14.4 ENVIRONMENTAL QUALIFICATION ANALYSES

14.4.1 Basis of Discussion

The following is applicable to both units unless otherwise noted. A feedwater line break is not part of the Unit 1 current license basis for the standby safeguards analyses contained in Section 14.2, Standby Safeguards Analyses. However, postulated feedwater line breaks are part of the environmental qualification analyses required for both units contained in this section.

14.4.2 Postulated Pipe Failure Analysis Outside Containment

Presented below is a discussion of analyses associated with high energy line breaks (HELBs) outside of containment.

Effect of the Replacement Steam Generators on Unit 1

The effect of the Unit 1 RSGs on postulated pipe failure analysis outside of containment has been evaluated. An evaluation of high energy line breaks outside of containment for the RSGs concludes that the difference in mass between the original and replacement steam generators will not have a significant effect on the calculated steam enthalpy for the steam line break outside containment. Also, the RSG has no effect on the pipe breaks outside of containment from a structural loading standpoint. Finally, steam generator replacement does not adversely affect any of the plant equipment and instrumentation required to shutdown the reactor listed in Table 14.4.2-1A. Therefore, the material presented in Section 14.4.2 is not affected by steam generator replacement.

14.4.2.1 High Energy Systems Definition

High energy piping systems are defined as those having a normal service temperature above 200°F, a normal operating pressure above 275 psig, and a nominal diameter greater than one inch.

The systems of the Donald C. Cook Nuclear Plant that fall under the above definition are:

1. Main Steam System
2. Feedwater System
3. Steam Generator Blowdown System
4. Chemical and Volume Control System
5. Steam Supply to Auxiliary Feedwater Pump Turbine
6. Other Systems as tabulated in Table 14.4.2-2

Specific high energy lines in each of the above systems are presented in Section 14.4.2.6.

14.4.2.2 Criteria for Pipe Rupture Postulation

14.4.2.2.1 Definition of a Piping System

A piping system is defined as having pressure-retaining components consisting of straight or curved pipe and pipe fittings such as elbows, tees and reducers. The boundaries of a system are described in terms of a piping run. A piping run interconnects components such as pressure vessels, pumps, valves, or structural anchors that may restrain pipe movement. A branch run differs from a main piping run only in that it originates at a piping intersection as a branch of the main pipe run.

14.4.2.2.2 Design Basis Breaks

Design basis breaks in high-energy piping systems are defined in this section. The criteria for considering the effects of pipe whip are considered in subsequent sections.

1. Break Location Based on High Stress Points

   There is no ASME Section III Code Class I piping outside the containment for the Donald C. Cook Nuclear Plant. The criteria used to determine the design basis piping break locations are as follows. Piping breaks were postulated to occur at the following locations in each piping run or branch run:

   A. The terminal ends except NUREG-800, "Standard Review Plan (SRP)", section 3.6.2 break exclusion zones are established between the containment penetration and the first outboard isolation valve in the steam generator blowdown and chemical and volume control system letdown lines

outside containment based on piping support modifications and revised stress analyses (Reference 1).

B. At intermediate locations where:

i. the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 ($S_h + S_A$)[1]; and

ii. for cases where no stress analysis was performed, breaks were postulated at any orientation, and the most adverse locations were examined.

2. Size and Orientation

Once a design basis break location has been established, as defined above, the break orientation and size depend upon the following additional conditions:

A. Longitudinal breaks in piping runs or branch runs were examined for pipes of 4" nominal diameter and larger. A longitudinal break is parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location with the length of the break equivalent to twice the inside pipe diameter. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

B. Circumferential breaks were examined in piping runs and branch runs exceeding a nominal 1-inch diameter. A circumferential break is perpendicular to the pipe axis, and the break area is equivalent to the cross-sectional flow area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to cause pipe movements perpendicular to the plane of the break.

[1] S_h is the stress calculated per ANSI B31.1.0-1967. S_A is the allowable stress range for expansion in stress calculated by the rules of the USA standard code of pressure piping. ANSI B31.1.0-1967.

14.4.2.2.3 Design Basis Crack

A design basis crack is defined as a single open crack of a size of one-half the pipe inside diameter in length and one-half the pipe wall thickness in width. The location of this crack can be anywhere along the length of the pipe.

Crack Location

Where high-energy pipes are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, a single postulated crack in the pipe system has been postulated at the most adverse location except NUREG-800, "Standard Review Plan (SRP)", section 3.6.2 crack exclusion zone is established in the steam generator blowdown piping located in the steam generator normal blowdown flash tank room based on piping, piping support modifications and acceptable stress analysis (Reference 1). The criteria for evaluating the effects of jet impingement and resulting steam-air environment are discussed below.

Pipe rupture induced loads, such as pipe whip, jet impingement, and compartment pressure are analyzed below in Sections 14.4.5, 14.4.7 and 14.4.8, respectively.

14.4.2.3 Section Not Used

14.4.2.4 Section Not Used

14.4.2.5 Section Not Used

14.4.2.6 Pipe Rupture Locations and Evaluations

Rupture locations were established in each of the high energy system piping based on the stress analyses presented in Section 14.4.4. Major break locations, which establish the design basis, are tabulated in Table 14.4.2-5.

The consequences of each postulated break were evaluated with respect to (1) high compartment differential pressure, (2) jet effects, (3) potential pipe whip damage, and (4) environmental effects. All assessments of potential damage are expressed in terms of the postulated rupture not causing damage to any of the equipment (or supporting electrical cables, structures, etc.) required to assure safe shutdown of the plant. The consequences of each crack are evaluated with respect to jet effects and environmental effects. The evaluations are discussed below in Sections 14.4.5 through 14.4.8. The requirements to achieve safe shutdown in the event of these ruptures are discussed in Section 14.4.3 and environmental qualification of equipment required for safe shutdown is discussed in Section 14.4.11.

14.4.2.6.1 Main Steam Line Routing and Rupture Evaluation

Routing Description

The main steam piping from steam generators Nos. 1 and 4 have similar routing and were treated identically for the purpose of the rupture analysis. Each steam pipe exits the containment from the east side of the containment whereupon it enters its main steam stop valve and pipe rupture restraint. The two pipes then run around the south side of the containment wall (Unit 2), or north side of the containment wall (Unit 1) and into the West Steam Enclosure. They then enter the main steam accessway. The main steam piping from steam generators Nos. 2 and 3 exit the containment at the west main steam enclosure, enter the main steam stop valves and pipe rupture restraints. They then enter the main steam access way where they run horizontally to the turbine building. The above routing is shown isometrically in Figure 14.4.2-1 (Unit 2) and Figure 14.4.2-1A (Unit 1).

Rupture Locations

Potential design basis pipe break locations are shown on the Main Steam Isometric, Figure 14.4.2-1 (Unit 2) and Figure 14.4.2-1A (Unit 1) and are described below.

a. The terminal points of the main steam lines at the turbine stop valves in the Turbine Building and at the containment wall, and at the 5-way restraints located in the east and west main steam enclosures.

b. The branch point connection in the main steam line for the turbine bypass headers and the terminal point on this branch line.

The orientation and location of a design basis crack can be anywhere along the piping, shown on the Main Steam Isometric, Figure 14.4.2-1 (Unit 2) and Figure 14.4.2-1A (Unit 1).

14.4.2.6.2 Feedwater Routing and Rupture Evaluation

Routing Description

The 24" discharge of each main feedwater pump is routed through two check valves and a motor operated valve before joining in the Turbine Building into a single 30" line. The 30" line divides into two 20" lines, each of which goes to two feedwater heaters. After the two 20" lines leave the second feedwater heater, they join into a single 30" line. The single 30" line leaves the Turbine Building and divides into two 20" lines in the main steam accessway. One 20" line is routed through the Exterior Pipeway where it divides into two 14" lines which enter the Main Steam Enclosure (East). One 14" line supplies steam generator #1, and the other 14" line supplies steam generator #4. The other 20" line is routed through the Main Steam Line Accessway where it

divides into two 14" lines that enter the Main Steam Enclosure (West). One 14" line supplies steam generator #2, the other 14" line supplies steam generator #3. Each 14" line is provided with a flow nozzle, regulating valve, manual isolation valves, and a check valve before entering the containment.

Rupture Locations

Postulated design basis pipe break locations are shown on the feedwater isometric, Figure 14.4.2-2 (Unit 2) or Figure 14.4.2-2A (Unit 1). The locations have been determined on the basis of ANSI B31.1 calculated stress values based on the as-built system. These consist of the terminal points, at the high pressure feedwater heaters and containment penetrations. The orientation and location of a design basis crack can be anywhere along the piping, shown on the feedwater isometric, Figure 14.4.2-2 (Unit 2) or Figure 14.4.2-2A (Unit 1).

No intermediate locations in the Auxiliary Building or Turbine Building exceeded the 0.8 (S_h + S_A) stress criterion.

14.4.2.6.3 The CVCS Letdown Line Routing and Rupture Evaluation

Routing Description

The 2" CVCS letdown line exits the containment from the loop quadrant, passes through an isolation valve, and runs in an east-west direction through the auxiliary building into the letdown heat exchanger room. The high energy portion of the CVCS system is the letdown line from the point where it leaves the containment to the point where it enters the letdown heat exchanger.

Rupture Locations

A break is postulated to occur at the heat exchanger. The orientation and location of a design basis crack can be anywhere along the piping.

