would occur) the exfiltration rate could be increased to infinity and the doses at the site boundary would not reach 300 rems in 2 hours. This calculation was for the maximum reactor building airborne halogen radioactivity following the postulated 100 percent core melt and includes continued leakage from the primary containment during the exfiltration of the reactor building. With a wind speed above 20 to 22 miles per hour (far below the wind speed at which any exfiltration will occur), even if all the reactor building airborne radioactivity were released in 2 hours, the maximum thyroid doses at the site boundary would not exceed 300 rems. Figure 12-1 shows the exfiltration rates which will give 300 rem to the thyroid at the site boundary as a function of wind speed with no wind direction diversity. The figure also shows the expected actual exfiltration rate. The calculations indicate that a substantial margin exists between the expected exfiltration rates and those which could result in excessive doses.

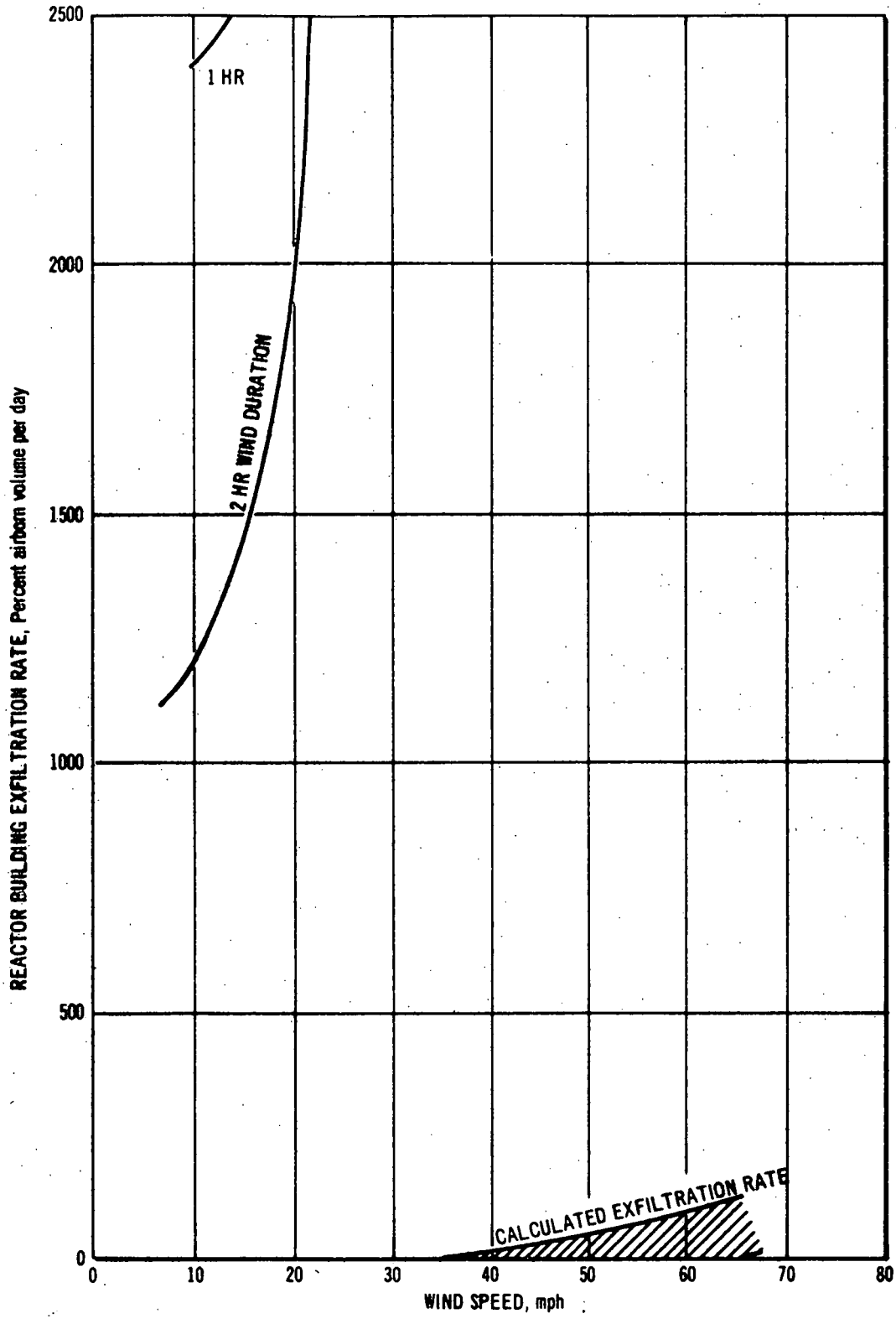


Figure 12-1. Reactor Building Exfiltration Rate Giving 300 rem Thyroid Dose Following a 100 Percent Core Melt

QUESTION

13. Provide an analysis of the amount of radioactive material which, during an accident, might leak to the reactor building via leakage from the suppression pool cooling equipment. Discuss the relative importance of this leakage in the determination of off-site consequences for the 100 percent core meltdown case.

ANSWER

The containment and core spray systems are tight systems and are designed to have little liquid leakage. The pumps are operated only for tests (so the seals should be tight) the internal pressure is relatively low and the coolant temperature is low, so the leakage rate will be quite low. Nevertheless, assuming a maximum average leakage rate of 1 gallon per hour (a conservative maximum based on pump operating experience) for each of the 10 pumps operating, the leakage of water from the containment would be 10 gallons per hour. All this equipment is located in the pressure suppression chamber room below ground level, with sumps drained to the radioactive waste disposal system, and the only openings to the rest of the reactor building are two stairwells. The water leaked to this chamber would be pumped to the waste disposal system with little holdup in the suppression chamber room. The only halogens present in the containment cooling water are in solution and would largely remain in solution in the cool water. The pressure suppression chamber room, below ground with low inleakage, would have a very low air exchange rate, so that, with the large halogen partition factor, only a small fraction of the halogens in solution would become airborne.

Nevertheless, conservatively assuming that 10 percent of the halogens contained in the leaked water become airborne and are drawn off by the reactor building standby gas treatment system, the release of halogens from this source would be 50 times less than the leakage of airborne halogens from the primary containment air space (0.5 percent per day leakage rate). Therefore, the halogens leaving the primary containment in leaked water are negligible compared to the airborne halogens leaking from the primary containment.

QUESTION

14. Provide an analysis of the protection system which demonstrates that, under all conditions of allowed bypassing, no single failure of the protection system will remove the automatic protection capability against any accident which could result in fuel damage. Possible single short circuits at the rod drives themselves should be included in the analysis in terms of causing a withdrawal and simultaneously bypassing the rod-block interlock.

ANSWER

Bypassing of individual protective system signals is only allowed in those conditions where adequate protection is afforded by the reactor's condition and by other signals.

Bypassing is allowed on the reactor protection system to avoid scram signals from those inputs which are not normal during the start-up phase of operation. These include a "Dump Volume High Water Level" signal, a "Low Condenser Vacuum" signal, and a "Main Steam Line Isolation Valve Closure" signal. In order to close the scram dump valve, terminating flow into the scram dump tank so the tank can be drained, the scram dump tank high water level scram can be bypassed. But an interlock prevents rod withdrawal on high scram dump tank water level. These bypasses are removed as soon as start-up proceeds to the point where these inputs become normal for an operating reactor. Removal of the bypass is forced by the operating "Mode" switch and other interlocks making it impossible to operate at a significant power level without protection. No allowable bypass in combination with a single equipment malfunction or operator error can lead to an accident causing fuel damages.

Bypassing is allowed on the Intermediate Range Monitors. In this case, there is an excess of monitors over that required to satisfy the basic input requirements of a Dual Bus Protection System. There are eight IRM's and only four are needed. Consequently, the operator is allowed to bypass one on each of the two protection buses provided the two are not in the same quadrant. The bypassing is interlocked so that the above considerations must be met. After the Average Power Range Monitors are on scale and the reactor is at substantial power, the IRM's no longer serve a safety function that is not better met by the APRM's. Because of this the IRM's are bypassed in the "Run" mode so long as the APRM's located in the same quadrants are reading upscale. Assuming that two IRM's are bypassed and another one malfunctions, the reactor is still protected by five IRM's. There is no likelihood of an accident leading to fuel damage at this low power level.

Bypassing is allowed on the APRM's. As with the IRM's, there are eight when only four are needed to satisfy the needs of the Protection System. The operator is allowed to bypass one on each of the two protection buses provided the two are not in the same quadrant. The bypassing is interlocked so that the above considerations must be met. Assuming that two APRM's are bypassed and another one malfunctions, the reactor is still adequately protected by five channels. Under the worst combination possible within the above assumptions, there are still six valid pairs of scram signals from the eight channels to protect the reactor against gross power level damage.

Local fuel damage requires special considerations. If a single channel is bypassed the other APRM in that quadrant would sense a high neutron flux and block rod withdrawal at neutron flux levels well within design. Additionally procedural controls dictate rod withdrawals which keep operating conditions well within design. Thus, the criteria that there be no fuel damage due to an allowable bypassing in combination with an equipment malfunction or an operator error has been met.

The effect of a short circuit on a control rod drive withdrawal circuit has been analyzed. The withdrawal sequence requires that the rod be jogged in before it can be withdrawn. To accomplish a withdrawal would require two carefully selected short circuit paths, and the first short must be removed before the application of the second short, the time interval being within fairly close limits. Unless all these conditions are met, the rod would not move out continuously. It is possible under the right combination of circumstances to move one rod one notch with one short circuit. No plausible malfunction is foreseen that could logically lead to fuel cladding damage.

