CP&L

Brunswick Nuclear Plant
July 26, 1994

94 JUL 29 AM: 01

Serial: BSEP 94-0285

Mr. Stewart D. Ebneter
Regional Administrator
United States Nuclear Regulatory Commission
101 Marietta Street, N. W., Suite 2900
Atlanta, GA 30323

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-324 / LICENSE NO. DPR-62
RESPONSE TO IE BULLETIN 80-13
INSPECTION RESULTS OF BRUNSWICK UNIT 2 CORE SPRAY SPARGERS

Dear Mr. Ebneter:

Pursuant to NRC IE Bulletin 80-13, Carolina Power & Light Company (CP&L) hereby submits the results of the inspections performed for Brunswick Unit 2 core spray spargers and associated piping. The reactor pressure vessel internal piping and spargers associated with the Core Spray (CS) System were visually examined with a remotely operated underwater camera during the B211R1 refueling outage which ended on June 26, 1994.

Enclosure 1, provides relevant portions of the Engineering Evaluation Report (EER), which documents the analysis of the examination.

The analysis concludes that the as found condition of the Brunswick Unit 2 core spray spargers and associated piping is acceptable and no postulated scenario will affect the safe operation of the plant, and design margins for the core spray system will be maintained during the cycle 11 operation. Therefore, the condition of the internal core spray spargers and associated piping does not impose any restrictions on plant operations for the next operating cycle.

Please refer any questions regarding this submittal to Mr. G. Honma at (910) 457-2741.

Yours very truly,

R. P. Lopriore
Manager
Regulatory Affairs Section

SHC/shc (corespar.u2)

Enclosure

cc: NRC Document Control Desk
Mr. P. D. Milano, NRC/NRR Senior Project Manager - Brunswick
Mr. R. L. Prevatte, NRC Senior Resident Inspector - Brunswick

ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-324
OPERATING LICENSE NO. DPR-62
RELEVANT PORTIONS OF ENGINEERING EVALUATION REPORT (EER) 94-0137
UNIT 2 CORE SPRAY INVESSEL PIPING EVALUATION FOLLOWING IVVI EXAMINATIONS

This EER documents the Unit 2 Core Spray Sparger In-Vessel Visual Inspections (IVVI) performed during Refuel Outage 10 (B211R1); bounds the inspection results by previous evaluations performed on Unit 1 & 2 Core Spray piping and spargers by General Electric (GE); and provides justification to use the spargers for another operating cycle in the as-found condition. This EER is identified as Q-list on Form 2. The internal core spray piping and spargers are designed to ANSI B31.1.0 - 1967 Power Piping Code and as a reactor vessel internal component, are in compliance with the applicable portions of ASME Section III, 1965 through Summer 1967 Addenda. They are not pressure boundary components. however, they are essential to the safety of the plant according to the definition of Criteria I of the NRC General Design Criteria, 10 CFR 50, Appendix A. This classification is based on the function of the core spray spargers to provide a flow path to direct water to the core region during a Loss of Coolant Accident (LOCA). 0PT 90.1 (Core Spray/Feedwater Visual Examination) is performed each refueling outage to satisfy the requirements of ASME Boiler and Pressure Vessel Code, Section XI, 1980 Edition through the Winter 1981 Addenda, Table IWB-2500-1, Category B-N-2, B13.21 (as applicable), Technical Specification 4.0.5 for Unit 1 or Unit 2 (as applicable), and Provisions of Inspection and Enforcement Bulletin 80-13.

1.0 HISTORY OF CORE SPRAY PIPING & SPARGER NON-DESTRUCTIVE EXAMINATIONS

1.1 Introduction

1.1.1 In accordance with IE Bulletin 80-13 (Ref. 1) the reactor pressure vessel internal piping and spargers associated with core spray (CS) system are visually examined with a remote operated underwater camera during each refueling outage as part of Periodic Test PT-90.1 (Ref. 2). The inspection is recorded on video tape for a documentation record.

1.1.2 Cracking in the in-vessel CS spargers and piping is an industry concern. The first instance of cracking in the CS spargers occurred between 1978 and 1980 at Oyster Creek and Pilgrim nuclear power stations, which eventually resulted in the issuance of IE Bulletin 80-13 in 1980.