14.4.2.6.4 Steam Generator Blowdown Line Routing and Rupture Evaluation

Routing Description

The steam generator blowdown lines enter the auxiliary building from the containment through four individual penetrations. Each line is provided with an isolation valve and a blowdown control valve. The 2" blowdown lines combine to a single 3" line before entering the steam generator startup blowdown flash tank. The blowdown fluid flows to the normal blowdown flash tank via a four-inch line.

Rupture Locations

Terminal end breaks at the containment penetrations are eliminated based on the piping support modifications and revised stress analyses. The orientation and location of a design basis crack can be anywhere along the piping. (Note: the flash tanks operate at pressures below the HELB threshold and are therefore not considered terminal ends).

14.4.2.6.5 Main Steam Supply to Turbine-Driven Auxiliary Feedwater Pump

Routing Description

The steam supply to the auxiliary feedwater pump turbine is taken from the two main steam leads at the west steam enclosure. A four-inch pipe branches from each of the two 30" main steam lines between the containment wall and the main steam stop valves. The two four-inch lines each contain a motor-operated valve and a check valve before they join to a common 4" line which passes through the diesel generator pipe tunnels and before entering the auxiliary feedwater pump room. The routing of this piping is shown in Figure 14.4.2-1 (Unit 2) or Figure 14.4.2-1A (Unit 1).

Rupture Locations

Postulated design basis pipe break locations have been determined based on the criteria given in Section 14.4.2.2. These consist of the terminal points which are located at their branch connections on the main steam lines and at the auxiliary feed pump throttle valve. The orientation and location of a design basis crack can be anywhere along the piping shown on the Auxiliary Feedwater Isometric, Figure 14.4.2-1 (Unit 2) or Figure 14.4.2-1A (Unit 1).

14.4.2.6.6 Other High Energy Systems / Lines

Other high energy systems and lines are tabulated in Table 14.4.2-2.

14.4.2.7 Flooding

The Seismic Class I areas of the auxiliary building were reviewed for non-seismic Class I piping whose failure might furnish a flooding potential. The worst-case source of flooding water is considered to be an ESW line break with a subsequent pump runout at 13,000 gpm. ESW is conservatively chosen even though it is a seismic class I system. This is because ESW is an open system which could pump an unlimited amount of water into the auxiliary building if there were no operator action. Calculations show that it would take 21.4 minutes to flood the auxiliary building to a 12-inch depth on the 573'-0" elevation. This is the maximum flooding depth above which there is the potential for safety systems to be adversely affected. DLA-700, located in the auxiliary building sump, would trigger in 1-2 seconds into the event, which would alert operators

and provide ample time for operator action in the event of a flooding condition. A review of the east steam enclosure and other Seismic Class I areas connected to this enclosure via openings indicated that failure of the main feedwater non-Class I seismic piping might furnish a flooding potential. Sufficient drainage paths exist to preclude flooding of any equipment required for safe shutdown of the reactor. In addition, the auxiliary building arrangement was reviewed for any potential damage to the required equipment due to the cascading water as it passes through floor penetrations and stairwells toward the lowest level. The floor drains, openings and stairwell are located such that no water will drip down on required equipment from the floor above.

14.4.2.8 Reference for Section 14.4.2

1. Letter from Mr. John F. Stang of Nuclear Reactor Regulation to Mr. Robert Powers of Indiana Michigan Power dated November 21, 2000 on subject "Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments (TAC Nos. MA8893 and MA8894).

14.4.3 Safe Shutdown Following Pipe Rupture

14.4.3.1 Control Room Habitability

The control room will be maintained habitable and its equipment functional for all design bases events. Thus, the capability to bring the reactor to a cold shutdown condition from the control room will be maintained.

14.4.3.2 Redundancy

The capability to mitigate the consequences of an accident and to bring the reactor to a cold shutdown condition is assured. Loss of redundancy of equipment required for hot shutdown for a particular accident is not permitted in the protection system (as defined in IEEE-279) or for engineered safety features equipment, cable penetrations, and their interconnecting cables. Loss of function of equipment required for cold shutdown is not permitted. Environmentally-induced failures caused by a leak or rupture, which would not in itself result in protective action but might disable protective equipment, was also considered. In this regard, a loss of redundancy will be permitted but a loss of function will not be permitted. For such situations the capability for bringing the plant to cold shutdown is assured.

14.4.3.3 Separation Criteria

Separation criteria for cable systems are described in Chapter 7.

14.4.3.4 Emergency Procedures

General emergency procedures allow for evaluation of the specific incident and determination of appropriate actions to be taken to achieve a safe shutdown condition. Prompt achievement of hot shutdown will be assured by adherence to the aforementioned criteria; maintenance of hot shutdown will be accomplished by adherence to general emergency procedures. These procedures also allow for placing the reactor in a cold shutdown condition.

14.4.3.5 Analysis of Emergency Conditions

The analyses of emergency conditions listed below are general in nature since it is deemed appropriate to allow for assessment of the incident prior to ultimately bringing the reactor to cold shutdown.

Effect of the Replacement Steam Generators on Unit 1

The effect of the RSGs on the analysis of emergency conditions has been evaluated. The analysis contained in Section 14.4.3 is general in nature and presents the expected method of operation

associated with the high energy line breaks outside of containment. Steam generator replacement does not adversely affect any of the plant equipment and instrumentation required to shut down the reactor listed in Table 14.4.2-1a. Therefore, the material presented in Section 14.4.3 is unaffected by steam generator replacement.

Effect of the RTD Bypass Elimination on Unit 1

Evaluation performed to support RTD Bypass Elimination demonstrates that the conclusions of the accident analysis remain valid.

14.4.3.5.1 Main Steam Line Rupture

The following systems provide for the necessary safeguards system response to a steam pipe rupture outside the containment.

1. Safety injection system actuation from any of the following:
   a. Two out of three low pressurizer pressure signals.
   b. Low main steam line pressure (two out of four lines).[2]
   c. High differential pressure between any two SGs.
2. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.
3. Redundant isolation of the main feedwater lines. Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves, a safety injection signal will rapidly close all feedwater control valves, and trip the main feedwater pumps.
4. Trip of the fast acting main steam isolation valves occurs on any of the following:
   a. Low steam line pressure (two out of four lines).[3]

[2] This "HYBRID" Steamline Break Protection was installed in Unit 1 during the refueling outage of 1997. The previous "OLD" Steamline Break Protection required high steam line flow in two out of four main steam lines, in coincidence with either low-low reactor coolant system average temperature (two out of four loops) or low main steam line pressure (two out of four lines).

[3] This "HYBRID" Steamline Break Protection was installed in Unit 1 during the refueling outage of 1997. The previous "OLD" Steamline Break Protection required high steam flow in coincidence with low steam line pressure.

b. High steam flow in any two steam lines in coincidence with low-low reactor coolant system average temperature in any two loops. Each steam line has a fast-closing stop valve capable of stopping flow in either direction. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. In addition each main steam line incorporates a 16 inch diameter venturi type flow restrictor which is located inside the containment. These components limit the rate of release of steam for an outside break.

   Steamline isolation is complete 11 seconds after the setpoint is reached. The isolation time allows 8 seconds for valve closure plus three seconds for electronic delays and signal processing.

5. Safety injection actuation will also initiate automatic start of the two motor-driven feed pumps. The low-low-level signal in any two steam generators will start the turbine-driven feed pump.

The plant is designed to accept the steam line rupture outside the containment with concurrent loss of offsite power (diesel power available only) and a single active failure in a required system. For small steam line breaks at power which do not cause the reactor power to reach a point at which an immediate reactor trip would occur, no reactor core safety limit will be violated. The small break will result in a continued loss of water from the secondary side of the plant and will eventually result in condenser hotwell low level. This low level will result in a loss of main feedwater, and the reactor will be tripped on low-low steam generator level or feed/steam flow mismatch. After the trip, steam release through the break will cause reactor coolant system cooldown. The cooldown would occur until the steam generator feeding the break empties. The cooldown will automatically initiate safety injection on low pressurizer pressure. Initiation of safety injection will isolate all main feedwater by tripping closed the main feedwater control valves, tripping closed the main feedwater isolation valves, and tripping the main feedwater pumps. The trip of the feed pump turbines initiates closure of the feed pump discharge valves.

Should the plant be at hot standby or subcritical at the time of a small steam line break, the plant will be cooled down by the operator who would have other systems available following the incident to facilitate an orderly shutdown of the reactor. Hence the method and procedure to be used for shutdown will be determined by the operator based on the equipment available.

14.4.3.5.2 Feedwater Line Rupture

The following systems provide necessary protection against a loss of normal feedwater:

1. Reactor trip on low-low water level in any steam generator.
2. Reactor trip on steam/feedwater flow mismatch coincident with low water level in any steam generator.
3. Two motor driven auxiliary feedwater pumps (450 gpm nominal each) which are started on:
   a. Low-low level in any steam generator
   b. Trip of all main feed pumps
   c. Any safety injection signal
   d. Blackout signal
   e. Manually

Each of these pumps feeds two steam generators in its unit.

4. One turbine driven auxiliary feedwater pump (900 gpm) which is started on:
   a. Low-low level in any two steam generators
   b. Reactor coolant pump bus undervoltage
   c. Manually

The turbine driven auxiliary feedwater pump feeds the four steam generators on its unit.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. Following the initiation of the low-low level trip, the auxiliary feedwater pumps are automatically started, reducing the rate of water level decrease. The capacity of the auxiliary feedwater pumps is such that the water level in the steam generators does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the reactor coolant system relief or safety valves. The plant is designed to accept this failure (feedwater line rupture) with concurrent loss of off-site power (diesel generator power available only) and a single active failure in a required system. In addition, all required systems are operable from the control room or accessible for manual operation.