QUESTION

15. Discuss the manner in which Figure III-2-1 in Amendment 2 can be interpreted to include the case of a control rod drop at rated power. Discuss the effects of control rod worth, operating temperature and the Doppler coefficient at the higher temperature on this curve. Include an interpretation of the meaning of peak enthalpy.

ANSWER

Figure III-2-1 of Amendment 2 of the Plant Design and Analysis Report is intended to show the variation in the peak fuel enthalpy (heat content) which exists in the core following a 0.025 Δk rod drop accident as a function of the initial moderator density. The moderator density is chosen as a convenient variable to indicate the state of the reactor.

The peak enthalpy which exists at the end of the excursion before any significant heat transfer from the fuel to moderator has taken place consists of the sum of two components, the enthalpy that was added during the excursion and the enthalpy that was present prior to the excursion. The reference enthalpy is 0 cal/gm at 20°C.

In these analyses the assumption is made that prior to the excursion the heat flux is in equilibrium with the neutron flux. For this not to be the case, the reactor would have had to be operating on a very short period; shorter than the characteristic period for heat transfer from the fuel rods, which is of the order of several seconds.

At hot standby the power level is very low (assumed to be 10^{-6} of rated) and the fuel temperature is essentially in equilibrium with the moderator (286°C). Initial fuel enthalpy is about 20 cal/gm uniformly distributed throughout the core and across each fuel rod. The postulated excursion adds a maximum of about 180 cal/gm. Peak fuel enthalpy in the hottest fuel segment is, therefore, about 200 cal/gm, including the effects of gross axial, radial and local (interbundle) power peaking factors characteristic of the excursion geometry. The enthalpy distribution across the fuel rods is initially flat and remains flat throughout the excursion (adiabatic approximation).

For accidents occuring at rated power it is assumed that the hottest fuel rod in the region which surrounds the postulated dropped control rod is initially running at the core average power level. This is a conservative assumption since the fuel in which the maximum enthalpy is deposited during the excursion is initially operating significantly below rated power due to the presence of the inserted control rod. At the core average heat flux, the volumetric average fuel temperature is about 530°C, which corresponds to a fuel enthalpy of about 35 cal/gm. At the core average heat flux, the peak fuel temperature at the center of the fuel rod is about 670°C or about 45 cal/gm. The nuclear excursion adds about 50 cal/gm to the initial peak fuel enthalpy resulting in a final peak fuel enthalpy of 95 cal/gm. Again, this represents the highest fuel enthalpy in the core since the excursion

energy increment includes the axial, radial and local power distributions and the initial fuel enthalpy represents the centerline enthalpy of the peak single rod in the excursion zone.

A comparison of an accident at hot standby with one at rated power shows that while the latter case has a higher initial fuel enthalpy, the enthalpy added during the excursion is much smaller, 50 cal/gm rather than 180 cal/gm.

The smaller energy increment added at power is due primarily to two factors. These factors are (1) the higher initial power which makes the Doppler coefficient more effective through immediate sensible heating of the fuel and (2) the flatter or less peaked power distribution which exists at the lower moderator densities characteristic of power operation. The magnitude of the Doppler coefficient in the two cases is essentially the same since the reduction of Doppler coefficient at power due to higher fuel temperature is offset by the inverse dependence of the coefficient on moderator density (see Figure 7-3).

The maximum rod worth of 0.025 Δk was used for the entire range of reactor states shown in Figure III-2-1 of Amendment 2. The constraints of thermal limits and reduced number of rods in the core at operating power, however, will preclude rod worths of this magnitude existing in the core at power. Therefore, the maximum peak fuel enthalpy at power, resulting from the drop of a rod of less worth, will be less than that shown in Figure III-2-1. The trends of peak enthalpy as a function of rod worths below the 0.025 Δk maximum may be seen from Figure 62 in the Dresden Unit 2 Plant Design and Analysis Report, Volume 2.

QUESTION

16. Provide information that demonstrates that the T. I. P. system containment isolation system will be designed such that, under conditions of 100 percent core meltdown, no single control malfunction within the isolation system (failed valve, or motor, or line, etc.) will result in excessive increase in fission product release. Simultaneous rupture at the core of the maximum number of potential in-service tubes should be assumed. Credit taken for operations of the T. I. P. system drives should be for manual, remote operation only. Credit for operation of the valves should be for automatic or remote operation only.

ANSWER

See Answer to Question No. 5.

QUESTION

17. Provide the Dresden 1 data which leads to the use of the 10^{-5} concentration ratio of halogens in steam to water in the steam line break accident.

ANSWER

INTRODUCTION

Data obtained at Dresden Unit 1, a typical BWR, for the transport of radioiodines, indicate that a decontamination factor, DF, of 3×10^4 is obtained between the reactor moderator water and condensate and that an additional decontamination factor of 2×10^2 exists between the condensate and off-gas. The data from which these conclusions were reached are summarized in the subsequent sections of this answer.

BRIEF DESCRIPTION OF DRESDEN 1

The net generating capacity of Dresden Nuclear Power Station Unit 1 is 210,000 KW. The dual cycle boiling water reactor steam generating system produces steam at two pressures for the double admission turbine. Primary steam is generated in the reactor vessel as the recirculating water coolant passes through the fuel assemblies in the core from the bottom entrance to the steam dome turning vane above. The steam-water mixture passes through the risers to enter the primary drum where the mixture is separated. The steam leaves the drum at 990 psia and enters the turbine pressure regulating control valves at 965 psia (rated load). These valves automatically adjust the flow to maintain the pressure constant at the primary drum.

The recirculating water, approximately seventeen times the quantity of the primary steam at rated load, flows from the primary drum through the downcomers to the suction of the four reactor recirculating pumps to be pumped through the associated secondary steam generators where a portion of its heat is removed to produce secondary steam. The recirculating water continues on to return to the bottom of the reactor. The secondary steam enters the secondary load control valves at the turbine. The pressure of this steam is inverse to the turbine load. It approaches primary drum pressure at zero load and decreases to a design pressure of 475 psia at full load.

The primary and secondary steam from the turbine passes to the main steam condenser where it is condensed and deaerated and returned to the reactor via the primary and secondary condensate feed pumps. Gases are removed from the condenser through a slotted 16-inch pipe extending the full length of the condenser and located at the top of each of the two tube banks above the air cooling sections. The two 16-inch pipes join in a 20-inch pipe leading to the two steam air ejectors and mechanical vacuum pumps.

The two steam air ejectors, with separate inter- and after-condensers, remove the non-condensable gases from the condenser during operation and discharge the gases into a 118-foot-long, 30-inch pipe. The size of the pipe normally provides a 20-minute holdup to allow decay of the radioactive gases carried with the steam from the reactor to the condenser. The off-gas then passes through three absolute particulate filters located at the end of the off-gas holdup pipe and is discharged to the 300-foot stack.

ENVIRONMENTAL MEASUREMENTS AND OBSERVATIONS

Carryover of Non-Volatile Species in Primary Steam System

The measurement of steam decontamination factors for two typical non-volatile radionuclides, Na-24 and Cu-64, were determined during plant operations in 1960. The steam decontamination factors were obtained by comparing the concentration of Na-24 and Cu-64 activities in filtered reactor water with the concentrations observed in the primary steam. The results are listed in Table 17-1.

TABLE 17-1

PRIMARY STEAM DECONTAMINATION FACTORS

Reactor Power MWt	Primary Steam Flow lbs/h	Decontamination Factor*	
		Na-24	Cu-64
300	1.15×10^6	$>7.7 \times 10^3$	-
630	1.4×10^6	$>1 \times 10^4$	1.3×10^5

* Decontamination Factor is defined as the concentration in the reactor water divided by the concentration in the unfiltered steam.

The practical limits of this type of measurement are dependent on the initial concentration of the active species in the reactor water.

Na-24 and Cu-64 were the only non-volatile species that meet this requirement. If one considers these two elements representative of all non-volatile species the steam decontamination factor ranges between 1×10^4 and 1.3×10^5 at full power. The value for Cu-64 may be more reliable than the value based on the Na-24 because of the higher activity levels found in the reactor water and the relative radiochemical separation procedures. However, copper activity could be plating out in the sample lines, etc. Thus, to be conservative, the Na-24 decontamination factor of 1×10^4 will be used as a basis of comparison with results from the iodine carryover tests.

Carryover of Radioactive Iodines in Primary Steam System

Although no specific program was established for investigating the iodine decontamination factor between the reactor water and condensate, several independent measurements have been made which indicate the magnitude of this parameter. The results of these measurements are summarized in

Table 17-2, together with approximate plant operating conditions. Some of the earlier DF data are greater due to the extremely low iodine concentrations in the steam or hotwell. As shown in Table 17-2, the decontamination factor for iodine between the reactor water and condensate is approximately 3×10^4 compared to a value of 1×10^4 for a "typical" non-volatile species, Na-24. The higher iodine DF is not considered significant since the sodium data is a single measurement. Thus, iodine behaves in a manner similar to sodium, a non-volatile species, with respect to carryover in the steam from the reactor water.

Carryover of Radioiodines from the Hotwell to the Off-Gas

Although it was plausible that a significant reduction of iodine would occur in the Dresden Unit 1 hotwell and the air ejector due to gas washing, accurate data to support this hypothesis has only recently become available. This was primarily due to the extremely low iodine concentrations in the off-gas.

Prior to the last refueling outage, Dresden Unit 1 was operating with several defective fuel assemblies in the core which resulted in a relatively high off-gas (3×10^4 to 5×10^4 μc/sec) and iodine-131 (3.5×10^2 μc/sec) release to the moderator.

