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September 27, 1994

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Methodology for Postulating Passive Failure Pipe Breaks

REFERENCE: (a) Letter from Mr. J. C. Brons (New York Power Authority) to Document Control Desk (NRC), dated September 7, 1988, Methodology for Postulating Service Water System Breaks

(b) Letter from Mr. J. D. Neighbors (NRC) to Mr. J. C. Brons (New York Power Authority), dated June 12, 1989, Methodology for Postulating Service Water System Breaks for Indian Point 3 (TAC No. 69404)

This letter submits, for the Commission's review and approval, Baltimore Gas and Electric Company's (BGE) revised methodology for postulating single passive failures of the Calvert Cliffs Nuclear Power Plant. As an example of the proposed methodology, we have evaluated single passive failures of the Saltwater (SW) System piping in the Service Water (SRW) Pump Room during the long-term period following a Loss of Coolant Accident (LOCA). Information supporting BGE's revised methodology is presented in Attachment (1).

BACKGROUND

In accordance with the Calvert Cliffs licensing basis and as stated in the Calvert Cliffs Updated Final Safety Analysis Report (UFSAR), the SW System was designed to provide a continuous flow of cooling water to those systems and components necessary for plant safety during normal operation and under

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abnormal and accident conditions. During accident conditions, the SW System must provide the cooling water necessary to allow the engineered safety features to perform their intended function when subjected to:

a. The single failure of any active component at any time after an initiating event, or

b. The single failure of any active or passive component used during the long-term recirculation phase.

The UFSAR single failure analysis describes the passive failure as a rupture of the affected system. The failure is analyzed to determine the ability of the system experiencing the failure to perform its accident-mitigating functions, and to evaluate the capability of other plant systems to cope with flooding effects resulting from the passive failure. The current analysis demonstrates that the facility will be able to mitigate the consequences of an accident assuming a passive failure (i.e., rupture) during containment sump recirculation; however, the analysis did not clearly define rupture size, location, or timing. In order to resolve questions which have been raised regarding the basis for these criteria, it is now necessary to clarify the technical basis used to formulate the criteria. Therefore, a methodology change is proposed to clarify these aspects of the single failure analysis.

DESCRIPTION OF CHANGE

Baltimore Gas and Electric Company proposes the utilization of limited size breaks (through-wall leakage cracks) in the analysis of passive failures of SW System piping. This passive failure is postulated to occur no less than 24 hours after a LOCA.

A flooding analysis was performed to evaluate the effects of pipe breaks in plant piping systems. The analysis assumes a "critical crack" of dimensions equal to one-half the pipe diameter in length and one-half the wall thickness in width, as defined in Standard Review Plan (SRP) 3.6.1. For this particular case, the critical crack is assumed to occur in the largest SW piping (36") in the SRW Pump Rooms. The analysis, which is applicable to both Units 1 and 2, concludes that flooding would be limited to the SRW Pump Rooms, and that sufficient time would be available to terminate the source of flooding before equipment essential to long-term mitigation of a LOCA would be affected. The analysis is discussed in greater detail in Attachment (1).

Updated Final Safety Analysis Report changes reflecting the revised methodology will be incorporated in the UFSAR upon approval of this position by the Commission. Additionally, passive failures in other moderate energy piping systems at Calvert Cliffs may be re-evaluated using the criteria described in this letter. These evaluations will be subject to the same methodology and timeframe as the SW System flooding evaluation example summarized in Attachment (1).

CONCLUSION

In summary, to clarify the CCNPP licensing basis, BGE requests approval of a change to the methodology used in the evaluation of passive failures. The proposed change will limit flooding effects from moderate

energy piping system ruptures to the amount of leakage flowing through a through-wall leakage crack equal to one-half the pipe diameter in length and one-half the pipe wall thickness in width. This crack is assumed to occur in the largest pipe in the area to be evaluated, at least 24 hours after the accident. It is requested that this methodology and timeframe be approved for use in systems meeting the SRP 3.6.1 definition of a moderate energy system.

