



Docket Nos. 50-500
and 50-501

March 2, 1979

Serial No. 2-141

LOWELL E. ROE

Vice President
Facilities Development
(419) 259-5242

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Case:

Under separate cover, we are transmitting Amendment No. 26 to the Application for Licenses for the Davis-Besse Nuclear Power Station Units 2 and 3. This Amendment consists of Revision No. 22 to the Preliminary Safety Analysis Report (PSAR). Three (3) original and twenty-five (25) conformed copies of the transmittal sheet are included together with sixty (60) copies of Revision No. 22 to the PSAR.

Revision 22 to the PSAR is being submitted to incorporate the PSAR changes which have been committed in letters since the Revision 21 submittal of September 22, 1978 and the PSAR changes discussed relating to main steam and feedwater line breaks outside containment. The "PSAR Revision Contents" identifies the associated letter for a given PSAR change in this revision.

Yours very truly,

LER/TJM

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APPLICATION FOR LICENSES

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNITS NO. 2 and 3

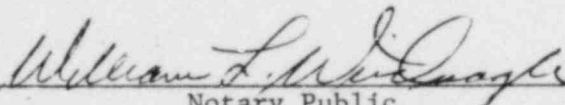
Docket Nos. 50-500
and 50-501

Amendment No. 26

Enclosed herewith amending the above Application are sixty (60) copies of Revision No. 22 to the Preliminary Safety Analysis Report.

By 
Vice President, Facilities Development

Sworn to and subscribed before me, this 28th day of February, 1979.


Notary Public

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WILLIAM L. WINDNAGLE
Notary Public, State of Ohio
My Commission Expires October 24, 1980

DIRECTIONS FOR INSERTING REVISION 22, MARCH 1979

During insertion of the revised pages, a dash (-) in the remove or insert column of the directions means no action is required. Asterisks (*) at the left of the instructions indicate an additional clarification at the end of the directions. The page revision indexes go in front of the applicable chapter.

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<u>Outstanding Item No.</u>	<u>Subject</u>	<u>TECo to NRC Letter Serial No./Date</u>	<u>PSAR Pages Affected</u>
14 of NRC Letter dated 2-3-78	Grouting	2-123 / 6-6-78	2C-87b(1)
3 in SER and NRC Letter dated 3-17-78	Protection of safety-related equipment against main steam and main feedwater line breaks out- side containment.	2-119 / 5-15-78 Meeting of 8-31-78, and Telecons Week of 2-19-79	3.2-11 3.2-12 3.6-2 3.6-5d 3.6-5g through 3.6-6b 10.3-1 10.3-5 10.4-18 Fig. 10.3-1
5 in SER	Cold Shutdown Criteria	2-138 / 12-15-78	5.5-12 5.5-12a 10.3-2 through 10.3-5
8 in SER	Hydrogen Recombiners	2-137 / 11-17-78	6-v 6.2-80 through 6.2-82 6.2-84 through 6.2-89 Fig. 6.2-9 Fig. 6.2-23b(Deleted) OI 21-1

Grout Pressures - Grout will be injected in two stages; for the lower stage, a packer will be set at approximately el 550+. Effective grout pressure at the packer will be held to no greater than 25 lb/in². The upper stage will then be grouted with an effective grout pressure at the surface packer no greater than 10 lb/in². Six borehole extensometers per station area will be used in conjunction with surface leveling to monitor rock movements during grout and ensure rock uplift or heave does occur.

18

Construction monitoring - All phases of the grouting program, including grout preparation, drilling of grout holes, injection of grout, and analysis of injection will be supervised by a resident geotechnical engineer knowledgeable in the subsurface conditions at the site. The grout holes will be logged under the direction of the geotechnical engineer by recording penetration rates, dropage of drill steel, and changes in the color of cuttings; and this data, together with grout take data, will be used to evaluate the effectiveness of the grouting program.

22

All foundation areas, including grouted and non-grouted areas, will be examined by a construction phase bedrock verification program as described below. If additional areas are determined to require remedial treatment, such treatment will be implemented following the procedures identified above.

21

The applicant will 1) notify the NRC staff when excavations which expose rock conditions are open for inspection, and 2) provide the NRC staff with a final foundation report prior to the commencement of placement of any structural foundation mats.

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i. Bedrock verification program

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(1) Introduction

Bedrock cavities at the site are expected to be limited to an upper zone above approx el 540. For those structures of Units 2 and 3 which are founded above this elevation, the bedrock will be grouted to el 540; see discussion in preceding section.

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CHAPTER 3.0

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Table 3.2-1 (cont'd)

Structure/System/Component	SAR Section	Seismic Category I
26. <u>Containment Air Cooling Units & Associated Service Water Piping Inside Containment</u>	6.2.2	Yes
27. <u>Main Steam System</u>	10.3	
Isolation valves		Yes
Piping upstream of isolation valves		Yes
Piping downstream of isolation valves		No***
Atmospheric vent valves		Yes
Safety valves		Yes
28. <u>Auxiliary Feedwater System</u>	10.4.9	
Auxiliary feedpump turbine		Yes
Auxiliary feedwater pumps		Yes
Piping and valves up to and including isolation valves		Yes
Other piping and valves		Yes
29. <u>Service Water System</u>	9.2.1	
Service water pumps		Yes
Service water strainers		Yes
Piping and valves associated with other safety related systems outside containment.		Yes
Other piping and valves		No
Ultimate heat sink transfer pumps		Yes
30. <u>Component Cooling Water System</u>		
Component cooling pumps		Yes
Component cooling surge tank		Yes
Component cooling heat exchangers		Yes

*Code and Quality group requirements will be met to the extent possible.

**Only pressure boundary meet this code.

***Piping to the auxiliary building-turbine building wall is seismic Category I.

3.6

PROTECTION AGAINST POSTULATED PIPING FAILURES

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The purpose of these criteria is to provide protection of nuclear power station equipment and structures from adverse effects due to pipe-failure accidents. It is intended to comply with 10 CFR Part 50 appendix A, General Design Criterion 4 - "Environmental and Missile Design Bases." Compliance with 10 CFR Part 100 is the overriding safety considerations for postulated pipe breaks. Protection against pipe failure effects is provided to fulfill the following design criteria:

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1. Reactor Coolant Pipe Failure

A pipe failure in a loop in the reactor coolant system pressure boundary, together with a simultaneous safe shutdown earthquake, will not cause:

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- a. Loss of integrity to another loop of the reactor coolant pressure boundary.
- b. Loss of integrity of the containment vessel.
- c. Loss of integrity of the main steam or feedwater system.

It shall not cause loss of function to systems required to mitigate the consequences of the loss-of-coolant accident, assuming the failure of a single active component.

2. Main Steam and Feedwater Pipe Failure

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A pipe failure in the main steam or main feedwater pump discharge boundary will not cause:

- a. Loss of integrity to the reactor coolant pressure boundary.
- b. Loss of integrity to unisolatable main steam or feedwater piping in another loop, considering a single active failure.
- c. Loss of integrity to the containment vessel or containment isolation system (subsection 6.2.4), if the pipe break occurs inside the containment vessel.

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It will not cause loss of function to systems required to mitigate the consequences of a steam or feedwater line break accident, assuming the failure of a single active component.

It shall not cause loss of function to the systems required to maintain the unit in a hot shutdown condition, assuming the failure of a single active component.

A pipe failure in the main steam or main feedwater pump discharge pressure boundary will not cause loss of function to the systems required to cool down the unit to at or near ambient conditions.

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3. Pipe Failure in other Systems than Reactor Coolant, Main Steam, or Feedwater

A pipe failure in systems other than the reactor coolant system, the main steam system, or main feedwater pump discharge lines shall not result in perforation of the containment vessel or control room or cause loss of integrity to the spent fuel pool. It shall not cause loss of function to any system required for hot or cold shutdown of the reactor from the control room.

In analyzing the effects of postulated piping failures, assumptions regarding the operability of systems and components will be in accordance with Section B.3.b through B.3.d of BTP APCS 3-1 (Reference 4).

The design criteria piping failures outside the containment are discussed in subsection 3.6.2.

Equipment protected from a piping failure is given in table 3.6-1. Methods of protection from postulated piping failures are given in Tables 3.6-3 and 3.6-3a for inside and outside containment respectively.

3.6.1 SYSTEMS IN WHICH DESIGN BASIS PIPING FAILURES OCCUR

3.6.1.1 High-Energy Fluid Systems Within the Containment Vessel

Pipe breaks within the containment vessel are postulated in accordance with NRC Regulatory Guide 1.46 with the exceptions noted in Subsection 3.12.46.

3.6.1.2 High-Energy Fluid Systems Outside Containment

The determination of high energy fluid systems outside the containment vessel is in accordance with Branch Technical Position APCS 3-1 (Reference 4).

Analysis for high energy piping breaks is required if either of the following conditions is met during normal reactor operation:

1. Maximum operating temperature in the line is higher than 200F.
2. Maximum operating pressure of the line is higher than 275 psig.

Dynamic effects must be considered for any line which exceeds 275 psig operating pressure except for piping in the "no break" zone. The environmental effects (i.e., compartment pressure, humidity, temperature, etc.) must be considered for any line with an operating temperature greater than 200F. The dynamic and environmental effects will be discussed in the FSAR.

Table 3.6-3a lists those systems assumed to be high energy systems.

3.6.2 DESIGN BASIS PIPING BREAK CRITERIA

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3.6.2.1 Piping Break Locations

3.6.2.1.1 Inside Containment Vessel

Break locations for piping inside the containment vessel are postulated in accordance with NRC Regulatory Guide 1.46 with the exceptions noted in subsection 3.12.46 and the criteria of this section.

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3.6.2.1.1.1 Systems in Which Pipe Breaks are Postulated

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Pipe breaks are postulated to occur in high energy piping systems or portions of systems. High energy piping systems are defined as systems that have either a maximum operating temperature exceeding 200 F or a maximum operating pressure exceeding 275 psig. Maximum operating temperature and pressure are defined as the maximum temperature and pressure in the piping systems during normal plant conditions; i.e., startup, operating at power, hot standby or reactor cooldown to the cold shutdown condition.

3.6.2.1.1.2 Pipe Break Criteria

Primary system break locations and types are postulated in accordance with subsection 3.6.2.1.1.3. A listing of RCS break locations is provided in Table 3.6-2. All postulated breaks are systematically analyzed for potential damage resulting from pipe whip, jet impingement, and subcompartment pressurization. If the damage is unacceptable in terms of relevant protection criteria, protective measures are taken such as installation of pipe whip restraints and/or jet impingement shields. Rerouting of piping, relocation of equipment, or providing additional enclosures to avoid the need for protective measures are considered.

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3.6.2.1.1.3 Break Locations for Primary System Piping

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Breaks in ASME Code, Section III, Class 1 piping are postulated to occur at the following locations in each piping or branch run:

- a. At terminal ends of the pressurized portions of the runs, and
- b. At intermediate locations between terminal ends selected by either of the following criteria:
 1. At each fitting (e.g., tee, cross, flange, and nonstandard fitting), welded attachment, and valve; or
 2. At locations where the maximum stress ranges for normal and upset plant conditions and for operating basis earthquake (OBE) exceeded $2.4 S_m$ (1) calculated by both Eq. (10) and either Eq. (12) or Eq. (13) in Paragraph NB-3653 of the ASME Code, Section III; and

at locations where the cumulative usage factor $U^{(2)}$ derived from the piping fatigue analysis, under the loadings associated with OBE and operational plant conditions, exceeds 0.1.