1.1.3 In 1982 CP&L notified the NRC that during the IE Bulletin 80-13 visual inspection of Unit 2 a crack was located on the A-loop upper clockwise sparger arm near the weld to the T-box at 170° azimuth. The crack was approximately 20 mils wide and extended cricumferentially approximately 120° around the pipe. Although continued operation without corrective action was judged to be safe, a clamp was installed over the cracked area as a precaution. Installation of the clamp provided full structural reinforcement to the sparger equivalent to a welded joint.

Inspections continued on Unit 2, in accordance with IE Bulletin 80-13, without evidence of further core spray piping and sparger cracking until 1988. During the performance of a 1988 visual inspection in accordance with IE Bulletin 80-13 (PT 90.1), a crack indication was identified on the north core spray line outside the shroud on the piping adjacent to the junction box in the heat affected zone near the weld. Upon completion of the IVVI, the reactor pressure vessel was deflooded and supplemental liquid penetrant (LP) and ultrasonic tests (UT) were performed on the crack. These examinations revealed the crack was through wall and approximately 3.5 inches in length along the inside diameter and 1.75" along the outside diameter. An analysis concluded that the unit could be safely operated during the next fuel cycle with no operational changes or restrictions (Ref. 5). The unit operated for two refuel cycles before brackets were installed during the Refuel Outage No. 9 (B210R1). The brackets provide full structural reinforcement to the core spray piping.

The Unit 2 piping has previously been analyzed (Refs. 3 & 5) for both structural adequacy and the effect of potential leakage through the cracks on the ability of the CS system to deliver cooling water to the core. The conclusion was that the existing cracks with the addition of the clamp and brackets was acceptable for continued operation. The amount of leakage through the maximum predicted crack size was within the design margin of the core spray system. Core spray piping outside the shroud has been analyzed for postulated cracks in all four T-box welds and the predicted leakage is still within the design margin of the CS system.

1.1.4 The BNP Unit 1 CS piping had no reported indications until Refuel No. 8 (B109R1). During the performance of PT 90.1 two linear indications were found in the Unit 1 in-vessel core spray piping (B-loop), by visual examination using a remote operated underwater camera. One indication is in the heat affected zone of a circumferential weld which is located in the in-vessel piping between B-loop inlet nozzle and the sparger, approximately 18" downstream of B-loop T-box. This linear indication is approximately 4" long. The other indication is located on a tee-to-sparger arm circumferential weld on one of the lower B-loop spargers. This indication is approximately 3" long. Both indications were evaluated by GE (Ref. 4) and determined to be acceptable in the as-found condition. The Unit 1 core spray piping will be routinely examined during refuel Outage No. 9 (B110R1) as part of PT 90.1, and be re-evaluated and/or repaired based upon the results of the examination.

2.0 REFUEL No. 10 (B211R1) EXAMINATION RESULTS

2.1 Inspection Results

2.1.1 During the performance of PT 90.1 on the A-loop Core Spray spargers a crack was found at a seal weld to flow nozzle coupling, extending radially from the toe of the weld to approximately 1/2" into the sparger base material. The nozzle coupling is the third fitting counter-clockwise from the 240° sparger support bracket. See Figures 1 & 2.