An analysis of the SW System failure in the SRW Pump Room using the proposed methodology has concluded that the ability of the plant safety systems to mitigate the consequences of a LOCA will not be diminished. This methodology change does not adversely impact plant safety. Furthermore, the proposed methodology is in conformance with current industry standards and regulatory guidance on the evaluation of passive failures. A similar analysis will be performed for other plant systems prior to adopting this methodology for those systems.

Finally, it should be noted that the New York Power Authority submitted a request for approval of a similar methodology change for postulating passive failures in the Indian Point 3 Nuclear Power Plant's SRW System (Reference a). Approval of their request is documented in Reference (b).

SCHEDULE

This methodology change is requested to be approved and issued by March 15, 1995. However, issuance of this change is not currently identified as having an impact on outage completion or continued plant operation.

Should you have any questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

RED/NH/dlm

Attachment: Evaluation of the Passive Failure Pipe Break Criteria and Analysis for the Calvert Cliffs Nuclear Power Plant Saltwater System

cc: D. A. Brune, Esquire
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ATTACHMENT (1)

EVALUATION OF THE PASSIVE FAILURE

PIPE BREAK CRITERIA AND ANALYSIS FOR THE

CALVERT CLIFFS NUCLEAR POWER PLANT

SALTWATER SYSTEM

Baltimore Gas and Electric Company

Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318

September 27, 1994

ATTACHMENT (1)

EVALUATION OF THE PASSIVE FAILURE PIPE BREAK CRITERIA AND ANALYSIS FOR CALVERT CLIFFS NUCLEAR POWER PLANT SALTWATER SYSTEM

I. SUMMARY

The objectives of this report are to review current regulatory and industry guidance for postulating passive failures in moderate energy lines in order to formulate a position on postulating passive failures for the cooling water systems and to evaluate the flooding effects which would result from a critical crack of a Saltwater (SW) System pipe in the Service Water (SRW) Pump Room.

This report produced the following major results:

A. The crack locations and sizes postulated under the guidance of SRP Sections 3.6.1 and 3.6.2 (Reference F) are believed to be bounding in terms of the consideration of passive failures as addressed in SECY-77-439 and ANSI/ANS 58.9-1981 (References G and I, respectively), and should be applicable to the Calvert Cliffs Nuclear Power Plant (CCNPP) SW System pipe failure analysis.

B. Passive failures of plant cooling water systems, and other moderate energy piping systems which typically operate during normal plant operations, are postulated to occur no less than 24 hours after the initiating event.

C. The components in the SRW Pump Room essential to the post-recirculation mitigation of a Loss of Coolant Accident (LOCA), namely the SRW pumps, are capable of performing their intended safety function following a flooding event resulting from a critical crack in any SW System pipe.

II. BACKGROUND

Calvert Cliffs Nuclear Power Plant's Updated Final Safety Analysis Report (UFSAR), Section 9.5.5, Table 9-17A, evaluates the response of the plant cooling water systems (Component Cooling Water, SRW and SW) to single failures (Reference A). The analysis considers either a single active failure at any time (concurrent with a Design Basis Event) or a single passive failure occurring after the onset of recirculation flow from the containment sump. The single failure analysis includes an evaluation of the consequences of flooding as a result of cooling water system passive failures. The conclusion of the analysis is that these single failures will not prevent the engineered safety features from fulfilling their design functions.

One of the scenarios evaluated in Table 9-17A is the ability of the SRW System to withstand flooding in the SRW Pump Room due to a break in the SW System piping. A question was raised internally regarding the basis for the statements made in the UFSAR discussion describing the plant's response to the above scenario. The single failure analysis describes the failure simply as a "rupture" in the affected system; however, the exact nature of the rupture, (i.e., size) cannot be determined from the information contained in the UFSAR. A review of licensing documentation specific to CCNPP did not reveal any information detailing assumptions on the magnitude or timing of the pipe break, as this criteria was still under consideration by the NRC at the time of design and licensing of CCNPP.