If a rupture should occur in the feedwater line that would not directly cause a reactor trip, the operator has numerous devices (steam flow versus feedwater flow, steam generator pressure, steam generator level, etc.) to detect such an event. If deemed necessary by the operator, the reactor and main feedwater pumps could be manually tripped, resulting in automatic start of the auxiliary feedwater pumps.

14.4.3.5.3 Letdown or Charging Line Rupture

In the event of a break in the letdown line downstream of the letdown orifices' isolation valves and upstream of the letdown heat exchanger, control room indication would come from one or more of the following:

1. Sudden drop in pressure in letdown line
2. Sudden loss of letdown flow
3. Decreasing pressurizer level
4. Decreasing volume control tank level
5. Excessive makeup to the volume control tank

On any of the above indications the operator would have several methods to isolate the letdown flow.

In the event of rupture of the letdown piping the operator would have multiple indication of such an event and redundant means of isolation.

14.4.3.5.4 Main Steam To Turbine-Driven Auxiliary Feedwater Pump and Steam Generator Blowdown Line Rupture

Both of these postulated accidents are considered as small steam line ruptures and are analyzed under Section 14.4.3.5.1.

14.4.3.6 Safe Shutdown Equipment

A list of safe shutdown equipment was compiled to identify systems and components required to achieve and maintain a safe shutdown. The approach in compiling the list was to identify the safe shutdown functions, and the systems required to perform those functions considering single active failure and redundancy requirements. For each of the required systems, including the support systems, such as the power, service, instrumentation and ventilation systems, plant flow diagrams, system descriptions and one line diagrams were used to identify the required flow paths and operational characteristics that must be established to accomplish the desired safe shutdown

function. From this information, a list of the components which are required for the system to perform its safe shutdown function was developed. Equipment identification number, description, system, function, building, elevation and room were identified for each component.

The Safe Shutdown Lists developed for Cook Nuclear Plant contain the minimum required equipment necessary to safely shutdown the unit following a High Energy Line Break (HELB) outside the containment.

The assumptions used in generating the Safe Shutdown Equipment List (SSEL) are:

1. The unit is operating at 100% power prior to the occurrence of HELB and concurrent loss of off-site power.
2. The reactor is tripped either manually or automatically.
3. A single active failure (in addition to loss of off-site power) is assumed in systems used to mitigate the consequences of a postulated HELB.
4. No piece of equipment required for safe shutdown is assumed to be out of service.
5. No concurrent or sequential design basis accidents or transients are assumed to occur.

The safe shutdown systems selected are those that are capable of achieving and maintaining subcritical conditions in the reactor, maintaining reactor coolant inventory, achieving and maintaining hot shutdown conditions for an extended period of time, achieving cold shutdown conditions, and maintaining cold shutdown conditions thereafter.

The primary safe shutdown systems were determined to be as follows:

- Chemical & Volume Control System
- Safety Injection System
- Reactor Coolant System
- Main Steam System
- Auxiliary Feedwater System
- Residual Heat Removal System

The supporting systems were determined to be as follows:

- Component Cooling Water System
- Essential Service Water System
- Emergency Power Systems
- 4kV Electrical Distribution System
- 250 VDC Electrical Distribution System
- 600 VAC Electrical Distribution System
- 120 VAC Electrical Distribution System
- Diesel Generator System
- HVAC Systems

Other miscellaneous safe shutdown support functions/components were determined to be as follows:

- Blowdown System Isolation
- Feedwater System Isolation
- Control Air System
- Condensate Storage Tank
- Refueling Water Storage Tank

Safe shutdown components included in the list were:

1. Active components that need to be powered to establish, or assist in establishing, the primary flow path and/or the system's operation.
2. Active components in the primary flow path that normally are in the proper position but may affect availability of path.
3. Power-operated components that need to change position to establish or assist in establishing the primary flow path, whose loss of electrical or air supplies result in the component adopting the required safe shutdown position.
4. Major passive mechanical components that support safe shutdown (heat exchangers and storage tanks).
5. Active components and check valves that isolate those branch flow paths that must be isolated and remain isolated to assure that flow would not be substantially diverted from the primary flow path.

Table 14.4.2-1 (Unit 2) and Table 14.4.2-1a (Unit 1) provide this component list (HELB-SSEL).

14.4.3.7 References for Section 14.4.3

1. D. C. Cook Report, High Energy Line Break – Safe Shutdown Equipment List, AEP Report No. NED-2000-441-REP, Sargent & Lundy Report No. SL-5396, Rev. 0, 02/18/2000 (Unit 2).
2. D. C. Cook Report, "High Energy Line Break – Safe Shutdown Equipment List, (HELB-SSEL) Report for Unit 1," AE101-HELB-001 Rev. 0, 08/04/2000.

14.4.4 Stress Calculations

Stress analyses were made on the main steam line, the feedwater line, the main steam to auxiliary feedwater pump turbine line, CVCS letdown piping, blowdown piping, and portions of condensate and heater drain piping. On all other high energy lines, the postulated break was assumed to occur anywhere. Break locations were assumed to occur at locations based on the criteria of Section 14.4.2.2.

- The main steam piping system was analyzed from containment penetrations up to, and including, the 36" bypass header. The stresses for the main steam piping system were calculated with the aid of a computer program using general flexibility and response spectra model analysis technique. The combined stress values due to thermal expansion, pressure, weight, and seismic loading conditions have been computed. Postulated design basis break locations outside containment have been determined on the basis of ANSI B31.1.0-1967 calculated stress values and the criteria given in Section 14.4.2.2.
- The feedwater piping system was analyzed from the feedwater heaters to containment penetrations. The piping was analyzed for pressure, dead weight, thermal expansion (including anchor displacements, where required), building settlement, and seismic anchor movement. Both Code stress analysis and HELB stress analysis were performed. The branch piping was de-coupled from system model.
- The Steam Generator Blowdown (BD) piping was analyzed from containment penetrations to the normal and startup flash tanks. The piping was analyzed for pressure, dead weight, thermal expansion (including anchor displacements, where required), building settlement, and seismic anchor movement.
- The CVCS Letdown piping was analyzed from containment penetration to the heat exchanger. The piping was analyzed for pressure, dead weight, thermal expansion (including anchor displacements, where required), building settlement, and seismic anchor movement.
- The Main Steam supply line to the turbine driven Auxiliary Feedwater Pump (AFP) Turbine, which is 4" in diameter, was analyzed from where it branches off from the 30" headers in the auxiliary building to the AFP Turbine.

- The portion of condensate piping from the Main Feed Pump Suction back to the L. P. heaters was analyzed for pressure, weight, thermal expansion (including anchor displacements), seismic anchor movement and seismic OBE.
- The high energy piping, including heater drains, near the number 5 feedwater heater (adjacent to the 4kV switch gear room) was analyzed for pressure, dead weight, thermal expansion (including anchor displacements, where required), building settlement (if applicable), and seismic anchor movement.

In addition to the above design-basis-break locations, a critical crack was postulated.

14.4.5 Description of Pipe Whip Analysis

The reaction load resulting from pipe rupture has the duration and initial conditions to adequately represent the jet stream dynamics and the system pressure characteristics. The piping systems in which pipe ruptures were considered are defined in Section 14.4.2.1. The loads induced by pipe rupture include the effects of any line restrictions, for example, flow limiters, between the pressure source and the break location. If a whipping pipe impacts an adjacent pipe of equal or greater nominal pipe size and equal or heavier wall thickness, the impacted pipe will be considered to be free from rupture. Protection from pipe whip is not required if pipe rupture occurs in such a manner that the unrestrained movement of either end of the ruptured pipe about a plastic hinge, formed at the nearest restraint or anchorage, cannot impact any structure, system, or component required for that incident.

Each high energy system was reviewed to determine its proximity to other systems necessary to the safe shutdown of the reactor. Restraints are provided to prevent pipe whip where there is any possibility that whip following a pipe rupture would damage systems, components, or structures that are needed to mitigate the consequences of that pipe rupture.

14.4.5.1 Method of Analysis

The locations of breaks for pipe-whip considerations are those defined in Table 14.4.2-5. For extremely short, straight piping runs, with no changes in diameter or discontinuities, intermediate breaks were not postulated. At each intermediate break location, the determination of longitudinal and/or circumferential break types was based on the analysis of the stresses. Where the total hoop stress and the total longitudinal stress differed in magnitude by less than 20%, both break types were assumed and analyzed. Where the magnitude of these two stresses differed by 20% or greater, only the break type associated with the higher stress was assumed and analyzed.

The loads induced from pipe rupture include the effects of any line restrictions between the pressure source and the break location. Rupture restraint spacing was determined by calculating the allowable spans necessary for plastic hinges to form so as to prevent contact with any necessary structure or component essential to safety. Gaps were considered. The analyses utilized either a static analysis method or an energy-balance method.

Static Analysis Method:

The thrust force resulting from the blowdown was conservatively calculated by:

F = 1.2 x P x A

Where:

F = Thrust force in pounds

P = Maximum operating pressure in psig

A = Area = internal cross sectional area of the pipe in square inches.

The forces acting on the rupture restraint were taken to be twice the blowdown force. The potential for plastic hinge formation was evaluated for selected locations by a comparison of the energy necessary versus the energy available. The energy necessary was calculated from a material strength analysis. The energy available was calculated by using an isentropic expansion of the compressed liquid to atmospheric conditions. In addition to this comparison, the time duration of blowdown was calculated to determine if the force of blowdown is applied over a sufficient time period to cause hinge formation of the pipe.