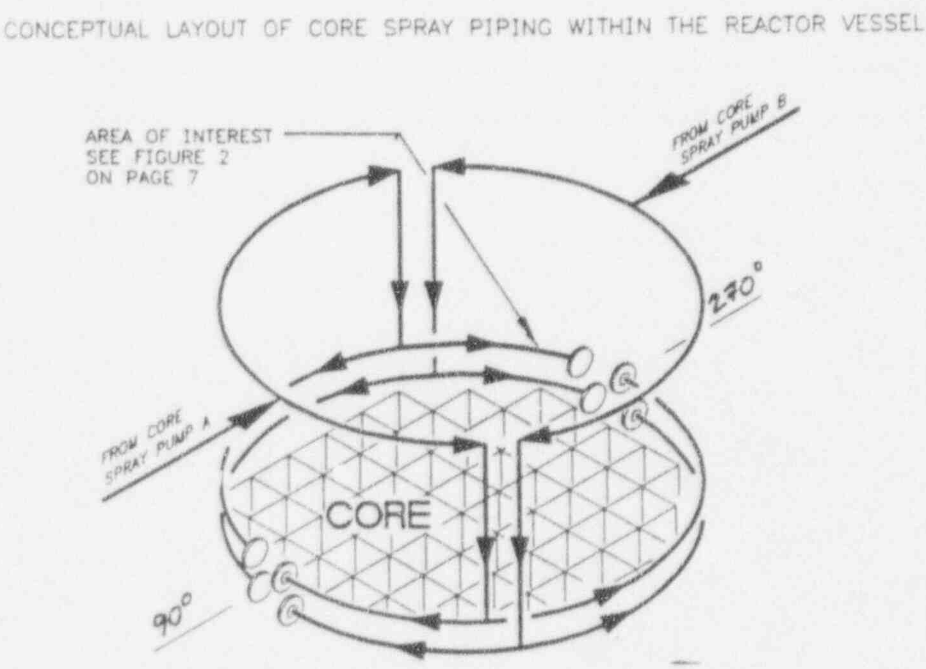
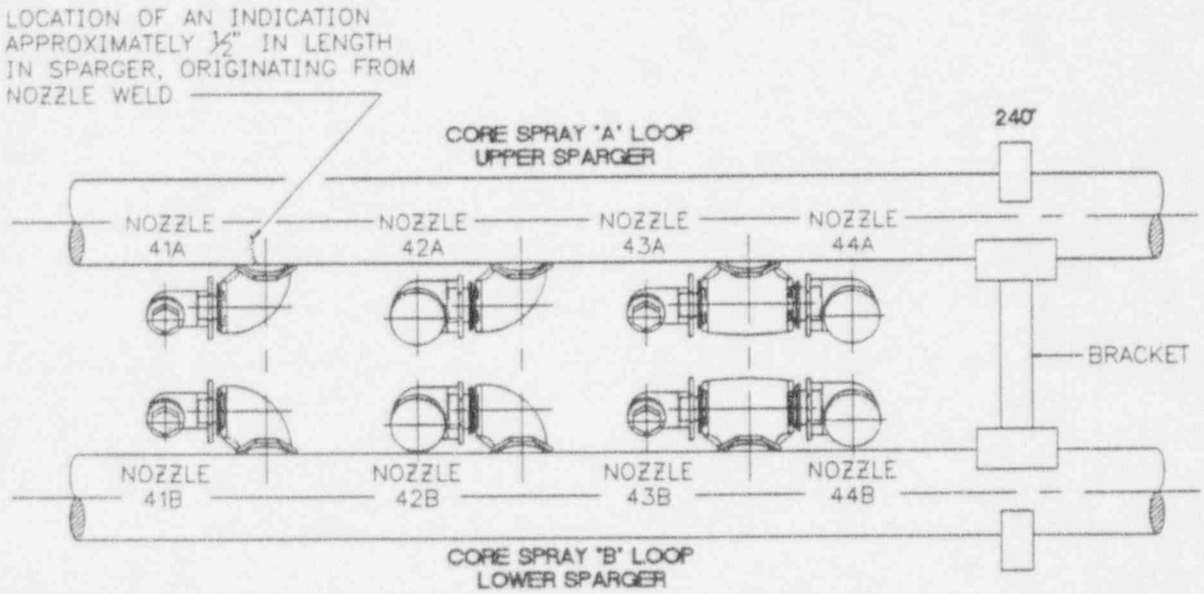


Figure 1 - Internal Core Spray Piping



VIEW FROM INSIDE THE REACTOR VESSEL LOOKING OUT

Figure 2 - Location of Linear Indication

3.0 EVALUATION FOR BOUNDING INDICATION BASED ON PREVIOUS ANALYSES

3.1 Similar indications in the Unit 1 & 2 Core Spray piping and spargers have been previously analyzed by General Electric Company (Refs. 3, 4 & 5) and determined to be acceptable in their as-found condition based on (1) the effect of the structural integrity of the in-vessel core spray piping; (2) the effect of leakage through assumed through-wall cracks and the impact to the ECCS analysis and (3) the effects of any postulated loose parts on the safety related equipment in the reactor pressure vessel or the effect on in-vessel components.

