July 2, 1986

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION

In the Matter of	)	
Carolina Power & Light Co.	)	
and North Carolina Eastern	)	Docket No. 50-400
Municipal Power Agency	)	(10 C.F.R. & 2.206)
	)	
(Shearon Harris Nuclear Power	)	
Plant)	)	

REQUEST FOR INSTITUTION OF PROCEEDINGS PURSUANT

TO 10 CFR 2.206

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OUTLINE CONTENTS

I. INTRODUCTION

The petitioners request that Mr. Harold R. Denton, Director of Nuclear Reactor Regulations require CP&L to respond to a show cause order pursuant to 10 CFR 2.202. In conforming with the requirements of 10 CFR 2.206, the petitioners will demonstrate that CP&L, by acts or ommission, has failed to meet the applicable standards required by 10 CFR et. al.. Petitioners will address the following issues: Emergency Planning, Plant Safety, Security, and Psychological Stress.

II.

Joseph Hughes and Steven Katz, are authorized by the Coalition for Alternatives to Shearon Harris, Calvin Regan, et. al., and Patricia Miriello, to assert the interest of the organizations' membership (1), (which includes CASH members residing in Chatham, Wake, Harnett, Lee, Durham, and Orange counties the principal population concentration of the organization lies within a 15 mile radius of Shearon Harris Nuclear Power Plant. See: Appendix A for organizational material.) Calvin Regan, et. al., (2) (see petition for CASH's representation of residence of persons living within the five mile zone at Appendix B), Patricia Miriello (3), (see documentation of Ms. Miriello's request for CASH's representation in these proceedings), and the interests of Joseph Hughes and Steven P. Katz (4). (Joseph Hughes and Steven Katz are CASH members and are responsible for developing legal strategy, and reside in Durham and Orange Counties respectively.)

On June 9, 1986, CASH filed documents with the NRC: first, a petition for leave to intervene pursuant to 10 CFR 2.714 (a) and 2.715 (a). A document in the form of a motion to State the Immediate Effectiveness of the Final Licensing Board Decision was filed on June 9, 1986 and this motion was joined and signed by Wells Eddelman, pro se. The motion complied with the procedural requirements of 10 CFR 2.788 and 10 CFR 2.764. In light of these filings, CASH's viability as a multicounty organization, CASH's representation of its membership, Mr Regan et. al., and Ms. Miriello, the petitioner clearly has the requisite interest to assert the following arguments.

III.
Standard Under 10 CFR 2.206 to Initiate a Proceeding

Section 2.206 provides a mechanism whereby members of the public may: 1. Request initiation of an enforcement action to modify, suspend or revoke a construction or operation licenses held by a utility; or; 2. for other such action as may be proper. The Director of the appropriate NRC office is vested with the authority to institute action pursuant to 10 CFR 2.202 Show cause order.

A show cause order, 10 CFR 2.202, should be issued by the Director where substantial health or safety issues have been raised. Consolidated Edison CL1758, 2NRC 173, 175 (1985). Additional health and safety requirements are set out in 10 CFR, and are relevent in determining whether adequate measures have been taken by the utility to protect public health and safety.

IV. Emergency Preparedness and Planning

A. Factual Background

On May 27, 1986, the Chatham County Commissioners passed a resolution rescinding prior approval of the Emergency Management Plan. (See: Appendix). The operative language is as follows:

> Now, therefore, be it resolved that the Chatham County Commissioners hereby rescind all prior approvals of the Shearon Harris Emergency Response Plan pending further critical study of the unresolved issues.

As a general proposition, local governmental entities are an integral part of emergency planning. See: 10 CFR 50.47.(b)(1); (primary responsibility for emergency response. . .by state and local organizations within the emergency planning zone (are) assigned, and specifically established and each organization has staff to respond and augment its initial response on a continuing basis). It is clear that Chatham County's emergency preparedness, as of this date, is fatally deficient. The Commissioners have rescinded their agreement to participate in the plan. Supporting organizations will not be staffed. Without staff mere notice of a radiological emergency occurrence would result in chaos. In short, there is no means of assuring that the population of Chatham County would be protected by any organization in the event of a radiological emergency.

B. Adequacy of the EMP

There can be no question that emergency preparedness, particularly in Chatham County is inadequate and fails to assure that any plan could be implemented. 10 CFR 50.47 (a)(2). Petitioner notes that FEMA found the E.M.P. adequate, as of May 1985. However, the FEMA finding has been mooted, by the Chatham County's rescision of May 27, 1986.

C. Requirement of Reasonable Assurance

It is clear that 10 CFR 50.47 (a) requires a finding made by the NRC that there be reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. FEMA did find that emergency planning was adequate in may 1985. Then presumption of adequacy and implementation is rebutted, due to the effect of the Chatham pullout. The EMP without Chatham County's participation cannot satisfy the requirements of 10 CFR 50.47 (b)(116) and 10 CFR part 50. (Supplemental documents will be forwarded to the Director analyzing the sixteen requirements for an EMP.)

D. One Year Test Standard Emergency Preparedness

New plants are required, to conduct a full scale exercise which tests the emergency plan. That plan is to be conducted within one year before issuance of the first full power operating license (10 CFR Part 50, Appendix E Section F1). It is understood that the emergency preparedness exercises are part of the operational inspection process and are not required for any initial licensing decision, 10 CFR 50.47(a)(2) however the language requiring a full scale exercise to be held within one year before full power operation. Union of Concerned Scientists vs. NRC, 735 Fzd 1437 (D.C. Cir. 1984), citing 10 CFR 50.47(a)(2). Here, the FEMA approval of the EMP was made in May of 1985. Chatham County participated in that test. Any plan which does not include Chatham County is clearly not the plan which was tested in May 1985. The plain meaning of, 10 CFR Part 50, App. E. (F) (1), requires a test of the plan one year prior to granting of an operation license. The Director should withhold granting of any operation license until such matter is resolved.

E.

The petitioners' final argument runs to the implementation phase of the EMP. 10 CFR Part 50 Appendix E. The applicable requirements of Plant Staffing assignments have not been clearly communicated to the operations staff. The requirements for "Activation of Emergency Organization" were tested during a given event occurring 28 June 1986. A preliminary analysis of the manner in which information is disseminated from the plant in the event of a siren system transmission, clearly there is a lack of preparedness with respect to activation of the notification systemboth onsight and within the affected communities. The events of 28 June 1986 are summarized as follows. See: affadavits at Appendix D.

An alarm siren was activated on June 1986 at 1:55 a.m. Numerous persons were awakened during the siren transmission. Persons living 2 miles north of the siren awoke and attempted to call various state and local authorities, and also called Shearon Harris Nuclear Power Plant. (See: affadavit of Barbara Keyworth and David Richardson). The Chatham County Sheriff Department dispatcher had not been informed by CP&L of the siren's purpose. The dispatcher stated that she had received other pohone calls from concerned residents of Chatham County. Calls were made to Shearon Harris Nuclear Power Plant. The proffered explanation was, that a shift whistle sounding at 2:00 a.m. had roused persons eleven miles from the plant. (See affadavit of Keyworth and Richardson). Confusion continued as calls were made to the N.C. Highway Patrol which resulted in a particularly uninformed and condescinding response.

Mr. Mac Harris, media manager CP&L, released a media piece which stated that vandals had tampered with the siren box setting the device off. This media release is contradicted by petitioners' affidavits which tend to prove that the siren which was allegedly tampered with had no visible signs of forceful tampering with either the security locks or the siren itself. (See affidavits of Frazier, Keyworth, Richardson and Thomas).

A continuing investigation of this matter continues. However, a number of inferences are readily apparent. First, security, if one chooses to believe CP&L's version of the incident, at the siren locations is not adequately provided. If vandals were able to set sirens off at will, the underlying reliability and value of the emergency warning system would be rendered useless. Second, there is apparently no method to secure information upon the activation of an emergency siren.

Clearly 10 CFR Part 50 App. E (c) requires the existence of message authentication scheme which includes notification of local emergency officials about unusual events, alerts, site area emergencies, and general emergencies. Note that 10 CFR Part 50 App. E. (D) (3), states that ". . . where there is substantial time available for state and local officials to make a judgement whether or not to activate the public notification system." Where there is a decision to activate. . . the state and local officials "will make the determination."

This incident implicates the unrefined information gathering and dissemination process which is the central thrust of any emergency notification scheme.

F. Conclusion and Requested Action:

The EMP approved by FEMA in May of 1985 is no longer viable. See Appendix E. It no longer provides for participation by Chatham County. The EMP has been flawed by an incident involving an emergency siren which sounded and residents of the EPZ were unable to ascertain definitive information with respect to the nature of the alarm or what action should be taken (evacuation, etc.; incidentally--no person from whom affidavits were taken turned to the emergency broadcast channel--petitioner will supplement this document as information becomes available).

Finally, 10 CFR 50.47(d) provides that a license authorizing fuel loading and/or low power operation may be issued after a finding that the state of emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. This standard has not been met. Therefore, petitioner moves that:

1. The Director should issue a 10 CFR 2.202 show cause order upon CP&L to demonstrate why CP&L should not be required to proceed with a complete Preliminary Safety Analysis pursuant to 10 CFR Part 50 App. E. II (in light of the Chatham County pull-out).
2. The Director should issue a 10 CFR 2.202 upon CP&L to demonstrate why CP&L should not be required to comply with the requirements of 10 CFR Part 50 App. E. III (in light of Activation of Emergency Notification System).
3. The Director should immediately revoke present or prospective authorization, for fuel loading and five percent testing of the Shearon Harris Nuclear Power Plant (lack of reasonable assurance that adequate measures can and will be taken in the event of a

radiological emergency due to Chatham County's pull-out).
4. That the Director proceed in a hearing upon the substantive issues raised by the petitioner in this and various pleadings filed with the NRC (pursuant to section 189 of the Atomic Energy Act).

V. Former CP&L Employee Investigation/Document Falsification/CP&L Quality Assurance

On January 1, 1986, Ms. Patty Miriello, a former CP&L worker at Shearon Harris and Brunswick nuclear reactors wrote to the presiding judge, James Kelley, Chairman of the ASLB panel in Docket 50400 OL alleging falsification of radiation exposure records and questionable practices relating to health physics and requested that her identity remain confidential. See Appendix F. The Chairman, however, ruled, pursuant to 10 CFR 2.780(b) that the allegations were to be treated as ex parte communications and disclosed the information to all parties in the case, including the Applicants. Although the NRC Office of Investigations (OI) has had documented evidence of Ms. Miriello's contentions since September 1985, the OI has yet to do a personal interview the the alleger. Moreover, the NRC OI has yet to issue a report of its investigation, which goes to the heart of the question of the Applicant's competence and integrity in operating the proposed Shearon Harris Plant.

1. As a worker exposed to radiation of the Applicant's nuclear reactors, the facts which have been brought forward by Ms. Miriello create serious close questions which would implicate the effectiveness of the Applicant's proposed radiation protection program for its employees. Moreover, the assertions which Ms. Miriello make, if substantiated by the Office of Investigation report which has yet to be completed, would result in a finding by the Commission that the Applicant's request for an operating license "may be revoked suspended or modified, in whole or part, for any material false statement of fact requireed of the Applicant." (10 CFR 50.100)

Miriello, a former employee of CP&L, alledged in September of 1985 that documents were falsified by the applicant. The OI has yet to complete this investigation. Among other allegations which have not been resolved, Miriello has been unable to obtain her complete record from the applicant and thus has been precluded from seeking positions within that field. Aside from the interest in freedom to pursue gainful employment, the applicant may be in violation of 10 CFR

50.100 (material false statements of fact), 10 CFR 0.735039(c) (disclosure of confidential information by the applicant), and a substantial possibility that the applicant may not have an adequate radiation protection program. All these issues may in combination or in part, amount to a substantial fundamental flaw in the final decision of the Licensing Board`s decision.

Moreover, petitioners allege that the Applicant may have a defective radiation protection program regarding the requirement for maintaining records of employee radiation exposure under 10 CFR 20.401. With regard to its former employee, Ms. Miriello, the Applicant may have violated 10 CFR 20.601 concerning falsification of employee monitoring records according to the attached affadavit. Moreover, when Ms. Miriello left employment with the Applicant, she was not provided with accurate exposure data as required under 10 CFR 20.408. In each of these instances, improper recordkeeping in the Applicant's radiation protection program could constitute adequate grounds for withholding or revoking the Applicant's proposed operating license.

Beyond concerns about the Applicant's radiation protection program, Ms. Miriello also has provided the NRC with documentary evidence of improper inservice ultrasonic inspectiions of the large reactor coolant line welds as part of the Applicant's quality assurance program. Attachments #1 and #2, which are both five pages long, show a discrepancy on the fourth page for the coolant line welds for numbers #9 and #12.

As Ms. Miriello notes in her letter to Judge Kelley of January 1, 1986, "two level III Nuclear Energy Services NDE inspectors argued over these ultrasonic results. They had conflicting opinions." When the new page four was revised as shown in Attachment #2, "note that the mention of a weld or repair weld was eliminated from page 4 of the original Mel Perry (NES corporate inspector) turned in." "Also removed was the listings of indications in this weld, referring specifically to indications #9 and #12."

According to Ms. Miriello's investigation, these design flaws in the Shearon Harris core coolant line are violations of the ASME Boiler and Pressure Vessel Code, Section XI, Articvle IWB3000, "Acceptance Standards for Flaw Indications", as quoted from the 1980 edition of the Shearon Harris PreService Inspection Manual.

According to the information which Ms. Miriello has observed and obtained, approximately 10% of the welds in the inservice inspection program at the Shearon Harris plant are defective and improperly documented. These inservice inspection records were altered and changed

without following the proper NRC procedure for record revisions on pipe welds.

We feel that these violations of NRC regulations in the Carolina Power and Light Company's Quality Assurance Program are sufficient grounds for withholding an operating license until these critical plant safety violations are investigated. Petitioner moves the Director to proceed in a 10 CFR 2.202 sho cause proceeding, and 189 hearing, to consider questions of material fact raised by this argument.

VI. Psychological Stress Argument

A. It is national policy that each federal agency shall utilize a "systematic, interdisciplinary approach which will insure the integrated use of natural and social sciences and environmental design arts", in order to assure that governmental action which affects the health and safety of the persons within a particular zone will be adequately protected. See: 42 U.S.C. 4331(2)(a). In order to implement this policy which the proposed action affects public health and safety, as a factor in determining whether the federal action significantly affects the human environment.

In People Against Nuclear Energy vs. U.S.N.R.C., 678 F2d 222 (D.C. Cir 1982), the Circuit Court was called to consider a novel health and safety issue, in light of the National Environmental Policy Act, 42 USC S4321, et. seq., and the Atomic Energy Act, 42 USC s 2133. The issue ran to the possibility that renewed operation of the plant at TMI would cause severe psychological distress to persons living within the vicinity of the reactor. The operation of the reactor would harm the stability, cohesiveness and well-being of the communities within the vicinity of the reactor. 678 F.2d at 226-227. The petitioners in PANE claimed that citizens had lost confidence that responsible institutions could function effectively during a crisis. That the area was becoming an undesirable location for residents and businesses; and, that the operation of the reactor was causing permanent damage to the economic and social health of the community were also alleged. Id. The Court in PANE held that the petitioners had alleged claimed within the meaning of the NEPA, and were allegations which rise to the level of environmental effects.

B. The central question in evaluating issues of psychological stress are the potential that particular governmental action may effect health. Language in the case supports the notion that there are occasions for considering when psychological stress is to be considered as a factor in evaluating the propriety of governmental action by a government agency. First, it is clear that

congress intended to include psychological stress as an element of the calclus for determining what effect a governmental action has on 'health' 678 F2d at 230. It is equally clear that the severity of psychological harm, and the cognizability of that harm under the NEPA will not be satisfied by "mere dissatisfaction arising from social opinions, economic concerns, or political disagreements with agency policies". Id. What does not seem clear is the extent to which psychological stress will preclude governmental action in light of the recent disaster at Chernobyl, and the recent failures of CP&L to adequately inform the public of thhe nature of an early morning siren which left numerous residents of the Emergency Management zone wondering whether to evacuate, and subsequently wondering whether the plan as designed could adequately assure the health and safety of their person in the event of a radiological emergency. In Metropolitan Edison v People Against Nuclear Energy 460 US 766, 75 Led2d 534, 103 S.Ct. 1556 (1983), the court held that the NEPA does not require the NRC to consider whether the risk of an accident at a nuclear power plant may cause harm to the psychological health and community well being of residents of the surrounding area. The Supreme Court in so holding did not affirmatively prohibit the consideration of psychological stress by the NRC in their determination of whether to order an Environmental Impact Statement or investigation.

C. Petitioners argument begins with the following premises: that the Commission must comply with the NEPA before it takes 'major federal action'. That such "major federal action' creates a statutory responsibility with the NEPA. A 'major federal action', includes, but is not limited to, new and continuing activities, including projects and programs, entirely or partially fininished, conducted, regulated or approved by federal agencies. 40 CFR s15.08.18(a). See also, 678 F.2d at note 14. (direct and immediate effect of psychological health or community well being). It is clear that responsibility to assure that nuclear power plants will operate without endangering the health and safety of the public lies with the Commission. Where the Commission takes 'major federal action' such action is continually reviewable in accordance with the standards set out in the NEPA.

The Commission is required to prepare a Supplemental Environmental Impact Statement upon the occurrence of either of the following conditions: first, where the agency makes substantial changes in the proposed actions that are relevant to safety concerns; and, second, where there are significant new circumstances or information which are relevent to environmental concerns and bears to the proposed action. 40 CFR 1502.9(c)(1). The petitioner argues that three significant new circumstances have developed within the time of the FEMA approval of the EMP

and this date. (as will be argued later the Chernobyl accident and the false siren, 28 June 1986, in Chatham County, and the Chatham County pull-out are such significant new circumstances).

D. The factors employed in determining whether an event rises to the level of a significant new circumstance are;

(a) the environmental significance of the new information;
(b) the probable accuracy of the information;
(c) the degree of care the agency used in considering the new information;
(d) the degree to which the agency supported its decision with additional data.

Warm Springs Dam Task Force v. Gribble, 621 F2d 1017 (9th Cir. 1980). These factors are relevent here to the degree that the Commission is required to take a 'hard look' at events which may rise to the level of significant new circumstances. Furthermore, in reviewing environmental allegations the Commission should take a 'hard look' where significant new circumstances are asserted. Alleged facts should be evaluated by the Commission, in a complete and comprehensive manner. See: 678 F2d 234, Note 20.

1. The twin disasters of Three Mile Island and Chernobyl have raised compelling questions with respect to the dispersal of radiation. The Director of Nuclear Reactor Regulation should take a 'hard look' at NUREG-CR-2239 and NUREG-CR-0956.

These documents concern data with respect to severe accidents (of the Chernobyl and TMI type). The issue concerns the quantities of radioactive material which affect persons. The NRC's failure to consider as part of its environmental assesment NUREG-CR-2239 and 0956, which is current and accurate information. In light of the particular argument NUREG 2239/0956, and the general argument that scientific understanding has been significantly advanced in light of TMI and Chernobyl (with respect to the dispersal of radiation), notions concerning the adequancy of a ten mile emergeny planning zone may be inadequate to protect the health and safety of those living around the Shearon Harris Nuclear Power Plant. Because the petitioner alleges a new, significant, environmental circumstance, supported by some particular data, it is moved, pursuant to 10 CFR 2.206, that the Director take action consistent with this new information and conduct an Environmental Impact Statement prior to any affirmative licensing action concerning Shearon Harris Nuclear Power Plant. The petitioner moves that the decision of the Licensing Board be stayed pending completion of the Environmental Impact Statement.

Wherefore, the undersigned, individually, and in their representative capacity prays that you institute a proceeding pursuant to 10 CFR 2.202, based upon the moved issues raised herein.

2 July 1986

Respectfully submitted,

Joseph T. Hughes, Jr.
604 W. Chapel Hill St.
Durham, N.C.
(919) 982-3818

Steven P. Katz
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Wells Eddleman, pro se
Durham, N.C.
(919) 688-0076

AAPPENDIX - A

Unanimously Approved April 19th 1986

DRAFT

MOTION CONSTITUTING THE COALITION FOR ALTERNATIVES TO SHEARON HARRIS (C.A.S.H.), CREATING AN INTERIM STEERING COMMITTEE, AND ESTABLISHING TWO THIRDS MAJORITY VOTE AS BASIS FOR DECISIONS.

Whereas the impending loading and operation of the Shearon Harris Nuclear Power Plant is a threat to our health, safety, and economic well-being and necessitates quick, creative, and concerted collective action both within and across our communities this Emergency Regional Assembly hereby constitutes itself as the Coalition for Alternatives to Shearon Harris (C.A.S.H.), membership in which is open to all individuals and groups which endorse the Apex Declaration. Further, until the convening of a second Regional Assembly, it creates an Interim Steering Committee to guide the Coalition's growth and activities to be comprised of representatives of those working groups which may be established to further the Coalition's aims and objectives, and representatives of those local organizations which may be created to implement them. Further, it establishes consensus as the ideal to be strived for in Coalition and Steering Committee decisions and specifies that in the event consensus is unattainable decisions shall be based on two-thirds majority vote.

Affidavit

My name is Ted Outwater. On Saturday, June 7, 1986, I contacted the following residents living within the Five Mile Zone around the Shearon Harris Nuclear Power Plant and obtained their signatures on the attached document.

I am a member of the Coalition for Alternatives to Shearon Harris (C.A.S.H.), serve on the C.A.S.H. Steering Committee, and work out of our Durham Office at 604 W. Chapel Hill St. Durham N.C. 27701.

Ted Outwater

Ted Outwater

State of North Carolina, Durham County

I, Julia Borbely-Brown, a notary public, due hereby certify that Ted Outwater the affiant personally appeared before me this day and acknowledged the due execution of the foregoing affidavit.

Winess my hand and notarial seal, this the 8th. day of June, 1986

Julia Borbely-Brown

notary public

State of North Carolina, Durham County

My commission expires: April 20, 1987

APPENDIX B

We would like the Coalition for Alternatives to Shearon Harris (C.A.S.H.) to represent us and to intervene on our behalf before the Nuclear Regulatory Commission in the matter of licensing the Shearon Harris Nuclear Power Plant. We do not believe that the interests of the residents living within the Five Mile Zone around the Harris plant have ever been recognized or represented.

NAME	ADDRESS	DO YOU LIVE INSIDE THE FIVE MILE ZONE?
Calvin P Rogers	RT 1 BOX 323 NEW HILL NC 27562	YES
Warren Thomas	Rt 1 Box 357 New Hill, N.C. 27562	YES
Rachel Thomas	Rt. 1, Box 357 New Hill NC 27562	
Mary Harris	P.O. Box 96 New Hill, N.C. 27562	YES
Susie Harris	P.O. Box 96 New Hill, N.C. 27562	YES
(signed for her by her husband - hurt hands)		
Wayne Dixon	Rt 1, Box 129 New Hill NC 27562	YES

APPENDIX I

Coalition for Alternatives To Shearon Harris c/o Durham Research Office 919-682-3818
604 W. Chapel Hill St. Durham NC 27701

APPENDIX C

A Resolution Concerning
the
Shearon Harris Nuclear Power Plant

WHEREAS, the nuclear power plant accident on April 26, 1986 in Chernobyl USSR has aroused widespread concern within the United States and throughout the world about the safety of nuclear power plants, and,

WHEREAS, there has surfaced within Chatham County a widespread and intense opposition to the nearly completed Shearon Harris Nuclear Power Plant constructed by Carolina Power and Light Company, and

WHEREAS, there are substantial and unresolved issues about the Chatham County evacuation plan,

NOW, THEREFORE, BE IT RESOLVED that the Chatham County Board of Commissioners hereby rescinds all prior approvals of the Shearon Harris Emergency Response Plan pending further critical examination of the unresolved issues.

This resolution shall be effective upon adoption.

This the 27th day of May, 1986.

Earl D. Thompson
Chairman

Hazel P. Boone
Clerk to the Board

STATE OF NORTH CAROLINA
COUNTY OF CHATHAM

AFFIDAVIT

From: Dan Frazier
Rt. 9, #1 Jones Branch Rd.
Chapel Hill, N.C. 27514
962-2267, 967-9057

This affidavit is to indicate that at 3:00 pm on 28 June 1986 I heard the first reports that some of my neighbors living about three miles south of my home in Chatham County heard a Shearon Harris emergency siren at about 1:55 am on 28 June 1986. I was concerned that part of the evacuation system upon which I rely had malfunctioned. I was also concerned that some of my neighbors were unable to find out what was happening for over 30 minutes. I was concerned enough to talk to some of the people who live near the siren and collect affadavits from them. I wanted to find out what happened and what effects the incident was having on those involved.

At 10:33 am on 29 June 1986 I called Shearon Harris, 362-8793, to find out what had happened. The man who answered the phone said that he was in the guard shack; that he didn't know anything about any siren or alarm Saturday morning and that there wasn't anyone for me to talk to. He was basically uncooperative, uncommunicative and uninformed. I had the distinct impression that he had been told that he didn't know anything.

At 10:35 am I called the Chatham Sheriff dispatcher, 542-2811, and he said that an emergency siren on Pea Ridge Rd. had gone off Saturday morning. He didn't know which of the two alarms on Pea Ridge Rd. had gone off.

At 12:10 pm I visited Barbara Keyworth and David Richardson on Hatley Rd. about 2 miles north of the siren which was reported to have gone off. At about 1:55 am Ms. Keyworth was awakened by a siren. She thought it was the Shearon Harris alarm because she had heard it before. She woke Mr. Richardson who also heard the siren. She estimates that the siren sounded about 3 to 5 minutes.

They feared that they might be in danger since they knew there was nuclear fuel at the plant. They called the Chatham Sheriff, Shearon Harris, and Raleigh State Patrol. Only Shearon Harris had an explanation: that they heard the shift change or break whistle. Ms. Keyworth did not accept this explanation since they live 11 miles from the plant.

More than 30 minutes after the siren sounded, they were finally told by the Chatham dispatcher that CP&L doesn't know why the alarm went off and that there was not an emergency.

Mr. Richardson then showed a cassette recording of the WRAL 11:00 news from 28 June 1986. In the newscast, Bill Lesley stated that CP&L officials had reported that vandals had broken into the Shearon Harris plant and set off an alarm. He also stated that CP&L planned to increase security at the plant. I was quite concerned since this was the third explanation that I had heard from CP&L. I was also a little amused.