Where no breaks are required to be postulated by application of the above stress and usage factor criteria, at least two breaks will be postulated at separate locations selected on the basis of highest cumulative usage factor or stress intensity.

- (1) S_m is the design stress intensity as specified in Section III of the ASME Boiler and Pressure Vessel Code.
- (2) U is the cumulative usage factor as specified in Section III of the ASME Boiler and Pressure Vessel Code.

3.6.2.1.1.4 Design Basis Break Types and Break Orientation

(1) Circumferential Breaks

Circumferential breaks are postulated in high energy piping at locations specified in Section 3.6.2.1.1.3 except:

- a. For nominal pipe size of 1 inch and less.
- b. Where breaks are postulated using the stress range or usage factor criterion, circumferential breaks are not postulated when the stress in the circumferential direction is at least 1.5 times that in the axial direction.

Where break locations are selected at pipe fittings without the benefit of stress calculations, circumferential breaks are postulated at the piping weld to each fitting, valve, or welded attachment. If detailed stress analyses or tests are performed the maximum stressed location in the fitting may be selected instead of the pipe-to-fitting weld.

For purpose of calculating pipe whip and jet impingement forces, circumferential breaks are assumed to result in pipe severance and initial separation amounting to a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by elastic or inelastic analysis.

The dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump controlled flow, and the absence of energy reservoirs shall be taken into account, as applicable, in the determination of fluid jet discharge.

Pipe whipping is assumed to occur at the directions defined by the stiffness of the piping configuration and jet reaction forces, unless limited by structural members or piping restraints.

(2) Longitudinal Breaks

Longitudinal breaks are postulated in high energy piping at the locations specified in Section 3.6.2.1.1.3 except:

- a. For nominal pipe sizes smaller than 4 inches
- b. Where breaks are postulated using the stress range or usage factor criterion, longitudinal breaks are not postulated when the stress in the axial direction is at least 1.5 times that in the circumferential direction
- c. At terminal ends
- d. At intermediate locations where the criterion for a minimum number of break locations must be satisfied
- e. At branch connections where the branch is a terminal end of another piping run. In this case a circumferential break of the branch is postulated.

Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are located (but not concurrently) at two diametrically-opposed points on the piping circumference such that a jet reaction causing out-of-plane bending of the piping configuration results. Alternately, a single split may be assumed at the section of highest stress as determined by detailed stress analysis, e.g., finite element analysis.

The dynamic force of the fluid jet discharge is based on a circular break area equal to the effective cross-sectional flow area of the pipe at the break location, and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the determination of fluid jet discharge.

Pipe deflection is assumed to occur in the directions defined by the stiffness of the piping configuration and jet reaction forces, unless limited by structural members or pipe restraints.

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3.6.2.1.2 Outside Containment Vessel

Pipe break and pipe break locations are postulated in accordance with the criteria stated in Sections B.1.a through B.1.e of BTP MEB 3-1 (Reference 5) with the clarifications noted hereafter. These clarifications are generally in response to the NRC letter of March 17, 1978, on the subject of protection from a high energy main steam or feedwater line failure.

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Concerning the steam and feedwater piping from the containment penetration to the first restraint beyond the main steam isolation valve (MSIV) or main feedwater isolation valve (MFIV) (this restraint may be a simple anchor, or if valve operability is required, a moment limiting restraint), the following applies:

- a. The piping, restraints, anchors, and valves will be seismic Category I in accordance with the discussion in subsection 3.12.29. The seismic analysis will be extended to the restraints. Quality assurance is adequately met, as discussed in subsection 3.12.29, without including the system between the isolation valve and the restraint on the "Q" list (See PSAR Section 17.1).

The applicant takes exception to the requirement that Quality Group B extend between the isolation valve and the restraint. The Quality Group B boundary will terminate at the isolation valve, inclusive of the valve. This is in compliance with the requirements of Regulatory Guide 1.26 and Section B.2.C (4) of BTP APCSB 3-1. The latter allows a classification change at the valve interface if the restraint is located at the valve. The restraint is considered to be at the valve since it will be located as close to the valve as possible. Between the isolation valve and the restraint the system is designed to meet the requirements of ANSI B 31.1 (Quality Group D) and is augmented by the following additional requirements.

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1. Material certification will be required.
 2. All circumferential and longitudinal pipe welds will be subjected to augmented inservice inspection (100% volumetric inspection) in accordance with Section B.2.d of BTP APCSB 3-1.
 3. The restraints for this portion of the system will be designed as if the system were nuclear class(ASME code), with the quality assurance, inspection, and tracability consistent with the discussion in PSAR subsection 3.12.29.
- b. The piping between the containment vessel and the first restraint beyond the isolation valve will meet the low stress limits of Section B.1.b of BTP MEB 3-1 as required by Section B.2.c of BTP APCSB 3-1 and will therefore qualify as a no-break zone. In addition, this piping will be designed such that the structure will maintain its integrity when subjected to the environmental

effects of a non-mechanistic pipe break, i.e., a break equivalent to the flow area of a single ended guillotine break of a steam or feedwater line without pipe whip or jet forces. In addition, any affected equipment that is required for safe shutdown will be capable of operating in the resulting environment or flooding conditions. In determining the conditions resulting from this assumed break, vent areas will be considered as required, as allowed by Section B.2.c(2) of BTP APCSB 3-1.

Concerning (1) the steam piping from the first restraint beyond the MSIV to the anchor beyond the control room or cable spreading room wall adjacent to the turbine building, and (2) the main feedwater piping from the first restraint beyond the MFIV to the first anchor outside the auxiliary building, the following applies:

- a. The subject piping will be seismic Category I.

The requirements of ANSI B31.1 (Quality Group D) are met, and augmented by the following additional requirements:

1. Material certification will be required.
2. All circumferential and longitudinal pipe welds will be subjected to augmented inservice inspection (100% volumetric inspection) in accordance with Section B.2.d of BTP APCSB 3-1.
3. The restraints for this portion of the system will be designed as if the system were nuclear class (ASME code), with the quality assurance, inspection, and traceability consistent with the discussion in subsection 3.12.29.

- b. Mechanistic breaks will be postulated according to the criteria of Section B.1.d(1) of BTP MEB 3-1, i.e., at the terminal ends and at each location where the stresses exceed $0.8 (1.25 S_A)$, but at not less than two separated locations chosen on the basis of highest stress. Limited pipe displacement and, therefore, limited break area will be considered as allowed by Sections B.3.a(3) and B.3.a(4) for circumferential breaks. For longitudinal breaks, piping movement may be limited as allowed by Section B.3.b(5).

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If the above breaks are not in the vicinity of a structure separating the high energy steam or feedwater line from an essential component required for shutdown after the pipe break, the applicant will postulate additional pipe breaks, including pipe whip and jet forces, according to the following criteria. For postulated breaks in the vicinity of structures separating the high energy steam or feedwater line from an essential component required for shutdown after the pipe break, the circumferential break(s) producing the greatest effect at the structure will be postulated. The break(s) producing the greatest effect will be that (those) break(s) between two restraints which produce(s) the greatest effect on the structure. The structure, in conjunction with the pipe whip restraint system, will be designed such that the shutdown function can be assured. Limited pipe displacement will be considered as allowed by BTP APCSB 3-1, Section B.2.b(1). The postulated circumferential breaks will be determined without regard to stress criteria.

Any affected equipment which is required for safe shutdown will be capable of operating in the resulting environment or flooding condition. In determining what equipment must be protected and environmentally qualified, redundant equipment or backup equipment will be assessed as discussed in subsection 3.6.2.3. In determining the conditions resulting from the assumed break, vent area will be considered as required, as allowed by Section B.2.c(2) of BTP APCSB 3-1.

Protection of the spent fuel pool, which is not required for shutdown, will be provided such that a postulated mechanistic break does not result in damage to spent fuel in the pool or to the integrity of the pool structure.

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Through-wall leakage cracks and the locations of through-wall leakage cracks are postulated in accordance with the criteria stated in Sections B.2.a through B.2.e of BTP MEB 3-1 (Reference 5) with the clarification that the requirement of Section B.2.c.i, which requires that cracks be postulated to occur individually at locations that result in the maximum effects from fluid spraying and flooding, is applicable only to piping designed to non-seismic Category I standards.

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3.6.2.2 Types of Breaks and Leakage Cracks

Once a design basis piping failure location has been established, as defined above, the types of breaks and through-wall leakage cracks that are postulated are in accordance with the criteria stated in Sections B.3.a through B.3.c of BTP MEB 3-1 (Reference 5).

Revision 16
July 1977

3.6.2.3 Analysis

In the analyses of postulated piping failures, the following assumptions are used:

- a. If the postulated pipe failure could result in separation of the turbine generator from the power grid, offsite power is assumed unavailable unless the assumption of the loss of offsite power is not conservative regarding shutdown capability.
- b. If the postulated pipe failure requires safety system response to the event, the analysis assumes a single active component failure in either the safety systems required to mitigate the consequences of the event or their auxiliary supporting features. This single active failure is in addition to the postulated pipe failure with its accompanying effects.
- c. Operator action credited to mitigate the consequences of the postulated pipe failure is analyzed for each specific event. The feasibility of initiating operator actions on a timely basis, as well as the accessibility provided to allow the operator actions, is demonstrated.
- d. In the analysis of pipe failures, the use of all plant systems, including nonseismic Category I systems, in bringing the plant to a cold shutdown condition, is allowed when the failure is in a seismic Category I system. When the failure is in a non-seismic Category I system, the use of only seismic Category I systems is allowed in the analysis.
- e. Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose, moderate-energy essential system, i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure, single failures causing loss of the other train or trains of that system are not assumed provided the system is designed to seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems. Examples of dual-purpose essential systems are service water system, component cooling water system, and decay heat removal system.
- f. Section B.2.c.(1) of BTP APCSB 3-1 requires that the steam and feedwater containment isolation valves be operable (capable of closure) after a high energy line break either upstream or downstream of the valve. The no-break postulation precludes the necessity of protecting the

valve from a break in the low stress zone. To mitigate the effects of feedwater pipe failure upstream of this zone, the feedwater isolation valve will be designed to close when subjected to the resulting environmental conditions and the loads due to the break. For the steam line isolation valves, operability of the isolation valve on the affected (broken) steam line is not required. Decay heat removal can be accomplished using the steam generator associated with the unaffected steam generator, even assuming a single active failure. The steam lines do not connect upstream of the turbine stop valves. Should the isolation valve on the unaffected steam line be assumed to fail to close, the turbine stop valves would isolate the unaffected steam generator. Credit could also be taken for the nonreturn valve on the affected steam line which will be located in a separate compartment from the isolation valve on the affected steam line.

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in the auxiliary building, the non-return valve in the affected main steam line will limit blowdown of the unaffected steam generator until the turbine stop valves and the unaffected main steam line isolation valve can close. The main feedwater flow to both steam generators will also be shut off by initiating closure of the main feedwater isolation valves.

In addition to the automatic closure, the MSLIV have the capability of being remote manually closed. The manual closure capability will also provide a means of testing the MSLIV during normal operation and isolating the main steam lines for maintenance purposes.

The main steam lines are equipped with a radioactive monitoring system with sensors located just downstream of the MSLIV. In the event of a steam generator tube failure, the sensors will activate an alarm in the main control room. Upon receipt of the alarm, the operator will then have the capability of isolating the faulty steam generator if so required.

The MSLIS components are qualified to serve in the environment specified in section 3.11. The radiation qualification level is given in Table 3.11-2. At this level of radiation, the MSLIS components will perform their intended functions.