3.2 For conservatism the crack is assumed to be through wall. The crack does not follow typical IGSCC cracking, i.e., the crack does not follow a circumferential path around the nozzle in the area of the heat affected zone, rather extends radially outward into the base material. The non-typical crack configuration is most likely the result of cold working in the base material during manufacturing of the sparger and the result of IGSCC. Thermal stresses in this part of the core are very low and Irradiation Assisted Stress Corrosion Cracking (IASCC) is not expected to be a contributor in the core spray spargers. An estimate of flux at the I.D. of the shroud wall for different elevations was performed for Unit 1 (Ref. 8). The estimated flux at +353.88" above vessel zero is less than 1.5 E 19 n/cm^2. The industry accepted IASCC initiation threshold value is 5.0 E 20 n/cm^2 for low stress. Even though Unit 2 fluence is expected to be 25% higher (1.875 E 19 n/cm^2) than Unit 1, the Unit 2 value is well below the threshold value required for IASCC initiation.

3.3 The new crack is considered bounded by previous General Electric analyses as follows:

3.3.1 Structural Effects

Effects on structural integrity of the in-vessel core spray piping have been previously analyzed by GE for cracks in higher stressed areas of the core spray piping (Refs. 3, 4 & 5). These analyses showed that all identified stresses expected during normal reactor operation were small. Based upon the review of the stresses, it was concluded that the structural integrity of the piping and spargers with the existing cracks would be maintained during core spray injection. Stresses considered included those due to downcomer flow impingement loads, seismic loading, pressure, weight and thermally induced loads.

The analyses determined that normal operating loads by themselves do not result in stresses which are sufficient to cause IGSCC initiation, however, the addition of weld stresses coupled with the local cold work could support initiation. Once IGSCC has been initiated, the normal load stresses and the residual stresses could cause subsequent growth of the induced crack.

In order to determine the integrity of the core spray line and spargers with the previously existing cracks, crack arrest evaluations were performed. The stresses due to pipe restraint were included in the evaluations. Because the applied normal loading of the components are predominantly displacement controlled, the stresses were found to relax as the cracks grow and the compliance (or flexibility) of the pipe and sparger increased. The compliance was reduced sufficiently to relieve almost all of the displacement controlled stresses below the threshold to sustain IGSCC crack growth when the crack reached 180° of the circumference. Therefore, the crack growth is expected to be negligible or at virtual arrest prior to reaching 180°. The current extent of cracking in the A-loop core spray sparger is less than 15° of the sparger circumference.

It is expected that the new crack will arrest once it has reached the end of the cold working zone. Cold working of surface material by grinding or machining accomplishes a number of material changes, such as: tears, fissures and smears or laps that can enhance corrosion. Without the additional stresses introduced by welding or primary loads, the stress levels drop below the threshold required to sustain IGSCC crack growth.

Based on the above, the new crack is considered bounded by previous structural analyses performed by GE.

3.3.2 Effects of Leakage Through the Crack

The crack in A-loop core spray sparger is located inside the shroud, therefore, any leakage through the crack would be delivered directly to the core region. A crack of this size is not expected to significantly affect the spray distribution. According to an analysis performed by GE (Ref. 6), if a crack or multiple cracks (in a single sparger) were to grow to the extent that all flow passed through the crack(s) and none through the nozzles, the ECCS performance during a postulated LOCA is not expected to suffer degradation. This conclusion was based on large scale test which confirm that Counter Current Flow Limiting (CCFL) breaks down soon after spray initiation. This breakdown causes downflow of the water inventory from the upper plenum with subsequent rapid delivery and rapid reflooding of the core. Following this, a residual pool of water remains in the upper plenum ensuring uniform coolant delivery to the individual bundles. Therefore, adequate core cooling from the core spray system is expected to be maintained as long as the spray water is injected into the upper plenum, regardless of its distribution through the spray nozzles.

Based on the above, the new crack is considered bounded by previous analyses performed by GE on the effects of crack leakage.

3.3.3 Effects of Loose Parts in the Reactor Vessel

Based on previous structural analysis performed by GE (Ref. 3, 4 & 5), no breaks are expected in the core spray line or core spray sparger piping and consequently, no loose pieces in the reactor. However, analyses of the possible consequences of a potential loose piece was previously performed by GE (Ref. 3, 4 & 5). The analyses evaluated two different types of loose pieces postulated for the core spray line (a section of core spray pipe and a small piece of core spray pipe) and three different types for the core spray sparger (a section of sparger pipe, a small piece of sparger pipe and an outlet nozzle). The analyses concluded that the probability for unacceptable corrosion or other chemical reaction due to loose pieces is zero. The potential for unacceptable flow blockage or other damage to the fuel assemblies was negligible. The potential for unacceptable control rod interference was negligible. Therefore, the evaluations concluded that no safety concern was posed by postulated loose parts.