Amusement turned to shear entertainment when I read in the 29 June 1986 News and Observer a fourth and all new explanation. Mac Harris, CP&L spokesman, was quoted: "We have clearly established that the siren was deliberately set off by some individual or individuals who vandalized the siren. Someone had to make a real effort to do it." I anxiously anticipate future explanations. I am really intimidated to have my well-being in the hands of people who have given me every reason to mistrust them.

At about 2:30 pm I visited with Mitchell Riley on Hatley Road about 2 miles from the siren. He and his wife Kay Riley were asleep at the time of the siren and were never awakened. They had their bedroom windows open and a quiet fan running. Mr. Riley stated that he had no faith in the evacuation plan and that they would probably move if the plant started up.

At about 3:30 pm I visited Ruth Thomas on Pea Ridge Rd. Her house is located across the street (about 200 feet) from the siren which sounded Saturday morning. She was awake after 1 am Saturday morning and heard the siren go off for about 5 minutes. Within two minutes after the alarm started she went outside to her front porch to see if CP&L was testing the siren. Although the siren isn't quite visible from the front porch because of the trees, she was convinced that no one was at the siren. She heard no one and heard no vehicles. Also, her high-strung dog didn't start to bark until she was outside. She felt sure that the dog would have barked if someone had been at the siren.

I was shocked that no one in her family was awakened by the siren. This includes her husband, Lieutenant Charles Thomas, of the Chatham Sheriff Department, and their son and daughter. The windows were closed and there were no fans or air conditioning running. It concerned me that one of the sirens, which we rely upon in case of a disaster, can't even wake people 200 feet away.

Ms. Thomas knows Anne Wilke who was the dispatcher for the Chatham Sheriff's Department at the time of the incident. Ms. Wilke told her that she was swamped with calls from people asking about the siren and had called in an extra dispatcher. Ms. Wilke also told Ms. Thomas that she had called CP&L to find out what had happened and that they said that the siren had been turned on accidentally.

Ms. Thomas and I then carefully examined the siren, the pole, the boxes on the pole and the area around the pole at 4:30 pm. I observed no breakage, no scratches or any physical damage at all. All of the locks were weathered. There were no parts that looked new or replaced. Ms. Thomas said that everything looked the same as always to her. She had examined the siren closely. She concluded that she really doesn't believe that anyone vandalized the siren.

I then drove to the south end of Pea Ridge Rd. to see the siren there about 1.5 miles from Ms. Thomas' house. I felt that since this siren was located further from houses than the Thomas siren, it would be a better choice for a vandal. I then drove to the siren on Big Woods Rd. about 3 miles from the Thomas siren. This siren is isolated far from any houses and would have been the best choice of the three for a vandal. I can't help but conclude that many of the other 66 sirens are also isolated. Why would a vandal pick the one across the street from a Lieutenant in the Sheriff's Department?

At about 5:30 I talked with Claire and Edward Thomas who live on Hatley Rd. about 2 miles from the siren. The siren woke him up and she was already awake. They thought it was a wreck or something. They never thought about Shearon Harris. The incident left them less secure about the evacuation plan.

At about 7:00 I talked to Radd Greenlaw on Hatley Rd. about 2 miles from the siren who was asleep and never heard the siren. Her husband Raymond Greenlaw woke up but didn't know why. Ms. Greenlaw very angry about the incident. She has never had any faith in the evacuation plan.

At about 7:30 I talked to Robert Hatley on Hwy. 64 about 1/4 mile from the siren. He was awake, heard the siren, knew exactly what it was and called the Chatham Sheriff (911). Anne Wilke, the dispatcher, didn't know anything and put him on hold. Anne then came back on the line and said they were investigating. Then the line was somehow cut off. Mr. Hatley got no explanation that night.

On 30 June 1985 at about 8:45 am I called Mac Harris, CP&L spokesman, 836-6189. I identified myself and said I lived near the siren and had collected affidavits from about twelve people and that I wanted to find out what happened from CP&L's viewpoint. The following is not verbatim, but accurately represents the ideas that were exchanged.

Harris: What are you going to do?

Frazier: I just want to find out what happened.

Harris: If you're getting signed affidavits you're obviously

taking action against CP&L. What orders are you going to bring against us? (very agitated)

Frazier: No kind of action. I was thinking of handing the affidavits over to the media.

Harris: Oh yes, oh well, okay, the press then. What is it you want to know?

Frazier: There are four contradictory explanations about what happened: (1) Shift change horn, (2) Error at the plant, (3) Vandals at the plant, (4) Vandals at the siren. Which is correct and how do you explain the other versions?

Harris: It is absolutely clear that someone forceably physically removed a lock (which was later replaced) on a control box at the siren and set off a 3 min. cycle at full volume. The 3 min. cycle cuts off automatically after 3 min. It was probably someone with a purpose and an agenda. We know this happened and I'm not interested in proving it.

I then asked Mr. Harris to address each of the other explanations mentioned above. He answers:

Explanation 1--It is reasonable that the people at the plant thought it was a change horn. People at the plant had no way of knowing the alarm went off (He was unaware of this explanation).

Explanation 2--He was also unaware of this version. He thought the Chatham sheriff had control of the switches. He doesn't know who the Sheriff's department talked to at Shearon Harris.

Explanation 3--WRAL got it wrong. He personally related explanation 4 to WRAL. It must have been changed in translation.

Mr. Harris stated that the sirens are fired by radio signal but can be set off from the box at the siren. There is no feed back from the alarms. The only way to know if an alarm goes off is to hear it.

Apparently, the next time one goes off like this the same thing will happen again.

Frazier: The alarm didn't awaken 3 people right under it. Will it be effective in an emergency?

Harris: That's just incredible. It's about 127 decibels. I don't know what those peoples' sleep habits are.

Harris later admitted that hot humid conditions like those of 28 June 1986 have great damping effect on sound and since the sirens weren't reliable under those conditions people within five miles were given special radios to warn them. Not all

of the people in the 5-10 mile zone are supposed to hear the sirens. He said that they aren't in as much danger anyway.

I asked about people (these people were just outside the 10 mile zone) not knowing who to call and not getting good answers. He replied that it is a real problem that people eleven miles from the plant don't know what to do. People in the 10-mile zone had been instructed to tune into the Emergency Broadcast System. Supposedly if they hear an alarm and turn on the radio and don't hear about an emergency then there isn't one. He said that the people who live eleven miles from the plant were a tough issue since they could hear the sirens but hadn't been informed of what to do. He said CP&L should do something about it.

When Mr. Harris heard about the siren at about 2:30 am 6/20/86 he thought about calling the press but didn't know who to call at that hour and so called no one.

I informed Mr. Harris of all the evidence (previously mentioned) that seemed inconsistent with the vandal at the siren hypothesis. He was agitated and said I'd just have to accept his version as fact.

Mr. Harris did not say how the siren was set off in the interest of not letting people know how to do it again. I asked if a system might be installed to notify some authorities immediately when an alarm goes off. He said he didn't know if such a system existed.

Mr. Harris took my number and said he would contact me if he got any new information. I thanked him and said goodbye. The information I learned from my neighbors leaves me very distressed. The sirens will not reliably awaken us and many won't know what to do if we hear it. There may be more false alarms and the authorities will not have any immediate answers. If I hear a siren I'll evacuate first and ask questions from Virginia.

Because of the four different explanations given for what caused the siren, I feel I cannot trust CP&L to let me know what is happening, even after the fuel is loaded. I have a strong fear that if there is a radiation leak from the plant that CP&L will take whatever action is in its interest and will not act primarily in the interest of threatened citizens.

I have written the above statement and believe that it is a true and accurate statement of the events and occurences described therein.

STATE OF NORTH CAROLINA
COUNTY OF CHATHAM

Addendum to Affidavit of Dan Frazier 6/28/86

Contemporaneous to the printing of this affidavit I learned of new information which indicated that the siren in front of Ruth Thomas' house may not have been the one which sounded on 6/28/86. It was probably the siren on Hank's Chapel Rd. near Ms. Thomas' which sounded. This information corroborates CP&L's explanation that a vandal set off the siren at the siren.

I have written the above statement and believe that it is a true and accurate statement of the events and occurrences described therein.

[illegible]

AFFIDAVIT

FROM: Barbara Keyworth
David Richardson
Route 4, Box 641
Pittsboro, NC 27312

This affidavit, taken by Dan Frazier at 12:30 p.m. on June 29, 1986, is to indicate that to the best of their recollection Barbara Keyworth and David Richardson heard a siren just before 2:00 a.m. on June 28, 1986. Their home is about two miles from the siren which was later reported to have sounded.

Ms. Keyworth heard the alarm first and thought it was the emergency alarm for Shearon Harris because she had heard it once before. She woke Mr. Richardson. Both noticed a moderate wind blowing from the Shearon Harris plant.
(BK) PR
At about 2:00 a.m. Mr. Richardson dialed the operator and asked for the police. He did not know which police he spoke with. The police expressed surprise and doubt that it was Shearon Harris.

Mr. Richardson had a very eerie feeling and really felt that something was wrong. Ms. Keyworth though that if the alarm was going off then something must be wrong at the plant.

Mr. Richardson called the operator and asked for a number for Shearon Harris. The number he tried was disconnected or no longer in service at that time.

Ms. Keyworth dialed 911 and talked to Anne Wilke, dispatcher for the Chatham Sheriff Department. She was surprised and did not know anything.

Mr. Richardson called the operator and got two numbers for Shearon Harris, 362-2320 and 362-8891. He dialed 362-2320 and reached Murdoch Jones in security. Mr. Jones said there had not been an accident and that the horn was for the shift change or break. When asked for his supervisor, he ignored the request and restated that it was the break siren. Mr. Richardson called 362-8891 and reached David Dean of the payroll office, who said that the siren was for the shift change and that it went off at 2:00 and 4:00 every morning. Mr. Dean expressed irritation and was sure that Mr. Richardson had heard the shift change whistle. Mr. Jones and Mr. Dean were told that Mr. Richardson and Ms. Keyworth live 11 miles from Shearon Harris.
(BK) PR
During these phone calls, Mr. Richardson and Ms. Keyworth wondered whether they should go ahead and evacuate or stay and keep trying for an explanation. They were aware that nuclear fuel was present at the plant. They really felt helpless.

Since the Chatham commissioners had pulled out of the

evacuation plan, Mr. Richardson thought he should contact the people who would take over the evacuation plan, the State Patrol. Ms. Keyworth called the operator for the State Patrol number. The operator asked, for what city? Ms. Keyworth said, for Pittsboro. The operator said there was not a patrol office there. Ms. Keyworth asked for Raleigh and got a number.

She dialed that number and reached Trooper Whitehouse, who laughed at her concerns and did not take her seriously. He said that he lived six miles from the plant and that there was nothing down there. She replied that there was nuclear fuel there. He asked, "Where did you hear that?" in a tone which implied that she was misinformed. He made no indication that he would do anything at all. Ms. Keyworth answered that it was public information and that there was the potential for a problem. She said that there had been problems at other new plants before fuel loading.

Ms. Keyworth asked Mr. Whitehouse to call Shearon Harris and ask what happened. He agreed to. He called back quickly and said it was the break bell. Ms. Keyworth said that that was impossible, since she lived eleven miles from the plant. Mr. Whitehouse replied that they were testing the sirens all the time. She answered that she had never heard one at night. Mr. Whitehouse suggested that maybe someone had pushed the wrong button, and then said that he was not going to argue at 2:00 a.m. Ms Keyworth was annoyed that she was involved in a discussion and having her concerns explained away when she had called in reporting an emergency alarm.
Ms. Keyworth told Mr. Whitehouse he had laughed at her. He replied, "No, I didn't." She asked his name. He replied, "Whitehouse, and I'm the night supervisor." She hung up, angry. Mr. Richardson called the governor's hotline, (800)662-9952, and got no answer. Ms. Keyworth called the Governor at 733-5811 and got no answer. She called WRAL radio and got no answer. She called the 94Z radio station and got no answer. She was very frustrated that, because it was 2:30 a.m., she could not reach anyone on the phone.
At 2:35 a.m. Ms. Keyworth called Anne Wilke, the Chatham Sheriff's dispatcher. Ms. Wilke said she had someone from CP&L on the line who wanted to know what the siren sounded like and how loud it was. Ms. Keyworth imitated the slowly oscillating, wailing sound. Ms. Keyworth then asked if there had been other calls. Ms. Wilke said "several," and then said it was not an emergency and CP&L did not know why the alarm had gone off.

Ms. Keyworth and Mr. Richardson got back to sleep after 3:30 a.m.

At 9:00 a.m. on June 28, 1986, Ms. Keyworth called 911, the Chatham Sheriff, about the alarm again. She was told that "someone down at the plant set it off accidentally."

BK
DR

Both Ms. Keyworth and Mr. Richardson were very angry and concerned that CP+L did not give any information by Radio or media that this was a false alarm, and in fact, did not respond at all. They feel the evacuation plan is totally unreliable and that CP+L and the N.C. State Highway Patrol were negligent in their responses.

To the best of my knowledge this statement accurately reflects the substance of my conversation with
BARBARA C. KEYWORTH
DAVID Richardson

I have read the above statement and believe it is a true and accurate statement of the events and occurrences described therein.
Barbara C. Keyworth
David Richardson

AFFIDAVIT

From: Ruth Thomas
Route 4, Box 835
Pittsboro, NC 27312
542-4030

This affidavit, taken by Dan Frazier at about 4:00 p.m. on June 29, 1986, is to indicate that on June 28, 1986 after 1:00 a.m., Ms. Thomas was awake and heard a siren go off. The siren is across the street from her house on Pea Ridge Road and is about 200 feet from her house. The siren sounded for about five minutes. She knew immediately that it was the Shearon Harris emergency siren and went outside to see if CP&L was testing the alarm. From her porch she saw and heard no one and no automobiles. There are trees that block the view of the siren from the front porch but she believed no one was there. Her excitable dog was sleeping outside in front of the house and did not start barking until she went outside. She felt sure that if someone had been present the dog would have barked.

She noted that the alarm did not seem as loud as it had when she had heard it previously. She did not feel that it was loud enough to awaken people. In fact, her husband Charles Thomas and their two children never awakened during the incident. Their windows were down and no fans or air conditioning were on.

She was not concerned that there was an emergency because she was monitoring a police scanner and she believed that the Sheriff's Department would have to have been called before the alarm could have been sounded. Since there was no news she assumed the siren to be a test or an error. She is worried that the sirens will not wake people up in an emergency.

Ms. Thomas does not believe that anyone vandalized the siren. She examined the siren, the pole, and the boxes on the pole

carefully at about 5:00 p.m. on June 29, 1986. She stated that everything looked normal to her and she saw no evidence of tampering. She had examined the siren previous to the incident.

At some time long after the siren sounded Ms. Thomas called Chatham Sheriff dispatcher Anne Wilke. Ms. Wilke told her that she had been swamped with calls from people concerned about the siren and had had to call in an extra dispatcher. She also stated that she had called CP&L and that they had said the siren was accidentally turned on.

To the best of my knowledge this statement accurately reflects the substance of my conversation with

_____. _____

I have read the above statement and believe it is a true and accurate statement of the events and occurrences described therein.

AFFIDAVIT

From: Anne Greenlaw
Route 4, Lot 2, Jordan Woods
Hatley Road
Pittsboro, NC 27312
542-3465

This affidavit, taken by Dan Frazier at 11:00 a.m. on June 30, 1986, is to indicate that Anne Greenlaw was awake at 1:55 on June 28, 1986, and heard a siren which was very faint. Her home is about two miles from the siren. Her windows were closed and an air conditioner and fan were on. She never considered that the siren might be from Shearon Harris.

When she learned that it was an emergency siren for Shearon Harris she felt much less safe because the alarm malfunctioned and because it was too faint to elicit an evacuation response.

To the best of my knowledge this statement accurately reflects the substance of my conversation with

_____. _____

I have read the above statement and believe it is a true and accurate statement of the events and occurrences described therein.

_____

AFFIDAVIT

From: Claire and Edward Thomas
Route 4, Box 638
Hatley Road
Pittsboro, NC 27312
542-3637

This affidavit, taken by Dan Frazier at about 500 p.m. on June 29, 1986, is to indicate that Claire Thomas was awake at about 2:00 p.m. on June 28, 1986, and heard a siren. Thomas Edwards was awakened by the siren. Their windows were open and no fans were running. Their home is about two miles from the siren. They thought the siren was from a wreck or something and never thought of Shearon Harris. When they learned that the alarm was from Shearon Harris, they felt less secure about the evacuation plan.

To the best of my knowledge this statement accurately reflects the substance of my conversation with

_____. _____

I have read the above statement and believe it is a true and accurate statement of the events and occurrences described therein.

_____

Comment on
Outdated Federal Guidance for
Size of the Emergency Planning Zone

Kenneth G. Sexton, Ph.D.
Research Associate
Dept. Environmental Sciences and Engineering
School of Public Health
University of North Carolina
June 30, 1986

Q. "IS A 10-MILE EVACUATION AREA ADEQUATE?"

A. NO ONE REALLY KNOWS.

Why not? There are many uncertainties in predictions of nuclear-power-plant-accident consequences. These result from uncertainties in the prediction techniques and in input data. The NRC is currently attempting to resolve major uncertainties for risk assessment. Generic rather than site-specific calculations were performed (using some outdated techniques and over-simplifying assumptions) to help determine the distance. The 10-mile evacuation plan is supposedly adequate to use as a base for evacuating additional areas outside the 10 miles as needed on a "ad hoc" basis when an accident does occur. No one knows if it will work until an accident happens because there are no required formal, predetermined, evacuation plans in place outside the 10-mile area to evaluate. No one claims that deaths and injuries will not occur outside the 10-mile EPZ in the case of a more severe accident.

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There are several important points that should be made very clear to all officials concerned about protecting the safety and health of the people in the counties surrounding any nuclear power plant. These facts come from reports and regulations from the Nuclear Regulatory Commission and the North Carolina Emergency Response Plan (NCERP). The immediate concern is with the Shearon Harris Nuclear Power Plant (SHNPP). However, the following discussion applies to any nuclear power plant of comparable size because the 10-mile EPZ is a generic distance which applies to all U.S. nuclear plants of comparable size.

The 10-mile emergency planning zone (or EPZ) is based on findings of a joint NRC-Environmental Protection Agency (EPA) Task Force which were published in 1978 (NUREG-0396). They concluded that the 10-mile EPZ was more than adequate to protect the public. However, it is also made clear that:

1) Although most early fatalities and injuries will occur inside the 10-mile EPZ, the NRC (NUREG-0396, pg 17; NUREG/CR-2239, pp 1-3 to 1-6) and the NC Emergency Response Plan (NCERP, Part 1, pg 1) acknowledge that some of the early severe health effects (injuries or deaths) which would result from the more severe accidents will occur beyond the 10-mile EPZ.

   "In addition, the EPZ is of sufficient size to provide for substantial reduction in early severe health effects (injuries or deaths) in the event of the more severe Class 9 accidents." (NUREG-0396, p 17)

2) The size of the EPZ and the emergency plan are not restricted to, nor designed specifically for protecting only the people in, the 10-mile EPZ. They are designed for the protection of all areas and all people that could be affected by an accident. The NRC assumes that any emergency plan deemed adequate for a 10-mile radius is sufficiently detailed to be adequate to cover emergency needs in areas beyond the 10-mile EPZ (NUREG-0396, pp 15-16). The NRC, CP&L, and NCERP acknowledge that emergency response outside the 10-mile EPZ may be needed. "The size of the EPZ represents a judgment on the extent of detailed planning needed to assure an adequate response base" (NCERP, Part 1, pg 1). The concept in the NCERP and NRC guidance is to use the EPZ planning as a "base for expansion of response efforts if necessary" (NCERP, Part 1, pg 1) and to respond on an "ad hoc" basis (NRC, NUREG-0396, pg 16).

3) The size of the 10-mile EPZ is "tempered" by probability (NUREG-0396, pg 15). Some amount of risk was determined by the NRC to be acceptable. Their decision was affected by low-probability estimates of the occurence and nature of severe accidents (NUREG-75/014). More recent NRC reports indicate that many of these earlier accident estimates may be too low (NUREG/CR-0400 cited in NUREG/CR-4199, pp 1; and NUREG/CR-4199, pp 8-9). There is much uncertainty in risk and probability estimates, as well as disagreement among experts on this matter (as indicated in different

NRC reports). The inclusion of a greater accident probability could result in the establishment of a larger EPZ upon reevaluation. Also, it should not be implied that the term "low-probability accident" indicates that a long time will pass before such an event occurs. It is therefore reasonable to expect that consideration of emergency plans be "tempered" by these uncertainties. Local officials should plan accordingly, especially when highly-populated areas are very near but beyond the presently-accepted 10-mile EPZ.

4) The latest NRC regulations published January 1, 1986 cite only this 1978 Task Force report as a basis for determining the EPZ (10 CFR 50.47 and its Appendix E). No report is cited which discusses or suggests a smaller EPZ for nuclear plants the size of the SHNPP. Simple techniques and information now known to be inappropriate, or at least not the best, were used for generic calculations used in determining the 10-mile EPZ. Furthermore, seemingly inconsistent NRC regulations do require "state-of-the-art" (current) computations be performed after an accident using site-specific information (eg. information specific to SHNPP) (NUREG-0654, Appendix 2, pp 2-2 and 2-3). "State-of-the-art" models (NRC-sponsored) have been used in recent years to estimate radiation doses to the public under a variety of accident and normal operation conditions, but evidently have not been used for reevaluation of the EPZ (NUREG/CR-2239, NUREG/CR-4199, NUREG/CR-3344, NUREG/CR-4000). Uncertainty is a major problem in accident predictions (NUREG/CR-2239, pp 2-7 to 2-10). There is, in fact, an on-going program for reevaluation of nuclear accident risk at the NRC, but work to date has been "greeted with skepticism... There is a disagreement over the credibility of some computer modeling codes that are the basis for all the predictions that will come out of NUREG-0956" (Science, April 1986, pp 153-154, attached). Therefore, there is justification in requesting the NRC to review and update the 1978 Task Force Report, and consequently the justification for the size of the EPZ. Current thinking would suggest that the NRC should require the SHNPP and all other plants to reevaluate the 10-mile EPZ using on-site and national weather service weather data specific to the area.

Local officals are responsible for deciding if this type and size of emergency planning is acceptable and adequate. There should be demonstrable assurance of ad hoc capability being adequate. For example and specifically related to the SHNPP, consideration should be given to the effect on local emergency response efforts if it were determined that Raleigh (and the state government) needed to be evacuated. Local officials must decide if they accept the very low NRC accident-risk and probability estimates which were determined before the Three Mile Island accident -- a serious accident which occurred despite its "low probability" of occurence.

Those responsible for assuring the health and safety of the public should be aware that current techniques have not been used in establishing the EPZ and that there are serious questions in regard to some of the assumptions under which it was established. The obvious implication is that these calculations and the resulting 10-mile recommendation are therefore suspect and uncertain for purposes of protecting public health.

ADDITIONAL DISCUSSION

The 10-mile Emergency Planning Zone (EPZ) is recommended by the Nuclear Regulatory Commission (NRC) as follows:

> "Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius, and the ingestion pathway EPZ shall consist of an area about 50 miles (80km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries." (10 CFR Part 50.47 "Emergency Plans")

This regulation recognizes that approximately a 10-mile radius is appropriate, but also implies that alternate sizes and configurations may be very significantly more appropriate. Although the regulation requires consideration be given to several area-specific factors, no mention is made of local meteorology. This is in contradiction to regulations for siting and post-accident calculations (10 CFR 100.10 and 10 CFR 50.47, respectively), and the findings of more recent accident-consequence estimates (NUREG/CR-2239, p 1-3), all of which consider local meteorology. Local officials must carefully determine local emergency response needs and the adequacy of emergency capabilities in approving a plan specific to a given nuclear power plant.

The 10-mile EPZ is based on the report of a joint NRC-Environmental Protection Agency (EPA) Task Force which was published in 1978. The report's principal meteorological references are dated 1968 and 1970 (USAEC, 1968; Turner, 1970). The report concluded that the 10-mile EPZ was more than adequate to protect the public. However, they used 1) meteorological techniques that are now outdated, and 2) nuclear-reactor-accident estimates developed before the Three Mile Island accident experience and before subsequent additional experiences with nuclear reactor problems. These early calculations and EPZ estimates depend on the estimates of the amount of radioactivity that would be released during accidents and the probabilities of different types of accidents occurring. Assumptions were made which now may be incorrect or inappropriate. Very simple assumptions were made concerning the behavior of the radiation plume that might be released in an accident. The atmosphere and its weather systems are very complex, and a wide range of plume behavior is possible. "The weather conditions at the time of a large release will have a substantial impact on the health effects caused by that release" (NUREG/CR-2239, pg 1-3). Given a plume released during an accident that would result in injury within the 10-mile EPZ, there are meteorological conditions which could result in significant exposure at distances beyond the 10-mile EPZ and even hundreds of miles "downwind". The plume can meander rather than travel in a straight line, making predictions of exposure difficult and allowing for multiple exposures to the population. Also, important considerations such as the effect of rain were mentioned but not included in calculations used in the final distance determination in the 1978 report (NUREG-0396, pp I-25 and I-26). The importance of the effects of rain on downwind radiation doses to the public are now documented by the NRC (NUREG/CR-2239; NUREG/CR-1244). Significantly-larger doses to the public can occur further downwind if the radiation release is "washed-out" of the air by rain (rain can clean the air of radioactive particulate as it falls, creating "hot spots" on the ground). On the official average, North Carolina receives rain on one of every three days. As another example, it was assumed in the report that the major dose exposure would occur within 2 hours after the accident. This assumption is debatable and has several implications. The evacuation time estimate for the NC Emergency Management Plan for the SHNPP is almost 4 hours. Sheltering in place until the released radiation passes may be the best strategy under some adverse conditions, but some meteorological conditions could result in long and uncertain sheltering times (waiting) while some lower-level exposure continues. Therefore, careful dose estimates and monitoring, accurate evacuation-time estimates, and good management by emergency personel are needed to minimize personal injury not only within the

10-mile EPZ but also at distances beyond the 10-mile EPZ. Unfortunately, beyond 10 miles these types of decisions and management will be performed ad hoc after an accident occurs. With a mean wind speed of approximately 7.5 mph in this area, there will not be much time (1-2 hours) before there could be a problem beyond 10 miles. It is prudent to be able to respond to problems beyond this distance for this reason, if for no other.