5.5.5.4 Evaluation

The MSLIS is designed to provide steam generator and containment vessel isolation under the following conditions:

1. A rupture of a main steam line inside and outside the CV.
2. A rupture of a main feedwater line inside and outside the CV.
3. A steam generator tube failure.
4. Normal operation.
5. Maintenance.

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Refer to System Operation for a description of how the above conditions affect the initiation of the MSLIS.

The provisions taken to assure the integrity of the MSLIV under the dynamic forces resulting from a quick inadvertent closure under maximum flow conditions are described in subsection 3.9.2.

The MSLIS is designed to seismic category I requirements as presented in section 3.7.

The MSLIV are designed to fail in the closed position.

5.5.5.5 Tests and Inspections

The MSLIS is periodically tested and inspected in accordance with the applicable codes and regulations. A definite program of tests and inspections will be provided in the FSAR.

5.5.6 REACTOR CORE ISOLATION COOLING SYSTEM

(Not Applicable to B&W NSSS)

5.5.7 DECAY HEAT REMOVAL SYSTEM

The capability to transfer heat from the reactor to the environment during the transition from normal reactor operating conditions to the decay heat removal system cut-in conditions, using only safety grade systems, and assuming only offsite or onsite power is available and the most limiting single failure (taken to mean single active failure) has occurred, is discussed below.

A second modulating atmospheric vent valve in each steam line, as discussed in Section 10.3, is intended to provide the capability, assuming the most limiting single failure with only onsite or offsite power available and a safe shutdown earthquake, to cool down to the decay heat removal system cut-in conditions within the time frame discussed below.

In order to demonstrate the capabilities of the total unit design, the design will be analyzed to confirm the capability to depressurize, borate, and cool down to the decay heat removal system cut-in conditions using only safety grade equipment with only onsite or offsite power available, and assuming a single active failure. Manual operation of pneumatic valves will be assumed since there is no safety grade air supply system. Remaining at hot shutdown for the time necessary to correct single failures in systems required for depressurization or boration will be assumed. Also, manual actuation of valves outside the control room, with and without single active failures, will be assumed. No repairs will be allowed to correct a mechanically induced single active failure of an atmospheric vent valve. This analysis will also confirm that sufficient boration capability exists to maintain the reactor coolant system at $1\Delta k/k$ subcritical margin. When considering boration requirements, the highest worth control rod is assumed to be at the fully withdrawn position. The analysis will also determine the time required to attain the decay heat removal system cut-in conditions considering the present design and the additional atmospheric vent valves. Should the analysis show that cooldown cannot be accomplished in approximately 36 hours, a cost-benefit analysis will be performed to determine the impacts of additional hardware changes required to attain approximately 36 hours cooldown. The results of the cost-benefit analysis will be presented to the NRC for its review.

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Davis-Besse Units No. 2 and 3 will reference a planned first-of-a-kind prototype steady state natural circulation qualification test to provide assurance of adequate boron mixing and natural circulation.

Specific procedures for cooling down using natural circulation will be provided at the Operating License review state.

A seismic Category I auxiliary feedwater supply is provided as discussed in subsection 10.4.9.2.1, which will permit cooldown to the DHR cut-in conditions, assuming the most limiting single active failure.

The design bases, description, evaluation, tests and inspections, and radiological considerations for the decay heat removal system are discussed in Section 9.3.7.

5.5.8 REACTOR COOLANT CLEANUP SYSTEM

The design bases, description, evaluation, tests and inspections, and radiological considerations for RC cleanup systems (purification system) are discussed in Section 9.3.4.

5.5.9 MAIN STEAM LINE AND FEEDWATER PIPING

The main steam piping is described in Section 10.3 and the feedwater piping is described in Subsection 10.4.7. Inservice inspection is described in Subsection 5.2.5.

5.5.10 PRESSURIZER

5.5.10.1 Design Basis

The pressurizer is designed to maintain the RC system at saturation pressure to prevent boiling of the coolant. Design data are presented in Table 5.1-8.

The pressurizer is designed in accordance with the ASME Code Section III for Class 1 vessels. The design also considers any structural effect resulting from vibration, whether induced by operational or environmental conditions.

The pressurizer steam and water volume are sized as follows:

- a. Water volume - The outsurge following a reactor trip from full power shall not uncover the lower level sensor in the lower shell of the pressurizer and shall not actuate the high pressure injection system assuming the transient starts when the water level is at the first low level alarm point. Furthermore, the high pressure injection system shall not be actuated by any decrease in pressure that may result from the insurge of makeup water required to permit heater operation assuming that 50% of the makeup water is mixed with the saturated water in the pressurizer.

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Multiplying G by the pipe area will give the release rate in lb/sec. For conservatism, the lines were considered fully open until the valves were completely closed. The accident signals will close the isolation valves within 5 seconds following the accident.

An equation for the release rate as a function of time was found. Integrating that equation over the valve closure time yields the total release during the 5-second interval immediately following the accident. The total release was used to calculate the offsite doses according to the methods outlined in subsections 15.2.3, 15.2.4 and 15.2.5 for thyroid, beta-skin, and total body gamma, respectively. This is consistent with the dose and X/Q assumptions of Regulatory Guide 1.4.

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Results: The incremental doses at the site boundary are 0.424 rem to the thyroid, 8.66×10^{-4} rem beta-skin, and 5.02×10^{-4} rem total-body gamma. These are very small fractions of the doses due to the continuation of the accident and will not raise the total doses above 10CFR100 limits.

6.2.4.4. Tests and Inspections

The containment vessel isolation system is designed so that each of its components can be tested periodically. Leakage testing shall be done accordance with appendix J of 10 CFR part 50, Reactor Containment Testing Requirements, except for the exceptions stated in subsection 6.2.1.4.

Each isolation valve will be tested periodically during normal operation and during shutdown conditions to ensure their operability when needed. Each automatic isolation valve has a manual override for testing purposes. In addition, the safety features actuation system will be tested periodically.

All welds of pressure retaining parts of the containment vessel isolation system will undergo examination after installation to meet the requirements of ASME Code, section III, class 2 components.

All piping components of the containment vessel isolation system that are directly connected to the primary reactor coolant system will undergo periodic in-service inspection of welds in accordance with ASME Code, section XI.

All mechanical and electrical components of the containment vessel isolation system including valves, valve actuators, cables, motors, sensing elements, position indicators, etc., will be periodically tested for operational functions.

6.2.4.5 Materials

Materials in the containment vessel used in or on the safety feature system that could contribute to radiolytic or pyrolytic decomposition products are discussed in subsection 6.2.5.6.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

The containment hydrogen control system is an engineered safety feature designed to serve as the combustible gas control system in containment.

Subsection 50.44 and General Design Criterion 41 of Appendix A to 10 CFR 50 require that systems to control the concentrations of hydrogen, oxygen, and other substances, which may be released into the reactor containment, be provided as necessary following postulated accidents to assure that containment integrity is maintained.

Following a loss-of-coolant accident (LOCA), hydrogen gas may be generated within the containment vessel as a result of the following:

- a. Metal-water reaction involving the zirconium fuel cladding and the reactor coolant.
- b. Radiolytic decomposition of the post-LOCA emergency cooling solutions (oxygen also evolves in this process).
- c. Corrosion of metals and paints by solutions used for emergency cooling or containment spray.

To insure that the hydrogen concentration is maintained below the lower combustible limit, the containment hydrogen control system is provided. The hydrogen control system is composed of a containment hydrogen recombiner subsystem, containment vessel gas analyzer subsystem, and a hydrogen purge subsystem.

6.2.5.1 Design Bases

Protection of the hydrogen control system from wind and tornado effects is discussed in Section 3.3. Flood design is discussed in Section 3.4. Missile protection is discussed in section 3.5. Protection against dynamic effects associated with postulated rupture of piping is discussed in Section 3.6. Environmental design is discussed in Section 3.11.

6.2.5.1.1 Safety Design Bases

Following a loss-of-coolant accident (LOCA), hydrogen gas may accumulate within the containment vessel from various sources. If a sufficient amount of hydrogen is generated, it may react with oxygen present in the containment vessel atmosphere at rates rapid enough to lead to high temperatures and significant overpressurization of the containment vessel. As stated in NRC Regulatory Guide 1.7, the lower flammability limit for hydrogen in air saturated with water vapor at room temperature and atmospheric pressure is assumed to be four volume percent.

The combustible gas control system components are designed to be operated as necessary to maintain the maximum hydrogen concentration in the containment vessel at or below 3.5 volume percent following a LOCA. Using the conservative assumptions of NRC Regulatory Guide 1.7 and Branch Technical Position CSB 6-2, a concentration of 3.5 volume percent is reached at 38 days after the LOCA.

Following a LOCA, hydrogen mixing is provided by the containment spray system (described in Section 6.2.2.1), the containment air coolers (described in Section 6.2.2.2), and the containment internal structure design which permits convective mixing and prevents entrapment of the hydrogen.

Containment vessel hydrogen concentrations are analyzed by the redundant containment vessel gas analyzers which sample from four regions of the containment vessel. Gas analyzer indicators and alarms are provided locally and in the control room.

One hydrogen recombiner will process the containment atmosphere at a sufficient rate to maintain the hydrogen concentration in the containment vessel below 4 volume percent hydrogen following a LOCA, as required by R.G. 1.7.

The hydrogen recombiner system will perform its design function assuming a single active failure. Isolation of containment penetrations is provided in accordance with GDC 56 and NRC SRP 6.2.4.

The hydrogen recombiner subsystem and the containment vessel gas analyzer subsystem are designed to sustain all normal loads as well as accident loads, including seismic loads and temperature and pressure transients from a LOCA. These subsystems will be designed to Quality Group B and seismic Category I.

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The recombiners will be protected from damage by missiles or jet impingement from broken pipes associated with the postulated LOCA.

The recombiners will be located away from high velocity airstreams or will be protected from direct impingement of high velocity airstreams.

Portions of systems located in the containment vessel, which are used to control post-LOCA combustible gases, will be designed to remain operable in the post-LOCA environment.

The hydrogen recombiner subsystem and the containment vessel gas analyzer subsystem will be tested and inspected periodically. The gas analyzers will also be calibrated periodically.

The sharing and transportation of portable external hydrogen recombiners, if this type is selected, will be described in the FSAR. The design of the external recombiner system installation will facilitate removal and replacement of portable units.

If external hydrogen recombiners are selected, the control console will be sufficiently shielded to protect operating personnel from radiation in the vicinity of an operating recombiner.

The hydrogen purge subsystem is a backup to the containment hydrogen recombiner subsystem for control of the hydrogen concentration in the containment vessel post-LOCA.

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6.2.5.1.2 Power Generation Design Bases

The combustible gas control system has no power generation design bases.

6.2.5.2 System Design

6.2.5.2.1 Containment Hydrogen Recombiner System

The objective of the containment hydrogen recombiner system is to limit hydrogen concentrations below flammability levels inside the containment after a postulated LOCA. The containment hydrogen recombiner system performs this function by catalytically or thermally recombining hydrogen and oxygen in the containment atmosphere to form water vapor. The operation of the containment hydrogen recombiner system does not result in the discharge of radioactive materials to the environment.

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6.2.5.2.2 Hydrogen Purge System

The hydrogen purge system (nonsafety-related, nonredundant system) is used as a backup to the hydrogen recombiner system after a LOCA and will only be used in the event that a hydrogen recombiner system is not operable.

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The containment isolation portion of the system is designed to meet seismic Category I requirements.