Energy Balance Method:

This method is consistent with that accepted by the NRC in Section III of the Standard Review Plan, Section 3.6.2. Principle elements of this method include:

- The blowdown thrust force is time-dependent and depends on the fluid conditions in the pipe.
- The kinetic energy of the ruptured pipe that is imparted to the restraint is balanced by/compared to the strain energy of the whip restraint structural steel.
- The restraint structural steel is allowed to yield in absorbing the energy of the ruptured pipe.

14.4.5.2 Pipe Restraints

Piping restraints were designed to accommodate the loading induced by the reaction or whipping forces from design basis breaks. For a specific break location, the pipe restraint will accommodate a longitudinal break extending one pipe diameter on each side of the high-stress point or a circumferential break at the high-stress point.

Effect of the Replacement Steam Generators on Unit 1

Each high-energy system was reviewed in the UFSAR to determine its proximity to other systems necessary for the safe shutdown of the reactor. The mass and energy releases during high energy line breaks are not increased by the RSGs. Also, the RSG has no effect on the pipe breaks outside of containment from a structural loading standpoint. Consequently, the thrust forces do not increase as a result of steam generator replacement. Therefore, the discussion in this section is not affected by steam generator replacement.

14.4.6 Compartment Pressures and Temperatures

Based on the break locations chosen according to the criteria of Section 14.4.2.2, representative break locations were analyzed for their effect on compartment pressures. The results from two computer codes (TMD and COMPARE) were used to determine the pressure and temperature associated with the postulated breaks.

The TMD computer code was used to analyze the resultant pressure and temperature from larger breaks (double ended main steam and feedwater line breaks). The COMPARE code was used to analyze the effects of the breaks postulated for the four-inch main steam to auxiliary feedwater line.

Appendix O of the original FSAR contains the results of HELB outside containment. In 1984, the issue of steam generator superheat with the MSLB outside containment was raised in NRCIE Information Notice 84-90 (Reference 2). This notice described a potential problem pertaining to plant analysis and equipment qualification with respect to a MSLB with releases of superheated steam. An analysis for the affected compartments was performed in response to this Notice (Reference 1) and updated to reflect changes to plant operating parameters. These EQ analyses were reviewed in response to the HELB program reconstitution. This later analysis only is limiting for temperature due to the superheat issue, and the original analysis remains bounding for pressure response on structures.

Evaluation of TPR, RSGs, and MUR on Unit 1

An evaluation for D.C. Cook Unit 1 related to the Steamline Break mass and energy release analyses outside containment analysis has been performed. Numerous plant changes for Unit 1 are addressed by this evaluation, including Thimble Plug Removal (TPR), Replacement Steam Generators (RSGs), and a Measurement Uncertainty Recapture (MUR) uprate program in conjunction with redefined containment volume distribution. This evaluation addresses the combined effect of these changes.

The evaluation includes analysis of the 4.6 ft^2, 1.0 ft^2, 0.6 $ft^2$and 0.1 ft^2 break areas at 3327 MWt NSSS power.

The current Analysis of Record (AOR) for MSLB outside containment is conservative for all Unit 1 operating parameters with the exception of the steam pressure and temperature. Therefore, the AOR remains bounding for mass and energy release data outside containment.

14.4.6.1 Compare Code

COMPARE is a computer code developed to analyze transient containment and other sub-compartment pressure response for nuclear power plants. The sub-compartments are treated as discrete volumes connected by junctions. The volumes are characterized as a homogenous mixture, assumed to be in thermodynamic equilibrium. The mixture consists of any one, or a combination of:

1. Steam
2. Two-phase water to its triple point
3. Any three perfect gases such as air, helium, etc.

Transient flow conditions are considered as quasi-steady approximations with a correction for fluid inertia effects. Several different methods can be used for handling fluid flow characteristics.

COMPARE was only used to analyze the effects of breaks postulated for the four inch steam supply line to the AFW pump turbines.

14.4.6.2 TMD Computer Code

The analytical methods used by the TMD computer code are described in WCAP-8078. The code has been modified to reflect the modeling required for: 1) the East Steam Enclosure and 2) the West Steam Enclosure/main steam accessway. The physical nodalization and the computer flow paths are shown in Figures 14.4.6-1 through Figure 14.4.6-4.

14.4.6.3 GOTHIC Computer Code

The GOTHIC (Generation of Thermal-Hydraulic Information for Containments) code is a general-purpose thermal-hydraulics computer program for design, licensing, and operating analysis of nuclear power plant containments and other compartments or buildings. The GOTHIC 6.0 (QA) computer code was used to analyze the environmental temperature and pressure response of a HELB. This current version of the computer code was developed by Numerical Applications, Inc. (NAI) of Richland, WA under contract to the Electric Power Research Institute (EPRI).

Within the GOTHIC computer code, conservation equations for the mass, momentum and energy are solved over a three-field fluid, consisting of a mixture of liquid water, liquid droplet, and water vapor. Non-equilibrium conditions may have existed simultaneously between the three fields. Gases may have been present in equilibrium in the vapor field. Mechanistic models are used to describe the mass, momentum and energy transfer between fields.

14.4.6.4 Input for Specific Break Analyses

14.4.6.4.1 Full Size Breaks or Main Steam Feedwater Lines

Full size main steam and feedwater line breaks were analyzed with regard to high compartment differential pressure in the main steam accessway and the steam enclosures. Conservatively representative terminal end breaks for both the main steam and feedwater lines were considered in the steam enclosures (see Figure 14.4.2-1 and Figure 14.4.2-1A, Figure 14.4.2-2 and Figure 14.4.2-2A). A conservatively representative feedwater line break was considered in the main steam accessway (Figure 14.4.2-2 and Figure 14.4.2-2A). Figures 14.4.6-1 through Figure 14.4.6-4 show schematics and TMD nodal networks for the enclosures. Table 14.4.6-1 and Table 14.4.6-2 give the volumes and vent areas of the TMD elements. Mass flow rates were determined for the following break locations:

a. A 4.6 ft^2 area main steam line break in east and west steam enclosures for the superheat issue and a 4.27 ft^2 break for the pressure response.

b. 14-inch main feedwater lines in east and west steam enclosures just outside the containment penetration.

c. 20-inch main feedwater line at tee with 30-inch main feedwater line in the main steam accessway.

The mass and energy releases for case (a) were calculated assuming a break area of 4.6 ft^2. The break for the east or west steam enclosure is in the same region of the pipe as the assumed break in the fan room discussed in Section 14.3.4.2.3. The mass and energy release to the east and west main steam enclosures (WMSE and EMSE) is shown in Table 14.4.6-5 and Table 14.4.6-5A. These data were used to generate a maximum temperature profile. The mass and energy releases used for the pressure response are provided in Tables 14.4.6-6a through Table 14.4.6-6c. The mass and energy release rates for case (b), feedwater break, were calculated for a complete double-ended break (break area for both forward and reverse flow was .85 ft^2). The mass velocity for forward flow was determined using the Moody model (CD = 1.0) assuming saturated liquid discharge of feedwater at 450°F. Backflow was determined using the Moody model (CD= 1.0) for saturated liquid discharge at a pressure corresponding to zero load steam generator pressure of 1020 psia, and was assumed to continue until zero load steam generator inventory plus the piping volume had been discharged. This assumes the worst condition on each side of the break location, leading to conservative pressure results. The blowdown is given in Table 14.4.6-6.

The mass and energy release rates for case (c), feedwater break, were calculated using a break area of 1.77 ft^2 for flow in both directions. Forward flow and reverse flow were determined using the Moody model (CD = 1.0) assuming saturated liquid discharge of 450°F. Backflow was terminated when the downstream piping volume was calculated to have emptied. This blowdown is given in Table 14.4.6-7.

14.4.6.4.2 Postulated Breaks Associated with the Four-Inch Main Steam to Auxiliary Feedwater Pump Line

One break location was analyzed with regard to the postulated consequences of high differential pressure associated with the break of the main steam to auxiliary feedwater line. The break location is inside the turbine driven feedpump room. Other break locations along this four-inch line were not analyzed for the specific differential pressures resulting from the break for one or more of the following reasons:

1. Intermediate breaks have been eliminated.
2. A more severe break had been analyzed in the potential break locations (at the terminal end inside the enclosure).

14.4.6.4.3 Other High Energy Lines

The consequences of high compartmental differential pressures associated with other high energy lines were not analyzed specifically because comparison with the analysis for the four-inch steam line to the auxiliary feedpump room indicated that pressure considerations along other, smaller, lines located in larger compartments would not be limiting.

14.4.6.5 Results of Analysis

Main Steam and Feedwater Line Breaks

The results of the analysis of the main steam and feedwater line break analyses with regard to peak compartment differential pressures are presented in Tables 14.4.6-9 through Table 14.4.6-13. These pressures are defined in terms of the nodalization presented in Figure 14.4.6-2 and Figure 14.4.6-4. The relation between the nodalization used in the computer analysis and the affected slab locations is presented in Table 14.4.6-8. Detailed pressure transients for the breaks analyzed are presented in Figures 14.4.6-8, through Figure 14.4.6-11. For the case where a steam and feedwater line break was analyzed in the same compartment, only the limiting transient is shown.

Main Steam to Auxiliary Feedwater Pump Breaks

The results of this analysis show the peak differential pressure to be less than 1.75 psi and the maximum temperature to be 298°F. A comparison with steady-state hand calculations has shown this value to be a very conservative estimate of the differential pressure in the compartment.