All nuclear units operating in this country are subject to the same type of plan. The calculations used for determining the 10-mile EPZ were performed for hypothetical accidents and meteorological systems. The generic 10-mile-distance calculations obviously do not use meteorological parameters or other factors specific for the Shearon Harris site and power plant. There are now better methods for modeling a specific site which result in more appropriate calculations. The NRC now uses more up-to-date (more correct) techniques and computer models to estimate site-specific radiation releases and doses to the public. Several of these models were developed by the NRC itself but evidently have not been used for reevaluation of the 10-mile EPZ. Even with these improved techniques, it is recognized and documented by the NRC that the reliability of the risk and dose estimates is still limited by the uncertainty of the amounts of radiation that will be released during accidents (NUREG/CR-4199, p 8). These uncertainties are further increased by the uncertainties of the meteorological estimates (NUREG/CR-4199, p 9; NUREG/CR-2239, p 1-3).

The obvious implication is that these calculations and the resulting 10-mile recommendation are therefore suspect and uncertain for purposes of protecting public health. Reevaluation with more current methodologies and recent experience could result in a larger EPZ distance which would require modification of the emergency plan and required participation outside a 10-mile radius before licensing of a plant. Part of demonstrating that an emergency plan is adequate is to show that the size of the area affected by the plan is appropriate. The problems and limitations of the older methodologies are now well documented. Those responsible for assuring the health and safety of the public should be aware that current techniques have not been used in establishing the EPZ and that there are serious questions in regard to some of the assumptions under which it was established. Consequently, the emergency plan may not be adequate to protect the health of the public in general. This is especially serious in the case of the SHNPP because heavily-populated areas including the state government systems exist so close to the presently-accepted 10-mile EPZ.

An appendix is being prepared which further documents these statements, includes additional findings and comments, and contains references which document the widely accepted criticisms of the older and simpler assumptions, dispersion parameters, and methodologies. These criticisms are found in 1) reports from the NRC, EPA, AMS (American Meteorology Society), a joint AMS-EPA workshop, and a Department of Energy (DOE) -sponsored DOE-AMS workshop; and 2) a statement from Herschel Slater, formerly of the Monitoring and Data Analysis Division, Office of Air Quality Planning and Standards, EPA, a meteorologist who co-authored the guidance document for EPA Air Quality Models in 1978 (This statement is attached).

Statement by the author:

I am a research associate in the Department of Environmental Sciences and Engineering at the School of Public Health, University of North Carolina, Chapel Hill, where I received my Ph.D. My research field is atmospheric chemistry and computer modeling of photochemical smog. This report represents an independent study not done in connection with my work at UNC.

My personal interest in the emergency plan for the Shearon Harris Nuclear Power Plant (SHNPP) is in regard to the techniques used to establish the size of the emergency planning zone. My reason for preparing this report is a sincere concern that the present plan and zone may be less than adequate to protect the general public in the event of an accident at the SHNPP. I am neither an anti-nuclear activist nor a member of the Coalition for Alternatives to Shearon Harris Steering Committee.

Kenneth G. Sexton

Kenneth G. Sexton, Ph.D

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Statement Concerning
the Procedures for Selecting the
Size and Configuration of an
Emergency Planning Zone (EPZ)

Herschel H. Slater, Consultant
Air Pollution and Meteorology
Chapel Hill,NC 27514
June 28, 1986

(I am a meteorologist, specializing in air pollution matters with experience and training that spans four decades. My experience includes service with the US Weather Bureau; US Air Force, as a career officer; Environmental Protection Agency; Adjunct Associate Professor, School of Public Health, UNC-CH; and Logistics Manager for Project GALE for NCSU and the Natonal Center for Atmospheric Sciences.)

Abstract

I am concerned about the size and configuration of the emergency planning zone (EPZ) as it applies to the Shearon Harris Nuclear Power Plant. CPL and the State of North Carolina apparently have accepted the Nuclear Regulatory Commission's suggested plume exposure pathway EPZ. NRC suggests an essentially circular area having a radius of about 10 miles.

Fortunately, meteorological data and analytical techniques have been developed over the past decade that enable more definitive configurations of EPZ's. CPL has the data and the competence to apply more sophisticated methodologies to this problem than the generic approaches suggested in NRC-promulgated regulations. CPL should be required to re-evaluate the proposed boundaries of the EPZ. I expect the result would be a more realistic and effective emergency response plan.

Discussion

Since the NRC regulations that pertain to the size of an EPZ were issued, most nuclear power facilities collect meteorologica data on site. Not only are the date site-specific, but they are designed to be applied directly to the problem of estimating the transport and dispersion of a cloud or plume of radioactive material.

Until such weather data began to be collected by commercial nuclear facilities, the weather data used to assist in choosing the boundaries of an EPZ usually came from the nearest official National Weather Service station. In the case of SHNPP, this is the station at the Raleigh-Durham Airport.

Data collected at RDU is of highest quality. The equipment is well-designed, excellently maintained and the observers are well-trained and dedicated civil servants. The problem is two-fold: 1) The data are not observed where, in the event

of an accident, the radioactive plume will generate and 2) The equipment is not designed to sense the meteorological phenomena that determine the rate that a plume of nuclear material will disperse. The equipment and observation procedures used at RDU are designed to meet the needs of aircraft operations and safety and to meet the needs of forcasters in preparing forecasts for the general public. The scales (or size) of atmospheric motion sensed for these purposes are much larger than those which control the dispersion of a plume.

The wind equipment at the airport is designed to be insensitive to the small gusts that are significant in determining the dispersion process. Observations are generally made at hourly intervals. This is much less frequent than needed to characterize the power of the atmosphere to disperse pollutants and to sense the rapid changes of gustiness during periods of the day when this phenomena changes rapidly. Also, the wind observations are made at 10 meters, about 32 feet, above the ground, far below the height that a plume likely may travel.

CPL has a body of meteorological data gathered by sensing equipment specifically designed to study and estimate the dispersion and transport of clouds or plumes of pollutants. Unlike the equipment at RDU it is sensitive to the important small-scale motions of the atmosphere. Also, some data are sensed at heights where a plume is most likely to occur.

The rate a cloud disperses is often determined by the character of the surrouding topography. The character of the gustiness is influenced markedly by the roughness and the thermal response of the surrounding surface. Is it farmed or forested? Plowed or covered with vegetation? Is a body of water nearby? The nearby SHNPP lake must have a significant affect on the way the atmosphere would disperse pollutants in the event of an accident. The lake's effect varies with season, time of day and cloud cover. With these considerations, good judgment dictates the use of available on-site data rather than data from a distant point when developing the optimum EPZ.

NRC documents stress the importance of rainfall on peak concentrations. A shower may immediately create a surface "hot spot". If a plume is emitted into a rain situation, little of the radioactive material may leave the site itself. With rain occurring on the average of about one day in three in central North Carolina (except in 1986!), careful analysis of rainfall statistics may dictate EPZ boundaries different than a circle.

Notwithstanding current NRC regulations, CPL and the State can take the initiative to fine tune the configuration of the SHNPP EPZ. CPL has the data and the professional competency to do so. In light of the concerns of so many, it is prudent for CPL so to do.

Roughly $60 million of the new funds sought for this year are to be transferred from the Pentagon to DOE, presumably for one or more underground tests in Nevada, beyond the two to four tests already scheduled for this fiscal year at a cost of $157.8 million. In fiscal year 1987, the underground testing account will jump to $226 million, or enough for three to five explosions. (The budget for underground testing of the weapons has exceeded that for laboratory research for several years.) In addition to the x-ray laser, a variety of nuclear-driven weapons such as particle beams, microwaves, hypervelocity pellets, and optical lasers are also under investigation and may eventually be tested.

"These nuclear power sources, if you want to consider them that way (they are explosions but they act as power sources)," may ultimately be unnecessary for a ballistic missile defense, Wagner testified. But "the first stages of the SDI program, which . . . may last decades, I believe and the Department believes will have this nuclear component, this new kind of nuclear-driven directed energy weapon as one of its very important options." ■ R. JEFFREY SMITH

Briefing:

New Shuttle Director Promises Emphasis on Safety

A new emphasis on safety will be the hallmark of the space shuttle's operations when flights resume, according to Rear Admiral Richard Truly, the new associate administrator for space flight at the National Aeronautics and Space Administration (NASA). Speaking on 25 March before an enthusiastic crowd at the Johnson Space Center in Houston, Texas, Truly outlined a series of activities that he said are "required to establish a realistic and achievable launch rate that will be safely sustainable."

Specifically, the entire budget and program management "philosophy, structure, reporting channels and decision-making process will be thoroughly reviewed," he said. All shuttle components considered vital to the safety of the orbiter and the crew will be reassessed, as will all waivers of engineering redundancy. Inspection and test requirements will be reviewed, and the booster joints, widely recognized to have been the cause of the shuttle accident in January, will be redesigned under the direction of the Marshall Space Flight Center in Huntsville, Alabama.

In addition, new launch criteria will be established at the outset, Truly said. "When it's time for the first flight, we are going to do it as safely as possible. We are going to launch in the daytime from Kennedy [Space Center in Florida], we're going to have a conservative flight design, [and] we're going to have a repeat payload, one that we have experience with." No civilians will fly during the first year, and all flights will occur in warm weather, he indicated.

Truly explained that the rules are necessary to restore the agency's credibility in the wake of the Challenger disaster (*Science*, 28 March, p.1495). The agency's present plan is to conduct roughly nine flights a year, beginning a year from now. First priority will be given to launching military satellites, as well as a tracking and communications satellite destroyed by the accident. "We cannot print enough money" to make the flights risk-free, Truly added. "But we certainly are going to correct any mistakes that we may have made in the past, and we are going to get going again just as soon as we can." ■

R. JEFFREY SMITH

Panel Sees Decline in Undergraduate Education

A National Science Board committee report says that the nation's undergraduate programs in science, mathematics, and engineering "have declined in quality and scope to such an extent that they are no longer meeting national needs." This poses a "grave, long-term threat to the nation's scientific and technical capacity, its industrial and economic competitiveness, and the strength of its national defense," the panel warns.

On the basis of evidence gathered in its inquiry, the committee pinpointed three areas that require highest priority attention.

■ Laboratory instruction was described as "often uninspired, tedious, and dull." Instrumentation and facilities were found to be obsolete and inadequate—the need for new instruments was put at $2 billion to $4 billion.

■ Faculty members in too many cases were seen as unable to maintain their teaching skills, currency in their disciplines, and command of new technology. Serious shortages of qualified faculty were noted in some disciplines.

■ Courses and curricula were described as "frequently out-of-date in content, unimaginative, poorly organized for students with different interests, and (they) fail to reflect recent advances in the understanding of teaching and learning."

According to the report, institutions of all types in all regions of the country are affected. The problems of engineering disciplines were said to be most serious.

The committee was formed last May to assess the state of undergraduate education in science, mathematics, and engineering and make recommendations on the role the National Science Foundation should take in improving it. Its chairman was Homer A. Neal, provost of the State University of New York at Stony Brook. The committee reported to the National Science Board, which is the policy-making body for the foundation.

In its recommendations, the committee said that NSF lacks the resources to solve the problems itself, but should take a leadership role in stimulating the states and the private sector to increase their investment in undergraduate science, engineering, and math education. The panel does recommend that NSF expenditures in the field be increased by $100 million a year in "leveraged" program support. Some $5.5 million for college instrumentation is the only program in undergraduate education in the NSF budget this year. NSF director Erich Bloch is charged with converting the committee recommendations into proposals to be incorporated in next year's NSF budget. ■

JOHN WALSH

Nuclear Meltdown: A Calculated (and Recalculated) Risk

For years, the nuclear industry has been trying to persuade the government to see a silver lining in the cloud that gathered over Three Mile Island. Broadly, the argument is that the 1979 nuclear accident was much less dangerous than official risk estimates would have led people to expect. Therefore, the risk studies should be rewritten. Eventually, if analysis confirms what the accident at Three Mile Island suggested, safety regulations may be adjusted to reflect a calmer view of what would happen in a meltdown.

An exercise of this kind has begun at the Nuclear Regulatory Commission (NRC), called the "source terms" review (*Science*, 5 April 1985, p. 31). The phrase refers to mathematical terms used to calculate leakage from radioactive sources. This project was inspired by the fact that radiation escaping from Three Mile Island was only a fraction of what might have been expected. Also, radioactive iodine was less volatile during the accident than many had predicted. Rather than venting to the atmosphere in a pure

form, virtually all of it combined with other chemicals and stayed in the plant.

On 26 March, NRC heard a staff report on the work done so far in the source term review. The NRC staffers said they definitely could see a glimmer in the darkness, but they could not be sure whether it was the glint of a silver lining or just another lightning bolt. Despite their uncertainty, they promised to have some new risk estimates ready for publication this fall.

Last year, the NRC released the first draft of a source term document that is meant to serve as the new scientific basis for work in the area. The report, called NUREG-0956, does not deal at all with risks. (These will be calculated in a separate document due in October, designated NUREG-1150.) Instead, the scientific document provides detailed forecasts of how radioactive chemicals might behave in 16 types of accidents and in six types of reactors. When it is complete in July, it will serve as the starting point for the risk analysis.

While the future version of this NUREG report may be sound, the present edition has been greeted with skepticism. The nuclear industry, which has sponsored its own research, calls it outdated and alarmist. The antinuclear groups see it as underplaying hazards. And a number of scientists describe it as simply unripe. In this regard, the file of public comments reveals an inherent problem that may keep the project unripe for a long time. This is a disagreement over the credibility of some computer modeling codes that are the basis for all the predictions that will come out of NUREG-0956.

There are two levels of disagreement. First, some researchers challenge the codes on a mechanical basis. The codes are so complex, tedious to review, and obscure, critics say, that they have been reviewed by almost no one except those paid to do so, that is, by NRC contractors. There may be a hidden bug in these models that no one has detected. Furthermore, it is impossible to "validate" the codes fully, for no one is going to stage nuclear accidents to see how well the numbers represent reality. For this reason, it is important that they be thoroughly vetted by independent scientists. Several commissioners stressed this point during the briefing.

Last year, a committee of the the American Physical Society (APS) reviewed some of this work, issued a report, and then disbanded—long before the game was over, it turns out. These APS members were consulted, according to the NRC staff, about the final version of NUREG-0956. But some of the APS group felt the consultation was perfunctory and fell far short of full peer review.

For example, one member of the APS committee, Fred Finlayson of the Aerospace Corporation, wrote to the NRC in January to explain why he considered the task unfinished. The codes have not been thoroughly peer-reviewed, Finlayson wrote, and their technical assumptions have not been adequately disclosed. He concluded that there were "too many uncertainties to provide a reasonable basis for revised risk analysis at this time." Nothing has changed his opinion since January.

Another, broader problem with the codes is that they distort natural phenomena by simplifying them. (The codes must be simplified to suit the computer.) Thus, knotty problems are sometimes omitted. However, these knotty ones could be important in an accident. For example, one such hard-to-model event is the scenario in which a molten core interacts with a limestone concrete floor to produce volumes of gas, heat, and a radioactive aerosol. In the right circumstances, these fumes could burst through the containment and pose a serious threat to public health.

Indeed, the codes are inadequate to cope with fuel-concrete interactions, one NRC official says, because the technical issues are unresolved. Research on this topic is now in progress in West Germany and at the Sandia National Laboratory in New Mexico. Similar uncertainties plague the issues of containment building integrity, high-pressure ejection of fuel from the reactor vessel, hydrogen production, iodine and lanthanum chemistry, and revaporization of deposited fission products. All are being researched. Citing the code's deficiencies in dealing with chemistry, R. Potter, a British official at the Atomic Energy Establishment at Winfrith, wrote of the treatment of iodine chemistry: "At best this is an oversimplification, and at worst, wrong." Unless this and other aspects were improved, he concluded that it would be "difficult to have the necessary confidence in the results."

The NRC staff, including the acting executive director Victor Stello, assured the commission that corrections and emendations of document NUREG-0956 will be finished by July. Unresolved technical issues, such as the interactions of the fuel with concrete, will be handled by setting wide uncertainty margins around relevant terms in the analysis. Work on the risk estimates themselves has already begun and will be completed within 6 months. Finally, in the bureaucratic tradition, a policy paper issued by Stello also promised that the staff would begin to propose regulatory changes right away, or, in any case, "as soon as the available information warrants such changes." ■

ELIOT MARSHALL

Insurance Drought Fosters Self-Help Plan for Biotechnology Firms

The insurance crisis that is currently affecting a host of industries has not passed up biotechnology. Faced with exorbitant premiums and in many instances the inability to obtain insurance, small biotechnology firms are turning to insuring themselves. The Association of Biotechnology Companies (ABC) plans to set up an offshore insurance venture to provide liability coverage to 20 member companies.

Warren Hyer, managing director of ABC, says that this plan hopefully will solve the member companies' immediate insurance crisis. Furthermore, it also may pave the way for the insurance industry to provide at least limited supplemental underwriting to companies for upgrading general liability coverage, protecting corporate executives and directors as individuals, bringing new products to market, or scaling up experiments for field and clinical trials.

Insurance is hard to get, says Hyer, because the insurance industry "does not know much about biotechnology. The risk right now cannot be identified." But insurers may be more willing to take on biotechnology concerns, he says, after the association's new insurance operation starts functioning. Discussions with two New York–based international brokers—Marsh & McLennan, Inc. and Johnson & Higgins—indicate that coverage on potential liability claims exceeding $1 million might be available from private insurance companies in the future, says Hyer.

ABC's tentative plan calls for each member company to be insured for liability claims up to $1 million. Each company would pay an annual premium of $100,000. The companies will review each other's research portfolios and will establish "a strong risk-prevention program" that sets out general guidelines for the conduct of research. The affiliate of the trade association is likely to be located in the Bahamas or Bermuda, Hyer indicated, to avoid U.S. tax laws that would treat a surplus in the insurance entity's trust funds as a taxable profit.

The insurance crisis extends to biotechnology's larger players, including pharmaceutical and chemical giants. "Everybody is having insurance problems," says Susan Racca, an analyst at the Industrial Biotechnology Association. Member companies of the IBA are scheduled to meet next week to discuss a self-insurance plan. The association shelved the idea several months ago but is taking it up again, says Racca, "because things have gotten so bad." ■ MARK CRAWFORD

P.O. Box 28071
Raleigh, N.C. 27611
January 1, 1986

CONFIDENTIAL

Atomic Safety & Licensing Board (Appeals)
U.S. Nuclear Regulatory Commission
Washington, D.C. 20552

Dear Mr. Kelley,

I have turned over more names of the drug abusers at the Shearon Harris Nuclear Power Plant who have been involved in nuclear safety related work to the Attorney General's Office of the State of North Carolina. Drug abusers are breaking the law and the law is the law even for people working in nuclear power. The U.S.N.R.C was not formed with the intent to act as a shield for an industry with criminal activities overtly present.

I am very shocked to be aware that Larry Robinson of the U.S.N.R.C. Office of investigation has not gotten back to me over the concerns I voiced in September of 1986. These concerns included:

1. Problems with the inservice/preservice inspection of welds at the Shearon Harris Nuclear Power Plant;
2. Weld documentation of both construction and inservice/preservice inspections. These are QA records,
3. Radiation control at the Brunswick Steam Electric Plant,
4. Integrity of supervisory personnel in Health Physics at both the Shearon Harris Nuclear Power Plant and the Brunswick Steam Electric Plant,
5. Overexposure cover-up at the Brunswick Steam Electric Plant.

In this letter I will add to the above list and I hope you

will help me get some answers / get something done about these matters.

As you may or may not know I have offered to submit to a polygraph, PSE, or truth serum concerning the drug information and the information I give to Larry Robinson of the U.S.N.R.C. OI. I also believe that the results of such an exam should be made public to the rate payers so that they see what they are contributing to in the way of corruption.

I will now state once again my knowledge of problems at Carolina Power & Light nuclear plants. This is a partial knowledge, since I can certainly go into further detail and make it more complete.

(1). Improper inservice inspection, ultrasonic inspection, of welds on safety related piping at the Shearon Harris Nuclear Power Plant. A copy of attachment #1 to this letter was provided to Larry Robinson with the above indication of wrong doing. Note please, that attachment #1 describes a weld of 0.172" thickness on a 6" reactor coolant line. This is a serious matter. Two level III NES (Nuclear Energy Services) NDE inspectors argued over these ultrasonic results. They had conflicting opinions. Ultrasonic exams are a back up to radiography. Hopefully, the ultrasonic exam catches what the X-ray exam misses.

Inspection sheet A was turned into me and was an official document which could only be changed through a revision process according to plant procedures. Ron Saunders, the NES lead level III ultrasonic inspector at Shearon Harris decided it didn't look right. He took the sheet back. He then argued with the Level III ultrasonic inspector (NES corporate) about what was on the sheet. When it looked "right" the terminology that is; Ron Saunders returned it to to me as sheet B which is attachment #2.

Mr. Mel Perry ←

Note, that the mention of a weld or repair weld was eliminated from page 4 of the original Mel Perry turned in.

Also removed was the listings of indications in this weld, referring specifically to indications #9 & #12.

This is questionable. This is in violation of the ASME Boiler and Pressure Vessel Code, Section XI, Article IWB-3000, 'Acceptance Standards for Flaw Indications'. This is the law as indicated in the Code of Federal Regulations. A direct quote:
(1980 Edition for the Shearon Harris Preservice Inspection)

IWB-3112 Acceptance

(a) Components whose examination either confirms the absence of or reveals flaw indications that do not exceed the standards of Table IWB-3410-1 shall be acceptable for service, provided the verified flaw indications are recorded in accordance with the requirements of IWA-1400(h) and IWA-6220 in terms of location, size, shape, orientation, and distribution within the component.

IWA-1400(h) Owner's Responsibility

(h) recording of examination and test results that provide a basis for evaluation and facilitate comparison with the results of subsequent examinations;

Please look at this also:

IWA-1400(l) Owner's Responsibility

(l) retention of all inspection, examination, Test, and repair and replacement records for the service lifetime of the component or system

Even though IWA-6000, Records and Reports, is written more with the inservice inspection of an operating plant in mind; it is still emphasized that conditions be (reported) recorded. Indications found during a preservice inspection are conditions that must be truthfully recorded.

Further,

IWA-6200, Preparation, paragraph (b)-8 contains this:

(8) abstract of examinations, tests, replacements or repairs performed; conditions recorded; and corrective measures recommended or taken;

These are records.

Attachments #1 and #2 show that the indications in a large reactor coolant line weld have been removed at least on paper or on the inservice inspection report. The indications are replaced as being beam reduction. I sincerely hope that there are no indications in that weld. If there are; then the owner: CP&L; NES; and the inspector removing them should be prosecuted for violation of both the federal code and the laws of the State of North Carolina which pertain to the inspection of boilers, pressure vessels and components and piping.

Not only was the official revision process ignored in this matter; but Thomas Bromback who was in charge of the Preservice Inspection program was informed by me and shown these same sheets. Mr. Bromach testified before you Mr. Kelley concerning the steam generator inspections at the Shearon Harris plant. Mr. Bromback told me to let it go.

In my opinion, if two level III inspectors were disagreeing a third party should have been called in, a new exam documented, and then an official revision should have been made: All very much in the open and up front.

Of major concern to me is that this sort of thing occurred, to the best of my knowledge in at least 10% of the welds in the inservice inspection program at the Shearon Harris plant. This I indicated to Larry Robinson in September of 1985. I also told him that there was documentation to substantiate what I was saying at the Harris plant.

A file was kept by Ron Saunders of NES of all the inservice inspection records that were changed on the side without a revision as required by procedure to be made. All of this is under CP&L's so-called quality assurance program. The third party authorized nuclear inservice inspector, Jim Stark, raised hell about this revision process and that it shouldn't be circumvented. Jim Stark and I both raised hell and we both felt pressure and intimidation from CP&L, Namely Bromback and Temple. (Phil Temple was a CP&L engineer in Inservice Inspection)

I requested in Sept. 1985 that the U.S.N.R.C. OI obtain the file of the original reports that were replaced by other inservice reports (in the permanent plant record) and compare what was originally written about weld conditions with what was actually in the permanent plant QA records. This was never done.

I want to know Why, Mr. Kelley? I am an engineer with credentials, integrity, and evidence. Why is this being ignored? Are the originals in that file going to disappear? All I am asking is that the file be thoroughly examined. How much time does it take to look at a 100 sheets? Not much.

The USNRC is supposed to be looking at safety concerns, fairly. I have offered to submit to a polygraph, PSE or truth serum. I felt sexist discrimination by CP&L during the hearing with that revengeful lover crap. Would a man have been called a revengeful lover? I'll take a polygraph concerning the engineering facts and the law. Will anyone from NES, CP&L or CONAM do that? See Attachment # 3

The second allegation or second set of allegations I presented to Mr. Robinson was:

2./ Overt negligence in health physics practices at the Brunswick nuclear plant with jeopardized the safety of employees. Intimidation in the way of watching my movements and contacts at the plant. Falsification of health physics records.

I also discussed with Mr. Robinson this matter:

3./ That I was asked to lie about my dose, by Health physics supervision, at Shearon Harris on the NRC form 4 which is concerned with Occupational Radiation Exposure. The nuclear engineering department at the North Carolina State University had lost my exposure records. After requesting an estimate several times and having gotten no where; my supervision stated that I drop the matter and state (lie) on the form that I had no exposure at the university. I called Mr. Prewatte the U.S.N.R.C inspector on site at Shearon Harris and asked him what I should do with this ridiculous order by management

I also dicussed with Mr. Robinson this matter:

4./ That I was told to handle U.S.N.R.C inspectors in the Preservice Inspection the right way by primarily Tom Brombach. That is to make sure that everything was or looked right, in so many words. During one visit by the U.S.N.R.C to Tom Brombach; the inspector and Tom Brombach were in a discussion with the office door open. The authorized nuclear inservice inspector and I (Mr. Stark was the ANII) were discussing CP&L's policy of stating when pre-service inspection of a weld was complete. Mr. Stark, I believe, was stating that if construction documentation indicated that a weld had been re-surfaced after the pre-service inspection that it would require another preservice inspection and that both construction and pre-service inspection documentation must agree. Sometimes these documents didn't agree

The U.S.N.R.C. inspector, I was later told by Tom Brombach; had been listening to the ANII and I while Tom said

that he was trying to convince the U.S.N.R.C. inspector that there were no problems. Phil Temple, second in command to Tom Brombach, told me afterward that I had pissed Tom Brombach off as did the ANII by the discussion in front of the U.S.NRC. Mr. Temple said that if it happened again that I would probably be fired. Tom Brombach afterward told both me and Jim Stark to keep our discussions to ourselves when the U.S.N.R.C was around.