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6.2.5.2.3 Deleted

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6.2.5.2.4 Containment Vessel Gas Analyzer System

The containment vessel gas analyzer system is designed to monitor the hydrogen concentration within the containment vessel. The system is a redundant, two-channel system which draws samples from four sample points within the containment vessel, conditions the samples, analyzes the samples for hydrogen concentration, and returns the samples to the containment vessel. Instrumentation comprising the analyzer system includes local pressure indicators to monitor the pressure of the sample, hydrogen analyzer elements, signal transmitters, local recorders, local and control room alarms for high

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level, and local and control room indicators. Containment vessel isolation valves are located on both sides of the containment for each of the four sample lines, and one isolation valve on the outside of the containment vessel for each of the two sample return lines. The containment vessel isolation valves are actuated by hand indicating switches, located in the control room.

6.2.5.3 Safety Evaluation

6.2.5.3.1 Original Safety Evaluation

The following assumptions were made for the calculation of hydrogen generation:

1. All NRC Regulatory Guide 1.7 assumptions were used in the analysis.
2. The quantity of hydrogen generated from radiolysis of the primary coolant is calculated using the methods and formulas given in Appendix A of Standard Review Plan 6.2.5. The number of lb-moles of oxygen produced from radiolysis is taken as one half of this value.
3. The metal-water reaction occurs within the first 2 minutes.
4. An insignificant quantity of H_2 is produced due to noble gases in the post-LOCA containment atmosphere.
5. An insignificant quantity of H_2 is dissolved in the coolant or trapped in the pressurizer steam space.
6. All gases which evolve are mixed uniformly throughout the containment atmosphere.
7. No recombination of H_2 and O_2 occurs.
8. The LOCA considered is a double-ended break of a hot leg reactor coolant pipe.
9. An insignificant quantity of H_2 is present in the containment atmosphere before LOCA.
10. The average power level was assumed to be 2,828 MWt (102 percent of full power).
11. Deleted

12. Pre - LOCA conditions assumed:

- a. $T = 120^{\circ}\text{F}$
- b. $P = 14.4 \text{ psia}$
- c. Relative Humidity = 70 percent (no credit is taken for post-LOCA increase in relative humidity)

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13. Since galvanized ductwork may collapse upon rapid pressurization of the containment vessel, 20 percent of the interior of the ductwork is considered to be exposed to the spray solution.

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14. Thickness of the galvanized coating on the grating is 3.5 mils.

The following sources of hydrogen have been considered, as required by Regulatory Guide 1.7:

- 1. Metal-water reaction
- 2. Radiolysis
- 3. Aluminum corrosion
- 4. Galvanizing corrosion
- 5. Paint decomposition
- 6. Aluminum paint

The assumptions, methods, and data given in SRI 6.2.5, Regulatory Guide 1.7, and Branch Technical Position CSB 6-2 were used to determine sources 1, 2, and 3 above.

The metal-water reaction is calculated using 5 times the 10 CFR 50.46 computed value of 0.64 w/F. The total weight of active cladding is taken as 42088.8 lbs. The total number of lb-moles of zirconium that reacts is:

$$5 \times 0.0064 \text{ w/f} \times 42088.8 \text{ lbs clad} \times 1 \text{ lb-mole Zr/91.22 lbs Zr} \\ = 14.765 \text{ lb-moles of zirconium reacting to form hydrogen}$$

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For each lb-mole of zirconium that reacts, 2 lb-moles of hydrogen are produced. Therefore, the total amount of hydrogen generated from this source is:

$$2 \times 14.765 = 29.53 \text{ lb-moles of } \text{H}_2$$

This reaction is assumed to occur within the first two minutes of the accident.

Free hydrogen may be liberated due to the corrosive reaction between the chemicals in the containment spray solutions and corrodible metals and paints in conjunction with elevated containment vessel temperatures. Aluminum corrosion contributes significantly to hydrogen production. Galvanizing and zinc base paints also contribute to hydrogen production early in the accident. Table 6.2-15 lists the quantity of each of the materials which may be exposed to the spray solutions. Table 6.2-16 lists the hydrogen generation rates and corrosion rates for galvanizing and zinc based paints in relation to time after LOCA.

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Paints used inside the containment vessel are listed in Table 6.2-17.

Table 6.2-15Containment Metals and Paints Subject to Corrosion by Spray Solutions

<u>Material</u>	<u>Component</u>	<u>Exposed Area (ft²)</u>	<u>Approx. Thickness</u>
Galvanizing	Conduit, cable trays, etc.	12.854	1.7 mils
Galvanizing	Ductwork	17,704*	1.0 mil
Galvanizing	Grating	59,317	3.4 mils
Zinc	Paint	229,000	3 mils
Aluminum	Various	809	.1875 inch
Aluminum	Paint	20,000	1.5 mils

*This includes an additional 20% increase in the area to account for the possibility of collapsed ducting.

Table 6.2-16Corrosion Rates of Containment Metals and Paints

<u>Material</u>	<u>Time After LOCA</u>	<u>Corrosion Rate</u>
Zinc based paints	0 - 1 hrs.	7.938×10^{-7} lb-mole of paint/ft ² -hr
	1 - 6 hrs.	2.54×10^{-7} lb-mole of paint/ft ² -hr
	> 6 hrs.	No corrosion
Galvanizing	0 - 1 hrs.	8.5×10^{-7} lb-mole of zinc/ft ² -hr
	1 - 6 hrs.	5.7×10^{-7} lb-mole of zinc/ft ² -hr
	> 6 hrs.	No corrosion
Aluminum	Continuous	1.189×10^{-5} lb-mole of Al/ft ² -hr
Aluminum-based paint	0 - 2 min.	Total consumption within 2 minutes after LOCA

Table 6.2-17
Paints Used Inside Containment Vessel

<u>Generic Type</u>	<u>Substrate</u>	<u>Manufacturer's Designation</u>	<u>Dry Sp. Gr.</u>	<u>Area Ft²</u>	<u>Curing</u>
Epoxy zinc chromate primer	Steel	Dupont Corlar 825-8031	1.5	1000	Polyamide catalyzed air drying
Inorganic zinc primer	Steel	Carbo-Zinc 11 or, Dimetecote 6 or, Mobilzinc 7	3.66	228,000	Self curing air drying
Epoxy surfacer	Concrete	Amercoat Naklad 110AA	2.24	108,000	Catalyzed air drying
Epoxy topcoat	Concrete and/or primed steel	Phenoline 305	1.44	224,000	Catalyzed
		or, Amercoat 66 or, Mobil 89 series	1.44	108,000	Air drying
Modified phenolic coatings	Unprimed steel	Phenoline 368	1.44	3,000	Catalyzed air drying
Aluminum Heat resisting (1200F)	Unprimed steel	Red Spot-Federal Regulation TT-P-28d (April 26, 1967)	1.56	20,000	Air drying curing at 400F full hardness

DE-2,3

5.2-85

The generation rates of hydrogen from paint coatings and galvanizing listed in Table 6.2-16 are strongly dependent on the ambient temperature. ORNL has performed several tests and confirmed that dependence. Conversations with the lead investigator, Dr. H. E. Zittel, revealed that below 100 C the generation rate of hydrogen from the two sources, i.e., painted surfaces and galvanizing, became negligible. Upon examining the temperature history for the DBA for Davis-Besse Units 2 & 3 (see Figure 6.2-6 of the PSAR), it was found that the containment temperature dropped below 100 C after 6 hours, thus the generation of hydrogen from paint surfaces and galvanizing become negligible after that time. Prior to 6 hours, the generation rates stated in Table 6.2-16 have been used. The higher generation rates during the first hour account for the higher temperatures existing in the containment during that time period. The surface areas of paint and galvanizing are the same as those presented on Table 6.2-17 of the PSAR.

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Aluminum paint was considered as a source of hydrogen and the hydrogen evolved from that source was assumed to be evolved at the beginning of the accident. There are 20,000 ft.² of aluminum paint containing 0.5 pounds of aluminum per 300 ft.² of paint. Since there will be 0.0555 lb-moles of hydrogen evolved per pound of aluminum, there will be 1.85 lb-moles of hydrogen evolved from that source.

The contribution to the hydrogen concentration from each source is shown in Figure 6.2-21. Analysis indicates that the hydrogen recombiner is not required to be started until approximately 46 days after the accident when the hydrogen concentration in the containment reaches approximately 3.5 volume percent.

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Doses due to purging the entire containment atmosphere contents at various times were calculated. No credit for charcoal filtration is assumed. The X/Q's utilized were as follows:

1. Site Boundary: The annual average site boundary X/Q for the worst sector ($2.5 \times 10^{-6} \text{ sec/m}^3$).
2. LPZ: The annual average LPZ X/Q for the worst sector ($4.27 \times 10^{-7} \text{ sec/m}^3$).

Figure 6.2-22 shows the site boundary doses for an instantaneous purge of the entire atmospheric contents of the containment at any time after 30 days following the LOCA. Likewise, Figure 6.2-23A shows the LPZ purge doses. The contributors to the whole body and skin doses are long-lived isotopes, principally Kr-85, with a half-life of 10.73 years. The thyroid dose rapidly decreases due to the approximate 8-day half life of the contributor, I-131.

Containment entry would be required at some time after the accident. The containment would not be bottled up for a time necessary for the long-lived isotopes to decay substantially. The offsite doses due to purging for entry and/or opening the containment for access would be the same as the dose received from the postulated release through the combustible gas control system purge subsystem.

DB-2,3

Table 6.2-17A is Deleted

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Table 6.2-17B is Deleted

Table 6.2-18 is Deleted

The site boundary thyroid dose and whole body dose after 140 days are both well below the annual doses allowed from normal operation of 15 mrem and 5 mrem, respectively. The calculated dose due to post-LOCA purging, based on an improbable worst case LOCA with the associated very conservative core melt-down assumptions, is a small fraction of the unit overall dose. The purge dose definitely will not add to the radiological impact of the unit. The combined purge and LOCA doses are within 10 CFR 100 guidelines.

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The integrated activity released from the containment over a period of time Δt from t to $t + \Delta t$ is given by:

$$IAR_i = \left\{ [a + b(1 - \eta)] \lambda_{ie} + (1 - \eta_p) \lambda_p \right\} A_i(t) \frac{1}{\lambda_T^i} \left(1 - e^{-\lambda_T^i \Delta t} \right)$$

λ_{ie} = removal constant due to containment leakage

a = direct unfiltered function of λ_{ie}

b = direct filtered function of λ_{ie}

η = filter efficiency

η_p = purge filter efficiency

λ_p = removal constant due to purging

λ_d^i = radioactive decay constant

t = time at which H_2 purge is initiated

Δt = duration of H_2 purge

$A_i(t)$ = activity inventory at time t

$$\lambda_T^i = \lambda_{ie} + \lambda_d^i + \lambda_p$$

The doses are then given by:

$$\text{Skin dose: } D_{\text{skin}} = \sum_{\text{all isotopes}} D_{\text{skin}}^i = \sum_{\text{all } i} (IAR_i) (X/Q) (DCF_i)_{\text{skin}}$$

$$\text{Total body: } D_{\text{TB}} = \sum_{\text{all isotopes}} D_{\text{TB}}^i = \sum_{\text{all } i} (IAR_i) (X/Q) (DCF_i)_{\text{TB}}$$

$$\text{Thyroid dose: } D_{\text{THY}} = \sum_{\text{all halogens}} D_{\text{THY}}^i = \sum_{\text{all } k} (IAR_k) (X/Q) (DCF_k)_{\text{THY}} (BR)$$

$(DCF_i)_{\text{skin}}, (DCF_i)_{\text{TB}}, (DCF_k)_{\text{THY}}$ = dose conversion factors

(BR) = breathing rate $(3.47 \times 10^{-4} \text{ m}^3/\text{sec})$

It was conservatively assumed that upon purge initiation the entire atmosphere activity in the containment is released.