As discussed in subsequent sections of this report, current industry standards and regulatory guidance were reviewed to determine appropriate assumptions of fluid leakage resulting from a post-recirculation passive failure of cooling water piping, and a flooding analysis was performed based on this new assumption.

III. REVIEW OF REGULATORY GUIDANCE AND INDUSTRY STANDARDS

A. General Design Criteria

On July 10, 1967, the Atomic Energy Commission (AEC) published the proposed General Design Criteria (GDC) for Nuclear Power Plants (Reference B). These 70 criteria were issued for public comment. Baltimore Gas and Electric Company obtained a construction permit on July 7, 1969, to construct both Calvert Cliffs Units 1 and 2 (Reference C). The construction permit was obtained from the AEC after their review of our Preliminary Safety Analysis Report, which contained an assessment of our compliance with the draft GDC (Reference D).

Of these draft GDC, Criterion 41 applies to single failures of the SW System. This Criterion requires that:

> "Engineered Safety Features . . . shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component."

Criterion 41 (1967) did not require consideration of passive failures for engineered safety features.

In 1971, the final version of the GDC was issued as Appendix A to 10 CFR Part 50 (Reference E). The purpose of the final GDC is to " . . . establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission." The final GDC, which are to be used for applying for a construction permit, do not specifically define passive failures for the single failure criteria. However, by 1971, we already had our construction permit and, therefore, the final GDC were not used to establish the principal design criteria for Calvert Cliffs.

Notwithstanding the timing of the current GDC relative to the issuance of the construction permit of CCNPP, it appears that the application of passive failures was still under

consideration when the final GDC were issued. Criterion 44 is applicable to station cooling water systems. This Criterion states that:

> "A system to transfer heat from structures, systems and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems and components under normal operating and accident conditions."
>
> Suitable redundancy in components and features and suitable interconnections, leak detection and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

Within GDC 44, the single failure criterion is not specifically defined as active and/or passive failures. However, Footnote 2 in Appendix A, does state that: "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development." This appears to indicate an element of judgment on the part of the NRC when considering passive failure in fluid systems.

B. SECY-77-439

As further clarification for defining the types of passive failures to be considered for fluid systems in nuclear power plants, the NRC staff provided the following position in SECY-77-439 (Reference G).

On Page 4, the report asserts that:

> "In the study of passive failures, it is current practice to assume fluid leakage owing to gross failure of a pump or valve seal during the long term cooling mode following a LOCA (24 hours or greater after the event) but not pipe breaks. No other passive failures are required to be assumed . . . ."

On Page 9, the report states that:

> " . . . on the basis of the licensing review experience accumulated in the period since 1969, it has been judged in most instances that the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in application of the Single Failure Criterion to assure safety of a nuclear power plant."

On Page 9, the report continues:

> " . . . an example of the application of a passive failure requirement is the approach to long-term recovery subsequent to a loss-of-coolant accident. Applicants are required to consider degradation of a pump or valve seal and resulting leakages in addition to initiating failure (LOCA)."

C. NUREG-0800, Standard Review Plan (SRP)

A review of NRC regulations relative to passive failures indicates that whereas consideration of passive failures is required for high energy systems (SRP Section 6.3, Emergency Core Cooling System), the passive failure criteria is more relaxed for moderate energy lines (in particular for the SW System, refer to SRP Section 9.2.1). Furthermore, although limited size breaks, commonly referred to as "critical cracks", in moderate energy lines have been required, they have been taken as initiating events and not coincident with a Loss of Offsite Power or LOCA (SRP Sections 3.6.1, Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment and 3.6.2, Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping). The intent has been to eliminate or reduce the risk of affecting the operation of a system important to safety as a result of breaks in a nearby moderate energy system.

The NRC does not provide guidance in SRP Sections 3.6.1 and 3.6.2 for the evaluation of pipe breaks to support their review of a licensee's conformance with GDC 44. These sections address the review of postulated ruptures of piping systems and the evaluation of the impact of the dynamic effects associated with postulated rupture on structures, systems and components important to safety.