Iodine-131 activity levels observed in the off-gas during this period were approximately 5×10^{-4} μc/sec, or a total decontamination factor of 7×10^6 between the reactor water and the off-gas. If one assumes that a DF of 3×10^4 occurred between the reactor water and hotwell, the DF between the hotwell and off-gas would be 2×10^2.

CONCLUSIONS

Measurements performed at Dresden Unit 1 indicate an iodine decontamination factor of 3×10^4 to 10^5 between the reactor water and the condensate and a decontamination factor of 2×10^2 between the hotwell and the off-gas. A total iodine decontamination factor of 7×10^6 was measured between the primary water and the off-gas.

In the steam line break accident a large amount of water is carried from the vessel with the steam. To estimate the total quantity of halogens carried from the reactor, the quantities carried with the steam and with the water must be estimated. Since the masses of steam and water lost in the accident are comparable, but the water to steam decontamination factor is in the range of 3×10^4 to 10^5, the halogens carried by the steam are negligible compared to those carried by the water.

TABLE 17-2

IODINE DECONTAMINATION FACTORS BETWEEN REACTOR WATER AND CONDENSATE

	July 1961	5-22-63	1-13-64	3-3-65
Power (MWt)	620	620	530	375
Off-Gas (μc/sec)	3×10^2	2×10^3	4×10^3	6×10^4
Approximate number of fuel defects	2	5	6	15 Identified 28 Suspect
Type of fuel cladding	Zr	Zr + SS	Zr + SS	Zr + SS
Average Burnup (MWDt)		5,000	8,300	9,000
Conductivity, μmho				
Reactor Water	0.4	0.4	0.4	0.4
Condensate	0.3	0.3	0.3	0.3
pH				
Reactor Water	7	7	7	7
Condensate	7	7	7	7
Sampling Method				
Reactor Water	Grab	Grab	Grab	Grab
Condensate	Grab	In-Line	Grab	Grab
Analysis Method*				
Reactor Water	1	1	1	1
Condensate	1	3	2	1
μc/ml I-131				
Reactor Water	Isotopic Values	1.2×10^{-3}	1.6×10^{-2}	3.6×10^{-2}
Condensate+	not Reported	4×10^{-8}	6×10^{-7}	2.2×10^{-6}
μc/ml I-133				
Reactor Water	Gross Iodine	3.6×10^{-3}	-	2.0×10^{-1}
Condensate+	Only	$<1.2 \times 10^{-7}$	-	4.1×10^{-6}
Decontamination Factor				
I-131	Gross Iodine	3×10^4	3×10^4	2×10^4
I-133	Only	$>3 \times \times 10^4$		5×10^4
Average	$> 3 \times 10^4$	3×10^4	3×10^4	3×10^4

*Analytical Methods:
1. Radiochemical separation.
2. Gross decay and spectrometry data on untreated samples.
3. Gross decay, spectrometry, and ion exchange filtration.

+Values corrected for dilution by secondary steam.

QUESTION

18. With reference to the adequacy of the seismic design of the existing 300 foot stack, justify the use of the factor C equal to 0.033 in the formula for distribution of total base shear over the height of the structure. In addition, the effects of overturning motion combined with shear motion in determining the moment about the base of the stack, under earthquake conditions, should be discussed.

ANSWER

The existing 300 foot stack for Dresden Unit 1 Power Plant was designed in accordance with Bechtel Corporation's Specification No. C-2345-66, Revision 1 dated December 5, 1957. Section II, Paragraph 2.6 of that specification reads as follows: "Seismic forces shall be analyzed in conformance with report of the ASCE-SEAONC Joint Committee (1951), ASCE Separate No. 66, or Transactions Vol. 117, Page 716, which recommends that the total base shear be distributed over the height of the structure in accordance with the formula:

$$F_x = \frac{V\,W_x h_x}{\Sigma\,(wh)} \quad \text{and} \quad V = CW$$

Where

V = Total Lateral Base Shear
C = 0.033
W = Total Weight of Stack
W_x = Weight of Increment under consideration (use at least 10 increments).
h_x = Height of Increment above Base
$\Sigma\,(wh)$ = Summation of products of weight at each increment and its height above base
F_x = Lateral force at section under consideration."

Separate No. 66 specifies the value of C shall not be less than 0.03 nor more than 0.10 for structures other than buildings. This value of C = 0.033 was also used in the seismic design for other structures for Dresden Unit 1.

Separate No. 66 specifies that the horizontal shear due to earthquake at any horizontal plane shall be distributed to the various resisting elements in proportion to rigidities. Also the dead load moment of stability of every building or other structure shall not be less than one and one-half times the overturning moment caused by wind pressure. Provision for overturning moment shall be made for the specified earthquake forces in the top ten stories of buildings or the top 120 feet of other structures, and the moments shall be assumed to remain constant from these levels in to the foundations.

Section II, Paragraph 2.5 of Specification C-2345-66 reads as follows: "The chimney shall be designed to resist wind pressure in accordance with ACI 505-54 for a gust velocity of 110 miles per hour." Preliminary calculations indicate that the stresses produced by this wind force greatly exceeds the earthquake stresses, and that the method of treating shears and moments as specified in Separate No. 66 for earthquake loads is adequate.

John A. Blume and Associates have made a dynamic analysis on the Dresden Unit 1 stack using the response spectrum curves shown in Figure 54 of the Plant Design and Analysis Report. The method of analysis used was to treat the stack as a flexible cantilever system fixed at the bottom of the foundation. Thirty-one lumped mass points were considered to be supported by weightless elastic columns. Natural frequencies, mode shapes and dynamic response of the equivalent thirty-one mass system were computed. A total of ten modes were considered, and a damping value of five percent was assigned to each of the ten modes analyzed.

The results of this analysis are as follows:

1) Maximum shear at the base is 66.5 kips or 6.7 percent of the total weight of the structure. This produces a shearing stress distribution of the rock of only 74 psf, which is small and may be neglected.

2) Maximum overturning moment is 7300 foot-kips. This produces an overturning soil pressure distribution of 3,740 psf. Since this stack is founded on rock, this pressure is not excessive. This stack has a factor of safety of 3.92 against overturning.

3) The maximum computed displacement occurs at the top of the stack and has a magnitude of only 3.2 inches, indicating that the structure is relatively rigid. First mode period of vibration is 1.87 seconds.

4) Rocking was considered but it was found to contribute only negligibly to the overall response of this structure.

5) Stresses in the concrete stack were determined due to earthquake and were found to be very small - maximum compression of 335 psi.

QUESTION

19. Provide a table which lists the criteria for maximum allowable stresses as a fraction of yield stress for Class I and Class II structures under the following loading conditions where applicable:

a) dead loads plus live loads plus operating loads

b) dead loads plus operating loads plus seismic loads (0.1g)

c) dead loads plus operating loads plus complete loss of coolant loads plus seismic loads (0.1g)

d) dead loads plus operating loads plus complete loss of coolant loads plus seismic loads (0.2g)

ANSWER

The general criteria for maximum allowable stresses for Class I and Class II structures and equipment are given on pages II-15-1 and II-15-2 of Amendment No. 2 to the Plant Design and Analysis Report in the answer to Question II, 15.a.

There are a number of different structures and many items of equipment that are classified as Classes I and II. These items are made of various materials and are designed and constructed to various codes and standards. It is, therefore, impractical to include all the information for each of the Class I and II structures at this stage in the design. However, the maximum allowable stresses used for various loading conditions are given for representative examples of some of the most critical Class I structures in Tables 19-1, 19-2 and 19-3. The loading conditions considered are not in each case exactly as set forth in the question, due to the differences in the structures considered.

In Table 19-1 the term "Dead Loads" includes the weight of the drywell vessel and all appurtenances. "Operating Loads" include the gravity loads from equipment supports, the restraint to thermal movement of the drywell vessel due to the compressible material between the vessel and the concrete external to the drywell. The "Loss of Coolant Load" is the drywell design pressure of 62 psig. The summation of loads as stated in Condition 1 of Table 19-1 provides the design basis for the primary containment. For conditions of normal operation, the loads are much less than Condition 1 and will be determined during the design.

In Table 19-2, the term "Dead Loads" includes the weight of the structural components and the architectural appurtenances of the reactor building. "Operating Loads" consist of gravity loads from all equipment and piping, and for the weight of water over the reactor during refueling and in the storage pools. The "Live Loads" will include loads which can be expected throughout the reactor structure, i.e., roof and crane loads, movable equipment, the shipping cask, stored supplies, etc.

Table 19-3 is a tabulation of allowable stresses for Class I piping.

Both horizontal and vertical earthquake loads are included in the "Seismic Loads" for each representative example given. The vertical acceleration assumed will be equal to 2/3 the horizontal ground acceleration.

The Dresden Unit 1 ventilation stack, which is shared with Unit 2, although not specifically listed in the Plant Design and Analysis Report, page V-6-2, as a Class I structure, will be included in this category as a part of Unit 2 design.

The seismic design of Class II structures and equipment will be in accordance with applicable codes following the normal practice for the design of power plants in the State of Illinois. As a minimum, the design will meet the requirements of the "Uniform Building Code" for Zone 1, including the allowable stress specified for combined dead load, live load, and seismic loads.

TABLE 19-1

ALLOWABLE STRESSES

Loading Condition	General Membrane		General Bending Plus Local Membrane Plus General Membrane		General Membrane Plus Secondary Stresses	
	Allowable Stress psi	Percent of Y.S.	Allowable Stress psi	Percent of Y.S.	Allowable Stress psi	Percent of Y.S.
1. Dead Loads plus Operating Loads plus Loss of Coolant Loads plus Seismic Loads (0.1g)	19,250	50.7	28,875	75.97	52,500	138
2. Dead Loads plus Operating Loads plus Loss of Coolant Loads plus Seismic Loads (0.2g)	Safe shutdown of the plant can be achieved. (See Note 1 below.)					