6.2.5.3.2

Reassessed Safety Evaluation

The sources of hydrogen evolution have been reassessed in order to more adequately justify the conclusions reached by the Applicant. See Outstanding Item No. 21.

With respect to the experimental data supporting the corrosion rates and evolution of hydrogen for the galvanized materials and zinc based paints, the following is presented. Research conducted at Oak Ridge National Laboratory by Dr. H. E. Zittel on corrosion of zinc based paint indicates that hydrogen generation became negligible below 100C. Hydrogen generation was assessed using a temperature-time history of 300F (149C) for 5 minutes, 295F (140C) for 1 hour 45 minutes, and 225F (107C) for 22 hours 15 minutes. When the hydrogen generation for this temperature-time history was compared with hydrogen generation due to previous testing for longer exposure times, it was concluded that additional exposure did not result in significant amounts of evolved hydrogen. It should be noted that at all times the above temperature-time history is much more severe than the conditions postulated for Davis-Besse Units 2 and 3. The post-LOCA temperature for Davis-Besse Units 2 and 3 will decrease below 220F within approximately 6 hours rather than the 24 hour period of the test. The results of the Zittel testing clearly indicate that hydrogen generation below 100C is negligible.

While the Applicant firmly believes that the analysis reported in section 6.2.5.3.1 is valid, a second case has been evaluated in which certain variations have been introduced. This case is discussed below.

For zinc based paint thicknesses corresponding to that for Davis-Besse Units 2 and 3, the total experimental hydrogen yield was found to average $1\text{cm}^3/\text{cm}^2$ (H.E. Zittel, Nuclear Technology, Vol. 17, Feb. 1973). It should be noted that the manufacturer of the paint to be utilized at Davis-Besse quotes in his literature the above value of hydrogen yield obtained by Dr. Zittel for the subject paint. Should one assume that this hydrogen yield would be experienced essentially instantaneously (within the first 2 minutes after a LOCA), the total hydrogen calculated during this time for Davis-Besse Units 2 and 3 would be 21 lb-moles. This is a conservative method of determining the hydrogen generated from the zinc based paints. The value of 0.5 lb-moles resulting from the analysis presented in section 6.2.5.3.1 is considered more realistic.

It should be noted that Dr. Zittel's work was performed at a pH of 9.3. The pH values as a function of time for Davis-Besse Units 2 and 3 have been evaluated in section 6.2.3.2 for a variety of scenarios involving single failures. Under worst conditions, the spray pH for the first three hours after a postulated LOCA are calculated to range from 9.5 to 10.8 and then settle down to a value of 8.5.

Early in the accident when the temperature is high a maximum corrosion rate for galvanized material of 8.5×10^{-7} lb-mole/ft²-hr has been used, as reported in section 6.2.5.3.1. This rate has previously been utilized for other plants, such as Calvert Cliffs and St. Lucie. Although section 6.2.5.3.1 reflects the Applicants belief that hydrogen produced as a result of corrosion of galvanized material is negligible below approximately 200F, in order to be conservative a corrosion rate was applied to zinc coated surfaces even at temperatures below 200F. Accordingly, based on research performed by Westinghouse, a value of 2.41×10^{-8} lb-mole/ft² hr has been used for a long term hydrogen evolution rate. This research was done at a pH of 9.3 and a temperature of 200F. This value has been quoted in Westinghouse topical report WCAP-7153-A, which has been approved by the NRC staff. This conservative method of determining the hydrogen generated from galvanized material results in a total of 17 lb-moles of hydrogen at 300 days. The more appropriate value resulting from the analysis presented in section 6.2.5.3.1 was 0.38 lb-moles.

In accordance with Branch Technical Position CSB 6-2 the hydrogen generation analysis has been re-evaluated using corrosion rates for aluminum that were adjusted upwards to account for the higher predicted temperature and pH conditions expected earlier in the accident. The rates used are temperature dependent and range from a maximum of 10⁴ mils/year to a minimum of 200 mils/year. The high rate is based on a conservative application of the information in the Westinghouse topical report WCAP-7153-A, which has been approved by the NRC staff. In re-evaluating the contribution due to aluminum in the hydrogen generation analysis, the Applicant decided to change the aluminum ladder as discussed below. The amount of aluminum for the re-analysis therefore decreased from 809 ft² reported in section 6.2.5.3.1 to 534 ft², approximately a reduction of one-third.

The analysis has also been expanded to consider the effect of radiation. H₂ generated due to absorption of radiation by the organic paint topcoat inside the containment (229,000 ft² @ 3 mils thick) was taken as 2 cm³ per gram of paint at STP conditions (corresponds to a total dose of 10⁹ Rads). This information was obtained from Nuclear Engineering and Design--Parkinson and Sisman (1971) pp 247-280, North-Holland Publishing Co. Total hydrogen generated was determined to be 0.68 lb.-moles.

All surface areas used in the analysis are very conservative. This conservatism occurs for the following reasons:

1. All areas, as a minimum, were derived by adding quantities shown on the design drawings.
2. In some cases a safety factor was used to allow for system revisions.
3. The hydrogen evolution analysis assumed 100% coverage of the materials with the spray solution. This is clearly impossible since much of the areas used in the analysis will be shielded from the spray. According to WCAP-7153-A the corrosion rate when exposed to the vapor phase is less than 10% of the rate when exposed to the spray. We recognize that condensation on the galvanized duct work surfaces may result in a generation rate higher than 10% for the vapor phase. Therefore, for conservatism, 100% coverage was used in the reanalysis.

For specific types of materials:

a) Paint areas

Actual surface area painted = 196,000 sq. ft.

Surface used for analysis = 229,000 sq. ft.

Safety margin = 17%

Additionally the top part of the containment dome will not be wetted. This is a disk 34 feet in radius or nearly 3600 sq. ft. area. Thus the maximum possible wetted area will be 192,400 sq. ft. Thus a 19% safety margin on wetted area was used.

b) Galvanizing areas

Galvanized areas were tabulated as follows:

	<u>Sq. Ft.</u>
Ductwork actual outside surface	= 9,887
10% allowance for revisions	= 988
	= <u>10,875</u>
Inside Surface Area = Outside Area	= <u>10,875</u>
1. Total ductwork area	= 21,750
2. Conduit and cabletrays, inside and outside	= 33,250
3. Grating, all surfaces	= 27,292
4. Grating supports and ladders	= <u>13,646</u>
Total items 1 through 4.	= 95,938

A significant percentage of these areas will not be wetted by the spray. The above areas were used in the reanalysis.

c) Aluminum

Aluminum is the greatest contributor to hydrogen generation of all the metals. Accordingly an effort was made to eliminate aluminum from the containment whenever possible. As a result of this evaluation it was determined that an inspection ladder for the refueling canal will be fabricated from galvanized steel instead of aluminum as previously planned. This reduces the area of aluminum in the containment from 809 sq. ft. to 534 sq. ft. as noted earlier.

Upon re-evaluation of the parameters utilized in the hydrogen generation analysis, as discussed previously, it was found that the results did not change appreciably and that the conclusions reached in the analysis reported in section 6.2.5.3.1 remain the same. The time at which the hydrogen concentration reaches 3.5v/o is calculated to be 38 days after the LOCA as compared to 46 days reported in section 6.2.5.3.1.

Another refinement was introduced into the conservative evaluation. The additional case considers the actual weight of the 534 ft² of aluminum subject to corrosion. This allows the evaluation of the point at which no more hydrogen can be generated from aluminum corrosion because of the finite quantity available. The time at which the hydrogen concentration reaches 3.5 v/o is unchanged (38 days) since at this time the aluminum corrosion has not terminated.

The sensitivity of the analysis to the areas of corroding metals and the metal based paints was also evaluated. Since the hydrogen generation rate is directly proportional to the area exposed, a 10% increase in area would result in a 10% increase in the amount of hydrogen generated from that source. Increase in individual areas by 10%, all at the same time, leads to a 2% increase in the total hydrogen evolved through all modes. The number of days when 3.5 v/o is reached, however, remain unchanged.

After the re-analysis described here, it is concluded that the results reported in section 6.2.5.3.1 are valid. The variations in parameters used in the re-analysis do not result in significant effects on the times when the hydrogen recombiner system is needed. The radiological assessment of section 6.2.5.3.1 remains unchanged.

6.2.5.4 Testing Inspections

The combustible gas control systems will be given a preoperational test before the station produces power and may be tested any time during normal station operation for operability and performance. The equipment, piping, valves and instrumentation are arranged so that all items can be visually inspected, except for components inside the containment vessel.

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6.2.5.5 Instrumentation Application

The combustible gas control systems are designed for manual operation. The systems will be initiated from individual control switches when the containment vessel H_2 content reaches the predetermined value after a LOCA. 22

The H_2 content will be determined by two redundant gas analyzer systems external to the containment vessel. The analyzer system will result in an alarm on excessive H_2 concentrations.

6.2.5.6 Materials

Major components of the Combustible Gas Control System (CGCS) located within the containment vessel are constructed of stainless steel. Commercial names and quantity will be supplied with the FSAR. Such materials are selected to resist decomposition and corrosion, must function under the environmental conditions given in Section 3.11, and must comply with the ASME Code, Section III.

Major components of the CGCS located outside the containment vessel are constructed of materials that ensure component function under the environmental conditions given in Section 3.11 and must comply with Section III of the ASME Code.

Parts of major components located within the containment vessel such as seals, cables, wires, and instrumentation must resist radiolytic and pyrolytic decomposition, and their materials are selected to function under the environmental conditions given in Section 3.11. Commercial names and quantity will be supplied with the FSAR.

Parts of major components located outside the containment vessel such as seals, cables, wires, and instrumentation are constructed of materials that function under the environmental conditions given in Section 3.11. Commercial names and quantities of materials used in the CGCS will be supplied with the FSAR.