14.4.6.6 Structural Capability to Withstand Pressure

To determine the capability of the auxiliary building to withstand forces in addition to those imposed by the support of equipment, pipe, ducts, and electrical cable tray, an additional equivalent static analysis of the main steam accessway and east and west steam enclosures was made. The analysis considered taking the members from their working condition to 0.90 f_y for the reinforcing bars and 0.9 f_c' for the concrete.

The results of this analysis are presented in Tables 14.4.6-15 through Table 14.4.6-17 and summarized in Table 14.4.6-8. The analysis takes into account only the pressure capability once the initial peak has stabilized. Dynamic analyses accounting for the initial peaks and jet forces are included in the results shown in Table 14.4.6-18, Table 14.4.6-19 and Table 14.4.6-20.

Table 14.4.6-8 presents the slab capability in light of the postulated pressures once the initial peak has stabilized. These results show that with two exceptions, the panel static capability is capable of resisting anticipated design basis loads. However, as noted in Section 14.4.9.1, the slab was modified to accommodate the consequences of the postulated overpressurization.

Table 14.4.6-18 takes into account the combined effects of postulated pressure rise and jet effects due to postulated circumferential breaks. The results from this analysis have shown that, with two exceptions, existing portions of the east and west steam enclosures were capable of withstanding calculated peak differential pressures. One is the slab WSL-1 mentioned previously, the other is slab EW-4 which is discussed in Section 14.4.9.

Table 14.4.6-19 and Table 14.4.6-20 take into account the combined effects of pressure rise and longitudinal breaks at the postulated break locations. Slabs requiring protection are noted. For the break of the four-inch main steam to auxiliary feedwater pump line in the auxiliary feedwater pump room, the room peak pressure was evaluated and found to be within the structural capabilities.

14.4.6.7 References for Section 14.4.6.

1. MSLB Environmental Analysis, Donald C. Cook Units 1 and 2, Impell Report No. 01-0120-1524, Revision 0, September 1986.
2. NRC IE Information Notice No. 84-90, "MSLB Effect on Environmental Qualification of Equipment," December 7, 1984.
3. NS&L Calculation TH 90-07, December 12, 1990.
4. NS&L Calculation TH 93-01, March 4, 1993.
5. NS&L Calculation TH 95-15, December 12, 1995.

14.4.7 Description of Jet Impingement Load Analysis

The jet impingement load is defined as the load on a component (piping, equipment, or structure) of the undeflected jet from an instantaneous circumferential or longitudinal break in a high-energy pipe. The jet forces or loads at the point of rupture are consistent with those used in the pipe whip analysis, and were based on the most severe fluid pressure and temperature conditions occurring during normal operating modes. At the point of postulated rupture, full cross sectional area breaks were assumed to discharge the high energy fluid at a rate equal to the critical flow rate. Jet blowdown impingement forces were conservatively calculated by the following formula.

$$F = 1.2 \times P \times A$$

Where:

F = Impingement force in pounds

P = Maximum operating pressure in psig

A = Area = cross sectional area of rupture pipe in square inches

Based on the results included in NUREG/CR 2913, the effect of the jet impingement is limited within ten pipe diameters from postulated High Energy Line Breaks and ten diameter equivalent for critical cracks. The critical crack size is defined to be one-half the pipe diameter in length and one-half the wall thickness in width. The impingement effects for a critical crack will be modeled as a rupture of a pipe with a diameter equivalent to the critical area.

For pipes initially containing steam, the jet flow away from the point of rupture is assumed to diverge at an included angle (ϕ) of 20°. The angle of divergence (ϕ) for a jet from pipes initially containing saturated or sub cooled water is determined by the expansion ratio necessary to

establish equilibrium at ambient pressure and depends on the initial fluid conditions. Hence, the area of the jet at some distance from the point of a longitudinal break is:

$$A_1 = [L_1/2 + (L_3) \text{ Tan } (\phi/2)] \text{ x } [L_2/2 + (L_3) \text{ Tan } (\phi/2)]$$

Where:

A_1 = area of the jet

L_1 = width of break

L_2 = length of break

L_3 = distance to target

ϕ = angle of divergence

The locations that require protection from postulated jets have been determined. These locations are the intersections between cabling required to support operation of equipment listed in the SSEL (Table 14.4.2-1 Unit 2 and Table 14.4.2-1a, Unit 1) and the high energy lines. Protection is provided at these locations by either 1) installation of impingement barriers, 2) moving the cable to non-critical locations, or 3) other appropriate measures.

14.4.7.1 Jet-Impingement Pressures and Temperatures

Pressure

Trays bearing cables associated with equipment required to bring the reactor to a cold shutdown after a high energy line incident are protected where required from jet impingement forces.

Temperature

All control and instrument cables used in the auxiliary building are qualified for use inside the containment for accident conditions. The qualification test environments envelope the temperature excursion for the worst case high energy line break. Power cables used in the auxiliary building for Class 1E service are all rated at 90°C (190°F) for continuous operation. Production samples of power cable were aged at 121°C (250°F) for 168 hours to verify the retention of adequate physical properties. Power cable standards anticipate emergency operation at 130°C (266°F) for one 36-hour period per year without significant loss of life or function. These temperatures exceed the high energy line break requirements.

Effect of the Replacement Steam Generators on Unit 1

Evaluations for the RSGs have shown that the mass and energy releases during high energy line breaks are not increased by the RSGs. Also, the RSG has no effect on the pipe breaks outside of containment from a structural loading standpoint. Therefore, the discussion in Section 14.4.7 is not affected by steam generator replacement.

14.4.7.2 Jet Erosion of Concrete

The erosion of concrete by steam jets was evaluated in WCAP-7391, "Pressurized Water and Steam Jets Effects On Concrete" by Westinghouse Atomic Power Division. In summary, five reinforced concrete beams were subjected to steam jets with nozzle diameters of 1, 2, and 4 inches. The distances investigated between nozzles and beams were 1 foot and 4 feet; the initial system pressure was 2250 psi. The results are as follows:

Section 4, "Evaluation of Beam Behavior," pg. 4-2:

"4.2 EROSION EFFECTS

> The erosion under all beam tests was observed to be (at most) 30 mils of surface paste removal, with no significant loss of either fine or coarse aggregate. The resultant surfaces showed the same appearance as would be present after light sandblasting. It can thus be concluded that short-term erosion of concrete surfaces as a result of either a loss-of-coolant accident or steam line break is definitely not a design consideration."

Section 5, "Conclusions and Design Recommendations," pg. 5-1:

"5.1 EROSION

> As a result of the test program, no evidence was seen that concrete erosion should be a concrete design consideration. On this basis, it is recommended that no consideration be given to erosion effects in the design of concrete structures to withstand the blowdown loads for pressure and temperature conditions used in this study."

14.4.7.3 Impingement Barrier

The following requirements were met for the application of an impingement barrier at design basis break locations as a means of preventing the jet from impinging on needed safeguard systems in the event of a pipe break:

- The impingement barrier was designed to withstand the jet forces resulting from the escape of high energy fluid at the postulated pipe break location.

- The stresses imposed on the impingement barrier during dynamic pressurization shall be limited for membrane stresses produced by pressure to 90 percent of Yield Strength.
- The impingement barrier shall be a Class I structure.

Barriers are provided to protect needed equipment from adverse jet force loadings from all design bases events. The barriers provided will withstand the impingement loads, as well as the normal working loads.

14.4.8 Concrete Structure Evaluation

14.4.8.1 Containment Integrity

The present restraint system design precludes any functional damage to the containment building or to any penetrations, seals, and engineered-safety-features related piping or cables due to any of the previously described postulated breaks.

The containment shell was analyzed to verify the ability to maintain integrity with the presence of the short term pressure peak that occurs locally due to a postulated rupture of a main steam line in either the east or west steam enclosures.

Analysis has shown the ability of the structure to withstand a local pressure transient inside the containment in this same area of the containment shell (fan accumulator room) for a greater differential pressure than developed in this case for a break outside the containment.

Effect of the Replacement Steam Generators on Unit 1

Evaluations for the RSGs have shown that the mass and energy releases during a main steam line break or a LOCA are not increased by the RSGs. Therefore, the discussion in Section 14.4.8 is not affected by steam generator replacement.

14.4.8.2 Compartment Pressure-Loading Stress

The walls and slabs in the following areas were analyzed to determine the ultimate capability for a postulated high energy line break outside containment:

a. Main Steam Enclosure (East)

b. Main Steam Enclosure (West)

c. Auxiliary Building Adjacent to Main Steam Line Accessway.

Figure 14.4.8-1 and Figure 14.4.8-2 show the layout and identification marks for the walls and slabs analyzed.

To accommodate the consequences of the postulated main steam line breaks discussed previously, pressure relief panels are provided on the East and West Main Steam Enclosures that provide venting area to limit the maximum pressure developed in the enclosures. In addition, it was necessary to modify the roofs of the East and West Steam Enclosures to provide additional venting area. The extent and nature of these modifications are presented in Section 14.4.9.

In order to protect the equipment listed in Table 14.4.2-1 and Table 14.4.2-1a the (SSEL) from the consequences of high compartment differential pressure following the feedwater line break, it was necessary to analyze for the break consequences inside the main steam accessway and in the main steam enclosures. Details of the analysis are presented in Section 14.4.6. These analyses have shown that the postulated feedwater line break is less severe than that associated with the steam lines in the East and West Steam Enclosures.