Based on the above and the crack geometry, the new crack is considered bounded by previous loose parts analyses performed by GE.

4.0 JUSTIFICATION FOR USING PIPING/SPARGERS AS-FOUND

4.1 It has been determined from previous General Electric Company analyses that the crack found on the Unit 2 A-loop Core Spray sparger piping during the performance of PT 90.1 is bounded by the analyses.

4.2 The crack is in a non-typical location (extending radially into the base material) and is believed to be a result of cold working in the area. The crack is not expected to grow beyond the cold worked area since stress levels drop below the threshold to sustain crack growth.

4.3 Previously performed analyses have concluded that cracks of a similar nature do not degrade the piping such that the structural integrity of the piping would be compromised during core spray injection.

4.4 The crack is located within the shroud area, therefore, any leakage from the crack will be directed to the core. Analyses have concluded that if all of the flow from core spray header is passed through the cracks and none through the nozzles, the ECCS performance during a postulated LOCA is not expected to suffer degradation.

4.5 Analyses have been performed to evaluate the effect of loose parts as a result of structural integrity degradation of the core spray piping and spargers. Results concluded that the probability for unacceptable corrosion or other chemical reactions due to loose pieces is zero. The potential for unacceptable flow blockage or other damage to fuel assemblies and unacceptable control rod interference is negligible.

5.0 DISPOSITION OF PIPING/SPARGERS AS-FOUND IN REFUEL OUTAGE NO. 10

5.1 Based on previous analyses performed by General Electric Company (GE), the Unit 2 internal core spray piping is acceptable in the as-found condition for the next operating cycle of Unit 2. There is no postulated scenario involving the internal core spray piping that will affect the safe operation of the plant and all design margins for the core spray system will be maintained during the operating cycle.

5.2 The condition of the internal core spray piping does not impose any restrictions to plant operation for the next operating cycle.

5.3 The core spray piping is routinely examined every refueling outage as part of PT 90.1, therefore the piping/spargers will be reevaluated and/or repaired in Refuel Outage No. 11 (B212R1) based upon the results of the next examination.

REFERENCES

1. USNRC IE-Bulletin 80-13, "Cracking in Core Spray Spargers", May 12, 1980.

2. OPT-90.1 for Unit 2, Refuel Outage No. 10 (Outage B211R1), "Core Spray/Feedwater Visual Examination".

3. General Electric Company Report No. EAS-03-0190 (Supplement 1 of EAS-14-0388), Core Spray Line Crack Growth Analysis Update for Brunswick Steam Electric Plant Unit 2, January 1990.

4. General Electric Company Report No. GE-NE-523-97-0793, "Core Spray Crack Analysis for Brunswick Steam Electric Plant, Unit 1, July 1993.

5. General Electric Company Report, "Core Spray Sparger Crack Analysis at BSEP - Unit 2, NEDO 22171, July 1982.

6. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K, NEDO-20566, Volume II, 1975.

7. Design Basis Document (DBD) - 18, Core Spray.

8. EER 93-0536 - Evaluation of Unit 1 Core Shroud Indications and Operability Assessment of Unit 1 and Unit 2.

SAFETY REVIEW COVER SHEET

DOCUMENT NO. EER 94-0137 REV. NO. 0
DESCRIPTION OR TITLE: Unit 2 CS Invessel Piping Evaluation following IVVI Exam.

1. Assigned Responsibilities:
   Safety Analysis Preparer: George L. Frick
   Lead 1st Safety Reviewer: George L. Frick
   2nd Safety Reviewer: Steve Bartz

2. Safety Analysis Preparer: Complete PART I, SAFETY ANALYSIS

Safety Analysis Preparer [signature] 5/4/94
SIGNATURE DATE

3. Lead 1st Safety Reviewer: Complete Part II, Item Classification.

4. Lead 1st Safety Reviewer: III may be completed. If either question 1 or 2 is "yes," then Part IV is not required.

5. Lead 1st Safety Reviewer: Determine which DISCIPLINES are required for review of this item (including own) and mark the appropriate block(s) below.