Note 1: The stress criteria for Loading Condition 2 is as given in paragraph 2 on page II-15-1 of Amendment No. 2 to the Plant Design and Analysis Report, which refers to analysis to determine energy absorption capacity.

TABLE 19-2

ALLOWABLE STRESSES FOR REACTOR BUILDING

Loading Condition	Reinforcing Steel Max. Allowable Stress	Concrete Max. Allowable Compression Stress	Concrete Max. Allowable Shear Stress	Concrete Max. Allowable Bearing	Structural Steel Tension on the Net Section	Structural Steel Shear on Gross Section	Structural Steel Compression on Gross Section	Structural Steel Bending
1 Dead Loads Plus Live Loads, Plus Operating Load Plus Seismic Loads (0.1g)	0.5 Fy	0.45 f'c	$1.1\sqrt{f'c}$	0.25 f'c	0.60 Fy	0.40 Fy	Varies with Slenderness Ratio	0.66 Fy to 0.60 Fy
2 Dead Loads Plus Live Loads, Plus Operating Loads Plus Wind Loads	0.667 Fy	0.60 f'c	$1.467\sqrt{f'c}$	0.333 f'c	0.80 Fy	0.53 Fy	Varies with Slenderness Ratio	0.88 Fy to 0.80 Fy
3 Dead Loads, Plus Live Loads, Plus Operating Loads, Plus Seismic Loads (0.2g)		Safe Shutdown of the Plant can be Achieved (See Note 1 Below)						

Fy = Minimum yield point of the material

f'c = Compressive strength of concrete

Note 1 = The stress criteria for Loading Condition No. 3 is as given on page II-15-1 of Amendment No. 2 to the Plant Design and Analysis Report - which refers to analysis to determine energy absorption capacity.

TABLE 19-3

ALLOWABLE STRESSES
FOR CLASS I PIPING

Loading Condition	Allowable Stress
1. Thermal Expansion	S_A
2. M. O. L. + S. L.	S_h
3. M. O. L. + 2 × S. L.	Safe shutdown can be achieved. (See Note 1 of Table 19-1)

M. O. L = Maximum operating loads including design pressure and temperature, weight of piping and contents including insulation and the effect of supports and other sustained external loadings.

S. L. = Seismic loads due to the design earthquake (0. 10g).

2 × S. L. = Seismic loads due to twice the design earthquake (0. 20g).

$S_A = f(1.25 S_c + 0.25 S_h)$.

Where:

f = stress range reduction factor for cyclic conditions.

S_c = allowable stress in the cold condition per ASA B31. 1.

S_h = allowable stress in the hot condition (design temperature) per ASA B31. 1.

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DRESDEN NUCLEAR POWER STATION

UNIT 2

PLANT DESIGN AND ANALYSIS REPORT

AMENDMENT NO. 5

ANSWERS TO AEC QUESTIONS

Commonwealth Edison Company

3478

DRESDEN NUCLEAR POWER STATION

UNIT 2

PLANT DESIGN AND ANALYSIS REPORT

AMENDMENT NO. 5

ANSWERS TO AEC QUESTIONS

Commonwealth Edison Company

QUESTION

1. Details of stability analyses concerning operation with multiple jet pumps, including conditions where one pump has failed or malfunctioned.

ANSWER

Introduction

A description of the jet pumps is given in the Plant Design and Analysis Report, Appendix A and Figures 12, 18, 19, 30 and 67. A discussion of various aspects related to jet pump stability can be found in Amendment No. 4, page I-1 including stability of parallel pump operation. Additional discussion can be found in Amendment No. 2, page II-3-1. The following paragraphs will serve to provide some additional information.

Multiple Pump Operation

With respect to stability of multiple jet pump operation, the primary source of information at this stage of design has been the tests described in the reference above which confirmed the stability of parallel jet pump operation. That the jet pumps in parallel do operate in a stable manner was expected from analyses of the loop characteristics. Over 3/4 of the 550 foot head loss in the drive pump loop occurs at the jet pump nozzles. Therefore, it is difficult for changes or disturbances in other portions of the loop, or in the lines feeding the nozzles, to affect the flow through or among the various nozzles. If a malfunction or disturbance occurs which affects one of the injection nozzles, the others in parallel will have a tendency to compensate due to the negative slope of the drive pump head-flow characteristic.

The jet pumps at rated flow, and below the flow control range as well, are tightly coupled hydraulically to their respective injection nozzles. That is, even if one injection nozzle partially malfunctions due to an arbitrarily imposed flow restriction, the jet pump will continue to function. For example, the injection flow in a nozzle could be reduced to 40 percent of rated (when pumping against rated-core pressure drop) before flow through the diffuser would stop.

Thus, the entire system is "stiff" with a strong tendency to remain at equilibrium over a wide range of conditions and stable parallel operation of jet pumps results.

Cavitation Margins During Operation

It is also of interest to note the wide margins which exist between normal operating conditions and those which cause cavitation in either the jet pumps, or the driving pumps.

Normally subcooling is 20.4 Btu/lb. Tests on jet pumps indicate that incipient cavitation does not occur down to a subcooling of 3 to 4 Btu/lb. The drive pumps can operate at approximately the same conditions without cavitation. Based on this, a reduction of 85 percent in feedwater flow at full power could be tolerated without bringing about conditions condusive to cavitation. The feedwater control system is designed such that flow cannot be decreased more than 85 percent below normal at rated power, thereby assuring that feedwater flow changes will not cause cavitation.

The pressure existing at both the jet and drive pumps, above that required to cause cavitation, is about 90 psi. Pressure changes during maneuvering do not exceed ±10 psi maximum and hence are not of concern from the standpoint of cavitation.

It is difficult for large reductions in pressure to occur. For example, a sudden 15 percent bypass of steam results in only a 7 psi transient reduction since the pressure regulator will quickly adjust. An arbitrary 90 psi pressure reduction would reduce power to approximately 85 percent of rated power due to void formation within the core. Since the pumps would continue to function, they do not contribute to the power reduction.

If pressure were decreased arbitrarily to values which would bring about cavitation in the pumps, flow through the core would decrease. This would result in a reduction in power along the power-flow characteristic of the core. An additional power decrease would result from the pressure reduction itself. However, there would be no safety implications involved because of the nature of the cavitation process itself as discussed below.

With respect to cavitation in the jet pumps, it is highly unlikely, as discussed in Amendment No. 4 page 1-3, that bubbles formed within the pump would be injected into the reactor since they would collapse before leaving the diffuser. Bubbles, were they to enter the core, would result in an increase in void volume and consequently a decrease in power. Experience and tests with both centrifugal and jet pumps operating at high temperature and pressure indicates that cavitation is a gradual orderly process and is not of a "chugging" or oscillatory nature. In addition, these tests have shown the absence of a sudden drop-off in flow as cavitation is increased. Rather the flows drop gradually and in a continuous orderly manner as cavitation is increased. This is in contrast to the more common experience involving pump operation at low pressures of a few atmospheres. The large volume changes associated with a change of phase of the liquid do result in much sharper decreases in flow as cavitation proceeds, since both centrifugal and jet pumps are basically volume flow devices. Therefore, even if cavitation were to occur, it would not become a driving function with respect to stability. Core power would gradually drop along the power-flow characteristic and stabilize at the new flow.

Jet Pump Malfunction

The effects on the system of a malfunction of a given jet pump have been analyzed. For example, if, for the purpose of analysis, one of the injection nozzles is assumed to be plugged while operating at

full power, flow would reverse through the jet pump diffuser, bypassing about 2 percent of the core flow. An additional 4 percent reduction in rated core flow results due to the loss of the jet pump, resulting in a net decrease to approximately 94 percent of rated flow. Reactor power would decrease without safety implications to about 95 percent of rated as determined by the power-to-flow characteristic of the core as given in Figure 34 of the Plant Design and Analysis Report, a copy of which is attached.

With respect to malfunctions at the inlet of the diffuser, it should be noted that as long as the injection nozzle is functioning, at least the nozzle flow itself will be injected into the bottom plenum of the vessel. This corresponds to 1/3 of the rated jet pump flow.

Drive Pump Malfunction

If one of the drive pumps were to fail, half of the jet pumps would coast down and cease to function because the equalizer line between the two driving pump discharges is closed. Recirculation flow would decay to a lower than rated value. Flow would reverse through the 10 idle jet pump diffusers. The other 10 jet pumps would continue to function. Since the core pressure drop would be reduced at the lower flow, the jet pump flow ratio $\left(\frac{\text{inducted flow}}{\text{driving flow}}\right)$ would increase. The driving flow remains essentially constant since the driving pump loop characteristic has not changed. Calculations show the 10 jet pumps would provide nearly 150 percent of their normally rated flow at the lower core pressure drop. The total flow injected by the jet pumps would be 75 percent of rated. About 12 percent of rated flow bypasses the core through the idle diffusers resulting in a net of about 63 percent of rated flow going through the core itself. Core power would drop to and stabilize at about 70 percent rated as determined by the power-flow curve indicated above. This transient is less severe than the complete loss of pumping power transient which itself does not violate the MCHFR criteria. Power could subsequently only be raised by a flow increase or by rod motion. In either case there would be no violation of the MCHFR. Protection against excessive power generation at a given flow is provided by the rod block interlock, as shown on Figure 34, Plant Design and Analysis Report.

Conclusion

Based on tests and analyses of the parallel jet pump characteristics it is concluded that they will operate in a stable and predictable manner.

Loss of a jet pump or a driving pump does not result in a serious flow loss or unstable operation of the remaining pumps. A gradual power decrease without safety implications is the only result.