For moderate-energy fluid systems, SRP Sections 3.6.1 and 3.6.2 specify through-wall leakage cracks in piping and branch runs exceeding a nominal pipe size of one inch, where the crack opening is assumed as one-half the pipe diameter in length and one-half the pipe wall thickness in width. It should be re-emphasized that the review under SRP Sections 3.6.1 and 3.6.2 does not deal with individual system design requirements necessary to ensure that the system performs as intended, but rather considers the protection necessary to assure the operation of such systems in the event of nearby piping failures. In addition, the criteria for evaluating postulated breaks in piping considers breaks only as single initiating events occurring during normal plant conditions, and not as passive failures postulated during the recirculation phase of plant cooldown following a LOCA.

D. Industry Standards

Subsequent to the publication of the General Design Criteria, considerable confusion arose as to the definition of passive failures, and the acceptable means of evaluating these failures. To resolve the various conflicting interpretations of the intent of the GDC,

American National Standard (ANS 51.7), Single Failure Criteria for PWR Fluid Systems, was developed.

The Standard defines a passive failure as "a breach of a fluid pressure boundary or blockage of a process flow path." The following note was provided to clarify the leak rate assumed to flow through the pressure boundary breach:

> "As an example, if a system were located in protected areas, were subject to periodic pressure test and inspection, had seismic analysis where required and had been designed and maintained to the ASME Boiler and Pressure Vessel Code, Sections III, Rules for Construction of Nuclear Power Plant Components, and XI, Rules for Inservice Inspection of Nuclear Power Plant Components, a review of a system involving piping, heat exchangers, valves, flanged joints, and system interface barriers might result in definition of a design leak rate for passive failure evaluation based on maximum flow through a failed packing or mechanical seal rather than based on complete severance of the piping."

In accordance with the guidance provided in the Standard, the passive failure is not to be considered during the "short-term" period of operations. The short-term is defined as, "that period of operation up to 24 hours following an initiating event, except that the design of emergency core cooling and containment spray systems will consider the short-term as terminating after the system is transferred from the injection mode to the recirculating mode."

The criteria specified in ANSI N658-1976 (ANS 51.7) was later incorporated into ANSI/ANS 58.9-1981. ANS 58.9 provided the following clarification of the flow rate to be considered as a result of the passive failure:

> "The design flow for a passive failure shall be defined by analysis of realistic passive failure mechanisms in the system, considering conditions of operation and possible failure or leakage modes, as appropriate."

Again, failed valve packing or pump mechanical seals are provided as examples of "realistic" passive failures.

IV. FORMULATION OF PASSIVE FAILURE CRITERIA FOR CCNPP

A. Criteria Based on Regulatory Guidance and Industry Standards

As discussed previously, to evaluate piping failures in a moderate energy fluid system, such as CCNPP's SW System, consistent guidance regarding the location, size and timing of the postulated failure must be provided. Enveloping passive failures in fluid systems are those which result in the loss of structural integrity of the system; i.e., a pipe break of

undefined size. A review of industry standards for piping has shown that in determining the criteria for postulating passive failures in fluid systems, it is important to distinguish piping failures as initiating events from long-term passive failures subsequent to the initiating event. A crack in a moderate energy line which is evaluated according to criteria in SRP Sections 3.6.1 and 3.6.2 is considered an initiating event. To satisfy GDC 44, current industry standards ANS 51.7 (Reference H), and ANSI/ANS 58.9 (Reference I) require the consideration of a long-term passive failure during post-LOCA recirculation in addition to the initiating event (in this case a LOCA). However, when supported by an analysis, the long-term passive failure is limited to the "maximum flow through packing or mechanical seals rather than based on complete severance of the piping" (References G and H). Further, no passive failures need be postulated in the short-term.

It should be re-emphasized that the review under SRP Sections 3.6.1 and 3.6.2 does not deal with individual system design requirements necessary to ensure that the system performs as intended, but rather considers the protection necessary to assure the operation of such systems in the event of nearby piping failures, which occur during normal plant conditions. These conditions notwithstanding, the criteria which have been developed for determination of pipe rupture locations and sizes are based on governing conditions of stress and fatigue.