DISCIPLINES Required:	(Print Name)	Signature/Date (Step 7)
[ ] Nuclear Plant Operations		
[ ] Nuclear Engineering		
[✓] Mechanical	George L. Frick	[signature] 5-4-94
[ ] Electrical		
[ ] Instrumentation & Control		
[ ] Structural		
[✓] Metallurgy	MICHAEL W. GUTHRIE	Michael W. Guthrie 5-5-94
[ ] Chemistry/Radiochemistry		
[ ] Health Physics		
[ ] Administrative Controls		

6. A QUALIFIED SAFETY REVIEWER will be assigned for each DISCIPLINE marked in step 5 and his/her name printed in the space provided. Each person shall perform a SAFETY REVIEW and provide input into the Safety Review Package.

7. The Lead 1st Safety Reviewer will assure that a Part III or Part IV is completed (see step 4 above) and a Part VI if required (see 9.d of Part II). Each person listed in step 5 shall sign and date next to his/her name in step 5, indicating completion of a SAFETY REVIEW.

8. 2nd Safety Reviewer: Perform a SAFETY REVIEW in accordance with Section 8.0.

2nd Safety Reviewer [signature] Date 5/6/94

DISCIPLINE: Mechanical

9. PNSC review required? If "yes" attach Part V and mark reason below: Yes [ ] No [✓]
   [ ] Potential UNREVIEWED SAFETY QUESTION
   [ ] Question 9 of Part IV answered "Yes"
   [ ] Other (specify):

PART I: SAFETY ANALYSIS
(See instructions in Section 8.4.1)
(Attach additional sheets as necessary)

DOCUMENT NO. EER 94-0137 REV. NO. 0

DESCRIPTION OF CHANGE: The invessel core spray piping and spargers were visually examined per PT 90.1 during the current refueling outage No. 10 (B211R1). EER 94-0137 was written to document the examination results and bound the as-found condition using previous analyses to allow the plant to operate another refueling cycle.

ANALYSIS: The invessel core spray spargers are designed to ANSI B31.1.0 1967 Power Piping Code and as a reactor vessel internal component meet the applicable portions of ASME Section III, 1965 through Summer 1967 Addenda. They are classified as Safety Related because the core spray system is part of ECCS. The spargers are inspected in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI, 1980 Edition through the Winter 1981 Addenda, Table IWB-2500-1, Category B-N-2, B13.21 (as applicable), Technical Specification 4.0.5 for Unit 1 or Unit 2 (as applicable) and Provisions of Inspection and Enforcement Bulletin 80-13.

The crack has been evaluated by comparison to previous analyses performed by GE for similar indications for, (1) the effects of structural integrity of invessel piping (2) the effect of leakage through the indication and the ability of the core spray spargers to deliver cooling water to the core during core spray initiation and (3) the effect of any postulated loose parts on safe operation of the Unit. For conservatism the crack is considered to be through wall.

Comparisons show that A-loop sparger is bounded by the previous analyses and based on those analyses, the crack does not degrade the spargers such that the integrity of the spargers would be compromised during a core injection. All structural design margins will still be met at the end of another eighteen month operating cycle.

PART I: SAFETY ANALYSIS

(See instructions in Section 8.4.1)

(Attach additional sheets as necessary)

DOCUMENT NO. EER 94-0137 REV. NO. 0

ANALYSIS: (Cont.) The location of the crack is within the shroud region, therefore any leakage through the crack would be to the core region. The crack will not significantly affect the spray distribution. Analyses have concluded that if all the flow from the core spray header is passed through cracks and none through the nozzles, the ECCS performance during a postulated LOCA is not expected to suffer degradation.

The effect of postulated loose parts from the invessel core spray spargers and piping has been previously analyzed by GE and there was no safety concerns identified. The effect of any loose parts on the safety related reactor vessel control rod drive components, fuel assemblies, or any other reactor vessel internal components was considered negligible.

The conclusion reached by review of previous analyses is that the Unit 2 Core Spray Spargers, in their as-found condition, will be acceptable for another 18 month operating cycle.

REFERENCES:

UFSAR Sect. 3.6, 3.7, 3.9, 3.11, 5.2, 5.3, 5.3A, 6.1, 6.2, 6.3, 7.0 and 15.0

Tech. Specs. 3/4.3.3, 3/4.4.3, 3/4.4.8, 3/4.5.3 and associated Bases.