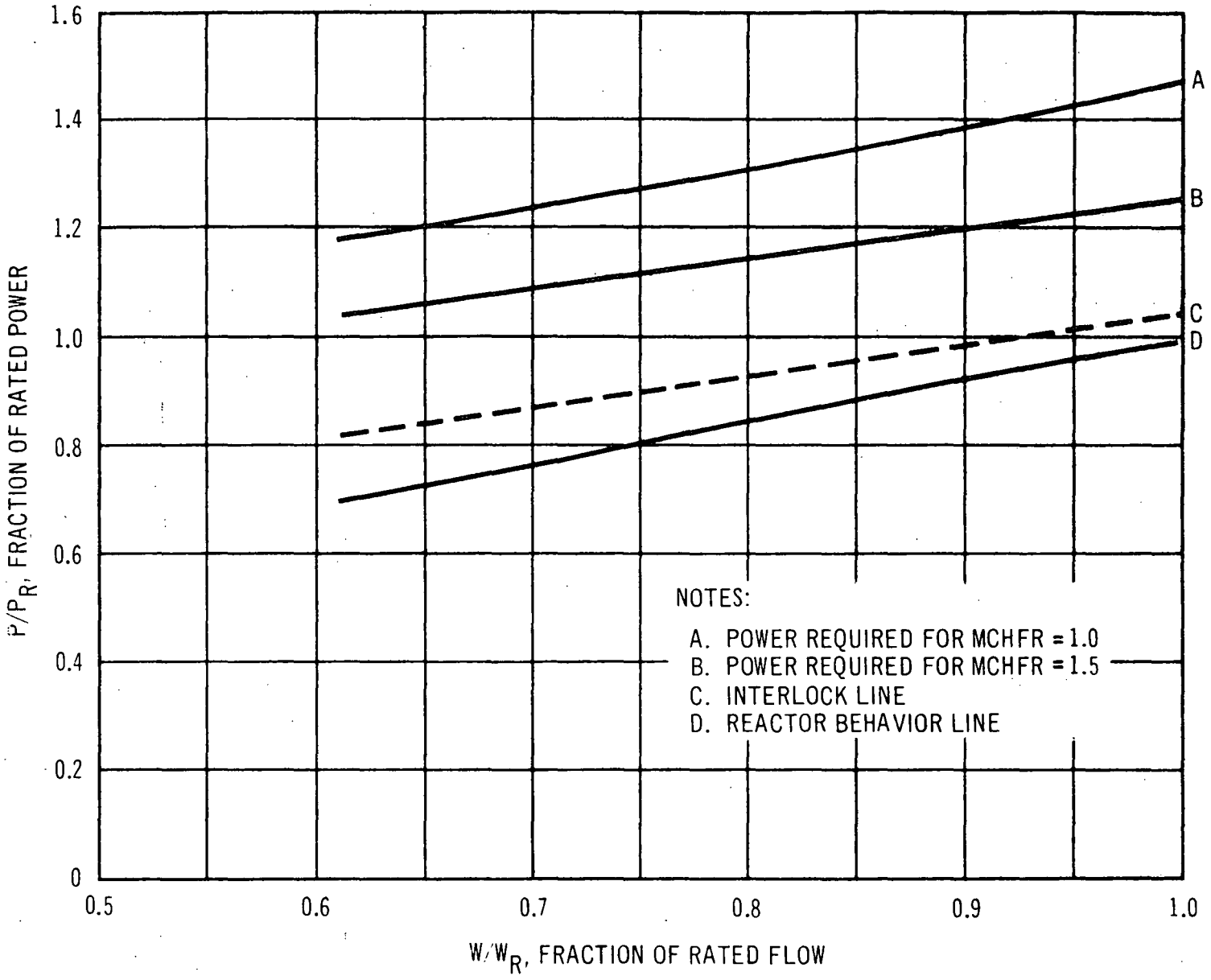

Figure 34. Typical Power - Flow Relationship

QUESTION

2. Justification for the mechanism for termination of the metal-water reaction at the point selected.

ANSWER

In reply to question III-5, Amendment No. 2, page III-5-1, the analytical model and digital code used in arriving at the amount of metal-water reaction is discussed. Because of the detailed representation of the core and conservative assumption that the core will receive all the steam required for the reaction, the in-core metal-water reaction estimated should be conservative in both quantity and time over which it occurs.

The reaction terminates because the Zircaloy melts, runs down the hot fuel surfaces, falls through the end plate of the fuel, and into the water below the core where it is quenched. Water is expected to be present at the bottom of the vessel, entering either through the core spray system, the feedwater system, or control rod drive system.

Calculations of the droplet diameter based on surface tension and the fuel end plate dimensions gave droplet sizes 3/8 inch in diameter. Molten drop reaction rates in water temperatures of interest given in ANL 6548, Figure 26, indicate that a reaction depth of 60 microns correlated with observed droplet reaction test data. Application of the 60 micron reaction depth to the mean diameter calculated resulted in a 4 percent reaction of the molten Zircaloy leaving the core. Thus a total estimate 24.5 percent in core and 3 percent i.e., $0.04 \times (100 - 24.5)$, post-melt reaction resulted in the total of 27.5 percent reaction in a minimum time of 30 minutes.

Zircaloy rod meltdown tests, evaluated since submitting Amendment No. 2, have shown that molten Zircaloy will indeed leave the core in drops of definite sizes. These tests were conducted in a test assembly with simulated fuel end plates and four induction heated Zircaloy rods. One test with nine rods and an actual end plate was also conducted giving similar results. All these experiments show that a normal statistical distribution of droplet sizes occur, ranging from a minimum of 0.137 inch diameter to a maximum of 0.477 inch diameter with a mean diameter of 0.269 inch. Application of the 60 micron reaction depth to the various sized drops showed an overall reaction of 5 percent, not appreciably different from that calculated initially for Amendment No. 2. In addition, the tests clearly showed that the molten drops were cooled by the water thus terminating any further reaction.

The realistic estimate above is well within the capability of the containment system, which from Figure II-4-1 Amendment No. 2, attached, is seen to be a 42 percent reaction when it occurs in a half hour, the minimum time over which the reaction is expected to occur. It is important to note that if the reaction were to take place over a longer period of time, which can be very likely, the containment capability increases. For example, the capability is 55 percent metal-water reaction

if it occurs over a one hour period, 66 percent if over a 2 hour period, and 74 percent if over a 4 hour period. This increase in containment capability results from the heat removal capacity of the containment cooling system.

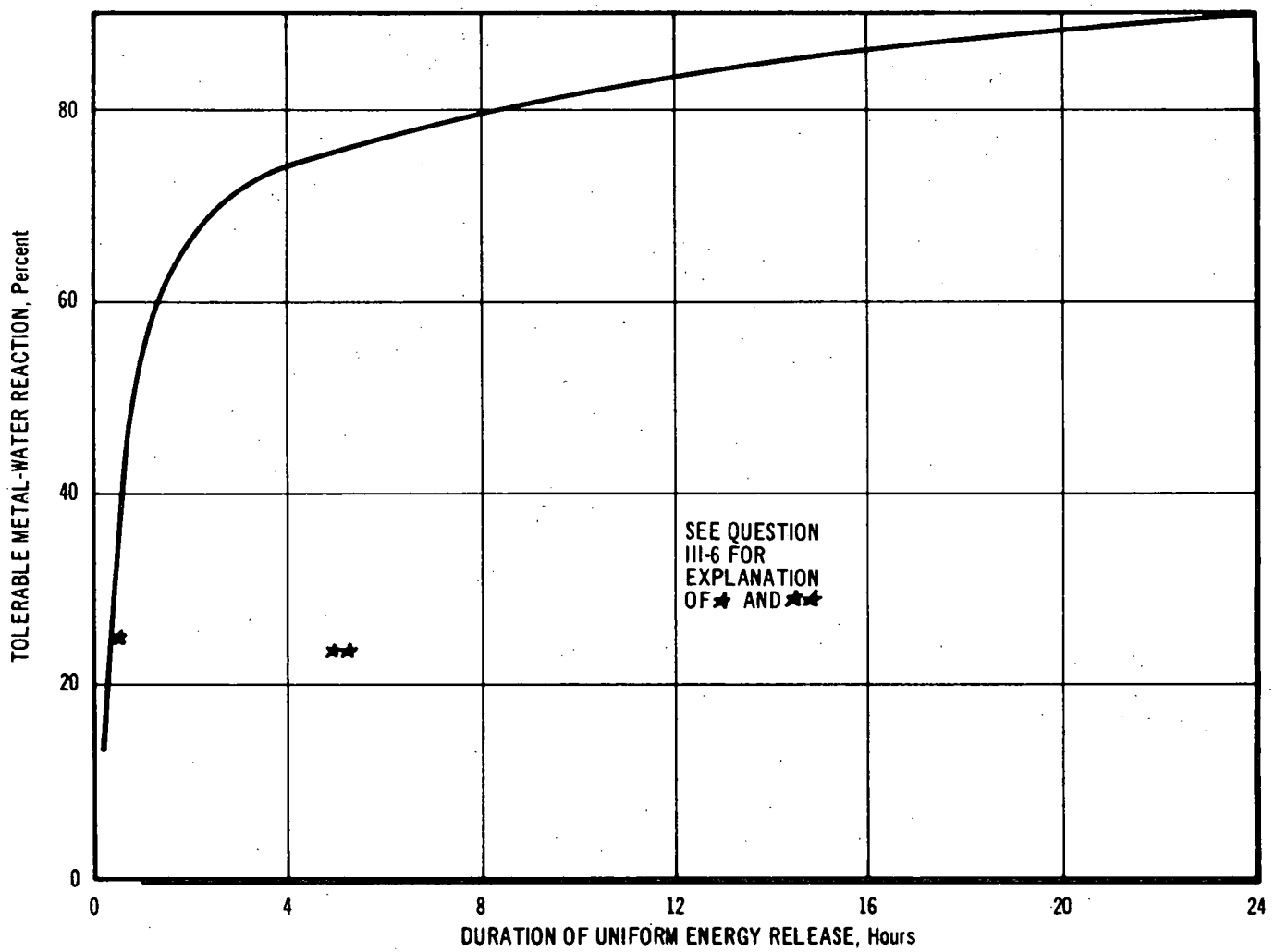

Figure II-4-1. Containment Capability - Tolerable Metal-Water Reaction as a Function of Duration of Energy and Gas Release