Similar analyses for the postulated feedwater line breaks have been performed in the main steam accessway and the results of the analyses are also presented in Section 14.4.6. As a result of these analyses, it was found that no additional structural modifications in the main steam accessway were required to accommodate the consequences of the feedwater break.

The combined time-history representing the compartment pressure and the jet impingement force was used for the non-linear dynamic analysis. Where a slab or wall panel was not affected by jet impingement, only the compartment pressure time-history was used. No other load was assumed to be acting simultaneously.

14.4.8.3 Structural Resistance to the Loading

The resistance or ultimate load capability of each wall or slab panel was determined as follows:

a. The ultimate moment capacity was determined using principles presented in ACI 318-71. The exact expression used is given in Table 14.4.8-1.

b. Failure by flexure occurs when the yield line pattern (collapse mechanism formed by plastic hinge lines) giving the lowest over-burden load is formed.

c. The ultimate shear capacity and permissible shear stress were determined according to the provisions of ACI 318-71. The exact expressions used are given in Table 14.4.8-1 and Table14.4.8-2.

d. The ultimate load capability determined by the above requirements defines the yield force of the elastic-plastic resistance function used for the non-linear dynamic analysis.

e. The allowable limit of the ductility factor, defined as the ratio of the maximum displacement to elastic displacement, is 10.0 in panels where flexure governs the failure and 3.0 in panels where shear or diagonal tension governs.

14.4.8.4 Analysis

The following analyses were performed for walls and slabs identified in Figure 14.4.8-1 and Figure 14.4.8-2.

14.4.8.4.1 Evaluation of Ultimate Pressure Capacity

The ultimate pressure capacities of the structural elements were evaluated using yield line theory. The yield line method assumes that after initial cracking of the concrete at points of maximum moment, yielding spreads until the full moment capacity is developed along the length of the cracks on which failure will take place. Tests indicate that the actual location and extent of these lines differs only slightly at failure from theoretical ones. Use of the idealized yield lines results in little error in the determination of the ultimate resistance, and that error is on the side of safety.

To assure ductile behavior, the actual tension reinforcement ratio "p" was compared with the balanced reinforcement ratio "pg," thereby indicating sufficient ductility of the structure. The ultimate load was evaluated by assuming a yield line pattern and using the virtual work principle. For the most general cases of (i) panels with all four edges fixed, and (ii) panels with three edges fixed and one edge free, solutions were obtained for minimum uniformly distributed loads, presented in Figure 14.4.8-3 and Figure 14.4.8-4. For nonstandard cases, solutions were worked out separately. In all cases, the possibility of local collapse of a part of the structural element was considered. For the pressure capability obtained by yield line analysis, the ultimate shear stresses were evaluated at critical sections to check the possibility of premature shear failure. The shear expressions are listed in Table 14.4.8-2. The expression for the permissible ultimate shear stress is shown in Table 14.4.8-1. Where shear governed, the pressure capability was reduced in proportion to the ratio between the permissible ultimate shear stress and the calculated ultimate shear stress.

The presence of openings was considered in the analysis. The effect of openings depends on their location, size, and shape. Small compact openings located away from regions of high stress were ignored. When they were located near yield lines, the length of the yield line in the virtual work equation was reduced, by conservatively assuming the edge of the openings to be located on the yield line. In some cases, the possibility of gross relocation of yield lines due to large openings was also considered. Solutions of such cases were obtained by modifying the yield line pattern for minimum energy and, hence, minimum load.

14.4.8.4.2 Evaluation of Ultimate Impingement Force Capacity

To adapt the use of yield line analysis to this situation, the jet impingement loads were assumed to be concentrated point loads. Yield line patterns assumed for the standard cases are presented in

Figure 14.4.8-5, Figure 14.4.8-6, and Figure14.4.8-7. Punching shear was checked using the impingement area as the bearing area of the concentrated load.

The effects of openings in panels were considered to be negligible because heavy bands of reinforcement were placed around all openings. A jet directed at an opening will pass through the opening to affect an adjacent area, but will have insignificant effects on the panel itself.

14.4.8.4.3 Capacity Under Combined Pressure and Impingement Force

For any panel, the pressure capacity is generally different from the impingement force capacity. From yield line theory, it was determined that a straight-line interaction curve between the pressure and impingement force capacities gave a conservative estimate of the combined capacity. The combined capacity due to a particular combination of pressure and impingement force, equivalent to a point on the interaction curve, was estimated by taking the weighted average of the pressure and impingement force capacities based on the proportions of applied pressure and impingement force. This combined capacity defines the yield force of the elasto-plastic resistance function used for the non-linear dynamic analysis, when this combined loading acts on the panel under consideration.

14.4.8.5 Structural Components

Existing Seismic Class I structures have been reviewed for their adequacy, and where required, these structures were modified to the extent necessary to ensure their integrity. Seismic Class I structural elements such as floors, interior walls, exterior walls, building penetrations and the building as a whole, were analyzed for possible reversal of loads due to the postulated accident.

Failure of any structure, including seismic Class II or Class III structures, caused by the postulated accident was reviewed to assure that it would not cause failure of any other structure, system, or component in a manner to preclude the capability to bring the plant to a safe shutdown condition.

The auxiliary building, a Seismic Class I structure, was designed to Working Stress Criteria (WSD) in accordance with Building Code ACI 3l8-63 for all operating load conditions including the operating basis earthquake. The structure was designed to Ultimate Strength Criteria (USD) in accordance with ACI 3l8-63 for all loading conditions which include the design basis earthquake and/or the design basis pipe breaks.

The structural elements which would be exposed to a pressure differential across the element are analyzed for this additional load in accordance with USD concepts of the ACI Building Code.

14.4.9 Plant Modifications

14.4.9.1 Structural

To accommodate the consequences of the postulated main steam line break in the enclosure, it was found necessary to modify the roofs of the east and west steam enclosures to provide additional vent area. The modified roof design is presented in Figure 14.4.9-1 and Figure 14.4.9-2. The modified roof design provides 491 additional square feet of vent area. This modification results in pressures that are below the design pressures in all but three areas of the enclosures. These exceptions are slabs W-SLl in the west steam enclosure (Figure 14.4.8-1) and slab EW4 (Figure 14.4.8-1) in the east steam enclosure. Slab W-SL1 was modified to accommodate the consequences of the postulated accident without failure of that segment of the slab which is safety related. The lightly reinforced section of E-W4 can fail and still allow the plant to be safely shut down.

Analyses taking into account the combined effect of jets and pressure rise have shown that jets produced by postulated full-area circumferential breaks would only cause slab W-SL1 to fail in a manner where safe shutdown of the plant would not be assured. As indicated above, this slab was modified to accommodate the consequences of the postulated accidents, such that safe shutdown was assured.

Postulation of full area longitudinal breaks will, however, result in significant design loadings on the areas indicated in Table 14.4.6-19 and Table 14.4.6-20. These walls are protected from the direct effects of a jet.

CCW Pump Protection

Main Steam (MS) and Feedwater (FW) pipes are located within a pipe chase in the vicinity of the CCW pump room. A critical crack in any of these pipes could impact any or all of the three (3) access doors, and cause them to fail to remain closed. This would create a harsh environment in the CCW pump area, for which the pumps are not qualified. Therefore, the critical crack could conceivably disable all of the CCW components for both units. In order to resolve this concern, a modification was made to install door restraints on all three (3) of the pipe chase access doors, so that they would not fail following a pipe rupture. (Reference 1)

West Main Steam Enclosure Door Enhancement

Doors 1-DR-AUX428 and 2-DR-AUX429 which open to the Auxiliary building roof from the West Main Steam Enclosures of Unit 1 and Unit 2 (respectively) were modified to ensure that the doors would open before 1 psid is reached following a Main Steam line break. These modifications

were completed to ensure that the EQ requirements for the equipment within the enclosures are met during HELB conditions.

Turbine Building Block Wall Reinforcement

To preclude an adverse consequence from the postulated steam line break in the Turbine Building to the Essential Service Water Pumps, roof vent area is provided to assure the Turbine Building pressure does not exceed the structural capability of the block wall separating the Turbine Building from the Screenhouse. The block wall has been reinforced to achieve the required structural capability.

14.4.9.2 Mechanical

Based on the pipe rupture analysis, it has been concluded that:

1. No changes in pipe routing of any high-energy lines are required.
2. No movement of the mechanical equipment or valves listed in the SSEL (Table 14.4.2-1 and Table 14.4.2-1a) are required; instrumentation was relocated or protected as necessary.
3. Sufficient drainage capacity was provided to preclude flooding of any equipment required for safe shutdown of the reactor.
4. Two additional major pipe restraints were required for each unit. One restraint was added along the main feedwater line to steam generators #1 and #4 where the line splits from a single 20-inch line to two 14-inch lines outside the containment. The second restraint was at the location inside the main steam accessway, where the 20-inch feedwater line branches from the 30-inch feedwater line. These restraints prevented pipe movement from damaging the equipment listed in the SSEL. Gaps between piping and restraints were measured at temperatures. (Note: These restraints are no longer required since intermediate breaks were eliminated on feedwater piping).

AFW Pump Room Protection

The AFW pump rooms have been modified to protect them from the consequences of a HELB in the Turbine Building as well as a break of the 4 inch main steam piping in the TDAFW pump room. The rooms have been completely isolated from each other as well as the Turbine Building by maintaining the doors closed, removing the ventilation fans, and sealing opening. Safety related coolers are installed in each TDAFP room to maintain its ambient temperature at or below 85°F

normally and 104°F with the pumps running. Safety related coolers are installed in all MDAFP rooms. Each MDAFP Room Cooler is designed to maintain the room temperature at or below 85°F with the AFP in standby and at or below 115°F with its AFP running. The coolers reject heat to the ESW system and are powered from safety related sources.