5./ I also told Larry Robinson that Brombach said one morning around 7:30AM that an NRC inspector was coming and that everything was or was to look right before he arrived in an hour or so. I questioned Brombach as to how he knew about the USNRC visit. It sounded as though it might have been an unannounced visit and that Tom Brombach might have been tipped off. Tom Brombach went further and said that he knew the head man in Atlanta and that he had been called by him. I want to know if that was an unannounced visit. A simple Yes or No will do. There is no inspection report in the Wake County Public Library (for the last couple of years the reports aren't there) I would appreciate knowing if the visit was unannounced and a copy of the report. I gave Mr Robinson the approximate time of the visit and I described the inspector. Mr. Robinson recognized who I was discussing. How much effort is needed by the USNRC to clear the air on this concern I brought into the open? There were only a few U.S.N.R.C visits to the Preservice Inspection at Shearon Harris while I was in Preservice Inspection at Shearon Harris.

To the best of my knowledge, I have presented you with the truth. During the Oct. ASLB hearing both you, Mr. Kelley, and I were aware of CP&L beating a dead horse over and over again. I am referring to the statement that I have a Master's degree. Well I am still saying that. It's accepted practice even according to the U.S. O.P.M. to do so. The OPM will give an individual credit for having a Masters degree completed if it is within 6 months of completion. That is directly printed on the attached documentation to the federal register application to help clarify the basis for different GS levels. Besides it being accepted practice; I did allow my resume to become part of the USNRC ASLB hearing record. It states in black and white on the resume that the thesis was in review.

Since CP&L was so concerned with what I said, There is a copy of Mr Bromback's testimony on attachment #4 which includes problem areas with Mr Bromback's testimony. Throughout the testimony there are notations indicating problems with the truthfulness or lack of truthfulness of Tom Bromback

I have been informed by an executive of a nuclear firm concerned with Unit 1 at Three Mile Island in the last month that Unit 1 has steam generator problems. Mr. Kelley; Conam did all of Unit 1's steam generator eddy current inspections. Are there leaks in areas of tubes which CONAM has just inspected in late 1984 and early 1985? Was anything missed because people were high? I'm curious, aren't you?

I also believe that the ASLB judges may have missed something which occurred during my testimony and questioning by the CP&L attorney: Thomas Baxter. I believe Mr. Baxter was covering up and I allege he was doing so which is in conflict with his being an officer of the court. It can also be considered an obstruction of justice, since this was a hearing before representatives of the United States and the State of N. Carolina being a participant.

To explain more fully to the best of my recollection; Mr. Baxter initiated a line of questing concerning what I saw at lunchtime in the Daniel or construction parking lot (over lunchtime, I guess from somewhere in 11:30 to about 12:30) Mr. Baxter stated that Daniel construction workers couldn't go out to the parking lot at lunch time. I replied or asked him, "then who did I see?" He immediately afterward changed the subject and I believe he started to question me about something else. At that point I realized that he didn't want anymore to come out. He knew Daniel construction workers who were CRAFT couldn't be there in the lot at lunchtime, I allege. But, contractor engineers like myself could go out to the lot as could CP&L personnel who also parked in the construction lot. I think I am correct in stating that Daniel construction engineers, Westinghouse, Epasco (if present), T&B and other contractor and supervisory personnel were permitted to go to their cars at lunch. I believe the questioning line was changed to keep that out of the record. It was probably engineers and supervisory personnel out in the lot getting high at lunch and not the Daniel Craft.

Since I have received no results from the U.S.N.R.C. OI in Atlanta; I request that you seek to have the following matter investigated.

6.) During the Sept. 1985 ASLB (appeals) hearing and in your presence I was questioned by Thomas Baxter the CP&L legal representative. He asked me about attending graduate school in nuclear engineering at the North Carolina State University (NCSU). I did let it slip that I would be working on a thesis project with Dr. Murty. That is in the hearing record.

In October of 1985, directly after the ASLB hearing on drug abuse; I went to see Dr. Murty who is the graduate student admissions administrator in nuclear engineering at NSCU. I asked when I would be given admission once again in writing. He said that he didn't know if I would be re-admitted. He said that CP&L called him after/during the hearing and cancelled a research contract with him. (It was I believe Dr. Bullard from CP&L he said) Dr. Murty also stated that Dr. Turinsky had concocted some slander that I had cursed him personally over a bad reference that he had given about me.

This was incredibly embarassing and unprofessional. Immediately, I went next door to Dr. Turinsky's office and I asked him what was going on. I brought up the subject of my testimony at the ASLB hearing the previous week. He said that it would come out in time that I was lying according to what he had learned. I said, I had offered to take a polygraph. He then, to the best of my knowledge, went on to say that nobody in the industry would hire me. He said that I shouldn't come back into graduate school in nuclear engineering and that he would not work with me on a thesis topic and that he doubted that any other professors would have anything to do with me. I said then that N. Carolina has a blacklist law and that the honest people should remain in the industry.

In 1983 Dr. Turinsky gave a negative reference to Knolls Atomic Power Lab concerning me. Prior to that reference I had no conversation with Dr. Turinsky (the NCSU nuclear engineering department head) except for the welcome for a new student. Since I was informed by KAPL as to the content of the reference, (this is documented) I requested that Dr. Turinsky not give anymore negative references out based on hearsay. I stated that if he couldn't agree to that he had better cease with that nonsense since it cost me an opportunity to work for the United States and $30,000 year salary. Dr. Turinsky was never cursed. I allege that he decided to concoct this slander after conferring with CP&L in retaliation for my testimony at the ASLB. We had agreed the matter would be dropped in 1983. Most conveniently for CP&L it should reappear and disrupt my educational goals. I believe it is retaliation by CP&L, your licensee, in collusion with the NCSU nuclear engineering department another USNRC licensee to harass a witness at a federal hearing from giving further testimony and from backing up what has been given. It was also convenient for CP&L to drop Dr. Murty's research contract.

I hope that the U.S.N.R.C will look into this matter and get it stopped. I can't return to graduate school and take this kind of treatment. Oh, yes; Dr. Turinsky also stated that I was emotionally disturbed. My only reply is that I am not emotionally disturbed to the extent that I must abuse drugs at a nuclear facility, ignor the abuse as CP&L does in its worthless drug abuse measures, or to condone the abuse at a nuclear plant.

Attachment #5 plainly shows that harassment/intimidation doesn't seem to effect me. I just ask more questions. Isn't it cheaper for the U.S.N.R.C to clean up the nuclear business with a strict policy of regulation rather than what is currently going on?

Sincerely
Patty Murillo

Attachment 1
Sheet A

IST 501-0394 CAB 1-22-85
~~IST 501-0396~~

Plant/Unit SHNPP #1
Comp/System Reactor Coolant
ISO 1-151-RC-23 Loop N/A

CALIBRATION DATA SHEET

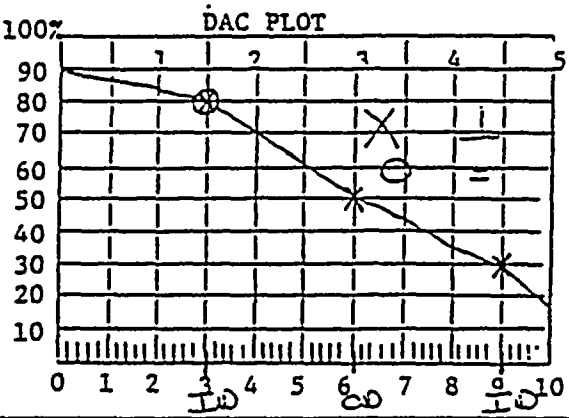
Page 1 of 5
Data Sheet No. IST 501-0394
Procedure No. IST-501
Subject: Piping Welds
Rev/Change No. 0 01
Calibration
Block No. UT-20-1
Fabrication No. N/A
Surface O.D.
Block Temp S/W J84-08 59 °F
Comp. Temp S/W J84-08 70 °F
Thickness .772"
CRT Calibrated in Inches of Metal Path
Each Maj. Screen Div= .357"
Per procedure paragraph 7.2.4.

INSTRUMENT SETTINGS	
Mfg/Model No.:	KB USL-38
Serial No. :	211596
Sweep Length :	7.80
Sweep Delay :	7.80
Pulse Length/Damping:	Fixed
Freq.: 2.5 Rep. Rate:	N/A
Filter: N/A Video:	N/A Jack: R
DEC/Gate Switch:	Range 2.5
Mode Select:	SFD Reject: Min
Gain (coarse) 40/40 (fine)	1/= 8/18
Scan Sensitivity	58/68 1/=

SEARCH UNIT	
Scan Angle: 45°	Mode: Shear
Fixturing (if any):	Lucite Wedge
Style or Type No. :	Gamma
Size & Shape:	.25" Round
Frequency :	2.25 MHZ
Serial No/Brand:	Aerotech IA 11343
Measured Angle	45° material 43°
Cable Type & Length:	RG-58 (1X6')
Couplant Brand:	Ultragel II
Couplant Batch:	# 8336

INSTR. LINEARITY CAL.					
Amplitude					
	High	Low		High	Low
1	100	51	5	60	30
2	90	45	6	50	25
3	80	40	7	40	20
4	70	35	8	30	14

DAC PLOT



SCAN AREA	
0° WRV	N/A
0° Metal	N/A
= To Weld	X
⊥ To Weld	X
Calibration	
Axial 48 dB	X
Circ 58 dB	X

AMPL. CONTROL LINEARITY		
Initial	dB	Result
80	-6	39
80	-12	19
40	+6	82
20	+12	82

CALIBRATION CHECKS	TIME
Initial Cal.	0820
Intermediate	—
Intermediate	—
Intermediate	—
Final Cal.	1300

EXAMINATION WELD/AREA	Recordable Indications Yes.	No	Geom	COMMENTS/REASON FOR INCOMPLETED SCAN (S)
1-RC-SW-A3	N/A	N/A	X	Complete Scan from Both Sides, Pipe to Reducer

EXAMINERS 1 [signature] Date 1-19-85 Level III
2 Philip M. Haines Date 1-19-85 Level I

REVIEWERS 1 _____ Date _____
2 _____ Date _____
3 _____ Date _____

ADDITIONAL SHEETS?		CHECK BOX	
Continuation	N/A	Beam Plot	N/A
Supplements	X	None	N/A

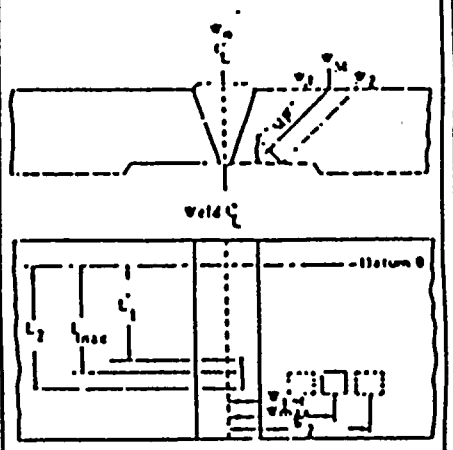
NES NUCLEAR ENERGY SERVICES, INC.

NUCLEAR ENERGY SERVICES, INC.

Project No. 5565	Site SHNPP #1	L_o Location Datum 0	Date: (Mo/Day/Yr) 11/19/95
Item Identification WELD: 1-RC-SW-A1 SIB: 1-151-RC-23		W_o Location Weld 8	Page 2 of 5
Examiner: TC-1A Level III		Angle Used: 0° / 45° ⊥ / 45° ∥ / N/A	Attached Cal. Data Sheet EST 501-0394
Examiner: TC-1A Level I		Scanning Dir: N/A / 50% / N/A / N/A	Thickness .777"
			Diameter (nom.) 6"

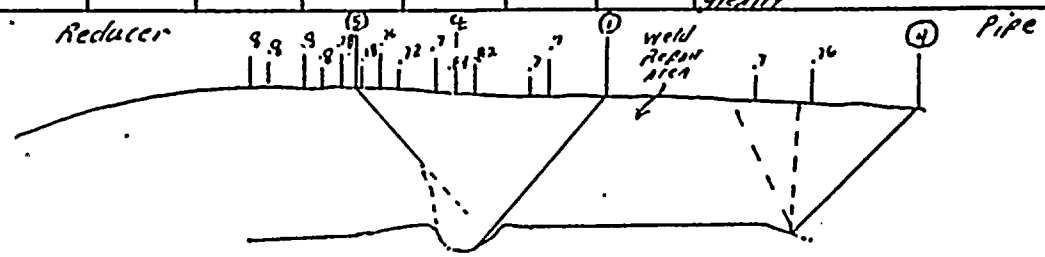
MP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)



Ind. No.	% of DAC	FWD W	FWD MAX MP	FWD 50% W_1	FWD MAX MP	BACKWARD 50% W_2	BACKWARD MAX MP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
2	30	0.8"	1.04"	0.65"	0.98"	0.95"	1.14"	*	0.5cw	*	N/A	Pipe	Seen at same time as #3, both 360° intermittent at varying amplitude
4	15	2.45"	1.57"	2.25"	1.43"	2.60"	1.68"	*	0.5cw	*	N/A	Pipe	Seen intermittently for 360° at varying amplitudes
5	10	0.53"	0.87"	0.1"	0.19"	0.64"	0.97"	*	Datum 0	*	N/A	Reducer	Indications 5, 6 and 7 seen at same time for 360° intermittently but their amplitude relationship varies greatly

DRAW FULL SCALE PLOT HERE:



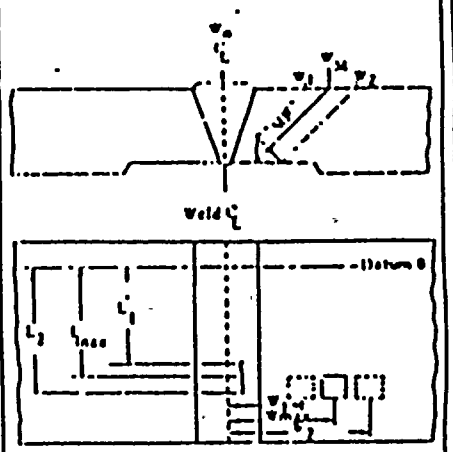
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

—— Sound Path
- - - Possible sound paths when there exist more than one option.

EST-501-0394

NUCLEAR ENERGY SERVICES, INC.

Project No. 5565	Site SHNPP #2	L_o Location Datum Ø	Date: (Mo/Day/Yr) 1/19/85
Item Identification Weld: 1-RC-SW-A3 ESB: 1-151-RC-23		W_o Location weld ℄	Page 3 of 5
Examiner: TC-1A Level III		Angle Used: 0° / 45° ⊥ / 45° ∥ / N/A	Attached Cal. Data Sheet IST SDI-0394
Examiner: TC-1A Level I		Scanning dB: N/A / 55dB / N/A / N/A	Thickness .772"
			Diameter (nom.) 6"

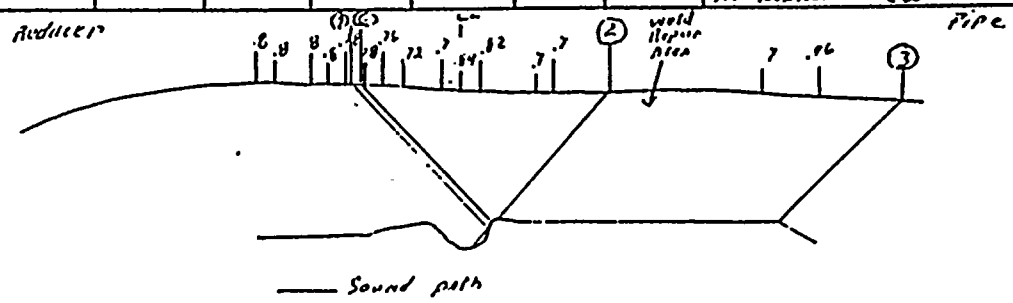


MP Metal Path
RBR Remaining Back Reflection
L. Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)

Ind. No.	% of DAC	W MAX W	W MAX MP	FWD 50% W_1	FWD MAX MP	BACKWARD 50% W_2	BACKWARD MAX MP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
2	15	0.8"	0.86"	0.60"	0.72"	0.95"	0.93"	*	0.5" cw	*	N/A	Pipe	Seen at same line as #3, both 360° intermittently at varying amplitude.
3	20	2.35"	0.86"	2.15"	0.72"	2.45"	0.93"	*	0.5" cw	*	N/A	Pipe	Seen intermittently for 360° at varying amplitudes.
6	20	0.53"	1.00"	0.4"	0.89"	0.7"	1.18"	*	Datum Ø	*	N/A	Reducer	Indications 5, 6 & 7 seen at same line for 360°. Their amplitude relationships vary. Indication 7 exceeds 40% DAC at location 1" cw
7	20	0.58"	1.22"	0.45"	1.44"	0.72"	1.29"	*	Datum Ø	*	N/A	Reducer	

DRAW FULL SCALE PLOT HERE:



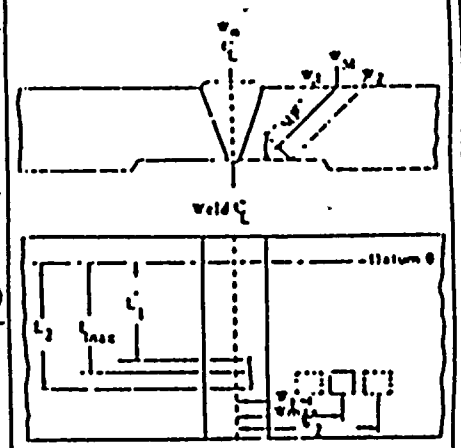
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

--- Possible sound paths when there exist more than one option

IST SDI-0394

NUCLEAR ENERGY SERVICES, INC.

pp #2	L_o Location: Datum Ø	Date: (Mo/Day/Yr) 11/19/85	
1-RC-501-A3	W_o Location: Weld Ø	Page 4 of 5	
151-RC-23	Angle Head: 0° / 45° L / 45° = / N/A	Attached Cal. Data Sheet TST 501-0394	
el III	Scanning dB: N/A / 52dB / N/A / N/A	Thickness .777"	
evel I		Diameter (nom.) 6"	

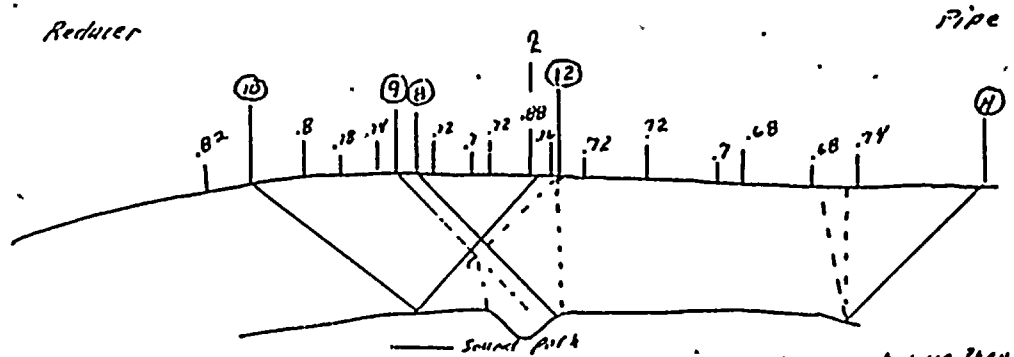


MP Met...
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)

Ind. No.	% of DAC	W MAX W	W MAX MP	FWD 50% W_1	FWD MAX MP	BACKWARD 50% W_2	BACKWARD MAX MP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
8	35	0.6	1.14	0.55	1.07	0.75	1.22	*	4"ccw	*	N/A	Reducer	All indications seen 360°
9	16	0.7	0.97	0.60	0.93	0.80	1.00	*	4"ccw	*	N/A	Reducer	interpretation at varying amplitudes
10	20	1.5"	2.11"	1.35"	2.04"	1.60"	2.22"	*	4"ccw	*	N/A	Reducer	weld ≈ 10 is CO geometry, deepens on ℄
11	20	2.4"	1.65"	2.25"	1.57"	2.60"	1.72"	*	4"ccw	*	N/A	Pipe	#11 could be CO or bottom of weld
12	20	0.15"	0.70"	0.05"	0.68"	0.25"	0.72"	*	4"ccw	*	N/A	Pipe	repair ~~weld~~ m Peay 11/19/85
													#9 + #12 could be either small
													indications or geometry prior been
													reflection.

DRAW FULL SCALE PLOT HERE:



REVIEWER _____ DATE _____
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

TST 501-0394

NUCLEAR ENERGY SERVICES, INC.

Project No.	Site	L_0 Location	Date: (Mo/Day/Yr)
5565	SHNPP #1	Column Ø	11/8/95

Item Identification: WELD: 1-RC-SG1-A3

U_0 Location: weld Center Line

Page 5 of 5

Attached Cal. Data Sheet TST-501-0394

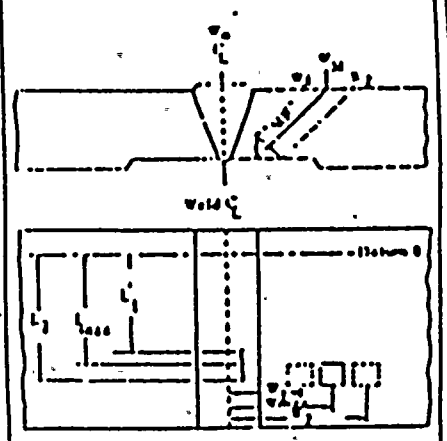
Thickness .272"

Diameter (nom.) 6"

Examiner: TC-IA Level III

Examiner: TC-IA Level II

Angle Beam	0°	45° L.	45° =	N/A
Scanning	N/A	N/A	0.8.18	N/A

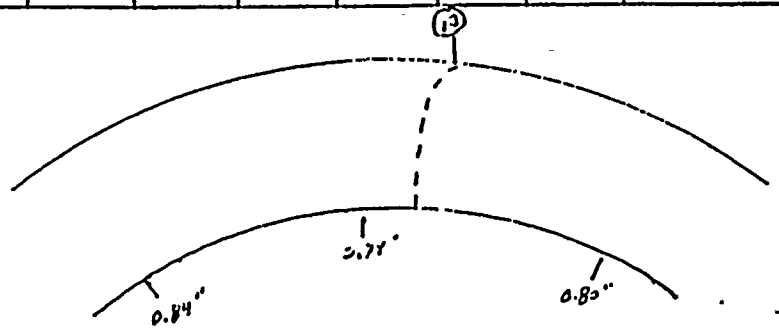


MP Metal Path
RBR Remaining Back Reflection
L. Distance from Datum 0

W max Distance from C_L to S.U. at maximum response.
W_1 Distance from weld C at 50% of Max amp. (fwd)
W_2 Distance from weld C at 50% of Max amp. (backward)

Ind. No.	% of DAC	W	MAX W	MAX MP	FWD 50% W_1	FWD MAX MP	BACKWARD 50% W_2	BACKWARD MAX MP	L_1 50% MAX	L. max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
13	50	℄	0.86"	℄	0.68"	℄	0.88"	*	6" ccw	*	N/A	℄	Beam Reflection; seen intermittently for 360° in both directions	

DRAW FULL SCALE PLOT HERE:



REVIEWER Ronald Saunders DATE 1-26-95

REVIEWER _____ DATE _____

REVIEWER _____ DATE _____

RM 1:23-85

TST-501-0394

Attachment 2
Sheet B

IST 501-0394 CAS 1-22-85
~~IST 501-0394~~

Plant/Unit SHUPP #1
Comp/System Reactor Coolant
ISO 1-151-RC-23 Loop N/A

CALIBRATION DATA SHEET

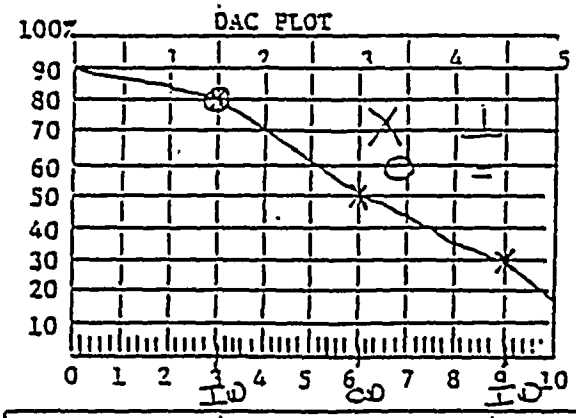
Page 1 of 5 0394 JP 1-26-85
Data Sheet No. IST 501-0394
Procedure No. IST-501
Subject: Piping Welds
Rev/Change No. 01
Calibration
Block No. UT-20-1
Fabrication No. N/A
Surface O.D.
Block Temp S/N J84-03 59 °F
Comp. Temp S/N J84-03 70 °F
Thickness .772"
CRT Calibrated in Inches of Metal Path
Each Maj. Screen Div = .357"
Calibration is per procedure, paragraph 22.(4)

INSTRUMENT SETTINGS	
Mfg/Model No.:	KB USL-38
Serial No. :	211596
Sweep Length :	7.80
Sweep Delay :	7.80
Pulse Length/Damping	Fixed
Freq.: 2.5 Rep. Rate:	N/A
Filter: N/A Video:	N/A Jack: R
DEC/Gain Switch	Range 2.5
Mode Select:	SEO Reject: Min
Gain (coarse)	40/40 (fine) 1/= 8/8
Scan Sensitivity	58/68 1/=

SEARCH UNIT	
Scan Angle: 45°	Mode: Shear
Fixturing (if any): Lucite	Wedge
Style or Type No. :	Gamma
Size & Shape:	.25" Round
Frequency	2.25 MHZ
Serial No/Brand: MeroTech	A11343
Measured Angle	45° material 43°
Cable Type & Length	RG-58 to MOT 1X6'
Couplant Brand	Ultragel II
Couplant Batch:	F 8336

INSTR. LINEARITY CAL.					
Amplitude					
	High	Low		High	Low
1	100	51	5	60	30
2	90	45	6	50	25
3	80	40	7	40	20
4	70	35	8	30	14

DAC PLOT



SCAN AREA	
0° WRV	N/A
0° Vertical	N/A
= To Weld	X
⊥ To Weld	X
Calibration	
Amp 43 dB	X
Size 58 dB	X

AMPLITUDE CONTROL LINEARITY		
Initial	dB	Result
80	-6	39
80	-12	19
40	+6	82
20	+12	82

CALIBRATION CHECKS	TIME
Initial Cal.	0820
Intermediate	—
Intermediate	—
Intermediate	—
Final Cal.	1300

EXAMINATION WELD/AREA	Recordable Indications Yes	No	Geom	COMMENTS/REASON FOR INCOMPLETED SCAN (S)
1-RC-SW-A3	N/A	N/A	X	Complete Scan from Both Sides, Pipe to Reducer

ADDITIONAL SHEETS?	CHECK BOX		
Continuation	N/A	Beam Plot	N/A
Supplements	X	None	N/A

EXAMINERS 1 [signature] Date 1-19-85 Level II
2 [signature] Date 1-19-85 Level I
REVIEWERS 1 [signature] Ronald Saunders Date 1-26-85
2 _____ Date _____
3 _____ Date _____

PSM 1-23-85

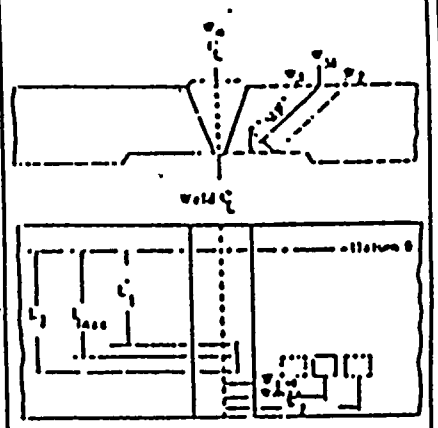
NES NUCLEAR ENERGY SERVICES, INC.