QUESTION

3. Justification for the assumption that there is no source of missiles which can violate the containment.

ANSWER

It is the design philosophy of the General Electric Company that there be no missiles which will penetrate the containment. This is accomplished in practice through the specific design of the containment and contained systems, which takes into account the potential for generation of missiles and minimizes the possibility of containment violation. As discussed in question 8 of Amendment No. 4, it has been concluded that with the application of conservative piping design and proven engineering practices, pipes will not break in such a manner as to bring about movement of pipes sufficient to damage the primary containment vessel. The design of the containment and piping systems does consider, however, the possibility of missiles being generated from the failure of flanged joints such as valve bonnets, valve stems, recirculation pumps, and from instrumentation such as thermowells. In considering potential missile sources of this nature at the current stage of detailed design, none have been found against which further design action is required. However, if such a case is found as the design progresses, appropriate design action will be undertaken regarding the potential missile source to maintain containment integrity.

The most positive manner to achieve missile protection is through basic plant arrangement such that, if failure should occur, the direction of flight of the missile is away from the containment vessel. The arrangement of plant components takes this possibility into account even though such missiles may not have enough energy to penetrate the containment. Analyses have led to the conclusion that instruments, ejected thermowells, etc., if they should become missiles, would not have sufficient energy to penetrate the containment. It has also been concluded that large, massive, rotating components such as the reactor recirculating pump motors would not have sufficient energy to move this mass to the containment wall.

Also, pump impellers and motor rotors upon failure would be contained within their housings and not generate missiles. There is the potential for valve bonnets to become missiles based on the assumption of failure of all bonnet bolts. This requires instantaneous, clean severance of all bolts, without any overturning motion. The damage potential is dependent upon the size of the valve and system in which the valve is located. Therefore, valve arrangement is important and is taken into consideration in the overall plant design. Large valve bonnets, such as are on the recirculation line valves, appear to have the capability to contribute to potential missiles from bonnet bolt failure. If it is necessary to locate such valves where containment protection is required there are many design actions available, such as welding the bonnets, providing deflector plates, adding bracing or keepers, etc. Although, to date, analyses have shown no need to take any extraordinary design actions, the designs are reviewed constantly for such a possibility.

As noted in Section V-3 of the Plant Design and Analysis Report, the design does consider the jet forces associated with a recirculation line severance. As a result of the design considerations for jet forces missile protection is obtained for items such as the vent lines and containment penetrations, all of which are designed against jet forces associated with failure of a recirculation line. Also, the containment is backed up by a concrete structure in most of its area and there is a gap of two to three inches between the containment and the concrete. The concrete structure limits the deflection of the containment.

This discussion has been directed towards internal missiles generated within the drywell and has not included the suppression chamber. The suppression chamber has no source of internal or external missile generation and the vent pipes joining it with the drywell are protected by the jet deflectors.

In addition to the care with which equipment is oriented with regard to missiles special care is taken in component arrangements to see that equipment associated with engineered safety systems such as the core spray and the containment spray are segregated in such a manner that the failure of one cannot cause the failure of the other or that the failure of any component which would bring about the need for these engineered safeguard systems will not render the safeguard system inoperable. Additionally, the control rod drive mechanisms are located in a concrete vault that provides protection from potential missiles.

The primary containment vessels are completely enclosed in a reinforced concrete structure having a thickness of 4-6 feet. (See Figure 3 of the Plant Design and Analysis Report.) This concrete structure, in addition to serving as the basic biological shielding for the reactor system, also provides a major mechanical barrier for the protection of the containment vessels and reactor system against potential missiles generated external to the primary containment.

QUESTION

4. Details concerning uncertainties and conservatism in calculating Doppler coefficient.

ANSWER

In assessing uncertainties in Doppler contribution to the safety of a Boiling Water Reactor (BWR) both the Doppler coefficient and resonance integral information must be considered. The significant parameter reflecting both these quantities is the reactivity decrement due to Doppler broadening. Based on evaluation of data and calculations discussed below it is estimated that design calculations have a nominal conservative bias of about 20 percent in predicted reactivity decrement.

Experimental data on UO_2 fuel against which the design model is compared is that of Hellstrand[1] and Pettus[2]. These particular measurements were chosen as being typical and more carefully performed than other information which has been reviewed. An overall assessment has been made of the accuracy of Doppler reactivity decrement data based on (1) review of the experimental data, (2) review and analysis of various Doppler calculational recipes, and (3) analysis of lattice parameter measurements and excursion experiments. The uncertainty in the data is assessed to be ± 10 percent with 67 percent confidence.

Summary comparisons of the Doppler reactivity decrement ($\Delta k/k$) and the Doppler coefficient ($1/I\ dI/dT$) are shown in Tables A5-I, -II, and -III for typical BWR fuel for both cold and hot reactor conditions. The comparison assumes no boiling and moderator temperature, T_m, initial fuel temperatures, T_1, and final fuel temperature, T_2, as indicated in the tables. Gross core spatial importance weighting is not included. The results show the design model predictions compared to Hellstrand and Pettus data. The Δ indicates the percentage difference between the experimental and design model. This comparison shows an average conservative bias of the design model of the order of 5 percent to 10 percent.

In addition to the data comparison above, the design model assumes a constant radial fuel temperature. It has been shown [3] that this can lead to a slight conservatism in estimating Doppler reactivity decrements. For typical BWR fuel it is estimated that about 3 percent conservatism is introduced by this assumption.

Finally, after approximately one fuel cycle for the remainder of reactor life the Doppler reactivity decrement is increased by 10 percent to 15 percent due to the presence of Pu-240 in the core[4].

On the basis of the above factors a nominal conservative bias of about 20 percent in the design calculations of Doppler effects is inferred.

REFERENCES

1. Hellstrand, E., Nuclear Science and Engineering #8, 497 (1960).

2. Pettus, W. G., and Baldwin, M. N., "Resonance Absorption in U-238 Metal and Oxide Rods," BAW-1244.

3. Dresner, L., "Some Remarks on the Effect of a Non-Uniform Temperature Distribution on the Temperature Dependence of Resonance Absorption," Nuclear Science and Engineering, 11, 39, (1961).

4. Crowther, R. L., "Approximate Method for Determination of Doppler Broadening of Pu-240 Resonances," Trans. ANS 7, 1 (June, 1964).

TABLE A5-I

DOPPLER REACTIVITY DECREMENT $(-\frac{\Delta k}{k})$

Initial Conditions: $T_m = 68°F$, $T_1 = 68°F$

Final - T_2	APED	Hellstrand	Δ%	Pettus	Δ%
200 °F	0.001629	0.001995	+22.5	0.001810	+11.1
1000 °F	0.009475	0.011196	+18.2	0.010162	+ 7.3
2000 °F	0.017001	0.019568	+15.1	0.017758	+ 4.5
3000 °F	0.023391	0.026348	+12.6	0.023911	+ 2.2
4000 °F	0.029143	0.032198	+10.5	0.029220	+ 0.3
5000 °F	0.034430	0.037423	+ 8.7	0.033962	- 1.4

TABLE A5-II

DOPPLER COEFFICIENT $(-\frac{1}{I}\frac{dI}{dT})$

200 °F	0.00016052	0.00019062	+18.8	0.00019806	+23.4
1000 °F	0.00010790	0.00011997	+11.2	0.00012434	+15.2
2000 °F	0.00008310	0.00008735	+ 5.1	0.00009035	+ 8.7
3000 °F	0.00007009	0.00007051	+ 0.6	0.00007283	+ 3.9
4000 °F	0.00006171	0.00005991	- 2.8	0.00006181	+ 1.6
5000 °F	0.00005578	0.00005249	- 5.9	0.00005410	- 3.0

TABLE A5-III

DOPPLER REACTIVITY DECREMENT $(-\frac{\Delta k}{k})$

Initial Conditions: $T_m = 547°F$, $T_f = 1000°F$

2000 °F	0.010165	0.011305	+11.2	0.010259	+ 0.9
3000 °F	0.018796	0.020461	+ 8.9	0.018569	- 1.2
4000 °F	0.026543	0.028398	+ 7.0	0.025740	- 3.0
5000 °F	0.033703	0.035419	+ 5.1	0.032143	- 4.6

Note: Δ indicates the percentage difference between the experimental value and the APED design model.

QUESTION

5. Information concerning local void coefficient.

ANSWER

A discussion of the design bases for moderator coefficients, including the void coefficient, is given in Section IV-2.2 and 2.3, of the Plant Design and Analysis Report. Additional information and preliminary data on these coefficients as a function of reactor state and fuel exposure are included in the answer to Question 7 of Amendment 4.

In summary, the principle limits and criteria imposed on the moderator coefficients, as discussed in the Plant Design and Analysis Report, are:

a) The moderator density (or void) coefficient averaged over the interior of the fuel element channels is designed to always be negative.

b) The magnitude of the negative operating void coefficient is always sufficiently small to assure no adverse contribution to hydrodynamic stability.

c) The magnitude of the negative operating void coefficient is sufficiently large to contribute significantly to power coefficient damping of internal power distrubances, such as xenon shifts, in the core.