The containment hydrogen recombiner system pipework and valves will comply with quality group B. Where other components are not available in quality group B, they will meet the highest and best quality standards at the time of purchase. 8 22

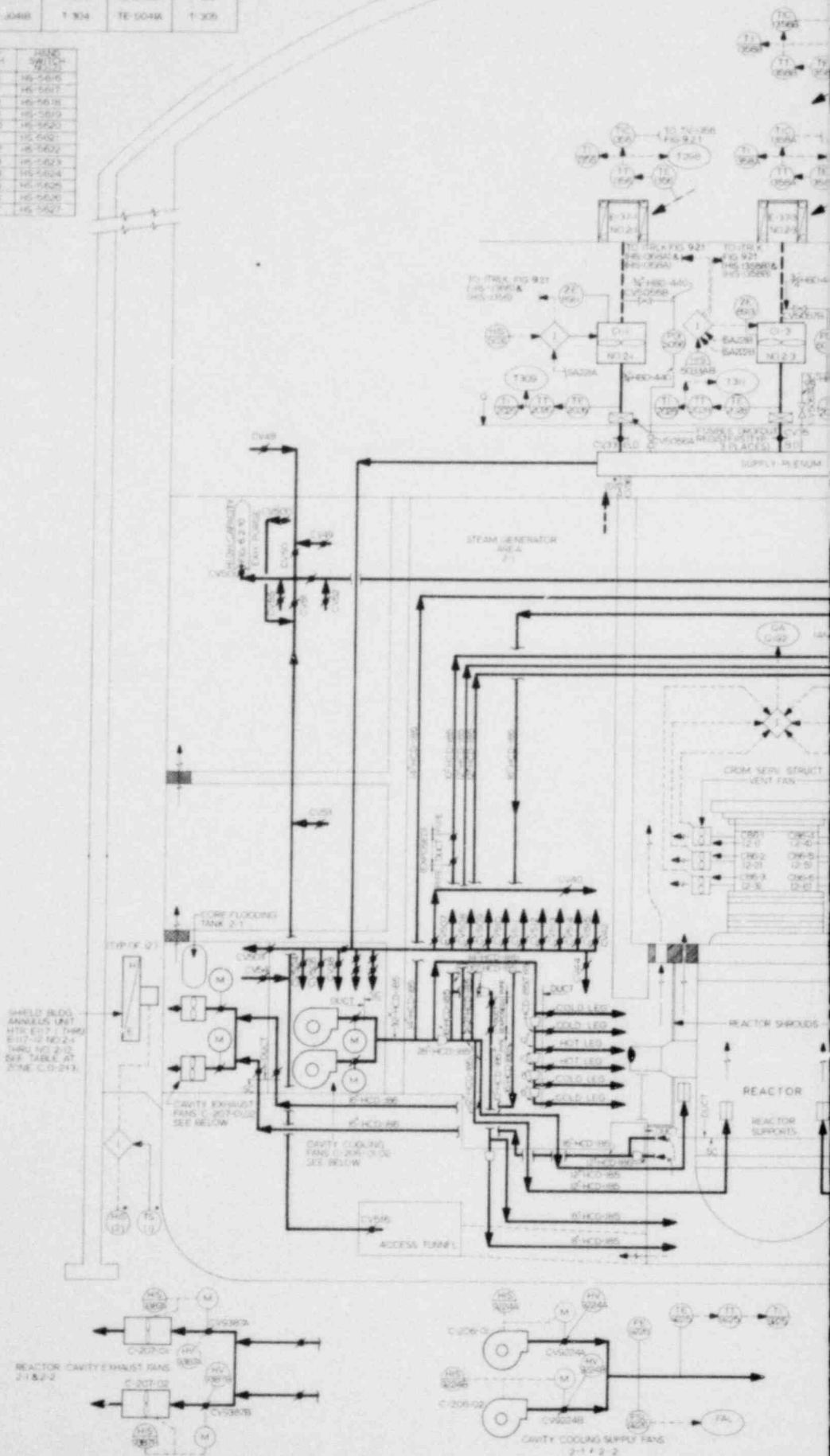
TABLE 6.2-19 (Sheet 1)

BLOWDOWN DATA FOR 4.24 ft² (0.6A) Hot Leg Break

TIME IN SEC	TOTAL INT MASS LBH	TOTAL INT ENERGY BTU	TOTAL MASS RATE LBH/S	TOTAL ENERGY RATE BTU/S
0.	0.	0.	0.	0.
.105000E-02	.548028E+02	.343006E+05	.521931E+05	.326672E+08
.203000E-02	.972081E+02	.607927E+05	.432707E+05	.270328E+08
.301000E-02	.137794E+03	.861372E+05	.414140E+05	.258618E+08
.406000E-02	.181504E+03	.113434E+06	.416204E+05	.259971E+08
.504000E-02	.223808E+03	.139863E+06	.431674E+05	.269681E+08
.602000E-02	.267946E+03	.167451E+06	.450393E+05	.281508E+08
.707000E-02	.317104E+03	.198108E+06	.468173E+05	.292730E+08
.805000E-02	.363766E+03	.227371E+06	.476145E+05	.297778E+08
.903000E-02	.410210E+03	.256415E+06	.473915E+05	.296370E+08
.100100E-01	.455827E+03	.284937E+06	.465482E+05	.291045E+08
.110600E-01	.503596E+03	.314798E+06	.454945E+05	.284392E+08
.120400E-01	.547321E+03	.342126E+06	.446166E+05	.278853E+08
.130200E-01	.590571E+03	.369155E+06	.441332E+05	.275802E+08
.140700E-01	.636730E+03	.398000E+06	.439612E+05	.274715E+08
.150500E-01	.679801E+03	.424915E+06	.439498E+05	.274645E+08
.160300E-01	.722914E+03	.451856E+06	.439923E+05	.274915E+08
.170100E-01	.766034E+03	.478803E+06	.440002E+05	.274966E+08
.180600E-01	.812155E+03	.507624E+06	.439245E+05	.274408E+08
.190400E-01	.855040E+03	.534423E+06	.437606E+05	.273453E+08
.200200E-01	.897732E+03	.561099E+06	.435627E+05	.272203E+08
.210700E-01	.943267E+03	.589550E+06	.433667E+05	.270966E+08
.220500E-01	.985610E+03	.616006E+06	.432070E+05	.269962E+08
.230300E-01	.102784E+04	.642388E+06	.430867E+05	.269198E+08
.240100E-01	.106995E+04	.668697E+06	.429704E+05	.268463E+08
.250600E-01	.111491E+04	.696786E+06	.428197E+05	.267511E+08
.260400E-01	.115667E+04	.722876E+06	.426172E+05	.266233E+08
.270200E-01	.119820E+04	.748818E+06	.423759E+05	.264710E+08
.280700E-01	.124242E+04	.776436E+06	.421100E+05	.263032E+08
.290500E-01	.128345E+04	.802067E+06	.418739E+05	.261542E+08
.300300E-01	.132433E+04	.827600E+06	.417150E+05	.260539E+08
.310100E-01	.136514E+04	.853089E+06	.416431E+05	.260084E+08
.320600E-01	.140888E+04	.880403E+06	.416516E+05	.260135E+08
.330400E-01	.144975E+04	.905931E+06	.417095E+05	.260498E+08
.340200E-01	.149117E+04	.931802E+06	.422631E+05	.263984E+08
.350700E-01	.153761E+04	.960825E+06	.442329E+05	.276407E+08
.360500E-01	.158202E+04	.988581E+06	.453135E+05	.283231E+08
.370300E-01	.162674E+04	.101654E+07	.456332E+05	.285245E+08
.380100E-01	.167125E+04	.104435E+07	.454155E+05	.283861E+08
.390600E-01	.171830E+04	.107376E+07	.448145E+05	.280055E+08
.400400E-01	.176138E+04	.110067E+07	.439529E+05	.274609E+08
.410200E-01	.180355E+04	.112701E+07	.430346E+05	.268812E+08
.420700E-01	.184778E+04	.115464E+07	.421218E+05	.263055E+08
.430500E-01	.188832E+04	.117995E+07	.413683E+05	.258305E+08
.440300E-01	.192859E+04	.120509E+07	.410864E+05	.256527E+08
.450100E-01	.196877E+04	.123018E+07	.410022E+05	.255994E+08

CONTAINMENT AIR COOLER FAN MOTOR BEARING AND WINDING TEMPERATURE INSTRUMENTATION SCHEDULE						
CONE COOLER FAN NO.	MTR PTE BRG TEMP	COMPUTER ANALOG INPUT	MTR O/B BRG TEMP	COMPUTER ANALOG INPUT	MTR STATOR TEMP	COMPUTER ANALOG INPUT
C1-1	TE-5036C	I-295	TE-5036B	I-296	TE-5036A	I-297
C1-2	TE-5040C	I-299	TE-5040B	I-300	TE-5040A	I-301
C1-3	TE-5046C	I-303	TE-5046B	I-304	TE-5046A	I-305

UNIT HEATER NO.	TEMP. SWITCH NO.	TEMP. SWITCH NO.
E117-1	TS-5610	HS-5610
E117-2	TS-5612	HS-5612
E117-3	TS-5618	HS-5618
E117-4	TS-5619	HS-5619
E117-5	TS-5620	HS-5620
E117-6	TS-5621	HS-5621
E117-7	TS-5622	HS-5622
E117-8	TS-5623	HS-5623
E117-9	TS-5624	HS-5624
E117-10	TS-5625	HS-5625
E117-11	TS-5626	HS-5626
E117-12	TS-5627	HS-5627



DB-2,3

FIGURE 6.2-23B HAS BEEN DELETED.

CHAPTER 10.0

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10.3 MAIN STEAM SUPPLY SYSTEM

The operating function of the main steam supply system is to convey steam from the steam generators to the high pressure turbine and other auxiliary equipment for power generation.

10.3.1 DESIGN BASES

10.3.1.1 Safety Design Bases

The system provides a means of discharging steam to the atmosphere in order to dissipate heat generated in the NSSS during startup, hot shutdown, or in order to permit step load reductions up to full load, in case the main condenser is not available.

The system provides isolation valving on steam lines leaving the containment.

The system provides an assured source of steam supply to operate the turbine-driven auxiliary feedwater pump for the reactor cooldown under emergency conditions.

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The main steam supply system components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, testing and postulated accidents, including loss of coolant accidents, in accordance with 10 CFR 50, Appendix A, General Design Criterion 4.

The system from the steam generators through the auxiliary building, including the isolation valves, is designed to remain functional after the safe shutdown earthquake.

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10.3.1.2 Power Generation Design Bases

The main steam supply system is designed to deliver steam from the steam generators to the high-pressure turbine for a range of flows and pressures varying from the warmup to the rated conditions. It also provides steam to the moisture separator reheaters, the main feed pump turbines, the turbine gland seal system, the steam jet air ejectors, and the turbine bypass system.

The portion of the main steam system which is constructed in accordance with ASME Section III, class 2 requirements is provided with access to welds and removable insulation as required for inservice inspection in accordance with ASME Section XI, Rules for Inservice Inspection of Nuclear Reactor Coolant System.

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10.3.1.3 Codes and Standards

The steam lines are constructed in accordance with the ASME Boiler and Pressure Vessel Code section III, class 2/MC up to and including the first steam generator isolation valve. Other piping and valves running to the auxiliary feedwater pump turbine are constructed in compliance with ASME Section III, class 3. The remainder of the main steam piping and valves are designed in accordance with the Power Piping Code ANSI B31.1.0. The spring loaded safety valves are designed and selected in accordance with the requirements of ASME Section III, class 2.

10.3.2 SYSTEM DESCRIPTION

10.3.2.1 General Description

The main steam supply system is shown schematically in figure 10.3-1. Major components are the main steam piping, main steam isolation valves, main steam safety valves, and power-operated relief valves.

10.3.2.2 Component Description

Main Steam Piping--Two 36-inch main steam lines carry the total steam flow of 11.76×10^6 lbs/hr. from the steam generators to the turbine. These lines are designed for a maximum pressure drop of 40 psi at valves wide open steam flow conditions and provide balanced steam pressures at the inlets to the turbine stop valves.

The design pressure-temperature of the main steam piping is 1050 psig, 600F, the same as for the steam generator feedwater-steam side. Each of the lines from the steam generators is anchored. The lines have sufficient flexibility to provide for relative movement of the steam generators due to thermal expansion.

Each line contains nine spring loaded safety valves, a power-operated atmospheric vent valve, a locally operated manual atmospheric vent valve, and a main steam isolation valve which are located outside of the containment vessel. All main steam connections are made downstream of the isolation valves with the exception of the connections for the auxiliary feed water pump turbine and low point drains.

Branch piping from the main steam lines provide steam to the reheaters, gland steam sealing system, steam jet air ejectors, main feedwater pump turbines, and the bypass steam to the condensers.

A drain line is connected to the low points of each main steam line. These drains are combined into a header and connected to permit discharge to the condenser and lines are sloped to promote adequate drainage. The drain lines also permit pressure equalization around main steam isolation valves during the plant start up.

Main Steam Line Isolation Valves - The main steam isolation valves are 36-inch diameter, 600-pound ANSI rated carbon steel angle globe valves

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Safety Valves - There are nine spring loaded safety valves installed in each main steam line. The lowest safety valve will be set at the design pressure of the steam generator (1050 psig). The highest safety valves setting is 1102 psig.