A single cooler has been installed in each of the East and West motor driven pump rooms. These coolers are powered from the same division as the motor driven AFW pump they serve. Their ESW supply is also from the same division as its power. Redundant coolers have been provided in the TDAFW pump room. The coolers are powered from opposite divisions and are supplied with ESW from the same division as their power. This ensures that a failure of a division of power will not impair both a motor driven AFW pump and the Turbine Driven AFW pump.

Blow out panels have been installed in the TDAFW pump room to prevent over-pressurization in the event of a line break in that room. The limiting room pressure has been calculated, and the room structure as well as the seals has been qualified for this pressure.

Any safety related equipment in the AFW corridor has been qualified for the HELB environment resulting from a worst case break in the Turbine Building, a steam line break in the TDAFW pump room which may cause the room doors to open, or a critical crack of the TDAFW pump steam piping in the corridor

Pipe Support Modifications

Main Steam (Unit 1 only), Feedwater, Steam Generator Blowdown, CVCS Letdown, Condensate and Heater Drain pipes were re-analyzed. The objective was to determine ways to reduce stresses, in order to eliminate the need to postulate any intermediate breaks in accordance with the criteria discussed in Section 14.4.2. It was determined that some of the existing pipe supports need to be modified or removed, and new supports need to be added to the piping. Modifications were implemented to carry out these changes. (References 3, 4, 5)

SWGR Rooms

The ventilation air from the East and West 600V Transformer Rooms, CRID Room and CRD Equipment room was exhausted to the Turbine Building through a normally open roll-up door. A steam line break in the Turbine Building would result in steam entering the rooms listed above. The resulting environment would impact the mild environment zone classification of these rooms.

The following modifications were made to alleviate these concerns. The roll-up door and the wire mesh barrier separating the West 600V Transformer Room and Turbine Building was replaced

with a single fire rated and security controlled door. HELB backdraft dampers were provided between the switchgear area and the Turbine Building. (Reference 6)

14.4.9.3 Electric and Instrumentation and Control

Based on the break location criteria defined in Section 14.4.2.2, it was concluded that minimal protective measures or revisions to present electrical cable routings, instrumentation locations, and controls routings were required. For trays containing cables associated with equipment, listed in the SSEL (Table 14.4.2-1 and Table 14.4.2-1a), they are limited in exposure to the environment for which they were designed (Section 14.4.7.1) by provision of shields, enclosures, or pipe restraints as necessary. Additional reviews were performed and no equipment or cables were identified which required protection from damage resulting from breaks or cracks.

Effect of the Replacement Steam Generators on Unit 1

Evaluations for the RSGs have shown that the mass and energy releases during high energy line breaks are not increased by the RSGs. Also, the RSG has no effect on the pipe breaks outside of containment from a structural loading standpoint. Therefore, the discussion in Section 14.4.9 is not affected by steam generator replacement.

14.4.9.4 References

1. CNP Design Change Package 2-DCP-4258, Structural Door Restraints for # 406, 407 and 408. (No Unit 1 Mod).
2. CNP Limited Design Change Package 2-LDCP-4614, West Main Steam Enclosure Door Enhancement. (1-LDCP-4736, Unit 1)
3. CNP Design Change Package 2-DCP-4259, Feedwater Support Modifications. (1-DCP-4790, Unit 1)
4. CNP Design Change Package 2-LDCP-4447, Support Modifications to CVCS Letdown and BD Piping. (1-DCP-4789, Unit 1)
5. CNP Limited Design Change Package 2-LDCP-4535, Heater Drain Line Piping Support Removal. (No Unit 1 Mod)
6. CNP Design Change Package 2-DCP-4247, HELB dampers/doors to 600V transformer Rooms. (1-DCP-4578, Unit 1).
7. CNP Engineering Change Package EC-0000051586, "Screenhouse - Turbine Building Masonry Wall HELB Boundary Modification".

14.4.10 Environment

14.4.10.1 Location of Required Equipment

The equipment required to bring the reactor to cold shutdown after a high energy line incident, and their locations in the plant, are listed in the Safe Shutdown Equipment List (SSEL) (Table 14.4.2-1 and Table 14.4.2-1a).

14.4.10.2 Equipment Capability for Operation During Incident

Motors

All motors have a Type "B," or better, insulation. Type "B" insulation consists of mica, asbestos, fiber glass, other inorganic materials, and synthetic resins capable of operation at a total temperature of 130°C (266°F). Motors for the equipment listed in the SSEL are located in areas that are not affected by the adverse environment. Total temperature includes expected temperature rise above ambient due to electrical current passing through the insulated wiring in question. Total temperature is important in that it is the maximum value that the insulation can be exposed to without degradation of the insulation. As a rule of thumb, each 10°C of temperature rise above the specified total temperature will approximately halve the effective life of Type "B" insulation. Degradation of the overall life of the motor after the accident is not of prime concern. There is no immediate problem of the device failing to function during the accident and for a reasonable time thereafter.

Solenoid Valves

All solenoid valves have a Type "H" insulation. This insulation consists basically of silicones capable of operating at an ambient temperature of 212°F. All solenoid valves exposed to an adverse environment are encased with either a weather proof or explosion proof water tight housing.

Electrical Cables

All control and instrument cables used in the auxiliary building are qualified for use inside the containment for accident conditions. The qualification test environments envelope the temperature excursion for the worst case high energy line break. Power cables used in the auxiliary building for Class 1E service are all rated at 90°C (194°F) for continuous operation. Production samples of power cable were aged at 121°C (250°F) for 168 hours to verify the retention of adequate physical properties. Power cable standards anticipate emergency operation at 130°C (266°F) for one 36-hour period per year without significant loss of life or function. These temperatures exceed the

high energy line break requirements. Because of this, no routing changes or cabling replacements were required.

Motor Starters

Motor starters are draw outs mounted in centralized motor control centers which are reasonably drip-proof, but not sealed from gradual penetration of temperature and humidity.

Instruments

The instruments required for the high energy line incident were enclosed and /or modified to withstand the anticipated adverse environment. The affected instrumentation supply and signal lines and cable routings were reviewed and relocated or protected as necessary.

14.4.10.3 Seals

The control room and electrical switchgear room are provided with seals on doors and penetrations which adequately protect these areas from the adverse environment associated with the high-energy line incident. For those areas containing equipment which is not qualified to perform its function under this adverse environment, seals are provided on doors and penetrations.

These areas contain equipment which is not qualified to perform its function under this adverse environment.

The only door from the above areas to the auxiliary building is an emergency fire exit from the back of the control room panel. This door is under strict administrative control, sealed to control room isolation criteria, and exits to an area containing no high energy lines.

14.4.10.4 Ventilation

Protection from Auxiliary Building Environment

No structural modifications were required to prevent the adverse steam environment from entering the electrical switchgear room or the control rod drive equipment room. Seals on the doors adequately control the steam input from a line rupture.

Protection from Turbine Building Environment

The Seismic Class I battery rooms, and the 4160 volt switchgear rooms are similarly isolated from any adverse environment resulting from postulated high-energy pipe ruptures in the turbine building by the inclusion of back-draft or fire curtain dampers in each ventilation duct penetrating its boundary and by sealed doors. The ventilation systems for Seismic Class I auxiliary feedwater pump rooms do not interface with the Turbine Building atmosphere.

Effect of the Replacement Steam Generators on Unit 1

Evaluations for the RSGs have shown that the mass and energy releases during high energy line breaks are not increased by the RSGs. Therefore, the discussion in Section 14.4.10 is not affected by steam generator replacement.

14.4.11 Electrical Equipment Environmental Qualification

Introduction

In December 1982, the NRC issued a Safety Evaluation Report (SER) on the Donald C. Cook Nuclear Plant electrical equipment environmental qualification program (Reference 1). Attached to this SER was a four-volume Technical Evaluation Report (TER) prepared by the NRC's consultant, Franklin Research Center (FRC). Initial responses to the FRC TER were submitted to the NRC on January 24, 1983, March 4, 1983, and March 29, 1983, via letter Nos. AEP:NRC:0775, AEP:NRC:0775B, and AEP:NRC:0775A, respectively (References 2, 3, and 4).

Between receipt of the TER and submittal of the responses to the NRC, the final rulemaking on electrical equipment environmental qualification, 10 CFR 50.49, was published in the Federal Register (Vol. 48, No. 15, dated January 21, 1983). This rulemaking went into effect as of February 22, 1983.

A list of equipment items believed to be within the scope of 10 CFR 50.49 has been provided to the NRC. This list, as currently revised, is provided in Reference 11).

On January 11, 1985, the NRC issued the final SER on environmental qualification for the Cook Nuclear Plant. Specific modifications to the facility have been scheduled pursuant to the SER and agreements with the NRC.

Analytical Bases

Detailed information regarding environment temperatures, pressures, radiation doses, and chemical sprays required for qualification of safety related electrical equipment is given here. The equipment must be qualified to demonstrate that it can perform its safety-related function following a high-energy line break (HELB). (See Section 14.4.2.2 for a definition of HELB). Loss-of-coolant accident (LOCA), main steam line break (MSLB), and feedwater line break are examples of HELBs. In addition, the environment resulting from a "critical crack" (see Section 14.4.2.2 for a definition of "design basis" or "critical" crack) in the 4" branch of the main steam line that feeds the turbine-driven auxiliary feedwater pump (TDAFP), was also considered. The location of the crack has been postulated at the most adverse location.