NUCLEAR ENERGY SERVICES, INC.

Project No. 5565	Site SUNPP #1	L_0 Location Datum Ø	Date: (Mo/Day/Yr) 11/14/85
Item Identification 1-151-RC-23		U_0 Location 42/0 ℄	Page 2 of 5
Examiner: TC-1A Level II		Angle Head: 0° / 45° L / 45° = / N/A	Attached Cal. Data Sheet SSF-501-0394
Examiner: TC-1A Level I Help in Learn		Scanning dB: N/A / N/A / 56:1B / N/A / N/A	Thickness .222"
			Diameter (nom.) 6"

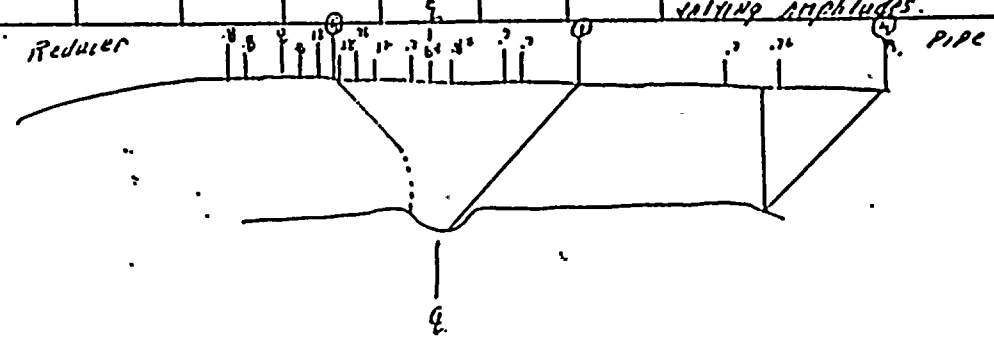
HP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

U max Distance from ℄ to S.U. at maximum response.
U_1 Distance from weld ℄ at 50% of Max amp. (fwd)
U_2 Distance from weld ℄ at 50% of Max amp. (backward)



Ind. No.	% of DAC	FWD U MAX	FWD U MAX HP	FWD 50% U_1	FWD 50% U_1 HP	BACKWARD 50% U_2	BACKWARD 50% U_2 HP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
1	30	0.8"	1.04"	0.65"	0.98"	0.95"	1.14"	*	0.5"W	*	N/A	Pipe	FD Geometric seen at same time & number.
2,4	15	2.45"	1.57"	2.25"	1.43"	2.60"	1.85"	*	0.5"W	*	N/A	Pipe	OD Geometric dependent on OD surface conditions 3,6 and 7 seen at same time but amplitude relationship varies
5	10	0.53"	0.89"	0.4"	0.79"	0.64"	0.97"	*	Datum Ø	*	N/A	Reducer	* seen 360° intermittently at varying amplitudes.

DRAW FULL SCALE PLOT HERE:



REVIEWER Ronald Saunders DATE 1-26-85
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

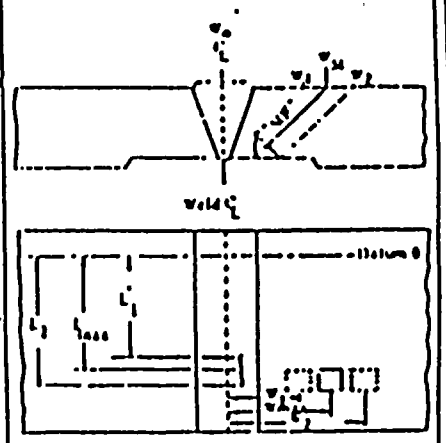
SSF-501-0394

NUCLEAR ENERGY SERVICES, INC.

Project No. 5565	Site SHNPP #1	L_0 Location Datum Ø	Date (Mo/Day/Yr) 1/19/85
Item Identification Line: 1-151-12C-23 Weld: 1-RC-SW-A3		W_0 Location Weld ℄	Page 3 of 5
Examiner: TC-1A Level II		Angle Beam: 0° / 45° L / 45° S / N/A	Attached Cal. Data Sheet TST 501-0394
Examiner: TC-1A Level I		Scanning Dir: K/A / SS/B / A/A / N/A	Thickness .772"
			Diameter (nom.) 6"

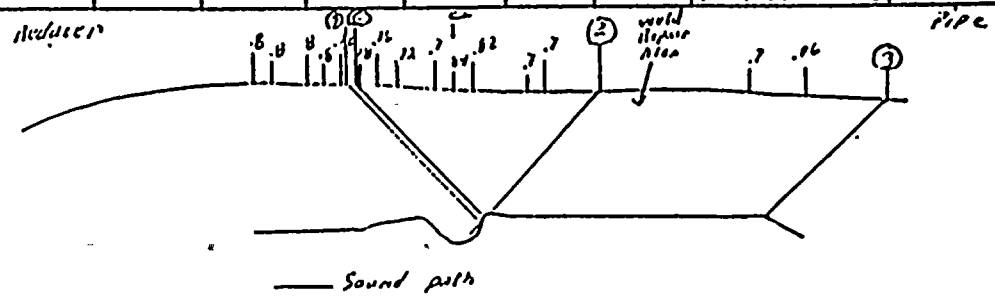
MP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)



Ind. No.	% of DAC	FWD W	FWD MAX W	FWD MAX MP	FWD 50% W_1	FWD MAX MP	BACKWARD 50% W_2	BACKWARD MAX MP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
2	15	0.9"	0.86"	0.60"	0.72"	0.95"	0.73"	*	0.5"cw	*	N/A	Pipe		Seen at same line as #7, both 360° intermittently at varying amplitude
3	20	1.35"	0.86"	1.15"	0.72"	1.45"	0.93"	*	0.5"cw	*	N/A	Pipe		Seen intermittently for 360° at varying amplitudes
6	20	0.53"	1.05"	0.4"	0.97"	0.7"	1.18"	*	Datum Ø	*	N/A	Reducer		Indications 5, 6 & 7 seen at same time for 360° Their amplitude relationships
7	20	0.58"	1.22"	0.45"	1.44"	0.72"	1.29"	*	Datum Ø	*	N/A	Reducer		will indication 7 exceeds 40% DAC at location 1" cw

DRAW FULL SCALE PLOT HERE:



--- Possible sound paths when there exist more than one option

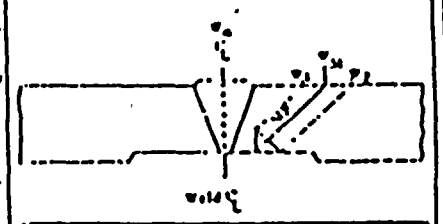
REVIEWER Ronald Saunders DATE 1-26-85
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

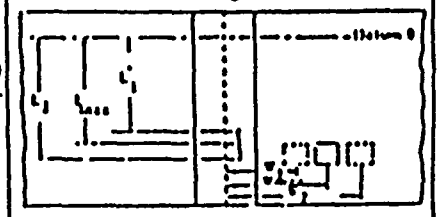
BM 1-23-85

TST 501-0394

NUCLEAR ENERGY SERVICES, INC.

TNPP #1	L_0 Location Pulsar Ø	Date: (Mo/Day/Yr) 1/17/85
0 1-151-RC-23 III	W_0 Location weld Ø	Page 4 of 5
Level	Angle Head: .0° / 45°L / 45°= / N/A	Attached Cal. Data Sheet SSI-601-017
A Level I	Scanning dB: N/A / 58 / N/A / N/A	Thickness .222"
2 M Alumel		Diameter (nom.) 6"



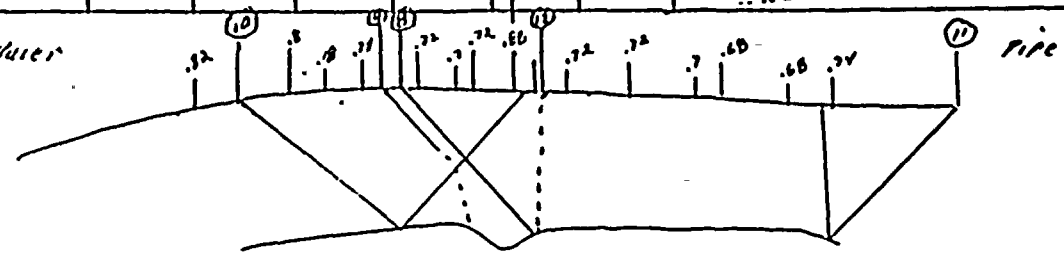


In. ... ath
RBR ... ing Back Reflection
L Distance from Datum 0

W max Distance from C_L to S.U. at maximum response.
W_1 Distance from weld C_L at 50% of Max amp. (fwd)
W_2 Distance from weld C_L at 50% of Max amp. (backward)

Ind. No.	% of DAC	W MAX W	W MAX HP	FWD 50% MAX W_1	FWD MAX HP	BACKWARD 50% MAX W_2	BACKWARD MAX HP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
8	35	0.6	1.14	0.55	1.07	0.75	1.22	*	4" CCW	*	N/A	Reducer	ID geometry
9	16	0.7	0.46	0.60	0.93	0.50	1.00	*	4" CCW	*	N/A	Reducer	Beam Reduction
10	20	1.5	2.11	1.35	2.04	1.60	2.22	*	4" CCW	*	N/A	Reducer	OD geometry - diagons on OD
11	20	2.4	1.65	2.25	1.57	2.60	1.72	*	4" CCW	*	N/A	Pipe	OD geometry - diagons on OD
12	20	0.15	0.70	0.05	0.68	0.25	0.72	*	4" CCW	*	N/A	Pipe	Beam Reduction
													* see 360° interpretation for varying amplitudes

DRAW FULL SCALE PLOT HERE:



REVIEWER Ronald Saunders DATE 1-26-85
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

SSI-501-039

NUCLEAR ENERGY SERVICES, INC.

Ject No.	Site	L_0 Location	Date: (Mo/Day/Yr)
5565	SHNPP #1	Column Ø	11/14/85

Item Identification: Weld: 1-RC-501-A3

W_0 Location: weld/center line

Page 5 of 5

Examiner: TC-IA Level III

Examiner: TC-IA Level II

Angle Head	0°	45° L	45° =	N/A
Scanning Jig	N/A	N/A	6.8.18	N/A

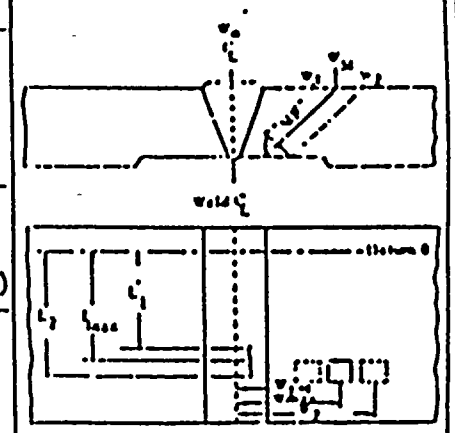
Attached Cal. Data Sheet TST-501-0394

Thickness .272"

Diameter (nom.) 6"

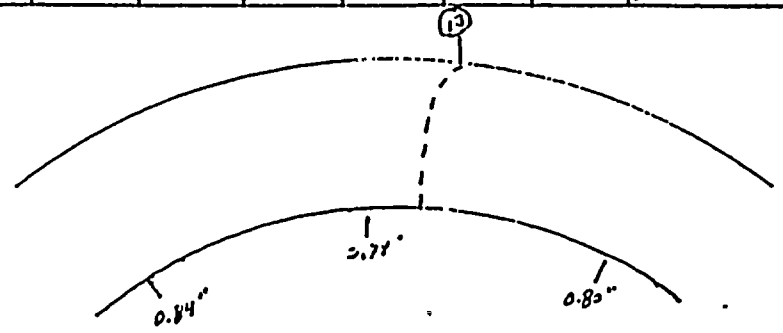
HP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)



Ind. No.	% of DAC	W	MAX HP	FWD 50% W_1	FWD MAX HP	BACKWARD 50% W_2	BACKWARD MAX HP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
13	50	℄	0.86"	℄	0.68"	℄	0.88"	*	6" ccw	*	N/A	℄	Back Reflection; seen intermittently for 360° in both directions

DRAW FULL SCALE PLOT HERE:



REVIEWER Ronald Saunders DATE 1-26-85

REVIEWER _____ DATE _____

REVIEWER _____ DATE _____

BM 1:2385

TST-501-0394

IST 501-0394 CAB 1-22-85
~~IST 501-0394~~

Plant/Unit SHUPP #1
Comp/System Reactor Coolant
ISO 1-151-RC-23 Loop N/A

CALIBRATION DATA SHEET

Page 1 of 5
Data Sheet No. IST 501-0394
Procedure No. IST-501
Subject: Piping Welds
Rev/Change No. 0 01
Calibration
Block No. UT-20-1
Fabrication No. N/A
Surface O.D.
Block Temp S/N J84-08 59 °F
Comp. Temp S/N J84-08 70 °F
Thickness .772"
CRT Calibrated in Inches of Metal Path
Each Maj. Screen Div = .357"
Per procedure paragraph 7.2.4.

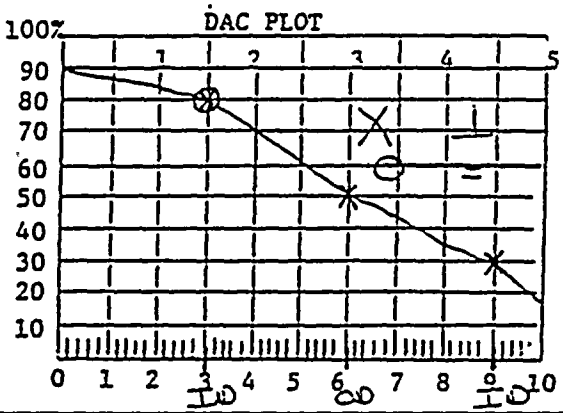
INSTRUMENT SETTINGS	
Mfg/Model No.:	KB USL-38
Serial No. :	211596
Sweep Length :	7.80
Sweep Delay :	7.80
Pulse Length/Damping:	Fixed
Freq.: 2.5 Rep. Rate:	N/A
Filter: N/A Video: N/A Jack:	R
DEC/Gate Switch: Off Range:	2.5
Mode Select: SFO Reject:	Min
Gain (coarse) 46/46 (fine) 1/5 .	8/18
Scan Sensitivity:	58/68 1/=

SEARCH UNIT	
Scan Angle: 45° Mode:	Shear
Fixturing (if Lucite any):	Wedge
Style or Type No. :	Gamma
Size & Shape:	.25" Round
Frequency :	2.25 MHZ
Serial No/Brand: AerroTech	A11343
Measured Angle	45° material 43°
Cable Type & Length:	RG 1X6' to MIDT
Couplant Brand:	Ultragel II
Couplant Batch:	# 8336

SCAN AREA	
0° WRV	N/A
0° Mat'l	N/A
= To Weld	X
⊥ To Weld	X
Calibration	
Axial 48 dB	X
Circ 58 dB	X

INSTR. LINEARITY CAL.					
Amplitude					
	High	Low		High	Low
1	100	51	5	60	30
2	90	45	6	50	25
3	80	40	7	40	20
4	70	35	8	30	14

DAC PLOT



AMPL. CONTROL LINEARITY		
Initial	dB	Result
80	-6	39
80	-12	19
40	+6	82
20	+12	82

EXAMINATION WELD/AREA	Recordable Indications Yes	No	Geom	COMMENTS/REASON FOR INCOMPLETED SCAN (S)
1-RC-SW-A3	N/A	N/A	X	Complete Scan from Both Sides, Pipe to Reducer

CALIBRATION CHECKS	TIME
Initial Cal.	0820
Intermediate	—
Intermediate	—
Intermediate	—
Final Cal.	1300

EXAMINERS 1 [signature] Date 1-19-85 Level III
2 [signature] Date 1-19-85 Level I

REVIEWERS 1 _____ Date _____
2 _____ Date _____
3 _____ Date _____

ADDITIONAL SHEETS? CHECK BOX			
Continuation	N/A	Beam Plot	N/A
Supplements	X	None	N/A

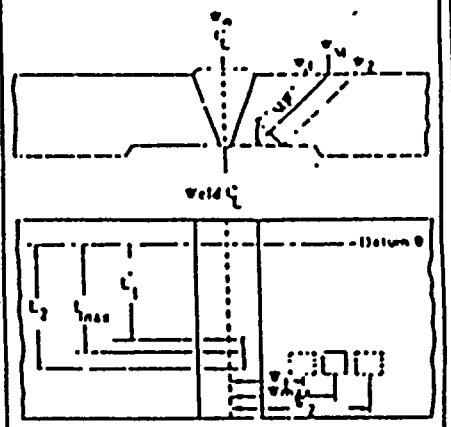
NES NUCLEAR ENERGY SERVICES, INC.

NUCLEAR ENERGY SERVICES, INC.

Project No. 5565	Site SHNPP #1	L_o Location Datum 0	Date: (Mo/Day/Yr) 11/8/95
Item Identification WEL#: 1-RC-SW-A1 CIP: 1-151-RX-23		W_o Location Weld 9	Page 2 of 5
Examiner: TC-1A Level II		Angle Used: 0° / 45° ⊥ / 45° = / N/A	Attached Cal. Data Sheet FST 501-0394
Examiner: TC-1A Level II		Scanning dB: N/A / 50dB / N/A / N/A	Thickness .722"
			Diameter (nom.) 6"

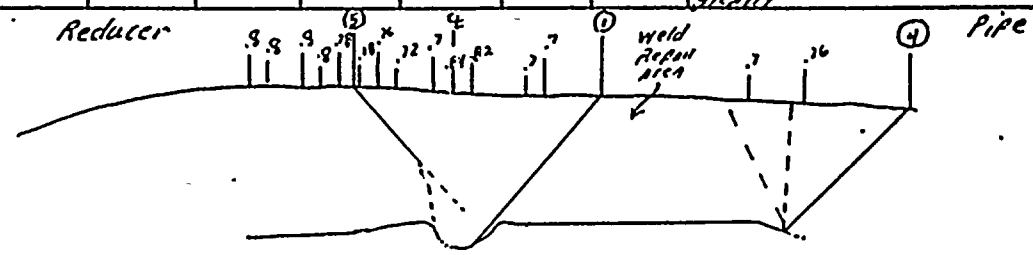
MP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)



Ind. No.	% of DAC	FWD W	FWD MAX MP	FWD 50% W_1	FWD MAX MP	BACKWARD 50% W_2	BACKWARD MAX MP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
2	30	0.8"	1.04"	0.65"	0.98"	0.95"	1.14"	*	0.5cw	*	N/A	Pipe	seen at same time as #2 both 360° intermittent at varying amplitude
4	15	2.45"	1.57"	2.25"	1.43"	2.60"	1.68"	*	0.5cw	*	N/A	Pipe	seen intermittent for 360° at varying amplitudes
5	10	0.53"	0.87"	0.1"	0.19"	0.64"	0.97"	*	0 (Ind 4)	*	N/A	Reducer	Indications 5, 6 and 7 seen at same time for 360° intermittent but their amplitude relationship varies greatly

DRAW FULL SCALE PLOT HERE:



REVIEWER _____ DATE _____
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

—— Sound Path
- - - Possible sound paths when there exist more than one option.

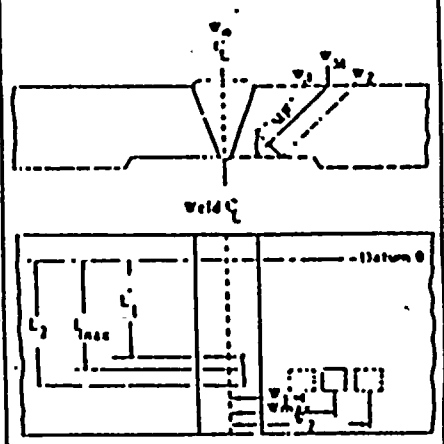
FST-501-0394

NUCLEAR ENERGY SERVICES, INC.

Project No. 5565	Site SHNPP #2 Weld: 1-RC-SW-A3	L_o Location Datum Ø	Date: (Mo/Day/Yr) 1/19/85
Item Identification FIG: 1-151-RC-23		W_o Location weld ℄	Page 3 of 5
Examiner: TC-1A Level III		Angle Used: 0° / 45°⊥ / 45°= / N/A	Attached Cal. Data Sheet TST SDI-0394
Examiner: TC-1A Level I		Scanning dB: N/A / 5E/dB / N/A / N/A	Thickness .772"
			Diameter (nom.) 6"

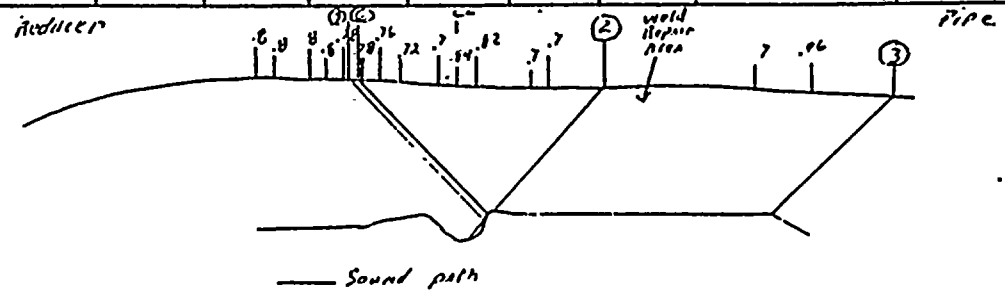
MP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)



Ind. No.	% of DAC	W MAX W	W MAX MP	FWD 50% MAX W_1	FWD 50% MAX MP	BACKWARD 50% MAX W_2	BACKWARD 50% MAX MP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
2	15	0.8"	0.86"	0.60"	0.72"	0.95"	0.73"	*	0.5"cw	*	N/A	Pipe	Seen at same line as #2, both 360°
													intermittently at varying amplitude.
3	20	2.35"	0.86"	2.15"	0.72"	2.45"	0.93"	*	0.5"cw	*	N/A	Pipe	Seen intermittently for 360° at
													varying amplitudes.
6	20	0.53"	1.00"	0.4"	0.89"	0.7"	1.18"	*	Datum Ø	*	N/A	Reducer	Indications 5, 6 & 7 seen at same line
													for 360°. Their amplitude relationships
7	20	0.58"	1.22"	0.45"	1.44"	0.72"	1.29"	*	Datum Ø	*	N/A	Reducer	vary. Indication 7 exceeds 40% DAC
													at location 1"cw

DRAW FULL SCALE PLOT HERE:



--- Possible sound paths when there exist more than one option

REVIEWER _____ DATE _____
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

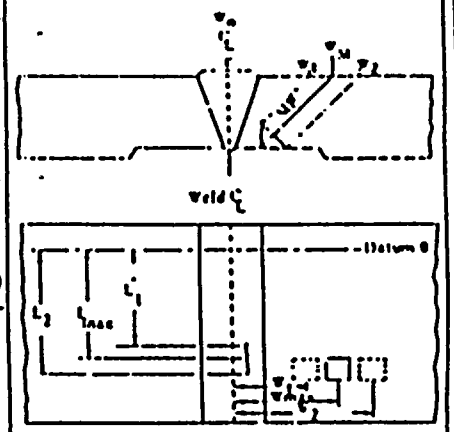
TST SDI-0394

NUCLEAR ENERGY SERVICES, INC.