The point in a given piping system where a rupture would most likely occur would be associated with points of high relative stress and high relative fatigue. These points can be predicted for any piping system for various operating conditions and design loadings; therefore, the criteria for selecting break sizes and locations are intended to provide the maximum practical protection by postulating breaks at those locations with the greatest potential for failure under loading conditions associated with specific seismic events and plant operational conditions. These same criteria are thus assumed to be applicable for the consideration of passive failures in piping during the recirculation phase of plant cooldown following a LOCA.

Since the SRP Section 3.6.1 and 3.6.2 criteria primarily are concerned with the protection of essential plant features from the dynamic effects associated with postulated pipe ruptures, only those portions of the SRP criteria dealing with the size and location of postulated ruptures can be considered appropriate for use in this review of passive failures. To evaluate the CCNPP SW System piping, a through-wall leakage crack equal to one-half the pipe diameter in length and one-half the pipe wall thickness in width is assumed to occur in the largest SW System pipe in the SRW Pump Room.

As stated previously, our UFSAR single failure analysis postulates a passive failure occurring after the onset of recirculation flow from the containment sump. Current industry guidance indicates that the passive failure should be considered only during the long-term period of operation following the initiating event. However, the definition of "long-term" differs, depending upon if the purpose of the system experiencing the failure. For emergency core cooling and containment spray systems, the long-term is considered to commence when these systems are transferred to the recirculation mode, whereas for other

plant systems, the long-term begins 24 hours after the initiating event. The rationale for applying more limiting criteria in the design of emergency core cooling and containment spray systems is a recognition of the relatively extended periods of required operation of systems that are expected to be on a standby status throughout the plant life. The likelihood of accelerated wear of such components as pump and valve seals would be increased in these systems due to the adverse conditions following a LOCA. However, for systems which operate continuously during normal plant operation, such as the cooling water systems, it is reasonable to assume that they will continue to operate for at least the first 24 hours after the initiating event without experiencing a passive failure. The extremely low probability of a pipe failure occurring during the first 24 hours after a LOCA is demonstrated below for the SW pipe break scenario.

B. Probabilistic Risk Assessment (PRA)

To support the position that a pipe failure occurring during the first 24 hours after a LOCA is overly conservative, the likelihood of catastrophic pipe failures has been reviewed.

The postulated scenario of concern is a SW pipe break in the SRW Pump Room. The values used to quantify the risk were extracted from the Individual Plant Examination (IPE) Summary Report, which was submitted to the NRC in December 1993 (Reference J).

The likelihood of any size SW pipe rupture (excluding leaks) occurring in the SRW Pump Room during the 24 hours following a LOCA was calculated from the value for the initiating event S226AR in Table 3.3.8.3-B of the IPE Summary Report. By dividing the annual frequency of all SW System pipe breaks in this room by 365 (days/year), the frequency of a break occurring during any 24-hour period was determined to be 9.8×10^{-7} per year. Therefore, the probability of a SW System pipe rupture in the SRW Pump Room occurring during any 24-hour period is considered low.

C. Safety Evaluation

Based on the arguments presented in this report with regard to the use of moderate energy piping failure criteria as delineated in SRP Sections 3.6.1 and 3.6.2, Baltimore Gas and Electric Company (BGE) feels that such criteria is applicable and bounding in the evaluation of passive failures in the CCNPP SW System piping.

Baltimore Gas and Electric Company has concluded that the margins of safety have not been reduced. This conclusion is based on the review of current NRC and industry standards and the Calvert Cliffs Probabilistic Risk Assessment. Data extracted from the Calvert Cliffs IPE Summary Report underscores the fact that the probability of failure of the SW System piping during any 24-hour period is low.