The moderator density coefficients are mathematically represented by

$$\frac{1}{k_{eff}} \frac{dk_{eff}}{d\rho} = \frac{1}{k_\infty} \frac{dk_\infty}{d\rho} - \frac{C \frac{dW}{d\rho}}{1 - CW} - \frac{\left(M^2 \frac{dB_g^2}{d\rho} + B_g^2 \frac{dM^2}{d\rho} \right)}{1 + M^2 B_g^2}$$

where

ρ = moderator density

C = control rod density function

W = control rod worth

M^2 = migration area

B_g^2 = geometric buckling

The terms on the right side of the equation represent the contributions of fuel multiplication, control rods and leakage, respectively, to the coefficients.

Any decrease in moderator density will increase neutron leakage from the region of the disturbance and will also enhance the strength of the control rods. These two contributions, therefore, are always negative everywhere in the core and monotonically increase in magnitude with moderator density reduction.

The k_∞ contribution, however, has a local spatial dependence. Within the fuel element channel a condition of undermoderation exists under all conditions and the local moderator density coefficient is negative. External to the fuel element, in the surrounding water gaps, a degree of overmoderation exists from the cold ambient condition through the lower end of the heatup range, and the local coefficient is slightly positive for these reactor states. The extent of this positive contribution is carefully limited by the design criteria. In the power range the gaps as well as the interior of the fuel assembly are undermoderated and local coefficients are everywhere negative.

Early in core life the control density is high and the control term contributes a strong negative effect to the coefficient. At high fuel exposures this term approaches zero due to removal of control. While the k_∞ term has a slight positive trend with exposure, the control term is by far the major factor in the reduction of magnitude of the coefficient with fuel exposure.

In a gross core power disturbance the k_∞ and control terms dominate since leakage is small. In a disturbance such as a control rod withdrawal, the control term in that region is zero but the geometric buckling and, therefore, leakage from the disturbed zone is large.

Near the end of core life where control is at a minimum a gross core transient that is sufficiently slow to allow water gaps to equilibrate with the flowing coolant (tens of seconds) may, below operating temperature, see a slight positive moderator reactivity effect. If, however, the transient accelerates so that voids are produced in the fuel element by fuel heating, or if voids should somehow be swept into the channels with the circulating coolant, an immediate negative k_∞ contribution would result such that the total moderator coefficient effect would be negative.

In a local disturbance the same phenomenon could occur except that a larger negative leakage effect would always be present. The local positive void coefficient external to the fuel channel in the water in the startup through the lower half of the heating range causes no difficulty because in operation there will be no voids formed in these water gaps. However, if it is assumed for the purpose of analysis that voids are formed in these water gaps at this low temperature level the positive contribution by the water gaps is very small. However, the over-all void coefficient will in all temperature conditions be negative since localized disturbances cannot be made small enough to generate voids in the water gaps only. Therefore, at no time will the formation of voids in the reactor lead to a positive reactivity feedback.

In summary, in those regions of the core where rapid moderator density changes can occur, density coefficients are designed for safety reasons to be always negative.

QUESTION

6. Information concerning local power variations in the core and their possible effect on stability and safe plant performance.

ANSWER

Introduction

Time-variant local power within the core can only occur because of internal causes such as a control rod motion, or to external causes such as changes in local subcooling or local flow to a given region of the core. How the core is designed to minimize local power variations is discussed in the following paragraphs.

Control Rod Motion

Control rod movement of any type is carefully controlled by procedures, and system design including interlocks. Therefore, control rod motion of an oscillatory nature cannot occur. Also, the core is strongly coupled hydrodynamically by selective orificing of different regions in the core. The effects of the coupling are calculated by a multi-channel power-hydraulic flow distribution code. It has been demonstrated on large cores such as Dresden Unit 1, Big Rock Point, and SENN that use of this code will result in local and gross stable operation even under conditions of local rod motion including rod oscillation.

Local Subcooling

Subcooling is expected to be relatively uniform and constant during operation because both power changes and feedwater flow adjustments are relatively slow. Carryunder of steam bubbles, as it affects subcooling, is not an oscillatory phenomena and will remain constant for a given set of operating conditions. This is supported by steam separator tests and by operation of natural convection boiling water reactors such as Humboldt Bay. The feedwater sparger provides a uniform distribution of feedwater around the vessel. The circumferential flow path between the vessel and the shroud is one of low resistance further assuring uniform flow to the jet pumps. The sparger is located 11 feet above the jet pumps thus providing a long quenching distance, compared to the 3-4 feet required for quenching the design carryunder value of 0.2 percent. Excellent mixing is provided both at the sparger and within the throat of the jet pumps themselves.

Local Flow

As discussed under "Local Subcooling" the reactor internals are designed in a manner to provide uniform flow. In calculating the flow into the various core regions and setting orificing requirements,

the thermal-hydraulic code employed also accounts for the resistance in the entrance plenum to each of the core regions. At most, the bottom plenum pressure drop is less than 5 percent of the total core pressure drop at rated flow. Because the pressure drop from the jet pumps to the core is such a small fraction of the total pressure drop, the local flow distribution to the core will not be affected appreciably by entrance perturbations. Also the geometry of the bottom plenum provides a vertical distance of nearly 10 feet to the core thereby permitting any flow maldistributions to readjust themselves. Thus local flows into the core cannot undergo large changes.

Because of the strong hydrodynamic coupling between core regions, flow into a given channel of the core will not be strongly affected by any arbitrary changes in the channel power. As an example of the strong hydrodynamic coupling within the core, computer runs for the Dresden Unit 2 core were made in which the power in a channel with the least inlet orificing was arbitrarily increased. These results showed that a 4 percent increase in power resulted in a 1 percent decrease in flow with the reactor at rated power. Similar calculations for channels with the tightest inlet orificing showed that a 25 percent increase in channel power was required to produce a 1 percent flow change.

Thus, it is concluded that local flow cannot be strongly influenced either by power shifts within channels nor by entrance flow perturbations. This has been confirmed by current operating experience.

Single Channel Stability

The stability of the reactor as a whole was discussed in Amendment No. 2, page II-2-2. However, the stability of individual channels within the core is also assured by making certain that they meet certain stability criteria. When these criteria are met, stable operation results such as has been shown in numerous test loops as well as in the Consumers Power Big Rock Point Power Plant rod oscillator tests. The stability of a single channel, operating in parallel with the core, is analyzed using the parallel channel option of the nuclear thermal-hydraulic stability code discussed in Amendment No. 2. Its response to arbitrary power and flow variations is studied on a frequency response basis. The amount of single phase damping required to meet the stability criteria is then determined. This is translated in terms of the minimum orificing or single phase pressure drop which must be designed into the core. This insures that the channel is well damped and will not sustain local power or flow variations.

As discussed in the reply to Question II, page II-2-4, Amendment No. 2, stability of the reactor and plant will be checked as part of the startup program. Any tendency for possible local power variations to occur will manifest itself well before actual instability results. In an actual core, local variations in power involve a sufficiently large number of bundles that the in-core instrumentation would detect such tendencies.

Conclusion

Local power variations due to external disturbances are controlled within an acceptably small range by the thermal-hydraulic characteristics of the core.

Local power changes due to internal sources will not result in instability or unsafe plant performance when the core is coupled hydrodynamically by the same technique applied to currently operating large boiling water reactors. The Dresden Unit 2 reactor is so coupled.

Application of currently available analytical models confirm the stability of individual fuel channels to arbitrarily imposed power oscillations.

QUESTION

7. Information concerning the means for providing valve redundance to assure containment for any failure in the steam lines.

ANSWER

One of the basic purposes of the primary containment system is to provide a minimum of one protective barrier between the reactor core and the environmental surroundings subsequent to an accident involving failure of the piping components of the reactor primary system. To fulfill its role as an insurance barrier, the primary containment is designed to remain intact before, during and subsequent to any design basis accident of the process system installed either inside or outside the primary containment. The process system and the primary containment are considered as separate systems, but where process lines penetrate the containment, the penetration design achieves the same integrity as the primary containment structure itself. The isolation valves in the process lines are designed to achieve the containment function when required.

The information which follows discusses the ability of the steam line penetration and the associated steam line isolation valves to fulfill the containment objectives under several single failure conditions of the steam line.

Figure A5-7 attached represents the penetration and isolation valve configuration for the main steam line. The steam line as it passes through the drywell containment vessel and the concrete biological shield is enclosed in a guard pipe that is attached to the main steam line through a multiple flued head fitting. This fitting is a one-piece forging with integral flues or nozzles and will be designed to meet all requirements of the ASME Pressure Vessel Code, Section VIII. The forging shall be radiographed and ultrasonically tested as specified by this code. The guard pipe and fittings are designed to the same pressure requirements as the steam line. The steam line penetration sleeve is welded to the drywell and extends through the biological shield where it is welded to a bellows which in turn is welded to the guard pipe. The bellows assembly accommodates the thermal expansion of the steam pipe and drywell relative to the steam pipe. The steam pipe is guided through pipe supports at each end of the penetration assembly to allow steam pipe movement parallel to the penetration and to limit pipe reactions of the penetration to allowable stress levels. Two isolation valves are provided. The external valve is located as close to the drywell penetration sleeve as practical, and the inside valve is located downstream of the reactor vessel safety valves.