PART II: ITEM CLASSIFICATION

DOCUMENT NO. EER 94-0137 REV. NO. 0

	Yes	No
1. Does this item represent:		
a. A change to the facility as described in the SAFETY ANALYSIS REPORT?	[ ]	[✓]
b. A change to the procedures as described in the SAFETY ANALYSIS REPORT?	[ ]	[✓]
c. A test or experiment not described in the SAFETY ANALYSIS REPORT?	[ ]	[✓]
2. Does this item involve a change to the individual plant Operating License or to its Technical Specifications?	[ ]	[✓]
3. Does this item require a revision to the FSAR?	[ ]	[✓]
4. Does this item involve a change to the Offsite Dose Calculation Manual?	[ ]	[✓]
5. Does this item constitute a change to the Process Control Program?	[ ]	[✓]
6. Does this item involve a major change to a Radwaste Treatment System?	[ ]	[✓]
7. Does this item involve a change to the Technical Specification Equipment List (BSEP and SHNPP only)?	[ ]	[✓]
8. Does this item impact the NPDES Permit (all 3 sites) or constitute an "unreviewed environmental question" (SHNPP Environmental Plan Section 3.1) or a "significant environmental impact" (BSEP)?	[ ]	[✓]
9. Does this item involve a change to a previously accepted:		
a. Quality Assurance Program	[ ]	[✓]
b. Security Plan (including Training, Qualification, and Contingency Plans)?	[ ]	[✓]
c. Emergency Plan?	[ ]	[✓]
d. Independent Spent Fuel Storage Installation license? (If yes, refer to Section 8.4.2, "Question 9," for special considerations. Complete Part VI in accordance with Section 8.4.6)	[ ]	[✓]

SEE SECTION 8.4.2 FOR INSTRUCTIONS FOR EACH "YES" ANSWER.

REFERENCES. List FSAR and Technical Specification references used to answer questions 1-9 above. Identify specific reference sections used for any "Yes" answer.

UFSAR Sect. 3.6, 3.7, 3.9, 3.11, 5.2, 5.3, 5.3A, 6.1, 6.2, 6.3, 7.0 and 15.0
Tech. Specs. 3/4.3.3, 3/4.4.3, 3/4.4.8, 3/4.5.3 and associated Bases.

PART III: UNREVIEWED SAFETY QUESTION DETERMINATION SCREEN

DOCUMENT NO. EER 94-0137 REV. NO. 0

	YES	NO
1. Is this change fully addressed by another completed UNREVIEWED SAFETY QUESTION determination? (See Section 7.2.1, 7.2.2.5, and 7.9.1.1)	[ ]	[✓]

REFERENCE DOCUMENT: ______ REV. NO. ______

	YES	NO
2. For procedures, is the change a non-intent change which only (check all that apply): (See Section 7.2.2.3)	[ ]	[✓]

[ ] Corrects typographical errors which do not alter the meaning or intent of the procedure; or,

[ ] Adds or revises steps for clarification (provided provided they are consistent with the original purpose or applicability of the procedure); or,

[ ] Changes the title of an organizational position; or,

[ ] Changes names, addresses, or telephone numbers of persons; or,

[ ] Changes the designation of an item of equipment where the equipment is the same as the original equipment or is an authorized replacement; or,

[ ] Changes a specified tool or instrument to an equivalent substitute; or,

[ ] Changes the format of a procedure without altering the meaning, intent, or content; or

[ ] Deletes a part or all of a procedure, the deleted portions of which are wholly covered by approved plant procedures?

If the answer to either Question 1 or Question 2 in PART III is "Yes," then PART IV need not be completed.

PART IV: UNREVIEWED SAFETY QUESTION DETERMINATION

DOCUMENT NO. EER 94-0137 REV. NO. 0

Using the SAFETY ANALYSIS developed for the change, test or experiment, as well as other required references (LICENSING BASIS DOCUMENTATION, Design Drawings, Design Basis Documents, codes, etc.), the preparer of the Unreviewed Safety Question Determination must directly answer each of the following seven questions and make a determination of whether an UNREVIEWED SAFETY QUESTION exists.