EVALUATION OF THE PASSIVE FAILURE PIPE BREAK CRITERIA AND ANALYSIS FOR CALVERT CLIFFS NUCLEAR POWER PLANT SALTWATER SYSTEM

V. FLOODING ANALYSIS

The flooding analysis is contained in BGE Calculation M-90-169, Revision 0 (Reference K). The evaluation is summarized in DS-001, Flooding Design Guidelines Manual, Section 8.17 (Reference L). The following discussion provides the methodology and assumptions used in evaluating the effects of a SW System pipe rupture in the SRW Pump Rooms.

A. Critical Crack Size

The flooding analysis assumes a crack in 36" SW piping in the SRW Pump Room. The only 36" SW piping in the SRW Pump Rooms is located at the SRW heat exchanger inlet and outlet nozzles. The majority of the remaining SW piping in the room is 24". The analysis considers the largest size piping in the room and is not dependent on the location of the crack.

The critical crack is assumed to be a slot break that has an area of half the pipe inside diameter by half the pipe wall thickness. This assumption agrees with SRP 3.6.1, which defines the critical crack as being half the pipe diameter in length and half the wall thickness in width. (SRP 3.6.1 does not specify whether inner or outer pipe diameter is to be used. However, ANSI/ANS 58.2-1988 (Reference M) states that the critical crack is based on pipe inside diameter.)

The inner diameter of the 36" SW pipe assumed in the analysis is 35.25", and the wall thickness is 0.375". The critical crack cross-sectional area, A, is:

$$A = (35.25\text{ in.}/2)\,(0.375\text{ in.}/2) = 3.3047\text{ in}^2$$

The critical crack versus a guillotine break was assumed based on the SW System being classified as a moderate energy system using SRP 3.6.1 criteria (maximum operating temperature is 200°F or less, and maximum operating pressure is 275 psig or less). SW System design conditions are 95°F and 50 psig. Service conditions are 95°F and 35 psig.

B. Method of Analyzing Flooding Effects

The analysis computes a maximum hypothetical flood height in the SRW Pump Rooms based upon a leakage rate through the critical crack and operator response time to terminate the flooding event. The flooding is assumed to be secured 30 minutes following indication of the flood via a Control Room alarm, which annunciates at a flood height of 5.0" ± 0.5". The flood height is determined by dividing the total volume of water released during the flooding event by the room area available for flooding. The computed flood height can be used to evaluate the effects on equipment located in the SRW Pump Rooms.

The leakage rate through the critical crack is determined by calculating the break area and assuming it to be an equivalent orifice. The methodology in ANSI/ANS 56.11-1988 (Reference M), American National Standard Design Criteria for Protection Against the

Effects of Component Flooding in Light Water Reactor Plants, was used to calculate this leakage rate, Q.

$$Q = 3.117\,AC\,\sqrt{2gH}$$

where:

- Q = volumetric flow rate from the critical crack (gpm)
- 3.117 = constant which converts ft-in^2/sec to gpm
- A = equivalent orifice cross-sectional area (in^2)
- C = discharge coefficient of equivalent orifice of cross-sectional area, A
- g = 32.2 ft/sec^2
- H = pressure drop across the orifice (feet of fluid)

The discharge coefficient, C, was calculated based on standard relationships for square-edge orifices. The pressure drop, H, was calculated by converting the service pressure (35 psig for SW) to feet of water. The overall floor area of the SRW Pump Room, less the area lost to equipment, tanks, and piping, was used to calculate the flood height based on leakage rate and time to secure flooding.

The analysis is conservative in that no credit is taken for any outflow of water via floor drains.

C. **Results of Analysis**

Based on a critical crack in 36" SW piping, the leakage rate was calculated to be 447 gpm. The time to actuate the SRW Pump Room high level alarm at 5.5" is 13.44 minutes. Given an operator response time of 30 minutes, the total duration of the flooding event is:

$$13.44 \text{ min} + 30 \text{ min} = 43.44 \text{ min}$$

This time was rounded up to 44 minutes for calculating the total water volume released.

Based on a leakage rate of 447 gpm and a flooding duration of 44 minutes, 19,668 gallons of SW are released to the SRW Pump Room. The resulting flood height is 18".