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The resultant total relieving capacity of all safety valves exceeds the maximum calculated rating flow of the steam generators. With a three percent accumulation, the maximum steam generator pressure while relieving is less than 110 percent above design pressure, which is within the maximum allowable in accordance with Article NC-7500 of ASME Code Section III, Nuclear Power Plant Components.

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Atmospheric Vent Valves - A power-operated relief valve is installed on the outlet piping from each steam generator. The atmospheric vent valves are installed for removal of post-shutdown reactor decay heat when the main condenser is not in service, when the unit is being started up or shut down, or when a turbine trip occurs on loss of vacuum or loss of electrical power to the turbine auxiliaries.

The power-operated atmospheric vent valves operate automatically at a point lower than the lowest set spring loaded safety valves to relieve the safety valves of duty during transients. The atmospheric vent valves' lift pressure can be remotely adjusted. The valves are automatically controlled by line steam pressure. Local manual actuators are also provided.

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A locally operated manual atmospheric vent valve is also installed on the outlet piping from each steam generator. These manually operated valves are provided to ensure the capability of cooling down the unit, assuming the most limiting single failure with only offsite or onsite power available and a safe shutdown earthquake, as discussed in Section 5.5.7.

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10.3.2.3 System Operation

During unit startup, rate of change of reactor power level is limited to 3/4 percent per minute while power is below 15 percent. This requires that power must be gradually increased to the self sustaining level before steam may be admitted into the turbine to begin rolling and placing in operation power generation equipment. Steam must be bypassed to the condenser during this interim period to prevent exceeding temperature limitations while increasing reactor power to the unit self-sustaining level. As steam is admitted to the turbine bypassing to the condenser is correspondingly reduced to avoid exceeding power or power increase rate restrictions.

The main steam supply system, operating in conjunction with the turbine bypass system (discussed in subsection 10.4.4), provides the capability described above. In the event the turbine bypass system cannot be used, the atmospheric vent valves may be used to bleed steam for unit start up.

As steam is admitted to the turbine, main steam is supplied to the steam jet air ejectors to establish and maintain condenser vacuum. The condenser evacuation system is discussed in subsection 10.4.2.

Main steam is then applied to the main feed pump turbine, and the main feedwater system is placed in operation. When the unit is operating above low power levels, the steam supply for the main feed pump turbine is shifted to the output of the moisture separator reheater. The main feedwater system is discussed in subsection 10.4.7.1.

At low power levels, the main steam supply system provides steam to the turbine gland sealing system. This prevent leakage of air into the condenser via the turbine glands. An external supply of steam to this system is not required at high power levels. The turbine gland sealing system is described fully in subsection 10.4.3.

Main steam is supplied to the second stage steam reheaters during power operation to raise the efficiency of unit. The reheaters are described in subsection 10.4.7.1.

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Once operation at high power is attained, the turbine bypass system draws steam as required from the main steam lines to increase plant reliability in the event of large turbine load rejection. If a large, rapid reduction in power demand occurs, main steam is bypassed directly to the condenser to prevent tripping the reactor. Should the condenser not be available as a heat sink, e.g., in the event of the loss of condenser vacuum, the turbine bypass will not open. Instead, the power-operated atmospheric vent valves and safety valves would eventually lift to exhaust the steam to the atmosphere and remove the heat energy as required.

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Under loss of offsite power conditions, steam from the main steam supply system is automatically provided to the auxiliary feedwater pump turbine driver. Thus, it is possible to feed the steam generators to produce steam and, thereby, remove decay heat from the reactor core. The auxiliary feedwater system is described fully in subsection 10.4.7.2.

The coordinated operation of the main steam supply system, the safety-relief valve system or turbine bypass system, and the auxiliary feedwater system as described above during a large turbine load rejection or loss of offsite power situation may also be employed to remove decay heat during normal shutdown operations.

10.3.3 SAFETY EVALUATION

Heat dissipation requirements during unit startup, hot shutdown, and cooldown are normally met by bypassing steam to the main condenser via the turbine bypass system described in subsection 10.4.4. If the bypass system is not available, the atmospheric vent valves are adequately sized to remove decay heat and for unit cooldown.

The locally operated manual atmospheric vent valves provide the capability, assuming the most limiting single active failure with only onsite or offsite power available and a safe shutdown earthquake, to cool down to the decay heat removal system cut-in conditions as discussed in Section 5.5.7.

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The steam generator code safety valves are adequately sized to permit load rejection from full power. The installed 25 percent turbine bypass valves will permit 40 percent load rejection without opening code safety valves.

Isolation valves are included in the system design and are described in subsection 10.3.2.2, and their performance under accident conditions is described in subsection 6.2.4, containment system, and in chapter 15.

The arrangement of the main steam supply system makes it possible to start automatically the steam driven auxiliary feed pump. The steam line to the auxiliary feed pump turbine is routed from a cross connecting header upstream of the main steam isolation valves. This arrangement will assure a supply of steam to the turbine-driven auxiliary feed pump even when the steam generators are isolated and a-c electrical power is not available. The auxiliary feedwater system is described in subsection 10.4.7.2. | 15

All safety related components in the main steam system are designed to perform their intended function in the normal and accident temperature, pressure, humidity, chemical, and radiation environment to which they will be subjected. Environmental design bases and qualification are discussed in section 3.11, Environmental Design of Mechanical and Electrical Equipment.

As shown on Figures 10.3-1 and 10.3-2, the main steam supply system is seismic Category I up to and including the anchor at the auxiliary building, turbine building wall, the auxiliary feed pump turbine, and the main steam safety valve outlets. Main steam isolation valve pneumatic actuation piping is seismic Category I up to and including the pilot valves. The definition of seismic Category I requirements is given in Section 3.7 and 2.5 and the techniques used are described in subsection 3.9.2. | 22
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10.3.4 TESTS AND INSPECTIONS

10.3.4.1 Preoperational Valve Testing

The steam safety valves located in the main steam piping at the outlet from each steam generator are individually tested during initial startup or shutdown operation by checking the actual lift and blowdown point of the valve as indicated by pressure gauges mounted in the main steam piping. Valves not being tested are gagged temporarily.

The lift and blowdown of the atmospheric vent valves are likewise checked during initial startup or shutdown operations.

A test of the local control for the manual atmospheric vent valves will be conducted prior to unit operation to demonstrate that the units can be cooled in a controlled manner, and that the valves can be operated safely and effectively. | 22

The main steam line isolation valves are hydrostatically tested in accordance with the requirements of Article NC-6000 of ASME Section III. Valve seat tests are in accordance with MSS-SP-61, except that seat leakage is not allowed to exceed 2cc/hr. per inch of diameter across the valve seat. | 15

The main steam isolation valves and turbine bypass valves are given an air seat test. Conditions of this test are 40 psi across the valve assembly with up and down stream temperatures of 60F. The test duration is one hour, and the maximum leakage rate is 34.3 cc per inch of diameter across the valve seat at standard conditions. | 15

Each main steam isolation valve will be tested in the shop to demonstrate that the closing time against a pressure of 1050 psig is less than 5 seconds using 80 psig actuating air and springs only. | 1

Using channels 1 and 2., respectively, each of the main stream isolation valves will be partially stroked periodically during normal unit operation to demonstrate the operability of the valves.

Periodic Functional Tests - Main steam isolation valves will be retested for closing times prior of cooldown for refueling. Inservice testing of the main steam line isolation valves is discussed in chapter 16 (subsection 4.11).

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The main steam line isolation valves are checked for closing time prior to initial startup.

10.3.4.2 Preoperational System Testing

The various alarm and pressure trip points to isolate the main feed pumps to prevent overpressurization are checked by comparing design setpoints with actual measured trip settings. The main steam line is hydrostatically tested to confirm leaktightness. Visual inspection of pipe weld joints confirms the exterior condition of the weld. Pipeline expansion and movement from the cold condition is checked by measuring movement from field bench marks such as steel columns or pipe supports as specified on design isometric piping drawings indicating calculated movements in the x, y, and z axes.