A WRITTEN BASIS IS REQUIRED FOR EACH ANSWER

	Yes	No
1. May the proposed activity increase the probability of occurrence of an accident evaluated previously in the SAFETY ANALYSIS REPORT?	[ ]	[✓]

Use of the core spray spargers in their present condition does not increase the probability of occurrence of any previously evaluated accident. For the next operating cycle the ability of the spargers to function as originally designed is unaffected by the indication and the structural integrity of the spargers is not jeopardized.

	Yes	No
2. May the proposed activity increase the consequences of an accident evaluated previously in the SAFETY ANALYSIS REPORT?	[ ]	[✓]

Based on a review of previous analyses performed by GE, the core spray spargers will function as originally designed during the next operating cycle and there is no postulated impact on any safety related reactor vessel components, therefore, using the spargers in their present condition will not increase the consequences of an accident previously evaluated in the UFSAR.

	Yes	No
3. May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAFETY ANALYSIS REPORT?	[ ]	[✓]

Using the core spray spargers in their present condition does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the UFSAR. The postulated effects of loose parts was considered and the consequences on safety related equipment in the reactor vessel is negligible. The ability of the core spray spargers to perform within their design margin of safety is unaffected.

	Yes	No
4. May the proposed activity increase the consequence of a malfunction of equipment important to safety evaluated previously in the SAFETY ANALYSIS REPORT?	[ ]	[✓]

The consequences of a malfunction of safety related equipment is not increased since there is no impact to any safety related system or component in the plant as a result of using the core spray spargers in their existing condition for the next eighteen month operating cycle.

PART IV: UNREVIEWED SAFETY QUESTION DETERMINATION

DOCUMENT NO. EER 94-0137 REV. NO. 0

5. May the proposed activity create the possibility of an accident of a different type than any evaluated previously in the SAFETY ANALYSIS REPORT? [ ] [✓]

The potential effect of any loose parts on any other safety related plant equipment is negligible. Since there is no predicted adverse effect on any safety related components, there is no possibility to create a new type of accident. Therefore, no possibility of an accident of a different type than any previously evaluated in the UFSAR exists.

6. May the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAFETY ANALYSIS REPORT? [ ] [✓]

The core spray spargers in their present condition can be used for another operating cycle without causing the malfunction of any invessel components. The spargers are within their structural and hydraulic design margins and will not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated.

7. Does the proposed activity reduce the margin of safety as defined in the basis of any Technical Specification? [ ] [✓]

No other system or equipment will be affected by the core spray spargers and all margins of safety for the system will be maintained. In the event of a core spray injection during the next operating cycle, no water will be lost from the shroud area. The structural integrity is not reduced below the design margin and crack growth is not expected to be any greater than growth experienced during the past. Therefore, no reduction of the margin of safety as defined in the bases of any Tech. Spec. will occur.

8. Based on the answers to questions 1 - 7, does this item result in an UNREVIEWED SAFETY QUESTION? If the answer to any of the questions 1-7 is "Yes", then the item is considered to constitute an UNREVIEWED SAFETY QUESTION. [ ] [✓]

PART IV: UNREVIEWED SAFETY QUESTION DETERMINATION

DOCUMENT NO. EER 94-0137 REV. NO. 0

9. Is PNSC review required for any of the following reasons? [ ] [✓]

If, in answering questions 1 or 3 "No", it was determined that the probability increase was small relative to the uncertainties; or, in answering question 2 or 4 "No", it was determined that the doses increased, but the dose was still less than the NRC ACCEPTANCE LIMIT; or in answering question 7 "No", a parameter would be closer to the NRC ACCEPTANCE LIMIT, but the end result was still within the NRC ACCEPTANCE LIMIT; then PNSC review is required.

REFERENCES:

UFSAR Sects. 3.0, 5.0, 6.0, 7.0, and 15.0
Tech. Specs. 3/4.3, 3/4.4, 3/4.5 and associated Bases.

This Unreviewed Safety Question Determination is for the following DISCIPLINE(s):
(Additional Part IV forms may be included as appropriate.)

[ ] Nuclear Plant Operations
[ ] Nuclear Engineering
[✓] Mechanical
[ ] Electrical
[ ] Instrumentation & Control
[ ] Structural
[✓] Metallurgy
[ ] Chemistry/Radiochemistry
[ ] Health Physics
[ ] Administrative Controls