The design of the pipe penetration support system takes into account the simultaneous stresses associated with normal thermal expansion, live and dead loads, seismic loads, and loads associated with a loss of coolant accident within the drywell to limit reactions on the penetration. For these conditions the resultant stresses in the drywell penetration do not exceed the code allowable design

stress. For failures of the steam pipe taken at random, the design takes into account the loadings given above in addition to the jet force loadings resulting from the failure. The resultant stresses in the pipe and penetration for this condition does not exceed 90 percent of the material yield stress.

The isolation valves are subject to specific requirements regarding reliability of performance. These requirements are stated in Nuclear Safety Criteria which have been established by the General Electric Company. The pertinent portions of these criteria are appended.

As noted above, the design provides a residual insurance barrier under assumed accident conditions. Figure A5-7 indicates the location of assumed independent breaks for which the sequence of protective functions are described below.

1. The failure occurs within the drywell upstream of the inner isolation valve.

   Steam from the reactor is released into the drywell and the resulting sequence is similar to that of a loss of coolant accident except that the pressure transient is less severe since the blowdown rate is slower. Both isolation valves close upon receipt of the high drywell pressure signal or the signal indicating low water level in the reactor vessel. This action provides two barriers within the steam pipe passing through the penetration and prevents further flow of steam to the turbine. Thus when the two isolation valves close subsequent to this postulated failure, containment integrity is attained, and the reactor is effectively isolated from the external environment.

2. The failure occurs within the drywell and renders the inner isolation valve inoperable.

   Again the reactor steam will blow down into the primary containment. The outer isolation valve will close upon receipt of the high drywell pressure or low water level signal, and the reactor becomes isolated within the primary containment as above.

3. The failure occurs downstream of the inner isolation valve either within the drywell or within the guard pipe.

   Both isolation valves will close upon receipt of either a high drywell pressure signal or a signal indicating low water level in the reactor vessel. The guard pipe is designed to accommodate such a failure without damage to the drywell penetration bellows, and the design of the pipe line supports protect its welded juncture to the drywell vessel. Thus the reactor vessel is isolated within the primary containment by means of the inner isolation valve, and the primary containment integrity is maintained by closure of the outer isolation valve. It should be noted that this condition provides two barriers between the reactor core and the external environment.

4. The failure occurs outside the primary containment between the guard pipe and the outer isolation valve.

   The steam will blow directly into the pipe tunnel until the isolation valves are automatically closed. Closure of the inner isolation valve places a barrier between the reactor core and the external environment. This barrier serves to isolate the reactor and complete the containment integrity. Closure of the outer isolation valve in this incident serves no useful purpose.

5. The failure occurs outside the primary containment and renders the outer isolation valve inoperative.

   The containment barrier and isolation of the reactor is achieved by the inner isolation valve and penetration configuration integrity as in 4 above.

6. The failure occurs outside the primary containment between the outer isolation valve and the turbine.

   The steam will blow down directly into the pipe tunnel or the turbine building until both isolation valves are automatically closed. This action isolates the reactor, completes the containment integrity, and places two barriers in series between the reactor core and the outside environment. The off-site consequences of failures 4, 5, and 6 are represented in the accident analysis as discussed in Section XI-5.3 of the Plant Design and Analysis Report.

7. There is a simultaneous failure of the steam line and guard pipe at a location between the two isolation valves.

   The steam will blow down into both the pipe tunnel and the interior of the drywell. The isolation valves will close by any of the signals indicating high drywell pressure, low water level in the reactor vessel, or high temperature in the steam tunnel. The inner isolation valve provides the barrier in the steam pipe to the further escape of steam or radioactive materials from the reactor vessel.

   It should be noted also that the turbine stop valves, located in the steam lines just ahead of the turbine will provide a back-up containment barrier in addition to the other isolation valves, for such breaks as 1, 2, and 3 as discussed above.

   In summary, the design of the steam line penetration and the provision of two isolation valves leads to several conclusions pertaining to containment integrity:

   a. The penetration components are specifically designed to withstand the occurrence of a steam line failure adjacent to the penetration or inside the guard pipe.

   b. The two isolation valves are separated by a concrete structure. This design feature and the provision for pipe anchors and guides makes the simultaneous failure of both isolation valves a highly improbable event.

   c. Analysis of postulated failures indicates that for all single failure events, the penetration and valve system will permit isolating the reactor so that a physical mechanical barrier is placed between the reactor core and the external environment subsequent to the failure event.

   d. The accidents involving steam line failures have been analyzed in Section XI-5.3 of the Plant Design and Analysis Report and the off-site consequences are significantly less than the guide line doses of 10 CFR 100.

NUCLEAR SAFETY CRITERIA
FOR BOILING WATER REACTORS

CONTAINMENT
STEAM LINE ISOLATION VALVE RELIABILITY

OBJECTIVE

To provide main steam line isolation valves which result in timely closure of extraordinary reliability.

CRITERIA

1. The overall quality levels and reliability attributes of main steam line isolation valves shall be equivalent to those normally provided in turbine stop valves, but this shall not be taken to mean that such valves must be of the same type or kind as turbine stop valves.

2. The piping arrangement pertinent to main steam isolation valves shall be such that periodic testing of these valves to full closure during normal plant operation is feasible.

3. Main steam line isolation valves shall be of a type which will assuredly achieve tight full closure under conditions of sudden and gross breakage of the steam pipe downstream. Gate valves will not be used.

4. Main steam line isolation valves shall be installed in tandem in each steam line, one inside the primary containment and one outside the primary containment.

5. One of the isolation valves in each of the steam lines shall utilize local stored energy (spring, compressed air, etc.) for closure motive power and shall not rely on continuity of any variety of electrical power for the motive force to achieve closure.

6. The speed of closure of the main steam line isolation valves shall be rapid, so that minimal loss of coolant is the result of breech of steam line outside the primary containment, and shall not exceed 10 seconds.

7. The speed of inadvertent closure of the main steam line isolation valves shall not be so rapid as to induce a more severe transient on the reactor plant than closure of turbine stop valves coincident with failure of the by-pass valves to open.

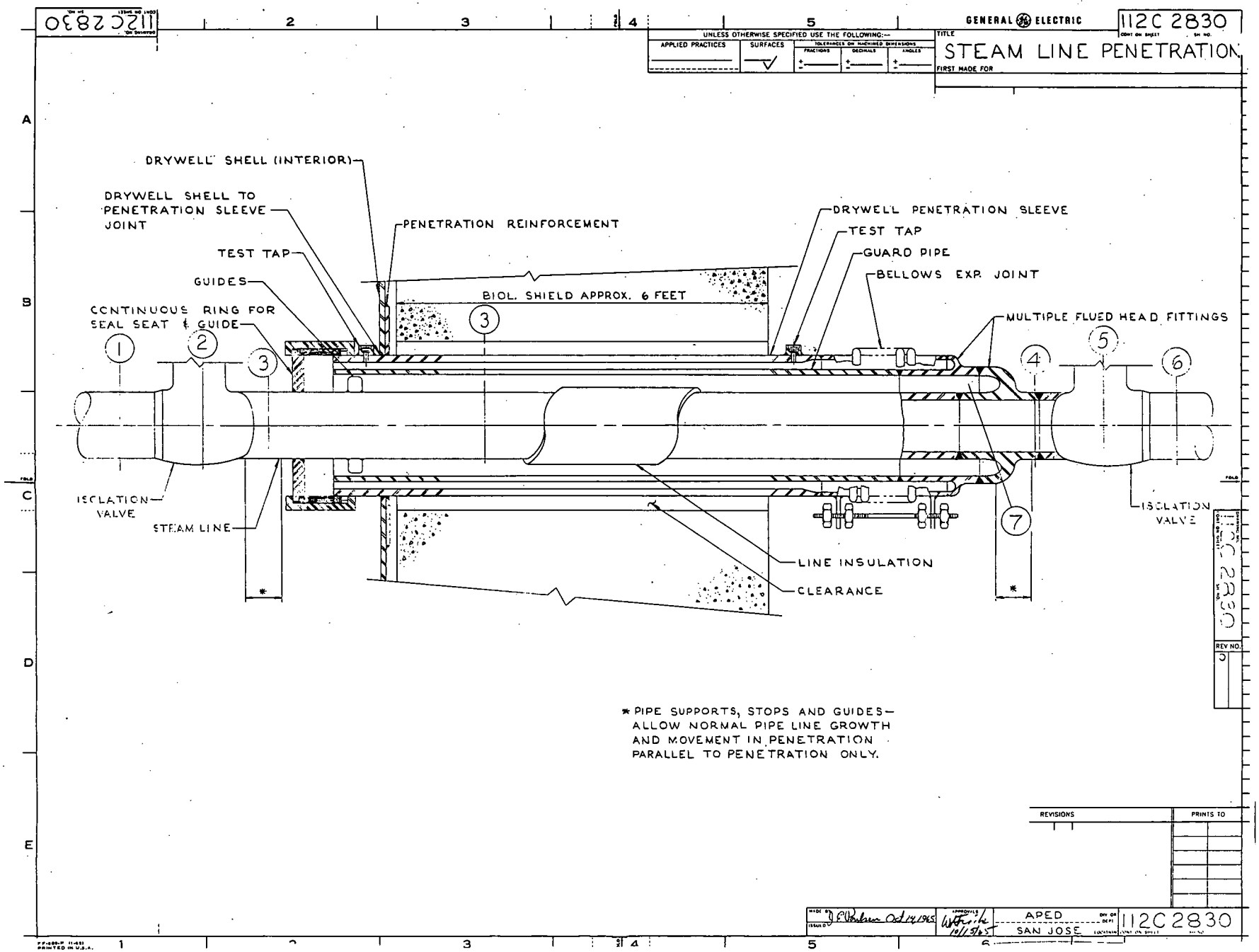

Figure A5-7. Diagram of Steam Line Penetration