Project No. 5565	Site SHNPP #2 weld: 1-RC-501-A3	L_o Location Datum Ø	Date: (Mo/Day/Yr) 11/19/85
Item-Identification 1-151-RC-23		W_o Location Weld ℄	Page 4 of 5
Examiner: TC-1A Level II		Angle Used: 0° / 45°⊥ / 45°= / N/A	Attached Cal. Data Sheet EST 501-0394
Examiner: TC-1A Level I		Scanning dB: N/A / 56dB / N/A / N/A	Thickness .772"
			Diameter (nom.) 6"

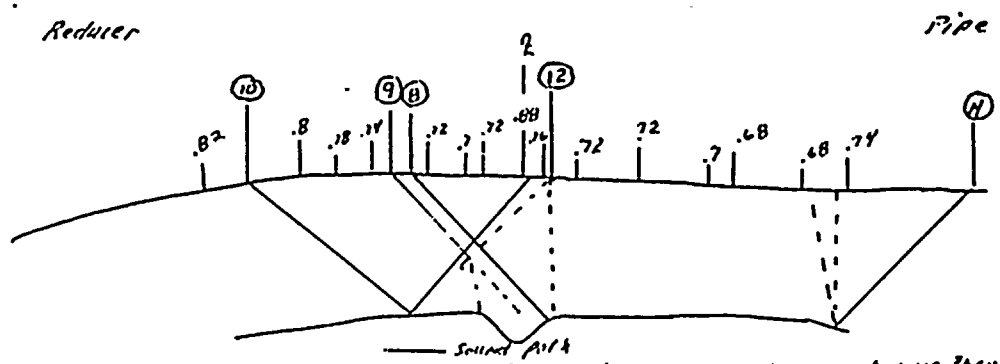
MP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)



Ind. No.	% of DAC	FWD W	FWD MAX W	FWD MAX MP	FWD 50% W_1	FWD 50% MP	BACKWARD 50% W_2	BACKWARD 50% MP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
8	35	0.6	1.14	0.55	1.07	0.75	1.22	*	4"ccw	*	N/A		Reducer	All indications seen 360°
9	16	0.7	0.97	0.60	0.93	0.80	1.00	*	4"ccw	*	N/A		Reducer	intermittently at varying amplitudes.
10	20	1.5"	2.11"	1.35"	2.04"	1.60"	2.22"	*	4"ccw	*	N/A		Reducer	weld ≈ 10 is OD geometric, dampens on ℄
11	20	2.4"	1.65"	2.25"	1.57"	2.60"	1.72"	*	4"ccw	*	N/A		Pipe	#11 could be OD or bottom of weld
12	20	0.15"	0.70"	0.05"	0.68"	0.25"	0.72"	*	4"ccw	*	N/A		Pipe	repair ~~weld~~ mpeung 11/19/85
														#9 + #12 could be either small
														indications or geometry from been
														reduction.

DRAW FULL SCALE PLOT HERE:



REVIEWER _____ DATE _____
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

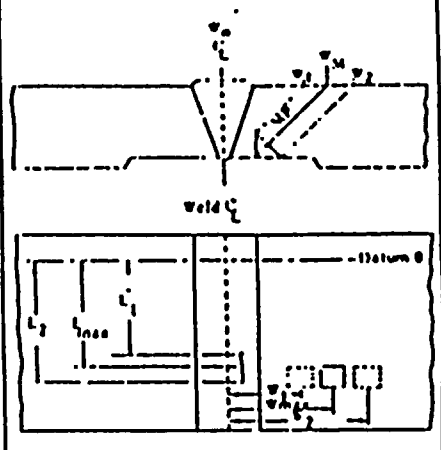
EST 501-0394

NUCLEAR ENERGY SERVICES, INC.

Project No. 5565	Site SHNPP A1	L_o Location Column Ø	Date: (Mo/Day/Yr) 11/8/85
Item Identification WELD: 1-RC-341-A3		W_o Location weld Center Line	Page 5 of 5
Examiner: TC-1A Level III		Angle Used: 0° / 45° L / 45° = / N/A	Attached Cal. Data Sheet TST 501-0394
Examiner: TC-1A Level II		Scanning dB: N/A / N/A / 4.8dB / N/A	Thickness .272"
			Diameter (nom.) 6"

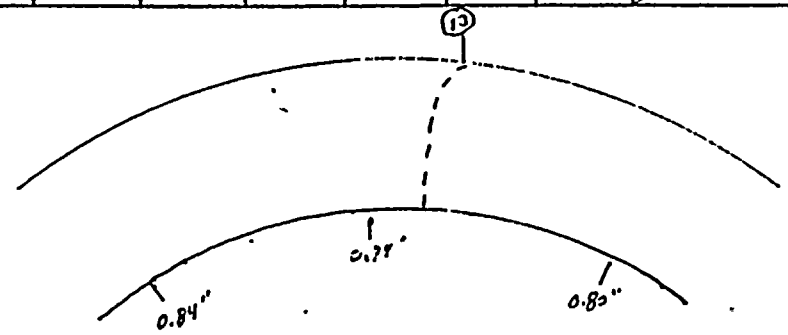
MP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)



Ind. No.	% of DAC	W	MAX MP	FWD 50% W_1	FWD MAX MP	BACKWARD 50% W_2	BACKWARD MAX MP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
13	50	℄	0.86"	℄	0.68"	℄'	0.88"	*	6" ccw	*	N/A	℄	Beam Redirection; scan intermittently for 360° in both directions

DRAW FULL SCALE PLOT HERE:



REVIEWER _____ DATE _____
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

TST 501-0394

IST 501-0394 CAB 1-22-85
~~IST 501-0394~~

Plant/Unit SHNPP #1
Comp/System Reactor Coolant
ISO 1-151-RC-23 Loop N/A

CALIBRATION DATA SHEET

Page 1 of 5 6394
Data Sheet No. IST 501-~~0394~~
Procedure No. IST-501
Subject: Piping Welds
Rev/Change No. 01
Calibration Block No. UT-20-1
Fabrication No. N/A
Surface O.D.
Block Temp S/N J84-03 59°F
Comp. Temp S/N J84-03 70°F
Thickness .772"
CRT Calibrated in Inches of Metal Path
Each Maj. Screen Div= .357"
Calibration is per procedure, paragraph 7.2.(4)

INSTRUMENT SETTINGS	
Mfg/Model No.:	KB USL-33
Serial No. :	211596
Sweep Length :	7.80
Sweep Delay :	7.80
Pulse Length/Damping	Fixed
Freq.: 2.5 Rep. Rate:	N/A
Filter: N/A Video:	N/A Jack: R
REC/Gate Switch	Off Range 2.5
Mode Select:	SFO Reject: Min
Gain (coarse)	40/40 (fine) +1/= . 8/18
Scan Sensitivity	58/68 +1/=

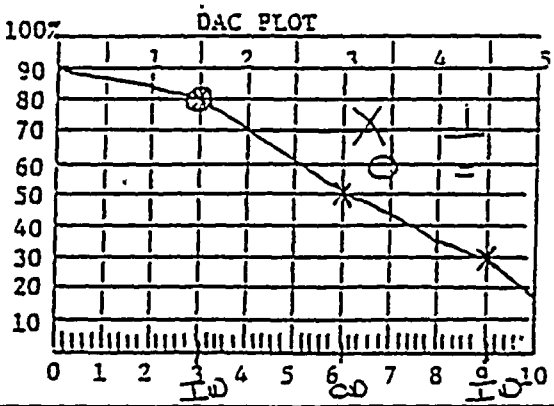
SEARCH UNIT	
Scan Angle:	45° Mode: Shear
Fixturing (if any):	Lucite Wedge
Style or Type No. :	Gamma
Size & Shape:	.25" Round
Frequency	2.25 MHZ
Serial No/Brand:	AeroTech A11343
Measured Angle	45° material 43°
Cable Type & Length	RG-58 (X6') to MDT
Couplant Brand	Ultragel II
Couplant Batch:	# 8336

SCAN AREA	
0° WRV	N/A
0° Vert	N/A
= To Weld	X
⊥ To Weld	X
Calibration	
Axial 48 dB	X
Circ 58 dB	X

INSTR. LINEARITY CAL.

Amplitude					
	High	Low		High	Low
1	100	51	5	60	30
2	90	45	6	50	25
3	80	40	7	40	20
4	70	35	8	30	14

DAC PLOT



AMPT CONTROL LINEARITY

Initial	dB	Result
80	-6	39
80	-12	19
40	+6	82
20	+12	82

CALIBRATION CHECKS	TIME
Initial Cal.	0820
Intermediate	—
Intermediate	—
Intermediate	—
Final Cal.	1300

EXAMINATION WELD/AREA	Recordable Indications Yes	No	Geom	COMMENTS/REASON FOR INCOMPLETED SCAN (S)
1-RC-SW-A3	N/A	N/A	X	Complete Scan from Both Sides, Pipe to Reducer

ADDITIONAL SHEETS?	CHECK BOX		
Continuation	N/A	Beam Plot	N/A
Supplements	X	None	N/A

PSM 1-23-85

EXAMINERS 1 [signature] Date 1-19-85 Level III
2 [signature] Date 1-19-85 Level I

REVIEWERS 1 Ronald Saunders Date 1-26-85
2 _____ Date _____
3 _____ Date _____

NES NUCLEAR ENERGY SERVICES, INC.

NUCLEAR ENERGY SERVICES, INC.

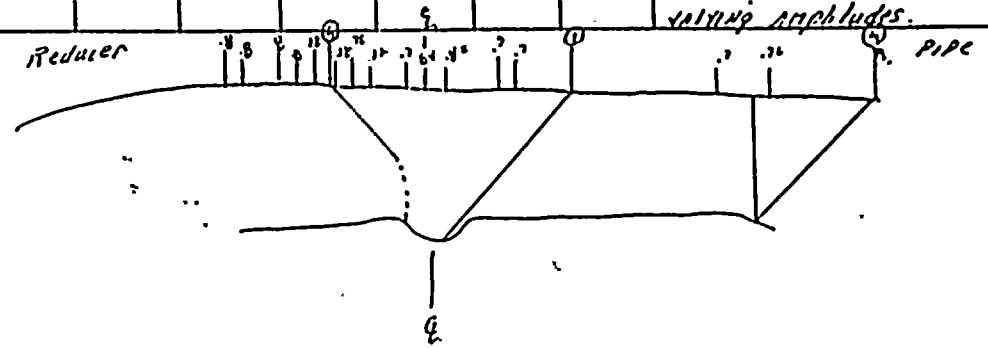
Project No. 5565	Site SUNPP #1	L_0 Location Datum Ø	Date: (Mo/Day/Yr) 11/19/85
Item Identification 1-151-RC-23		W_0 Location 442'10" ℄	Page 2 of 5
Examiner: TC-1A Level III		Angle Beam 0° / 45°L / 45°= / N/A	Attached Cal. Data Sheet ISI-501-0377
Examiner: TC-1A Level I Philip M Jones		Scanning dB N/A / 56:18 / N/A / N/A	Thickness .223"
			Diameter (nom.) 6"

HP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)

Ind. No.	% of DAC	W MAX W	W MAX HP	FWD 50% W_1	FWD MAX HP	BACKWARD 50% W_2	BACKWARD MAX HP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
7	30	0.8"	1.04"	0.65"	0.98"	0.95"	1.14"	*	0.5"cw	*	N A	Pipe	FD Geometry, seen at same time & number 8
8, 4	15	2.45"	1.57"	2.25"	1.43"	2.60"	1.68"	*	0.5"cw	*	N A	Pipe	OD Geometry, depends on OD surface
5	10	0.53"	0.89"	0.4"	0.79"	0.64"	0.87"	*	Datum Ø	*	N A	Reducer	indications 5, 6 and 7 seen at same time but amplitude relationship vary
													* seen 360° intermittently at varying amplitudes.

DRAW FULL SCALE PLOT HERE:

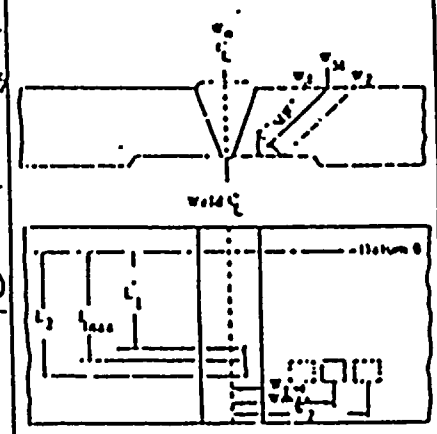


REVIEWER Ronald Saunders DATE 1-26-85
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

ISI-501-0384

NUCLEAR ENERGY SERVICES, INC.

Project No. 5565	Site SHNPP #1 Weld: 1-RC-1W-A3	L_o Location Datum Ø	Date: (Mo/Day/Yr) 1/19/85
Item Identification RB: 1-151-126-23		W_o Location Weld ℄	Page 3 of 5
Examiner: TC-1A Level III		Angle Used: 0° / 45°⊥ / 45°= / N/A	Attached Cal. Data Sheet TST 501-0394
Examiner: TC-1A Level I		Scanning dB: N/A / 55dB / N/A / N/A	Thickness .772"
			Diameter (nom.) 6"

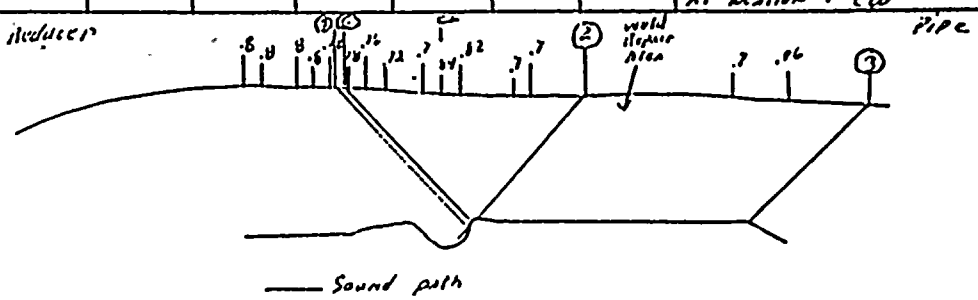


MP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)

Ind. No.	% of DAC	W MAX W	W MAX MP	FWD 50% MAX W_1	FWD 50% MAX MP	BACKWARD 50% MAX W_2	BACKWARD 50% MAX MP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
2	15	0.8"	0.86"	0.60"	0.72"	0.85"	0.73"	*	0.5"cw	*	N/A	Pipe	Seen at same line as #3, both 360° intermittently at varying amplitude
3	20	2.35"	0.86"	2.15"	0.72"	2.45"	0.93"	*	0.5"cw	*	N/A	Pipe	Seen intermittently for 360° at varying amplitudes
6	30	0.53"	1.03"	0.4"	0.97"	0.7"	1.18"	*	Datum Ø	*	N/A	Reducer	Indications 5, 6 & 7 seen at same line for 360° Their amplitude relationships
7	30	0.68"	1.22"	0.45"	1.44"	0.72"	1.28"	*	Datum Ø	*	N/A	Reducer	vary. Indication 7 exceeds 40% DAC at location 1" CW

DRAW FULL SCALE PLOT HERE:



REVIEWER Ronald Saunders DATE 1-26-85
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

--- Possible sound paths when there exist more than one option

BM 1-23-85

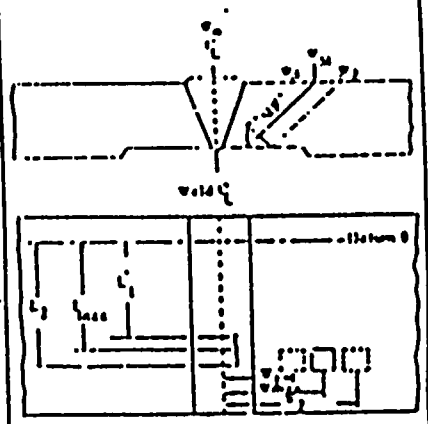
TST 501-0394

NUCLEAR ENERGY SERVICES, INC.

Project No. 5565	Site SNNPP #1	L_0 Location Pelvis ⊄	Date: (Mo/Day/Yr) 1/17/85
Item Identification 1-151-RC-23		W_0 Location Weld ⊄	Page 4 of 5
Examiner: TC-1A Level III		Angle Beam: 0° / 45° L / 45° = / N / A	Attached Cal. Data Sheet SSS-601-018
Examiner: TC-1A Level I Philip M Jones		Scanning dB: N / A / 58 / N / A / N / A	Thickness .222"
			Diameter (nom.) 6"

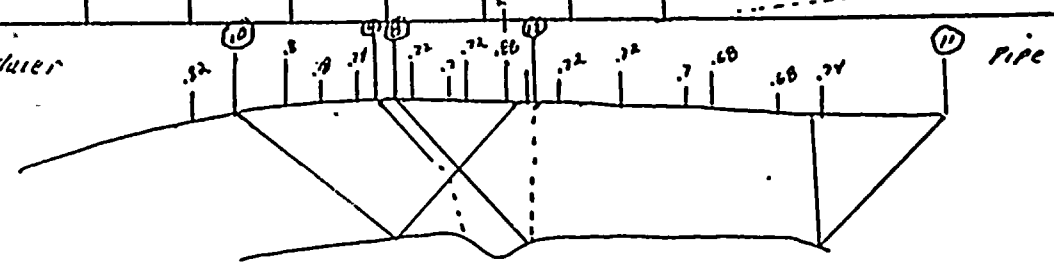
HP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0

W max Distance from ⊄ to S.U. at maximum response.
W_1 Distance from weld ⊄ at 50% of Max amp. (fwd)
W_2 Distance from weld ⊄ at 50% of Max amp. (backward)



Ind. No.	% of DAC	W MAX W	W MAX HP	FWD 50% MAX W_1	FWD 50% MAX HP	BACKWARD 50% MAX W_2	BACKWARD 50% MAX HP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
8	35	0.6	1.14	0.55	1.07	0.75	1.22	*	4" ccw	*	N / A	Reducer	I.D. geometry
9	16	0.7	0.46	0.60	0.93	0.50	1.00	*	4" ccw	*	N / A	Reducer	Beam Redirection
10	20	1.5	2.11	1.35	2.04	1.60	2.22	*	4" ccw	*	N / A	Reducer	OD geometry - depends on OD
11	20	2.4	1.65	2.25	1.57	1.60	1.72	*	4" ccw	*	N / A	Pipe	OD geometry - depends on OD
12	20	0.15	0.70	0.05	0.68	0.25	0.72	*	4" ccw	*	N / A	Pipe	Beam Redirection
													* see 360° independently at varying amplitudes.

DRAW FULL SCALE PLOT HERE:

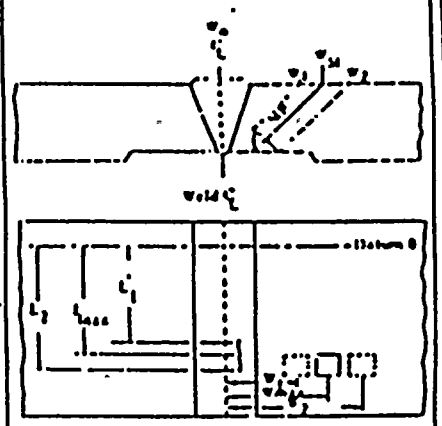


REVIEWER Ronald Saunders DATE 1-26-85
REVIEWER _____ DATE _____
REVIEWER _____ DATE _____

SSS-501-039

NUCLEAR ENERGY SERVICES, INC.

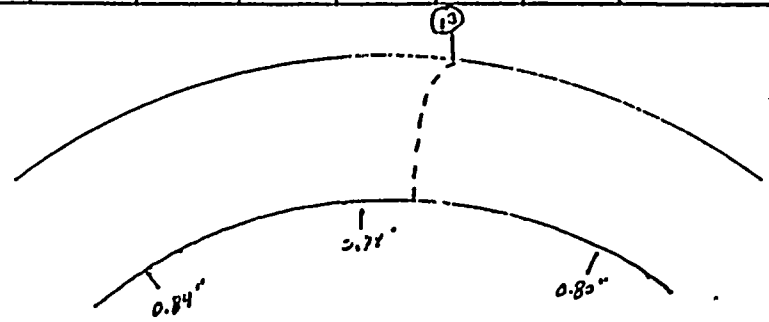
Project No. 5565	Site SHNPP #1	L_o Location 0314 ϕ	Date: (Mo/Day/Yr) 11/4/85
Item Identification WELD: 1-RC-501-A3		W_o Location weld center line	Page 5 of 5
Examiner: TC-IA Level III		Angle Head: 0° / 45° ⊥ / 45° ∥ / N/A	Attached Cal. Data Sheet TST-501-0394
Examiner: TC-IA Level II		Scanning dB: N/A / N/A / 68dB / N/A	Thickness .272"
			Diameter (nom.) 6"



HP Metal Path
RBR Remaining Back Reflection
L Distance from Datum 0.
W max Distance from ℄ to S.U. at maximum response.
W_1 Distance from weld ℄ at 50% of Max amp. (fwd)
W_2 Distance from weld ℄ at 50% of Max amp. (backward)

Ind. No.	% of DAC	W MAX W	W MAX HP	FWD 50% W_1	FWD MAX HP	BACKWARD 50% W_2	BACKWARD MAX HP	L_1 50% MAX	L max	L_2 50% MAX	RBR amp	S.U. Loc.	REMARKS
13	50	℄	0.86"	℄	0.68"	℄'	0.88"	*	6" ccw	*	N/A	℄	Beam Redirection: scan intermittently for 360° in both directions

DRAW FULL SCALE PLOT HERE:



REVIEWER Ronald Saunders DATE 1-26-85

REVIEWER _____ DATE _____

REVIEWER _____ DATE _____

BM 1:23-85

TST-501-0394

AMENDMENT NO. 37 JUSTIFICATION

2.4.8	The Cooling Water Canals and Reservoirs Section is revised to reflect the current design.
3.2.1	Table 3.2.1-1, Classification of Structures, Systems, and Components, is revised to delete most references to Note 4 in response to an NRC question and also to reflect the appropriate QA requirements for items classified as seismic. Other minor corrections update information to reflect current design.
3.7.1	This section is revised to add an appropriate reference.
3.9.3	Table 3.9.3-14, Non-NSSS Supplied Class 1, 2, and 3 Active Valves, is revised to designate a containment isolation valve in the safety injection system as active based on changing its normal position to open and to change information on valves for the RCPB Leak Detection Radiation Monitor as a result of design changes to increase flow and meet particulate sampling requirements.
5.2.5	This section is revised as a result of design changes for the RCPB Leak Detection Radiation Monitor.
5.4.13	Table 5.4.13-1, Pressurizer Valves Design Parameters, is revised to provide consistency between the FSAR and Technical Specifications concerning the Pressurizer PORV throat area.
6.2.2	This section is revised to reflect design changes to the Containment Heat Removal System.
6.2.4	Table 6.2.4-1, Containment Isolation System Data, is revised to show a safety injection valve as normally open based on results of startup testing and as a result of design changes for the RCPB Leak Detection Radiation Monitor.
6.2.5	This section is revised as a result of design changes for the RCPB Leak Detection Radiation Monitor.
6.2.6	This section is revised to delete references to a reduced pressure ILRT because this was not used for preoperational ILRT nor will it be used in the future. LLRT changes are made per IE Information Notice 85-71 to ensure determination of "As-Found" Type A Leakage Rate. Also, changes are made to clarify packing leakage and globe valve testing requirements to provide consistency with the preoperational and surveillance test programs.
6.5.2	This section is revised to reflect the as-built design of NaOH isolation valve logic.
7.3.1	Table 7.3.1-5, ESF Actuation Systems - Safety Injection Signals, and Table 7.3.1-7, ESF Actuation Systems - Containment Isolation Phase A, are revised as a result of design changes for the RCPB Leak Detection Radiation Monitor.

9.1.3	Table 9.1.3-2, Fuel Pool Cooling and Cleanup System Parameters, is revised to reflect final system parameters for Fuel Pool Cooling Pump flow rate and Total Developed Head (TDH). These values are consistent with those used in final system analyses.
9.1.4	Rewording in this section is provided to clarify intent, provide consistency with plant nomenclature and technical manuals, and correct typographical errors.
9.5.1	Commitment to provide on-site air for self-contained breathing equipment is revised to comply with NUREG 0800, 10CFR50, Appendix R and to reflect actual conditions.
9.5.5	This section is revised to provide additional details of the as-built design of the diesel generator cooling water system and support preoperational testing.
12.3.4	Editorial Change
13.1.1, 13.1.2 & 13.1.3	These sections are revised to reflect recent management organization changes and provide consistency with Technical Specifications.
13.2.2	The description of the Licensed Operator Requalification Training is revised to reflect 10CFR55 requirements.
14.2.12	This section is revised to provide compliance with IE Bulletin 80-06 and to provide consistency between design and testing requirements.
15.6.5	Typographical Error
15.7.4	This section is revised to incorporate changes as a result of NRC Technical Specification review-related concerns regarding containment ventilation isolation for a fuel handling accident.

2.4.8 COOLING WATER CANALS AND RESERVOIRS

The safety related cooling water channels (canals), reservoirs, and water control structures within the reservoir system of the Shearon Harris Nuclear Power Plant consist of the Main Reservoir, the Auxiliary Reservoir, the Auxiliary Reservoir Separating Dike, the Auxiliary Reservoir Channel, and the Emergency Service Water Intake and Discharge Channels. ~~and the Cooling Tower Makeup Water Intake Channel~~. 26 37

The design bases and operating modes of the reservoir system are described in relation to the safety-related Emergency Service Water System, Ultimate Heat Sink, and the Cooling Tower Makeup Water System; these discussions appear in Sections 2.4.11, 9.2.1, 9.2.5, and 10.4.5. 26

Shearon Harris Nuclear Power Plant complies with NRC Regulatory Guide 1.127 (refer to Section 1.8) and Ebasco Specification CAR-SH-CH-24, "Reservoir, Dams and Dike Instrumentation Program (Non-Nuclear Safety)." In addition, the North Carolina Utilities Commission requires a dam inspection program involving private consultants. As a minimum, the inspection program will include the water-control structures discussed in Section C.2 of Regulatory Guide 1.127. Periodic monitoring of embankment instrumentation will be performed. The Emergency Service Water Channels and Auxiliary Reservoir are monitored for sediment buildup.

The Shearon Harris Nuclear Power Plant reservoir system constitutes the only water bodies that are of concern regarding protection of plant facilities from flood and wave runup; discussion of the protection of channels and reservoirs is contained in Sections 2.4.2, 2.4.3, 2.4.4, and 2.4.5.

The only locations where potential blockage is of concern to safe plant operation are the Emergency Service Water Intake and Discharge Channels, ~~the Cooling Tower Makeup Water Intake Channel~~, and the Auxiliary Reservoir Channel. 37 These channels are Category I structures and are designed to remain stable when subjected to the Safe Shutdown Earthquake or the most severe cases of other postulated natural phenomena (see Section 2.5.6). In the unlikely event of a slide of the earth slopes, the size of the channels is sufficient to pass the minimum required service water flow at a maximum velocity of 2 ft. per second under the conditions of maximum drawdown of the Main Reservoir and the Auxiliary Reservoir, as indicated in Section 2.4.11. Channel plans and sections are shown on Figures 2.5.6-6, 2.5.6-7, and 2.5.6-8.