Limiting temperature and pressure for the LOCA and MSLB in various compartments are shown. Also shown are the predicted temperature and pressure profiles in containment (to which equipment is qualified), the radiological dose rates, and the chemical spray requirements. The postulated accidents and the environmental conditions calculated for the accidents are also

discussed in this section. However, the qualification conditions or testing profiles for each safety-related equipment item are not included.

Additional break locations exist within the containment structure as determined by High Energy Line Break (HELB) analyses performed per the allowances given in Generic Letter 87-11. These break locations were evaluated for their impact on surrounding structures and components. The results of these evaluations are contained within the HELB evaluation report. These breaks are of less magnitude (less mass and energy release) than the LOCA and MSLB and are bounded by those breaks.

14.4.11.1 Definition of Mild Environment

Section(c)(3) of 10 CFR 50.49 states that "Requirements for... environmental qualification of electrical equipment important to safety located in a mild environment are not included within the scope of this section. A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences". For Cook Nuclear Plant, a harsh environment is any location related to the pipe break analysis described in the FSAR, and having the potential for the temperature of operating electrical equipment or instrumentation to greatly exceed that of normal operating conditions, as well as a relative humidity of nearly 100%. For radiation considerations, a mild environment is one in which the integrated dose is less than 104 rads. For organic materials, radiation qualification may be readily justified by existing test data or operating experience for radiation exposures below 10^4 rads. For electronic components, however, failures in metal oxide semiconductor devices occur at somewhat lower doses. For this reason, radiation qualification for electronic components may have a lower exposure threshold.

14.4.11.2 HELB Inside Containment

The LOCA and the MSLB are considered inside containment, in addition to other smaller diameter High Energy Lines (HELs). The LOCA will result in maximum radiation doses, and elevated temperatures and pressures. The LOCA may also activate the containment spray system, producing an environment of chemical spray for some portion of the accident.

The MSLB will usually produce higher temperatures and pressures, but will release less radiation, although it may also activate the containment spray system. Radiation doses from the MSLB are essentially nil when steam generator tube integrity is maintained.

High Energy Line Break (HELB) evaluations for both Units for the following High Energy Lines inside the containment have been performed.

- 32” and 30” MS Piping
- 16” and 14” FW Piping
- 14” RHR Supply Piping
- 3” Letdown Line and 2” Drain Line
- 3” and 2” Letdown Piping
- 3” Normal Charging Line
- 4” Pressurizer Spray Line
- 3” and 2” Instrument Manifold Piping (Unit 2 only)
- 2” Steam Generator Blowdown Piping from SG #1 to CPN-6
- 2” Steam Generator Blowdown Piping from SG #2 to CPN-77
- 2” Steam Generator Blowdown Piping from SG #3 to CPN-78
- 2” Steam Generator Blowdown Piping from SG #4 to CPN-79

The postulated breaks for the above lines are terminal end circumferential breaks. In addition to the terminal end breaks, intermediate breaks were also postulated on the 3” and 2” Letdown Piping (U1), the 3” Normal Charging Line (Unit 2), the 2” SG Blowdown Piping from SG #1 to CPN-6 (Both Units) and the 2” Steam Generator Blowdown Piping from SG #3 to CPN-78 (Unit 1).

The LOCA and the MSLB provide bounding conditions for temperature, pressure, and radiation.

Environment conditions are further discussed below.

14.4.11.2.1 Temperature and Pressure

The long-term temperature and pressure profiles for the LOCA in Unit 1 and Unit 2 are shown in Figure 14.3.4-6 and Figure 14.3.4-7 of the respective Unit 1 and Unit 2 UFSAR Section 14.3.4. Temperature and pressure profiles for the MSLB inside lower containment are shown in Figures 14.3.4-11 through Figure 14.3.4-16 of the respective Unit 1 and Unit 2 UFSAR Section 14.3.4. Table 14.4.11-1 of the Unit 2 UFSAR tabulates the Unit 1 and 2 peak calculated temperatures and pressures for the LOCA and MSLB and feedwater line breaks inside containment.

14.4.11.2.2 Chemical Spray

Following the LOCA or MSLB, the containment spray system initially sprays a mixture of boric acid from the RWST and sodium hydroxide from the Spray Additive Tank. When the RWST low

level is reached, the containment spray pumps are realigned from the RWST to the containment recirculation sump. The containment recirculation sump water will consist of RWST water, sodium hydroxide, melted ice impregnated with sodium tetraborate, primary system water and accumulator water.

The Technical Specifications limits for capacity and boron concentration for the various contributors to the containment spray were used to evaluate the range of boron in the containment spray. Generally, containment spray will have a pH in the range of 7.0 to 10.0. This pH range is the result of a spray solution consisting of boric acid, sodium hydroxide, and after the transfer to cold leg recirculation, sodium tetraborate. If the Spray Additive Tanks are not isolated before the transfer to cold leg recirculation, the combination of recirculation sump water and sodium hydroxide from the Spray Additive Tank results in the maximum spray pH. The envelope for the maximum pH spray conditions is a pH of 13.1 and an operation time of 10 minutes.

14.4.11.2.3 Flooding Elevation

The flood level for the containment sump is 614'-0" for Unit 1 and 613'-6" for Unit 2 (Reference 9). Any safety related instrumentation located below the flood level will actuate before it becomes submerged except for containment high and low water level indicating switches, boron injection flow transmitters and their associated cable assemblies. This equipment is qualified to function after submergence.

14.4.11.2.4 Humidity

It is assumed that the containment atmosphere will be pure steam or a mixture of steam and noncondensibles at 100% relative humidity.

14.4.11.2.5 Radiation

Radiation doses inside containment are calculated by using the integrated gamma and beta radiation dose tables for either the upper or lower volume compartments of the containment. For devices above elevation 614'-0" (Unit 1); 613'-6" (Unit 2) the radiation doses in Table 14.4.11-2 are used. For devices that have been submerged below elevation 614'-0" (Unit 1); 613'-6" (Unit 2), the radiation doses in Table 14.4.11-3 are used.

Credit is taken for certain material covers to reduce the radiation dose. Only beta radiation is reduced by an attenuation factor. The attenuation factors for unit density material (e.g., water or organics such as cable jacket materials), for aluminum, and for steel are shown in Table 14.4.11-4, Table 14.4.11-5, and Table 14.4.11-6, respectively.

The inside containment integrated dose can then be calculated as follows:

Inside radiation dose (MRads) = Integrated Gamma (MRads) + Integrated Beta (MRad) x % attenuated Beta Factor.

Some equipment outside containment must also be qualified for radiation, for the LOCA event, due to radioactive liquids recirculating during the recirculation phase of the accident. Radiation doses to this equipment were determined from the calculations in Reference 10. The radiation exposure is based on the distance of a device to a discharge line. The majority of the outside containment radiation doses are calculated using a worst-case method, which assumes that the device is at the surface of a discharge line with the maximum outer diameter to wall thickness ratio. This worst-case method is used when the location of an outside containment device with respect to a discharge or sampling line cannot be ascertained. For those devices in which the distances to radioactive lines can be determined a reduction factor is used.

Table 14.4.11-7 lists the pipes and piping systems considered in calculating outside containment doses.

14.4.11.3 HELB Outside Containment

The categories of high-energy lines outside containment considered are identified in section 14.4.2.6. For each line, the criteria at the Cook Nuclear Plant require that breaks be considered at terminal ends and at intermediate points of high stress, and that "critical crack" be considered at any location.

Following the postulated break, equipment must be available to mitigate the consequences of the accident. No other accident other than the rupture need be considered.

Only temperature, pressure, and humidity need be considered for environmental consequences. Radiation is not a problem because of the limited amount of exposure. Letdown flow is radioactive, but was not considered to be a concern due to rapid isolation. Chemical environment is not a problem. Table 14.4.11-8 gives the limiting temperatures and pressures in various compartments due to a HELB outside containment.

Effect of Replacement Steam Generators on Unit 1

Evaluations for the RSGs have shown that environment temperatures, pressures, radiation doses and pH for safety related electrical equipment are not adversely affected by the use of the RSGs. Therefore, the discussion in Section 14.4.11 is not affected by steam generator replacement.

14.4.11.4 References

1. Letter dated December 30, 1982 (Varga to Dolan) enclosing the Safety Evaluation Report and Technical Evaluation Reports dated October 28, 1982.
2. AEP:NRC:0775, January 24, 1983 (Hunter to Denton).
3. AEP:NRC:0775B, March 4, 1983 (Hunter to Denton).
4. AEP:NRC:0775A, March 29, 1983 (Hering to Denton).
5. AEP:NRC:1067, "Reduced Temperature and Pressure Program Analyses and Technical Specification Changes," dated October 17, 1988.
6. Reference deleted.
7. Reference deleted.
8. Reference deleted.
9. Nuclear Safety and Licensing Calculation TH-97-16.
10. Nuclear Safety and Licensing Calculation RS-80-01.
11. DCC-QA105-QCN Revision 9, "Environmental Qualification Equipment List," February 13, 1996.
12. NED-2000-514-REP, “HELB Program – Target Evaluation Report”, Revision 0, April 20, 2000 (NDIS File MV-ENV-03).
13. Calculation SD-000727-001, “HELB: Identification of the Unit 1 High Energy Lines and Postulation of High Energy Line Breaks”.
14. Calculation SD-990825-003, “HELB: Identification of the Unit 2 High Energy Lines and Postulation of High Energy Line Breaks”.