The motor-driven AFW pump motor is below the 18" flood height (bottom of motor is at approximately 15") and is assumed to fail. However, flooding of the AFW pump motor will not result in loss of system functionality, because each unit has two independent, steam-driven AFW pumps located in a separate, water-tight compartment. The most limiting components (in terms of flooding height) in the SRW Pump Rooms required to mitigate the effects of a LOCA are the SRW pumps. The SRW pumps are considered essential to the mitigation of a Design Basis Accident. The bottom of the SRW pump motors is at a height of approximately 33". Therefore, this flooding event will not impact the operability of these pumps.

The analysis concludes that the postulated flooding scenario will be limited to the SRW Pump Rooms. There is one standard door in each SRW Pump Room that leads to the Radiation Exhaust Ventilation Equipment Room. This door is located 24" above the SRW Pump Room floor and is surrounded by a 6" curb in the Radiation Exhaust Ventilation Equipment Room. Therefore, flooding of the Radiation Exhaust Ventilation Equipment Rooms will not begin until the SRW Pump Room water level reaches 30". The analysis shows the flooding will be terminated before this level is reached. Personnel and equipment accesses to the 12' Turbine Building and the SRW Pump Room emergency escape hatch located nine feet above the SRW Pump Room floor and are all protected by watertight doors. Therefore, these areas are not considered to be flowpaths for flooding (in or out of the SRW Pump Rooms).

VI. CONCLUSIONS

A. The crack locations and sizes which were postulated under the guidance of SRP Sections 3.6.1 and 3.6.2 would be bounding in terms of the consideration of passive failures as addressed in SECY-77-439 and ANSI/ANS 58.9-1981, and are thus applicable to the CCNPP single failure analysis.

B. For plant cooling water systems, and other plant systems which typically operate during normal plant operations, the passive failure shall be postulated to occur no less than 24 hours after a LOCA.

C. The CCNPP SW System is capable of performing its safety function under active and passive failure conditions consistent with the design of the system.

D. A critical crack occurring in the long-term (i.e., 24 hours after a LOCA), at any location on the SW piping in the SRW Pump Room, will not affect the ability of any equipment essential to the mitigation of a LOCA.

E. The CCNPP UFSAR will be revised to reflect the new break criteria and analyses as discussed above.

VII. REFERENCES

A. Updated Final Safety Analysis Report for Calvert Cliffs Nuclear Power Plant, as revised

B. United States Atomic Energy Commission memorandum K-172, dated July 10, 1967, AEC Publishes General Design Criteria for Nuclear Power Plant Construction Permits

C. Notice from Mr. P. A. Morris (AEC), dated July 7, 1969, Notice of Issuance of Provisional Construction Permits

D. Preliminary Safety Analysis Report for Calvert Cliffs Nuclear Power Plant, Amendment No. 10, issued April 25, 1969

E. United States Atomic Energy Commission memorandum O-113, dated July 7, 1971, AEC Publishes Changes to General Design Criteria for Nuclear Power Plants

F. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, July 1981

G. SECY-77-439, Memorandum from Edson G. Case, Acting Director, Office of Nuclear Reactor Regulation to The Commissioners, dated August 17, 1977, Single Failure Criteria

H. ANS 51.7 (ANSI N658-1976), Single Failure Criteria for PWR Fluid Systems, dated June 21, 1976 (replaced by ANSI/ANS 58.9-1981)

I. ANSI/ANS 58.9-1981, Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems, dated February 1981

J. RAN 92-008, Revision 0, CCNPP Individual Plant Examination Report, dated December 1993

K. BGE Calculation M-90-169, Revision 0, issued April 28, 1991

L. Calvert Cliffs Nuclear Power Plant Design Standard DS-001, Revision 0, dated September 26, 1991, Flooding Design Guidelines Manual

M. ANSI/ANS 58.2-1988, issued October 6, 1988, Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture

N. ANSI/ANS 56.11-1988, issued November 4, 1988, American National Standard Design Criteria for Protection Against the Effects of Component Flooding in Light Water Reactor Plants