The use of screens for the Emergency Service Water Screening Structure and the Emergency Service Water and Cooling Tower Makeup Intake Structure, the location of the intake structures, and the maximum velocity of 2 ft. per second in the channels provide assurance that no blockage of the intake structures, damage to the intake structures or damage to the emergency service water pumps can occur.

The effects of failure of the Auxiliary Separating Dike are discussed in Section 2.4.4.

The design bases for reservoir operation during periods of low water level are discussed in Section 2.4.11.

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Structures	Design and Construction and Operations							Remarks	
	Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)		
Diesel Fuel Oil Storage Tanks and Tank Building	NA	-	-	I	B	A	Q	See Note (30)	~~26~~
Containment Air Locks, Equipment Hatch and Valve Chamber	2	ASME III	MC	I	B	A	Q	See Sec. 3.8 and Note (29)	~~26~~
Containment Internal Structures	NA	-	-	I	B	A	Q		~~26~~
Containment Crane Supports	NA	-	-	I	B	B	Q		~~26~~
Cooling Tower	NNS	-	-	-	-	E	-		
Electrical Manholes for Emergency Power and Control Cables	NA	-	-	I	B	A	Q	See Note (30)	~~26~~
Systems and Components									
Reactor Coolant System									
Reactor Vessel	1	ASME III	1	I	B	A	Q		
Steam Generator (tube side)	1	ASME III	1	I	B	A	Q		
(shell side)	2	ASME III	1	I	B	A	Q	See Note (4)	~~26~~ 37
Pressurizer	1	ASME III	1	I	B	A	Q		

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations							Remarks
	Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	
Reactor Coolant Hot and Cold Leg Piping, Fittings and Fabrication	1	ASME III	1	I	B	A	Q	
Surge Pipe, Spray Pipe, Fittings, and Fabrication	1	ASME III	1	I	B	A	Q	See Note (5)
Crossover Leg Piping, Fittings and Fabrication	1	ASME III	1	I	B	A	Q	
RTD Bypass Manifold	1	ASME III	1	I	B	A	Q	
Pressurizer Safety Valves	1	ASME III	1	I	B	A	Q	
Pressurizer Power Operated Relief Valves and Block Valves	1	ASME III	1	I	B	A	Q	
Valves of Safety Class 1 to Safety Class 2 Interface	1	ASME III	1	I	B	A	Q	
Pressurizer Relief Tank	NNS	ASME VIII	-	-	-	E	~~N~~ -	
Reactor Coolant Thermowell	1	ASME III	1	I	B	A	Q	
Auxiliary Reactor Coolant Piping (Drains, etc.)	2	ASME III	2	I	B	A	Q	
Pressurizer Relief Valve Discharge Lines (between Pressurizer Nozzle and Relief Valve Only)	1	ASME III	1	I	B	A	Q	

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations							Remarks
	Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	
Steam Generator Forging Type A	1	ASME III	1	I	B	A	Q	See Note (9)
Chemical & Volume Control System								
Regenerative HX	2	ASME III	2	I	B	A	Q	
Letdown HX (tube side)	2	ASME III	2	I	B	A	Q	~~See Note (4)~~
(shell side)	3	ASME III	3	I	B	A	Q	
Mixed Bed Demineralizer	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~
Cation Bed Demineralizer	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~
Reactor Coolant Filter	2	ASME III	2	I	B	A	Q	~~See Note (4)~~
Volume Control Tank	2	ASME III	2	I	B	A	Q	~~See Note (4)~~
Charging (High Head Safety Injection) Pumps	2	ASME III	2	I	B	A	Q	
Charging Pump Motors	IE	-	-	I	B	A	Q	
Seal Water Injection Filter	2	ASME III	2	I	B	A	Q	~~See Note (4)~~
Seal Water Return Filter	2	ASME III	2	I	B	A	Q	~~See Note (4)~~
Boric Acid Blender	3	ASME III	3	I	B	A	Q	~~See Note (4)~~
Letdown Orifices	2	ASME III	2	I	B	A	Q	

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations							Remarks	
	Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)		
Excess Letdown HX (tube side)	2	ASME III	2	I	B	A	Q		
(shell side)	2	ASME III	2	I	B	A	Q		
Seal Water HX (tube side)	2	ASME III	2	I	B	A	Q	~~See Note (4)~~	37
(shell side)	3	ASME III	3	I	B	A	Q		
Chemical Mixing Tank	NNS	ASME VIII	-	-	-	E	-		
Chemical Mixing Tank Orifice	NNS	-	-	-	-	E	-		
Boron Meter	NNS	ANSI B31.1	-	-	-	E	-		26
Boric Acid Tanks	3	ASME III	3	I	B	A	Q		
Boric Acid Filter	3	ASME III	3	I	B	A	Q		
Boric Acid Transfer Pump	3	ASME III	3	I	B	A	Q		
Boric Acid Transfer Pump Motors	IE	-	-	I	B	A	Q		
Boric Acid Batching Tank	NNS	ASME VIII	-	-	-	E	~~Q~~ -		37
Reactor Coolant Pump (RCP) Standpipe	NNS	ASME VIII	-	-	-	E	~~Q~~ -		37
RCP Standpipe Orifice	NNS	-	-	-	-	E	~~Q~~ -		37
RCP Seal Bypass Orifice	1	ASME III	1	I	B	A	Q	~~See Note (4)~~	37

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
System Piping and Valves									
a) Part of RCPB	1	ASME III	1	I	B	A	Q		
b) Required for reactor coolant letdown and makeup	2	ASME III	2	I	B	A	Q	~~See Note (4)~~	37
c) Required for providing boric acid for the letdown and makeup loop	3	ASME III	3	I	B	A	Q		
d) Normally or automatically isolated from parts of system covered by a, b or c	NNS	ANSI B31.1	-	-	-	E	~~X~~ -		37
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)	~~26~~
Operators for Safety-Related Active Valves	IE	-	-	I	B	A	Q	See Note (31)	
Boron Thermal Regeneration Subsystem									
Moderating HX (tube side)	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	
(shell side)	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	
Letdown Chiller HX (tube side)	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	
(shell side)	NNS	ASME VIII	-	-	-	E	-		37
Letdown Reheat HX (tube side)	2	ASME III	2	I	B	A	Q	~~See Note (4)~~	
(shell side)	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	
Thermal Regeneration Demineralizer	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	
Chiller Pump	NNS	-	-	-	-	E	-		

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations							Remarks
	Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	
Chiller Surge Tank	NNS	ASME VIII	-	-	-	E	-	
Chiller Unit	NNS	-	-	-	-	E	-	
a) Evaporator	NNS	ASME VIII	-	-	-	E	-	
b) Condenser	NNS	ASME VIII	-	-	-	E	-	
c) Compressor	NNS	-	-	-	-	E	-	
System Piping and Valves								
a) Not normally or automatically isolated from safety class components	3	ASME III	3	I	B	A	Q	
b) Other	NNS	ANSI B31.1	-	-	-	E	~~N~~ -	
Boron Recycle System								
Recycle Hold Up Tank	3	ASME III	3	I	B	A	Q	~~See Note (4)~~
Recycle Monitor Tank	NNS	AWWA D-100	-	-	-	E	-	
Recycle Monitor Tank Pump Casing	NNS	ASME VIII	-	-	-	E	-	
Recycle Evap. Feed Pump	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~
Recycle Evap. Feed Demineralizer	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
Recycle Evap. Feed Filter	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	37
Recycle Evap. Condensate Demineralizer	NNS	ASME VIII	-	-	-	E	-		26
Recycle Evap. Reagent Tank	NNS	ASME VIII	-	-	-	E	-		
Recycle Holdup Tank Vent Ejector	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	26 37
Recycle Evap. Condensate Filter	NNS	ASME VIII	-	-	-	E	-		
Recycle Evap. Concentrate Filter	NNS	ASME ~~III~~ VIII	X-	-	X-	E	-		
Recycle Evaporator Package									
a) Feed Preheater									
1) Feed Side	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	
2) Steam Side	NNS	ASME VIII	-	-	-	E	-		26
b) Gas Stripper	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	37
c) Submerged Tube Evap.									
1) Feed Side	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	37
2) Steam Side	NNS	ASME VIII	-	-	-	E	-		26
d) Evaporator Condenser									
1) Distillate Side	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	37
2) Cooling Water Side	3	ASME III	3	1	B	A	Q	~~See Note (4)~~	26

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks
e) Distillate Cooler								
1) Distillate Water Side	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
2) Cooling Water Side	3	ASME III	3	I	B	-	-	~~See Note (4)~~
f) Absorption Tower	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
g) Vent Condenser								
1) Gas Side	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
2) Cooling Water Side	3	ASME III	3	I	B	-	-	~~See Note (4)~~
h) Distillate Pump	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
i) Concentrate Pump	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
j) Piping								
1) Feed	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
2) Distillate	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
3) Concentrate	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
4) Cooling	3	ASME III	3	I	B	-	-	~~See Note (4)~~
5) Steam	NNS	ANSI B31.1	-	-	-	-	-	
k) Valves								
1) Feed	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
2) Distillate	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
3) Concentrate	3	ASME III	3	See Note (7)	B	~~B~~ A	~~R~~ Q	~~See Note (4)~~
4) Cooling	3	ASME III	3	I	B	-	-	~~See Note (4)~~
5) Steam	NNS	ANSI B31.1	-	-	-	-	-	

~~26~~ 37

AMENDMENT No. 37

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations							Remarks	
	Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)		
System Piping and Valves									
a) Not normally or automatically isolated from safety class components	3	ASME III	3	See Note (7)	B	A	Q	~~See Note (4)~~	37 ~~26~~
b) Other	NNS	ANSI B31.1	-	-	-	E	-		~~26~~
Safety Injection System									
Accumulators	2	ASME III	2	I	B	A	Q		~~26~~
Boron Injection Tank (BIT)	2	ASME III	2	I	B	A	Q		
Hydro Test Pump	NNS	-	-	-	-	E	-		37
System Piping and Valves									
a) Part of RCPB	I	ASME III	I	I	B	A	Q		

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
c) Piping and valves required for performance of safety functions of SC2 components and which are not in service during any normal mode of plant operation and are not testable	2	ASME III	2	I	B	A	Q		~~26~~
d) Operators for Safety-Related Active Valves	IE	-	-	I	B	A	Q	See Note (31)	~~26~~
Reactor Coolant Drain Tank Ht. Exchanger (shell side)	2	ASME III	2	I	B	A	Q		~~26~~
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)	
Containment Penetration Pressurization System									
System Piping and Valves Connected to Penetrations	2	ASME III	~~X~~ 2	I	B	A	Q		37 ~~26~~
Instrumentation	NNS	-	-	-	-	E	-		
Waste Processing Building (WPB) Cooling System									
WPB Cooling Pumps	NNS	-	-	-	-	E	-		~~26~~
Heat Exchanger (tube & shell side)	NNS	ASME VIII	-	-	-	E	-		
Piping and Valves	NNS	ANSI B31.1	-	-	-	E	-		

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations							Remarks
	Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	
Fuel Pool Cooling and Cleanup System								
Fuel Pool Heat Exchanger								
(tube side)	3	ASME III	3	I	B	A	Q	
(shell side)	3	ASME III	3	I	B	A	Q	
Fuel Pool Cooling Pumps	3	ASME III	3	I	B	A	Q	
Fuel Pool Cooling Pump Motors	IE	-	-	I	B	A	Q	
Fuel Pool Demineralizer Filter	NNS	ASME VIII	-	-	-	E	~~R~~ -	
Fuel Pool Demineralizer	NNS	ASME VIII	-	-	-	E	~~R~~ -	
Fuel Pool Refueling Water Purification Filter	NNS	ASME VIII	-	-	-	E	~~R~~ -	
Fuel Pool Strainers	3	ASME III	3	I	B	A	Q	
Fuel Pool Skimmer Filters	NNS	ASME VIII	-	-	-	E	~~R~~ -	
Fuel Pool Skimmer Pumps	NNS	-	-	-	-	E	~~R~~ -	
Fuel Pool and Refueling Water Purification Pump	NNS	-	-	-	-	E	~~R~~ -	
Fuel Pool Skimmers	NNS	-	-	-	-	E	~~R~~ -	
Fuel Pool Liner	NNS	-	-	I	B	B	Q	See Note (21)
Fuel Pool Nozzles	NNS	-	-	I	B	B	Q	See Note (21) and (21A)
System Piping and Valves								
a) Required for cooling and makeup to the fuel pools	3	ASME III	3	I	B	A	Q	
b) Makeup from RWST	3	ASME III	3	I	B	A	Q	

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations							Remarks
	Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	
c) Required for fuel pool cleanup and normally isolated from a)	NNS	ANSI B31.1	-	-	-	E	~~R~~ -	
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)
Fuel Handling System								
Manipulator Crane	NNS	-	-	-	-	E	-	
Reactor Vessel Internals Lifting Device	NNS	-	-	-	-	E	-	
Rod Cluster Control Changing Fixture	NNS	-	-	-	-	E	-	
Reactor Vessel Stud Tensioner	NNS	-	-	-	-	E	-	
Spent Fuel Handling Tool	3	-	-	I	B	A	Q	See Note (10)
Fuel Transfer System								
a) Fuel Transfer Tube and Flange	2	ASME III	2	I	B	A	Q	See Note (11)
b) Portions of Conveyor System and Controls in Fuel Handling Building	3	-	-	I	B	A	Q	See Note (12)
c) Remainder of System	NNS	-	-	-	-	E	-	
New Fuel Elevator	NNS	-	-	-	-	E	-	
New Fuel Racks	3	-	-	I	B	A	Q	
Portable Underwater Lights	NNS	-	-	-	-	E	-	

TABLE 3.2.1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
New Fuel Assembly Handling Fixture	NNS	-	-	-	-	E	-		
New Rod Cluster Control Handling Fixture	NNS	-	-	-	-	E	-		
Lower Internals Storage Stand	NNS	-	-	-	-	E	-		
Upper Internals Storage Stand	NNS	-	-	-	-	E	-		
Load Cell Linkage	NNS	-	-	-	-	E	-		
Spent Fuel Storage Racks	3	-	-	I	B	A	Q		
Refueling Cavity Seal Ring	NA	-	-	I	B	B	Q		
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)	
Liquid Waste Processing System	NNS	See Note (25)	See Note (25)	-	a	C	R		
Reactor Coolant Drain Tank Pump	NNS	ASME III	3	1	-	E	~~R~~ -	SEE NOTE (4)	37
Reactor Coolant Drain Tank Heat Exchanger									
(shell side)	2	ASME III	2	1	B	A	Q		
(tube side)	NNS	ASME VIII	-	-	-	E	~~R~~ -		
System Piping & Valves									
a) Not normally or automatically isolated from SC-3 components	3	ASME III	3	I	B	A	Q		
b) Other	NNS	B31.1	-	-	~~a~~	C	R		37
Gaseous Waste Processing System									
Gas Compressor	NNS	-	-	-	a	C	R		~~27~~
Gas Decay Tank	3	ASME III	3	1	~~a~~ B	A	~~R~~ Q		37

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks
Hydrogen Recombiner (Catalytic)	NNS	-	-	-	a	C	R	~~27~~
System Piping and Valves								
a) Not normally or automatically isolated from SC-3 component	3	ASME III	3	I	B	A	Q	
b) Other	NNS	B31.1	-	-	a	C	R	37
Solid Waste Processing System	NNS	See Note (26)	See Note (26)	-	a See Note (27)	C	R	
Containment Cooling System								
Containment Fan Coolers								
a) Fans and Casings	2	-	-	I	B	A	Q	
b) Supply Fan Motor	IE	-	-	I	B	A	Q	
c) Cooling Coils	2	ASME III	2	I	B	A	Q	
d) Ductwork and dampers up to concrete airshafts	2	-	-	I	B	A	Q	
e) Ductwork and dampers downstream of concrete airshafts	NNS	-	-	-	-	E	-	SEE NOTE (18) 37
Containment Fan Coil Units	NNS	-	-	-	-	E	-	SEE NOTE (18) 37
Instrumention	IE	-	-	I	B	A	Q	See Note (15)
Containment Ventilation System								
Airborne Radioactivity Removal System	NNS	-	-	-	-	E	~~R~~ -	See Note (18)

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
~~Containment Atmosphere Purge Exhaust System (other than between containment isolation valves)~~	~~NNS~~					~~C~~	~~R~~	~~See Note (18)~~	37
CRDM Cooling Systems	NNS	-	-	-	~~A~~ B	~~A~~ B	~~A~~ Q		
Containment Combustible Gas Control System									
Electric Hydrogen Recombiner	2	-	-	I	B	A	Q		
Instrumentation (in part)	IE	-	-	I	B	A	Q	See Note (15)	
Hydrogen Monitoring System (0-10% range capability)									~~27~~
a) Piping and Valves	2	ASME III	2	I	B	A	Q		
b) Hydrogen Analyzer Cabinet	IE	-	-	I	B	A	Q	See Note (15)	~~27~~
c) Remote Control Panel	IE	-	-	I	B	A	Q	See Note (15)	
d) Remote Sample Dilution Panel	NNS	-	-	-	-	E	-		
Containment Vacuum Relief (except blind flanges and valves for leak testing)	2	ASME III	2	I	B	A	Q		
INLET BELLMOUTH	3	ASME III	3	I	B	A	Q		37
Instrumentation	IE	-	-	I	B	A	Q		
Primary Shield Cooling System	3	-	-	I	B	A	Q		
Instrumentation	NNS	-	-	-	-	E	-		
Reactor Supports Cooling System	3	-	-	I	B	A	Q		
Instrumentation	NNS	-	-	-	-	E	-		

TABLE 3.2.1-1. (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
~~Reactor Cavity Pressure Relief Damper~~	~~3~~			~~I~~	~~B~~	~~A~~	~~Q~~		~~26~~ 37
Reactor Auxiliary Building (RAB) Ventilation Systems									
RAB Normal Ventilation System									
a) Tornado and Isolation dampers	3	-	-	I	B	A	Q		~~26~~ 37
b) All other components	NNS	-	-	-	-	E	-	SEE NOTE (18).	
RAB Steam Tunnel Ventilation	3	-	-	I	B	A	Q		
RAB Emergency Exhaust System	3	-	-	I	B	A	Q		
RAB ESF Equipment Cooling Systems	3	-	-	I	B	A	Q		
RAB ESF Battery Room Exhaust Fans	3	-	-	I	B	A	Q		
RAB Computer & Communications Room ~~Complex Isolation Valves~~ HVAC SYSTEM TORNADO PROTECTION DAMPERS	3	-	-	I	B	A	Q		
RAB Switchgear Room Ventilation System, including	3	-	-	I	B	A	Q		~~26~~ 37
a) Smoke purge isolation valves ~~room "B"~~									
b) Smoke purge makeup isolation dampers ~~and valves room "B"~~									
RAB Electric Equipment Protection Rooms Ventilation System, including	3	-	-	I	B	A	Q		~~26~~
a) HV-equipment room exhaust									
b) Smoke purge isolation valves and dampers									
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)	~~26~~

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
Waste Processing Building	NNS	-	-	-	-	E	-	SEE NOTE (18)	~~36~~ 37
Ventilation Systems									
MCC and Instrument Rack Area Local Cooler	3	-	-	I	B	A	Q		
Control Room HVAC Systems									
Normal Supply and Exhaust Subsystem									
a) Supply Fans & Casings	3	-	-	I	B	A	Q		
b) Cooling Coils	3	ASME III	3	I	B	A	Q		~~36~~
c) Electric Heating Coils	IE	-	-	I	B	A	Q		
d) Ducts and Dampers	3	-	-	I	B	A	Q		
e) Valves for Outside Air Intakes	3	ASME III	3	I	B	A	Q		
f) Chlorine & Radiation Detectors	IE	-	-	I	B	A	Q		~~36~~
g) Smoke Detectors	NNS	-	-	-	b	E	-		~~36~~
Control Room Smoke Purge and Exhaust									
Makeup a) Boundary Isolation Valves	3	ASME III	3	I	B	A	Q	~~See Note (1)~~	37
b) Other	NNS	-	-	-	-	E	-	SEE NOTE (18)	~~36~~
Control Room Emergency Filtration System	3	-	-	I	B	A	Q		
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)	~~36~~
Fuel Handling Building HVAC Systems									
Air Conditioning System for the Operating Floor									
a) Air Handling Unit	NNS	-	-	-	-	E	-		~~36~~
h) Offices & Kitchen Area Electric heating coils	NNS	-	-	-	-	E	-	SEE NOTE (18)	37
i) Exhaust Fans, Ducts and Dampers	NNS	-	-	-	-	E	-	SEE NOTE (18)	

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
b) Exhaust Fans	NNS	-	-	-	-	E	-		~~26~~
c) Ductwork and Dampers									~~26~~
1) Isolation Dampers	3	-	-	I	B	A	Q		~~26~~
3) ~~2)~~ Other	NNS	-	-	-	-	E	-	SEE NOTE (18)	37
2) DUCT IN UNLOADING AREA	3	-	-	I	B	A	Q		
Emergency Exhaust System for the Operating Floor	3	-	-	I	B	A	Q		
Normal Ventilation System for Areas Below Operating Floor									
a) Air Handling Unit	NNS	-	-	-	-	E	-		~~26~~
b) Exhaust Fans	NNS	-	-	-	-	E	-		
c) Ductwork and Dampers									
1) Isolation Dampers	3	-	-	I	B	A	Q		
2) Other	NNS	-	-	-	-	E	-	SEE NOTE (18)	~~26~~ 37
Spent Fuel Pump Room Ventilation System	3	-	-	I	B	A	Q		
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)	~~26~~
Fuel Oil Transfer Pump House Ventilation System	3	-	-	I	B	A	Q		~~26~~
Diesel Generator Building Ventilation System	3	-	-	I	B	A	Q		
a) DGB-Electric Room Ventilation									
b) DGB-F.O. Day Tank and Silencer Room Ventilation									
c) DGB - DIESEL GENERATOR ROOM VENTILATION SYSTEM									37

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
Chilled Water Piping and Valves									
a) Required to provide chilled water to safety related air handling units	3	ASME III	3	I	B	A	Q		~~26~~
b) Required only for RAB NNS Ventilation Systems and automatically isolated from a)	NNS	-	-	-	-	E	-	SEE NOTE (18)	~~26~~ 37
c) Operators for Safety-Related Active Valves	IE	-	-	I	B	A	Q		~~26~~
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)	~~26~~
Non-Essential Services Chilled Water System	NNS	-	-	-	-	E	-	SEE NOTE (18)	
Containment Atmosphere Purge and Makeup System									
Ductwork inside Containment Adjacent ~~up~~ to the isolation valves	~~NNS~~ 3	-	-	~~/~~ I	~~/~~ B	~~/~~ A	~~/~~ Q		~~26~~ 37
Containment isolation valves and piping	2	ASME III	2	I	B	A	Q		~~26~~
From isolation valves outside Containment to floor penetration at RAB Elevation 286 ft (PURGE MAKEUP) AND RAB EL 261 FT. (PURGE EXHAUST)	3	-	-	I	B	A	Q	~~See Note (4)~~	~~26~~ 37
Instrumentation (isolation valves only)	IE	-	-	I	B	A	Q	See Note (15)	~~26~~ 37
Other	NNS	-	-	-	-	E	-	SEE NOTE (18)	~~26~~

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
Operators for Safety-Related Active Valves	1E	-	-	I	B	A	Q	See Note (31)	~~26~~ ~~26~~ ~~26~~
Containment Hydrogen Purge (Exhaust) and Makeup System									~~26~~ 37
Containment Isolation valves and piping	2	ASME III	2	I	B	A	Q		~~26~~
From isolation valve outside Containment to floor penetration at RAB Elevation ~~286~~ 261 ft.	3	-	-	I	B	A	Q	~~See Note (4)~~	37 ~~26~~
Instrumentation (isolation valves only)	1E	-	-	I	B	A	Q	See Note (15)	~~26~~
Other	NNS	-	-	-	-	E	-	SEE NOTE (18)	~~26~~ 37
~~Operators for Safety-Related Active Valves~~	~~1E~~			~~I~~	~~B~~	~~A~~	~~Q~~	~~See Note (31)~~	
RAB OUTSIDE AIR INTAKES FOR CONTAINMENT VACUUM RELIEF AND PURGE SYSTEMS	2	-	-	I	B	A	Q		37
TURBINE BUILDING HVAC SYSTEMS	NNS	-	-	-	-	E	-		

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks
b) From the MSIV up to and including the last seismic restraint in the Turbine Building	NNS ~~3~~	See Note (16)	See Note (16)	I	B	B	Q	~~See Note (4)~~ 29 37
c) Downstream of last seismic restraint in Turbine Building	NNS	ANSI B31.1	-	-	-	E	-	
d) Operators for Safety-Related Active Valves	IE	-	-	I	B	A	Q	See Note (31)
e) Turbine Gland Sealing System	NNS	B31.1	-	-	-	C	R	
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)
Steam Generator Blowdown System								
System Piping and Valves								
a) From steam generator to and including containment isolation valves	2	ASME III	2	I	B	A	Q	
b) From containment isolation valves to RAB wall	3	ASME III	3	I	B	B	Q	~~See Note (4)~~ 29 37
Condensate and Feedwater System								
Condensate and Feedwater Pumps	NNS	-	-	-	-	E	-	
Electromagnetic Filter	NNS	ASME VIII	-	-	(27)	C	R	See Note (27)
Condenser Evacuation System	NNS	B31.1	-	-	-	~~C~~ E	~~R~~ -	37

[redacted] the MFIV check valve back [redacted] the last seismic restraint in the Turbine Building	NNS	SEE NOTE (16)	SEE NOTE (16)	[redacted]	B	B	Q	37
d) Upstream of last seismic restraint in Turbine Building	NNS	ANSI B31.1	-		-	E	-	

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations: Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	Remarks	
System Piping and Valves									
a) Feedwater piping from the steam generator back to and including the MFIV check valve; all branch connections from this section up to and including the first normally closed shutoff valve	2	ASME III	2	I	B	A	Q		
b) MFW control valves and bypass control valves; ~~flow control valves in the tempering lines to AFS nozzles~~	~~NNS~~ 3	ASME III	3	I	B	~~B~~ A	Q	See Note (4)	~~28~~ 37
~~c) Other~~	~~NA~~	~~ANSI B31.1~~	~~-~~	~~I~~	~~B~~	~~B~~	~~Q~~		~~29~~
e) ~~d)~~ Operators for Safety-Related Active Valves	IE	-	-	I	B	A	Q	See Note (31)	
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)	
Auxiliary Feedwater System									
AFW Pumps (Motor & Turbine Driven)	3	ASME III	3	I	B	A	Q		
AFW Pump Motors	IE	-	-	I	B	A	Q		
Condensate Storage Tank	3	ASME III	3	I	B	A	Q		
AFW Pump Turbine Driver	3	ASME III	-	I	B	A	Q	See Note (28)	
System Piping and Valves	~~2~~	~~ASME III~~	~~2~~	~~I~~	~~B~~	~~A~~	~~Q~~		37
a) From steam generator up to and including the containment isolation valves	2	ASME III	2	I	B	A	Q		

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Systems and Components	Design and Construction and Operations							Remarks
	Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)	
b) Other	3	ASME III	3	I	B	A	Q	
c) Operators for Safety-Related Active Valves	IE	-	-	I	B	A	Q	See Note (31)
Instrumentation	IE	-	-	I	B	A	Q	See Note (15)
Condenser Circulating Water System	NNS	-	-	-	-	E	-	
Demineralized Water Storage Systems								
Demineralized Water Storage Tank	NNS	-	-	-	-	E	-	
Reactor Make-up Water Storage Tank	3	ASME III	3	I	B	A	Q	~~See Note (4)~~ 37
Instrumentation (in part)	IE	-	-	I	B	A	Q	See Note (15)
Reactor Make-up Water Pump, Pipes/ Valves	3	ASME III	3	I	B	A	Q	~~See Note (4)~~ 37
Reactor Make-up Water Pump Motors	NNS	-	-	~~I~~ -	~~B~~ -	~~A~~ E	~~Q~~ -	~~20~~ 37
Chlorine Leak Detection (in part)	IE	-	-	I	B	A	Q	
Radiation Monitoring System								
Safety Area Monitors	IE	-	-	I	B	A	Q	See Note (15)

TABLE 3.2.1-1 (Continued)

CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

	Design and Construction and Operations							Remarks	26
Systems and Components	Safety Class (1)	Code	Code Class	Seismic Category (2)	Quality Assurance (3)	Quality Class (23)	Quality Assurance (24)		26
Piping and valves up to and including second isolation valve	2	ASME III	2	I	B	A	Q		26
All other piping	NNS	B31.1 ~~X~~	-	-	-	E	-		37 ~~26~~
Instrumentation	1E	-	-	I	B	A	Q	See Note (15)	
Inadequate Core Cooling System (in part)	1E	-	-	I	B	A	Q	See Note (15)	
Associated piping and valves	2	ASME III	2	I	B	A	Q		26

Notes to Table 3.2.1-1 (Continued)

(18) Those portions of this system whose failure may have an adverse effect on a nearby safety related component are seismically supported AND SEISMICALLY DESIGNED AND ARE SUBJECT TO THE APPROPRIATE QA REQUIREMENTS. 37

(19) The reinforced concrete mat and walls of the Unit 1 Turbine Building between column line 42 (approx.) and 43 (approx.) are designed and constructed to Seismic Category I requirements due to the presence of the diesel generator service water pipe tunnel and Class 1 electrical cable area above the pipe tunnel (see Figure 1.2.2-60). This area is designed and constructed to withstand the collapse of the Turbine Building concurrent with a SSE.