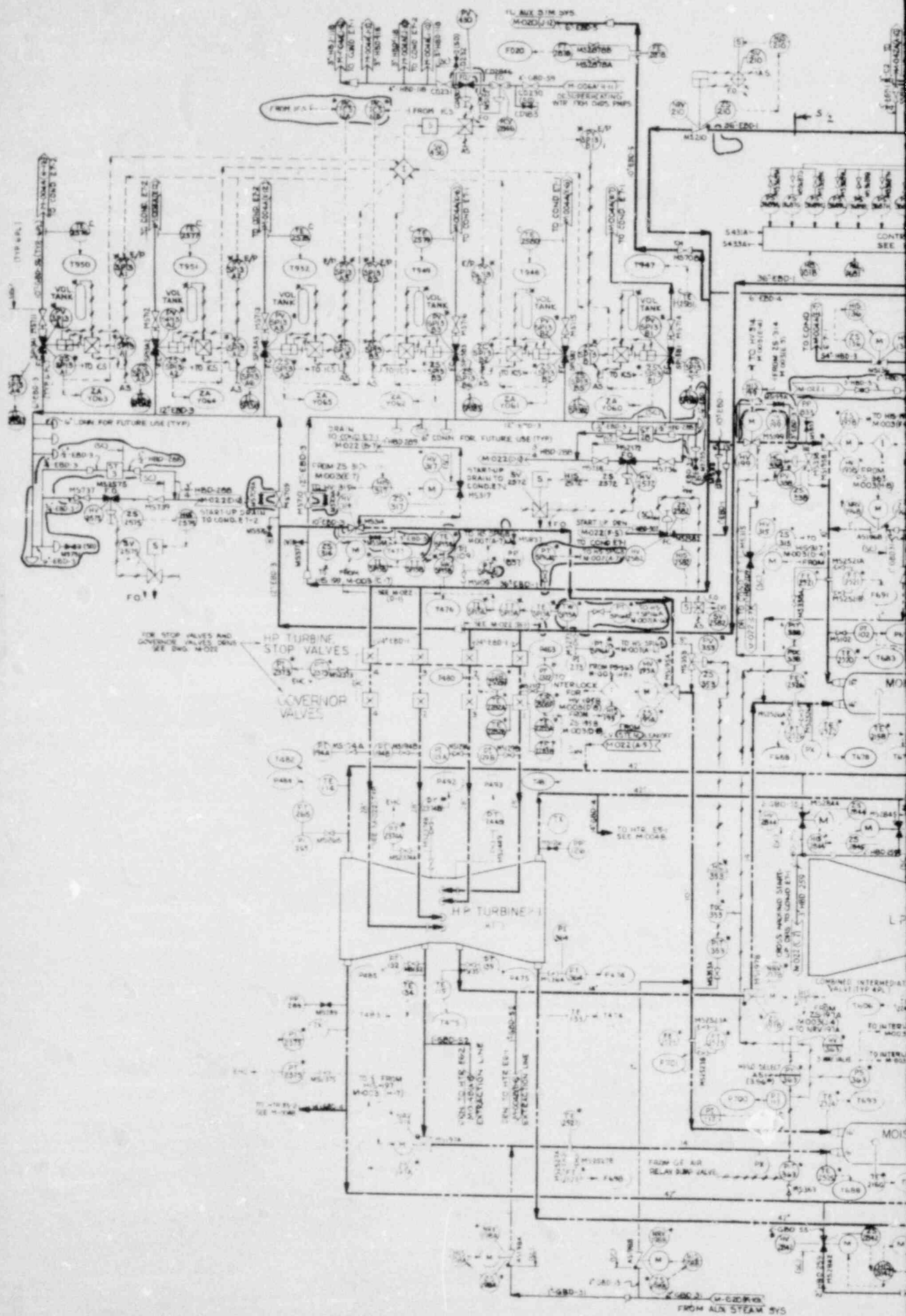
10.3.5 WATER CHEMISTRY

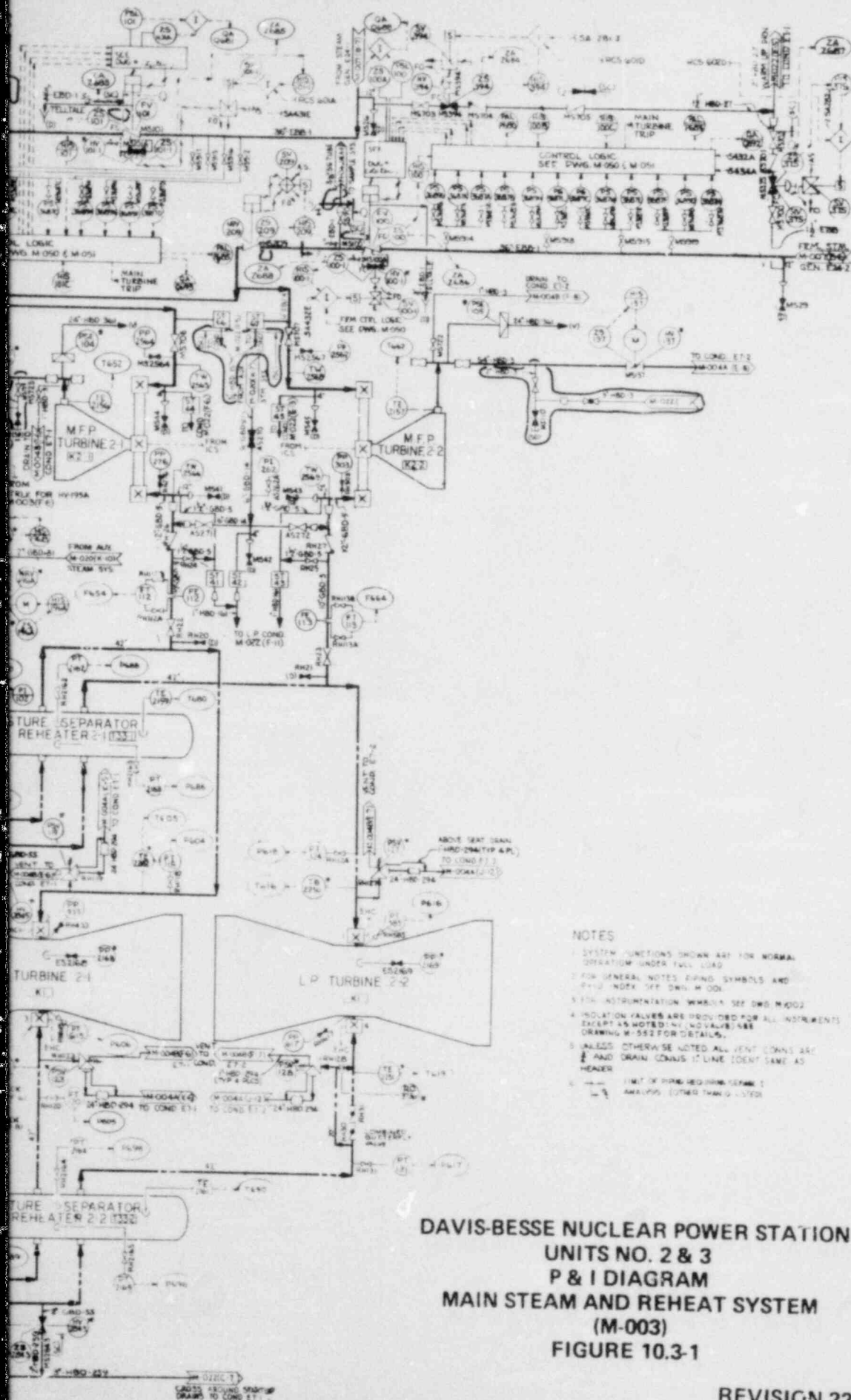
The feedwater chemistry requirements for the once-through steam generators and the materials used for the condensate and feedwater system are given in table 10.3-1. These requirements are met by using condensate polishing demineralizers and by using corrosion resistant tube material in the feedwater heaters.

Corrosion control of the steam and feedwater systems is maintained through the use of high purity feedwater containing pH, and oxygen controlling chemicals. Hydrazine, a reducing agent, is added to the deaerated feedwater to react with the remaining dissolved oxygen. The products of reaction are nitrogen and water with no solid products added to the system. The feedwater pH is regulated by the controlled addition of ammonia. The hydrazine and ammonia are injected immediately downstream of the condensate polishing demineralizers. Feedwater contamination due to a condenser tube leak is detected through a cation conductivity detector located upstream of the condensate polishing demineralizer.

10.3.6 INSTRUMENTATION APPLICATIONS

The main steam supply system instrumentation, controls, and protective devices are designed to ensure safe and reliable operation of the system. Pressure and temperature measuring instruments are provided for local and/or remote indication and alarm functions. Motor-operated valves are provided with remote position indication and control. Conductivity analyzers monitor general steam purity going to the turbines and annunciate any abnormal condition.





NOTES

1. SYSTEM FUNCTIONS SHOWN ARE FOR NORMAL OPERATION UNDER FULL LOAD.
2. FOR GENERAL NOTES, PIPING SYMBOLS AND P&ID INDEX, SEE DWG. M-001.
3. FOR INSTRUMENTATION, SEE DWG. M-002.
4. ISOLATION VALVES ARE PROVIDED FOR ALL INSTRUMENTS EXCEPT AS NOTED IN (NO VALVE) SEE DRAWING M-552 FOR DETAILS.
5. UNLESS OTHERWISE NOTED, ALL VENT COUPLERS ARE AND DRAIN COUPLERS IT LINE IDENT SAME AS HEADER.
6. LIMIT OF PIPING REQUIRED TO BE MAINTAINED ANALYSIS, OTHER THAN LISTED.

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4. Unit "End of Run" signal due to ion capacity exhaustion or head loss.
5. High differential pressure alarm across the entire demineralizer package.
6. Precoat pump flow.
7. Precoat tank low level alarm.
8. Backwash receiving tank high and low level alarms.
9. Backwash receiving tank high radioactivity alarm (two separate signals).
10. Holdup tanks level indicators.

Controls are provided for automatic flow balancing between units on stream. Remote manual operation is also available. Control panel signals indicate valve position and operating status of each demineralizer unit regarding which automatic sequence is in progress.

The hotwell instrumentation that gives warning of condenser inleakage of cooling tower water includes the following:

There are four sampling lines located at the low-pressure condenser shell which permit taking grab samples of the condensate in each of the four cross-over lines between the low-pressure and high-pressure condenser shells. In addition, the condensate pump common discharge is connected to the sampling system, permitting monitoring of the conductivity of the condensate downstream of the condensate pumps but upstream of the condensate demineralizers. Condenser inleakage of circulating water will be indicated by high conductivity of the condensate. The location of the leakage can be further identified by taking grab samples of the condensate in the cross-over lines.

10.4.6.6 Tests and Inspection

During normal operation, all of the equipment is readily available for inspection. The holdup tanks are located in a separate enclosed room which is isolated during periods when radioactivity has been detected in the secondary system and a backwash has been transferred to the tanks.

All pressure-containing components of the condensate polishing demineralizer system are hydrostatically tested before installation. Tests are performed after installation to verify performance requirements.

10.4.7 CONDENSATE AND FEEDWATER SYSTEM

The feedwater cycle is a closed system with deaeration accomplished in the main condenser and two one-half capacity deaerators. Six stages of feedwater heating (including the deaerators) are incorporated. Chemical injection is provided for Ph control and oxygen removal. Condensate polishing demineralizers provide impurity control.

10.4.7.1 Design Bases

10.4.7.1.1 Safety Design Bases

The main feedwater lines are designed so that failure in this piping will not damage the reactor coolant pressure boundary and also that failure in this main feedwater piping inside or outside the containment vessel will not prevent a safe shutdown of the unit.

The containment vessel isolation valves and all piping between the anchor at the auxiliary building, turbine building wall and the steam generator nozzles are designed to withstand the effects of a safe shutdown earthquake.

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10.4.7.1.2 Power Generation Design Bases

The system is designed to provide feedwater to the steam generators at the required temperature and pressure during all phases of operation.

The extraction lines and feedwater heaters are designed to minimize the possibility of water induction to the turbine and limit turbine overspeed due to entrained energy in the extraction system.

10.4.7.1.3 Codes and Standards

All components of the condensate and feedwater system which are subject to the system pressure are designed and constructed in accordance with the following codes and standards:

1. The design, materials, and details of construction of the feedwater heaters are in accordance with both the ASME Code, section VIII, Unfired Pressure Vessels and the standards for open and closed feedwater heaters of the Heat Exchange Institute.
2. All piping meets the requirements of ANSI B31.1, except as follows:
 - a. All feedwater piping from the main feed pumps to the containment isolation valves is upgraded for critical service.
 - b. All piping into the containment structure from and including the isolation valves meets the requirements of ASME, section III, nuclear class 2.

21. Combustible Gas Control (CSB)

By our letters dated January 17, 1977 and May 25, 1977, we informed the applicant that his proposed method for combustible gas control was not acceptable. The applicant appealed our position in a meeting held on September 29, 1977. As a result of this meeting, the applicant submitted new information in PSAR Revision 19 concerning hydrogen production and accumulation in the containment vessel following a loss-of-coolant accident.

Although the new information indicates that purging of the containment would not be required until about one year after the accident and, therefore, the incremental doses to the public would be very small, we still find that the proposed combustible gas control scheme is unacceptable. It has been our policy that repressurization and purging are not acceptable means of combustible gas control. Therefore, we require that hydrogen recombiners be included in the Davis-Besse Units 2 and 3 design.

The applicant has performed an analysis of hydrogen production and accumulation in the containment vessel following a loss-of-coolant accident. However, the applicant has not adequately justified the assumed corrosion rates of and quantity of materials that produce hydrogen.

Before we can conclude that the assumed rates are conservative, we must review the data upon which these rates are based. We require the applicant to (1) provide the experimental data used to support the corrosion rates and evolution of hydrogen for the galvanized materials and zinc base paints listed in PSAR Table 6.2-16; data should substantiate the statement that hydrogen generation from these two sources becomes negligible below 100 C; (2) justify the corrosion rate of aluminum in light of the statement in Branch Technical Position CSB 6-2 that the rate of 200 mils per year should be adjusted upward for higher temperatures; (3) for all corrosion used in the analysis, show how the application of the experimental data to the Davis-Besse Units 2 and 3 analysis is conservative with regard to pH and to temperatures of containment atmosphere, affected materials, and corrosive fluids; and (4) discuss the conservatism of the amount of surface areas assumed in the analysis.

Furthermore, PSAR Section 6.2.5.3 states that only 20 percent of the interior galvanized ductwork is considered exposed to spray solution. It is our position that 100 percent of the interior ductwork be considered exposed to the spray solution, due to circulation of post-accident vapors.

RESPONSE

Hydrogen recombiners have been included in the Davis Besse 2 and 3 design (see revised Section 6.2.5).

See subsection 6.2.5.3.2 for the reanalysis of hydrogen production and accumulation in the containment vessel following a postulated LOCA.