(20) Provides mechanical support for Safety Class 1 component.

(21) Will be designed and fabricated to the applicable portions of ASME III, although it is not classified as ANS Safety Class 1, 2, or 3.

(21A) Fuel Pool Nozzles will be considered from the Fuel Pool Liner to the first shop girth weld.

(22) Provides support to the Safety Class 1 pressure boundary conduit.

(23) Quality Classification (Operations Phase)

A - Safety related.
B - Non-safety seismic ally designed or falls under Regulatory Guide 1.97.
C - Radwaste.
D - Fire protection.
E - Non safety, non-seismic ally designed. ~~26~~ 37

(24) Quality Assurance Requirements (Operations Phase)

Q - QA requirements will meet 10CFR50 Appendix B criteria.
R - QA requirements will meet ETSB 11-1 QA requirements as a minimum. Optionally "Q" requirements may be imposed.
F - QA requirements will meet Fire Protection QA requirements as a minimum. Optionally "Q" requirements may be imposed.
- QA requirements of 10CFR50 Appendix B are not mandatory.

(25) The code and code class for individual components in the Liquid Waste Processing System can be found on Table 11.2.1-7.

(26) The code and code class for individual components in the Solid Waste Processing System can be found on Table 11.4.2-4.

(27) The ETSB 11-1 QA applies to components listed in Table 11.4.2-4 except those listed as manufacturer's standard.

(28) ASME III Code applies to oil cooler and trip/throttle valve only.

(29) Not Stamped.

direction. Two additional sets of statistically independent accelerograms, developed for the east-west and vertical directions, are presented on Figures 3.7.1-25 through 3.7.1-28.

A comparison of the spectral values of the SSE statistically independent horizontal east-west and vertical time histories, and the corresponding design response spectra, is presented on Figures 3.7.1-29 through 3.7.1-34, for two, four, and seven percent damping, using the frequency intervals discussed above. The comparisons discussed above show that none of the points fall below ten percent of the design response spectrum, and no more than five points fall below the design response spectrum.

The earthquake accelerograms used in the analysis of the Seismic Category I dams and dikes envelop the horizontal and vertical design response spectra presented on Figures 3.7.1-5 through 3.7.1-8. Figures 3.7.1-35 through 3.7.1-37 show the SSE horizontal accelerograms for one, two, and five percent damping.

To demonstrate that these time histories envelop the design response spectra, a high resolution response spectra analysis was performed. Each time history was analyzed at 247 discrete period points between the period range of 0.014 to 3.000 sec. These period points were spaced at 0.0005 sec. intervals at the short period end and at 0.1 sec. intervals at the long period end. These period intervals were established by performing response analysis at both half resolution (124 period points) and full resolution (247 period points). It was found that there was essentially no change in the general shape of the response spectra. Therefore, these 247 closely spaced period points are considered to be sufficient to detect all the peaks and valleys of the response spectra.

Comparison of these time histories with the horizontal design response spectra for the SSE are indicated on Figures 3.7.1-38, 3.7.1-39 and 3.7.1-40, for one, two, and five percent damping, respectively.

3.7.1.3 Critical Damping Values

The damping ratios, which are expressed as percentages of critical damping and used in the dynamic analysis of Seismic Category I structures, are consistent with those of Regulatory Guide 1.61, and are shown in Table 3.7.1-1.

For the Seismic Category I Main Dam, Auxiliary Dam and Auxiliary Separating Dike, the seismic analysis is presented in Section 2.5.6.

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For the Seismic Category I reactor coolant loop system, Seismic Category I piping systems, and Seismic Category I equipment not purchased as of March 1, 1977, the SHNPP complies with the damping values of Regulatory Guide 1.61. In accordance with the provision of Regulatory Position C2, documented test data have been provided to and approved by the NRC which justifies the use of a damping value higher than three percent critical for large piping systems under the faulted condition. A conservative value of four percent critical has been justified by testing for the Westinghouse reactor coolant loop, as presented in WCAP-7921-AR "Damping Values of

For Seismic Category I cable tray supports, damping ratios per 1978 Bechtel Power Corporation Cable Tray and Conduit Test Program (Report # 1053-21.1-4) is to be utilized.

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TABLE 3.9.3-14 (continued)

NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
1CS-V711SN	CS	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	2	RCPB Boundary
1CS-V70SN	CS	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	2	RCPB Boundary
2CS-V129SN	CS	RAB	(3)	Check	ΔP	Rockwell	2	1500	220 psig @ 200 F	2	Safe Shutdown
3CS-V222SN	CS	RAB	(3)	Check	ΔP	Rockwell	3	1500	150 psig @ 250 F	2	Safe Shutdown
3CS-V223SN	CS	RAB	(3)	Check	ΔP	Rockwell	3	1500	150 psig @ 250 F	2	Safe Shutdown
1SI-V39SA V45SB V51SA	SI	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	2	RCPB Boundary
1SI-V63SA V69SB V75SA	SI	RCB	(4)	Check	ΔP	Rockwell	1	1521	2485 psig @ 650 F	2	RCPB Boundary
2SI-V530B	SI	RAB	—	Globe	ΔP	Copes-Vulcan	2	1500	800 psig @200F	1	Containment Isolation

TABLE 3.9.3-14 (continued)

NON-NSSS SUPPLIED CLASS 1, 2 AND 3 ACTIVE VALVES

Tag Number	System	Location	Env. Qual.	Type	Operator	Manufacturer	Safety Class	Valve Design Rating (ANSI #)	System Design Conditions	Size (Inches-ID)	Function
3SW-V870SA-1	SW	RAB	(3)	Check	ΔP	Rockwell	3	600	150 psig @ 140 F	1	ESF Operation
3SW-V871SB-1	SW	RAB	(3)	Check	ΔP	Rockwell	3	600	150 psig @ 140 F	1	ESF Operation
2CS-V136SN	CS	RAB	(3)	Check	ΔP	Rockwell	2	1500	2735 psig @ 200 F	2	ESF Operation
2CS-V137SN	CS	RAB	(3)	Check	ΔP	Rockwell	2	1500	2735 psig @ 200 F	2	ESF Operation
2CS-V138SN	CS	RAB	(3)	Check	ΔP	Rockwell	2	1500	2735 psig @ 200 F	2	ESF Operation
2SP-~~V308SB~~ V448SA-1	SP	RCB	(5)	Globe	Solenoid	Target-Rock	2	600	90 psig @ 400 F	1	RCPB Leak Detection Rad. Monitor ~~H_2 Analyzer~~
2SP-~~V314SB~~ V449SB-1	SP	RAB	(3)	Globe	Solenoid	Target-Rock	2	600	90 psig @ 400 F	1	RCPB Leak Detection Rad. Monitor ~~H_2 Analyzer~~
2SP-~~V309SB~~ V450SA-1	SP	RCB	(5)	Globe	Solenoid	Target-Rock	2	600	90 psig @ 400 F	1	RCPB Leak Detection Rad. Monitor ~~H_2 Analyzer~~
2SP-~~V315SB~~ V451SB-1	SP	RAB	(3)	Globe	Solenoid	Target-Rock	2	600	90 psig @ 400 F	1	RCPB Leak Detection Rad. Monitor ~~H_2 Analyzer~~
3CH-B2SA-1	ESCWS Supply	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	4	Isolation
3CH-B4SB-1	ESCWS Supply	RAB	(3)	Butterfly	Diaphragm	ITT/Hammel Dahl	3	150	150 psig @ 125 F	4	Isolation

After collection in the containment sump, the collected leakage is pumped to the floor drain collection tank. The combined sump pump discharge flow is recorded in the Control Room. The sumps are also provided with level switches to alert the operator of high level conditions in the event of sump pump malfunction.

The sump discharge line may be sampled from outside of the Containment to provide additional aid in identifying the leakage source.

The system is designed to permit calibration and operability tests during plant refueling.

5.2.5.3.2 Containment Airborne Particulate and Gaseous Radioactivity Monitoring

The containment atmosphere radiation monitor is part of the safety related portion of the Radiation Monitoring System and is designed to provide a continuous indication in the Control Room of the particulate and gaseous radioactivity levels inside the Containment. Radioactivity in the containment atmosphere indicates the presence of fission products due to a reactor coolant system leak.

The monitor draws a continuous sample of containment air through a ~~sampler manifold assembly~~ sample point located inside the Containment. ~~Sampled points in the Containment are at the north reactor cavity, south reactor cavity, above each of the three steam generators, above each of the three reactor coolant pumps, and above the pressurizer. Normally, all points (except the pressurizer) are closed; on detection of high radiation, each point can be selected for further sampling by the operator.~~ The guidelines of ANS-13.1 have been followed to minimize biasing the particulate portion of the air sample. All sample lines are heat traced outside the Containment to prevent condensation within the lines up to 120 F and 100 percent humidity (non-condensing).

The monitor uses the airborne seismic category I and particulate and noble gas detector described in Section 11.5.2.6.5. The containment monitor is powered by RCPB the A bus. The monitor normally monitors the containment atmosphere for leakage as required by Regulatory Guide 1.45. A containment isolation signal will isolate the monitor from the Containment. The monitor provides a high radiation alarm when concentrations reach preset limits. The receipt of this alarm will also alert the operator to the presence of low level leakage so that ~~additional sampling (cycling through each sample point)~~ appropriate action can be ~~done~~ taken in order to locate the leakage source and initiate normal purge isolation when preset limits are exceeded.

TABLE 5.4.13-1

PRESSURIZER VALVES DESIGN PARAMETERS

Pressurizer Safety Valves		
Number		3
Maximum relieving capacity, ASME rated flow (lb/hr)		380,000
Set pressure (psig)		2485
Design temperature (F)		650
Fluid		Saturated steam
Transient Condition (F) :	Non-Faulted Conditions	673
	Faulted Conditions	682
Backpressure		
Normal (psig)		3 to 5
Expected during discharge (psig)		500
Throat Area (in^2)		3.64
Pressurizer Power Operated Relief Valves		
Number		3
Design pressure (psig)		2485
Design temperature (F)		650
Relieving capacity at 2350 psig, per valve (lb/hr)		210,000
Fluid		Saturated steam
Transient condition (F):	Non-Faulted Conditions	673
	Faulted Conditions	682
Throat Area (in^2)		~~2.45~~ 2.9
Pressurizer Spray Valves		
Number		2
Design Pressure, psig		2485
Design Temperature, F		650
Design Flow, for valves full open, each, gpm		350

c) During normal operation, the CCS is designed to ~~operate in the following manner to limit the containment temperature from 80 F to a maximum of 120 F:~~ maintain the containment average temperature below 120 F

~~1) When the service water temperature is 90 F or below, two of the four safety related fan cooler units will operate with both fans per unit operating at full speed along with three non-safety fan-coil units.~~

~~2) When service water temperature is above 90 F, in addition to the operation of safety and non-safety cooling units as discussed in 1) above, both standby safety related fan cooler units will be energized to operate with one fan per unit running at full speed. Operation of standby fan cooler units is anticipated approximately 370 hours a year.~~

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d) Mixing the containment atmosphere following an accident.

6.2.2.2.1.2 Design Description

The CCS consists of four safety related fan cooler units and three non-safety fan coil units.

Following a design basis accident only the safety related fan cooler units are required to operate. During normal power operation, safety related units operate in conjunction with the non-safety units to maintain required containment temperature. See Table 6.2.2-1 for major system components. Figure 6.2.2-3 describes the extent of essential portions of the ductwork and equipment for the CCS. Insert "B" From Page 6.2.2-5

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Two of the four safety related fan cooler units are located at Elevation 236', the remaining two safety related units are located at Elevation 286'.

Two separate trains are provided, each consisting of two fan cooler units with each unit supplying air to an independent, vertical concrete air shaft.

Train A Components		Train B Components	
Fan Cooler	AH-2	Fan Cooler	AH-1
Fan Cooler	AH-3	Fan Cooler	AH-4
Service Water	Loop A	Service Water	Loop B
Emergency Power	Diesel A	Emergency Power	Diesel B

Train selection of each fan cooler with its respective water supply is under administrative control.

Each fan cooler is served by water from the Service Water System. A detailed description of the Service Water System is given in Section 9.2.1.

Unit performance data is shown in Table 6.2.2-1

Each safety related fan cooler consists of cooling coil sections and two direct driven vane axial flow fans. ~~Each delivering 62,500 cfm at full speed.~~ Each fan is equipped with a two speed motor enabling half speed operation at ~~31,250 cfm.~~ A gravity damper is provided at the discharge side of each fan to prevent air flow in the reverse direction when only one fan per unit is required to operate. Both fans of the unit discharge into a common

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DBA conditions and integrated Leak rate test conditions.

TABLE 6.2.2-1

CONTAINMENT COOLING SYSTEM COMPONENTS

NOTE: All air quantities are actual cfm.

CONTAINMENT FAN COOLER SAFETY CLASS 2 UNITS

	Normal Operating Conditions	Design Basis Accident Conditions
No. of Units	2 fans per unit and 2 units operating	1 fan per unit half speed, 4 units starting and 2 units operating
Fan Cooler Unit Operating Capacity.ACFM	125,000	31,250
Actual Air Mixture Flow (ACFM) at Fan Inlet	62,500	31,250
Design Ambient Pressure, psig	0	45.0/39.1 (1) 27
Ambient Temp, F	120	258
Total Pressure, in. WG	7.9	5.1
Fan RPM	1770	870
Outlet Velocity, FPM	5800	2560
Brake HP	101.2	32.8
Motor HP	125	62.5
Cooling Water Flow - GPM	1500	~~1500~~ 1425 37
Entering Water Temp. F	95	95

NOTE: (1) 39.1 psig - steam line break pressure
45.0 psig - maximum containment design pressure 27

duct which is connected to a concrete air shaft through a locked open damper. ~~A pneumatically operated fail-close damper isolates the concrete air shaft from the fan cooler unit.~~ A branch duct connection ~~has been~~ is provided ~~upstream of the shaft isolation damper~~ to serve as a post accident discharge nozzle and is normally isolated by means of a separate pneumatically operated, fail open damper. 37

Insert "A" from Page 6.2.2-5

6.2.2.2.1.2.1 Post Accident Operation

During post-accident operation, four fan cooler units operate with one fan per unit running at half speed. The system can operate in this mode as long as both emergency diesel generators and both service water system trains are available.

In the event of failure of one of the emergency diesel generators or one service water system train only two fan cooler units will operate. ~~The damper in the main discharge duct will be closed, isolating the concrete air shaft and~~ The damper in the post-accident discharge branch duct will be opened. The post-accident discharge duct is provided with high velocity nozzles to diffuse air to accelerate the temperature mixing inside containment. These nozzles are directed to selected areas of heat release, to achieve thorough mixing of containment atmosphere. The high velocity nozzles direct turbulent air jets from discharge points at two levels inside containment where two separate trains of containment fan coolers are located. Two sets of nozzles are located at Elevation 286 ft. as shown on Figure 6.2.2-14, Sections C-14-1 and C-12-1, and the other two nozzles are shown on Figure 6.2.2-10 (plan at Elevation 221.00 ft.) as post accident discharge nozzles. Seismic Category I ductwork is used from the fan coolers to the discharge outlets. 37

As the post-accident containment atmosphere steam-air mixture passes through the system cooling coils, it is cooled and a portion of the steam is condensed. The combined cooling capacity of all four cooler units is adequate to prevent excursions beyond the peak design pressure and temperature of the Containment; however, in the event of a single active failure in one train, one containment spray pump and two containment fan coolers will provide the adequate cooling capacity. The fan cooler units receive electric power from the diesel generators 15 seconds after a LOCA through a timer-sequencer. An additional 10 seconds are required to bring the fans to the operational speed.

The containment fan cooler performance data, showing the energy removal rate as a function of containment atmosphere temperature, is shown on Figures 6.2.2-4 and 6.2.2-5 and Tables 6.2.2-2 and 6.2.2-3.

6.2.2.2.1.2.2 ~~Normal Operation~~

During normal power operation, two or all four of three non-safety fan coil units are in continuous operation along with ~~the~~ safety-related fan cooler units. The following describes their operation: 37

a) When ~~service water~~ containment average temperature is ~~90~~ 118 F or below: Only two fan cooler units will operate with both fans of the unit running at full speed. Each of the two vertical concrete air shafts is served by an operating fan cooler unit. In this mode of operation ~~each train is of 100 percent capacity (both air and water side) with~~ the idle train is serving as standby. ~~With 50 F service water entering temperature, each fan cooler has 2.341x10^6 Btu/hr heat removal capacity at 80 F entering air temperature. During this mode of operation two operating fan cooler~~ 37

Each shaft supply air damper is locked open and each nozzle damper is closed.

~~units will remove a total of 4.683×10^6 Btu/hr heat generated in the Containment. With 90 F service water entering temperature, each fan cooler has 2.83×10^6 Btu/hr. heat removal capacity and is rated at 125,000 cfm. During this mode of operation, two operating fan cooler units will supply at a total of 250,000 cfm and will remove a total of 5.66×10^6 Btu/hr. heat generated in the Containment.~~ 37

Insert "C" to Page 6.2.2-5

Each shaft supply damper is locked open and each nozzle damper is open. If containment average temperature continues to rise, the two standby fans of the fan coolers at elevation 236 ft. will be manually energized to operate at full speed and the nozzle dampers will remain closed.

and the other fan is on standby.

Each shaft supply damper is locked open and each nozzle damper is closed.

additional coolers will be operated

b) When the ~~service water~~ containment average temperature is above ~~90~~ 118 F ~~Two~~ One of the two fan cooler units located at floor Elevation 236 ft. will operate with ~~both~~ fans of the units running at full speed. ~~Each of the two vertical concrete air shafts is served by one of these two fan coolers.~~ The other two fan cooler units located at Elevation 286: will ~~be manually energized to~~ operate with ~~one~~ both fans per unit operating at full speed. ~~This operating mode is required to remove the excess heat generated by the CRDM ventilation system. Cooling air from these two coolers will be directed to the operating floor by automatic closing, on SIS, of pneumatically operated dampers at the concrete air shaft and by opening dampers at post-accident discharge nozzles. During this mode of operation both Trains A and B will be operating. With 95 F service water entering temperature, each fan cooler, operating with two fans at full speed, has 2.28x10^6 Btu/hr. heat removal capacity and is rated at 125,000 cfm. During this mode of operation, all four operating fan coolers will supply a total of 375,000 cfm and will remove a total of 7.3x10^6 Btu/hr. heat generated in the Containment.~~

Insert "C" from Page 6.2.2-4a

~~By means of a distribution network~~, Air is supplied to the steam generator and pressurizer subcompartments, the operating floor, the ground floor and the mezzanine floor. Figures 6.2.2-10 through 6.2.2-16 describe the plan and elevation drawings of the Containment showing the routing of air ~~flow guidance~~ distribution ductwork. A portion of supply air is tapped to serve the Reactor Support Cooling System and Primary Shield Cooling System described in Section 6.2.2.2.3.

Insert "B" to Page 6.2.2-3

~~The non-nuclear safety containment fan coil units operate in the following manner:~~

their air is directed to Reactor Coolant Pump and Steam Generator compartments.

The ~~system consists of~~ three non-nuclear safety fan-coil units are all located at the same elevation. These units are required to operate during normal plant operating conditions only. The fan-coil units are served by the Service Water System. A detailed description of Service Water System is given in Section 9.2.1. Each unit has cooling coil section and two one hundred percent capacity, direct driven, vane axial fans. ~~Each fan coil unit is rated at 91,000 cfm and will have the following cooling capacity based on entering service water temperature:~~ Unit performance is shown in Table 6.2.2-1.

Insert "A" to Page 6.2.2-4

~~a) With 50 F service water entering temperature, each fan coil unit has 2.082x10^6 Btu/hr heat removal capacity at 80 F entering air temperature. During this operation all three operating fan coil units will remove a total of 6.246x10^6 Btu/hr heat generated in the Containment.~~

~~b) With 90 F service water entering temperature, each fan-coil unit has 2.19x10^6 Btu/hr. heat removal capacity. During this operation all three operating fan coil units will supply a total of 273,000 cfm and will remove a total of 6.57x10^6 Btu/hr. heat generated in the Containment.~~

~~c) With 95 F service water entering temperature, each fan-coil unit has 1.866x10^6 Btu/hr. heat removal capacity. During this operation, all three operating fan coil units will supply a total of 273,000 cfm and will remove a total of 5.59x10^6 Btu/hr. heat generated in the Containment.~~

~~Air from the fan-coil units is directed to the RCP and steam generator subcompartments.~~

c) With (2) safety related fan cooler units and (3) non-safety related fan coil units operating at a service water temperature of 50 F, their total heat removal capacity is approximately ~~4,683,000 Btu/hr and 6,246,000 Btu/hr respectively for a total of~~ 11.1×10^6 ~~10,929,000~~ Btu/hr. These capacities are based on air entering the units at 80 F DB and between 48 F and 67 F WB. 37

The containment heat gain is 12.7×10^6 ~~11,902,100~~ Btu/hr. This includes heat contributed from equipment, lighting, piping, motors as well as fan motors. ~~These loads are based on a containment ambient of 120F and would be conservative for an ambient of 80 F. In addition, solar and transmission from the outdoors are not included since this omission constitutes conservatism.~~

Since heat gain is greater than the heat removal rate the temperature in the Containment cannot fall below 80 F.

6.2.2.2.2 Containment Spray System (CSS)

6.2.2.2.2.1 Functional Description

The purpose of the CSS is to spray borated sodium hydroxide solution into the Containment to cool the atmosphere and to remove the fission products that may be released into the containment atmosphere following a LOCA or MSLB. A summary of the design and performance data for the CSS is presented in Section 6.2.1. The fission product removal effectiveness and the pH control of the containment sump water of the CSS is described in Section 6.5.2.

6.2.2.2.2.2 Design Description

The CSS consists of two independent and redundant loops each containing a spray pump, piping, valves, spray headers, and spray valves. Figure 6.2.2-1 provides the process flow and instrumentation details of the system.

The operation of the CSS is automatically initiated by the containment spray actuation signal (CSAS) which occurs when a containment pressure of 12.0 psig (HI-3 signal) is reached. Section 7.3 describes the design bases criteria for the CSAS. Upon receipt of a CSAS, the containment spray pumps start operation and the containment spray isolation valves open.

The CSS has two principal modes of operation which are:

a) The initial injection mode, during which time the system sprays borated water which is taken from the refueling water storage tank (RWST). Section 6.2.2.3.2.3 describes the criteria used for sizing the RWST.

b) The recirculation mode, which is initiated when low-low level is reached in the RWST. Pump suction is transferred from the RWST to the containment sump by opening the recirculation line valves and closing the valves at the outlet of the refueling water storage tank. This switch over is accomplished automatically. See Section 7.3 for further details.