IE Files

AUG 1 1 1978

Docket Nos. 50-269/270/287

Mr. W. O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
422 S. Church Street
Charlotte, NC 28242

Dear Mr. Parker:

On August 10, 1978, Mr. F. J. Jablonski advised your Messrs. R. L. Gill and M. Curtis of NRC's plan to conduct confirmatory environmental qualification tests on electrical connector assemblies, and that Viking connectors of the type installed at Oconee Nuclear Generating Station Units 1, 2 and 3 have been tentatively selected to be tested.

At this time it is our intent to perform the qualification tests in accordance with certain information provided by you, to the Director, Office of Inspection and Enforcement, Region II, in response to IE Bulletin No. 77-05. In order for us to accurately duplicate the test conditions, further information concerning Viking connectors used in Class 1E circuits is requested from you including:

1. Part numbers (both male and female)
2. Drawings (Viking/Duke Power)
3. Viking Test Report No. QTP-124
4. All other qualification test reports (LOCA, radiation, aging, etc.)
5. Viking instruction books
6. Physical samples of the connectors
7. Purchase specifications/orders
8. Identification of cables used with the connectors (manufacturer, jacketing and insulating materials, number of conductors, size, etc.)
9. Connector/cable - assembly/termination procedures

50-269, 270, 287

OFFICE ➤						
SURNAME ➤						
DATE ➤						

To expedite our obtaining the information, we request a meeting at your corporate headquarters with members of your technical staff who have knowledge of Viking connectors. The meeting is requested at your staff's earliest convenience but on or before August 17, 1978. Please advise Mr. F. Jablonski of the meeting date and time. His telephone number is 312-858-2660.

Sincerely,

/S/

Harold D. Thornburg
Director
Division of Reactor
Construction Inspection
Office of Inspection and Enforcement

bcc:
RII Oconee Project Inspector

OFFICE ➤	IE:RIII	RCI:IE	AD:RCI:IE	D:RCI:IE		
SURNAME ➤	FJJablonski:	WRRutherford	GWReinmuth	HDThornburg		
DATE ➤	8/ /78	8/ /78	8/ 11 /78	8/ /78		

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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DECEMBER 27 1979

Docket No. 50-269

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

We have completed our review of your submittals dated October 18, November 21, November 27 and December 17, 1979, which provided your commitments and schedule for implementation of the TMI-2 Lessons Learned Task Force Recommendations for the Oconee Nuclear Station Unit No. 1. We have determined from our review and in discussions with your staff that you have committed to implement each Category A requirement (excluding requirement 2.1.7.a) prior to plant operation after December 31, 1979. I take this occasion to acknowledge, personally, your responsiveness to our requests.

Although you have committed to complete each item on schedule, I recognize that unforeseen circumstances may arise, such as equipment delays that may dictate that you revise your stated position. Should such an occasion arise, we request that you notify us promptly, within 48 hours, of the circumstances involved and any change to your present commitment.

As stated in my letter of October 30, 1979, the NRC will be performing a post-implementation review of your method of implementing each requirement. You should submit for our review by the required completion date, appropriate documentation of your actions. Category B lessons learned requirements, those scheduled for implementation by January 1, 1981, will be the subject of future correspondence.

Sincerely,

Original Signed by
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

AtD/DOR
DEisenhut
12/21/79

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OFFICE	ORB#4:DOR	C-ORB#4:DOR	ORB#4:DOR	A-AD-ORR:DOR	OELD	D:NRR
SURNAME	MFairtile/cb	RReid	CNelson	WGammill		HDenton
DATE	12/21/79	12/21/79	12/21/79	12/21/79	12/ /79	12/21/79

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

December 27, 1979

Docket No. 50-269

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

We have completed our review of your submittals dated October 18, November 21, November 27 and December 17, 1979, which provided your commitments and schedule for implementation of the TMI-2 Lessons Learned Task Force Recommendations for the Oconee Nuclear Station Unit No. 1. We have determined from our review and in discussions with your staff that you have committed to implement each Category A requirement (excluding requirement 2.1.7.a) prior to plant operation after December 31, 1979. I take this occasion to acknowledge, personally, your responsiveness to our requests.

Although you have committed to complete each item on schedule, I recognize that unforeseen circumstances may arise, such as equipment delays that may dictate that you revise your stated position. Should such an occasion arise, we request that you notify us promptly, within 48 hours, of the circumstances involved and any change to your present commitment.

As stated in my letter of October 30, 1979, the NRC will be performing a post-implementation review of your method of implementing each requirement. You should submit for our review by the required completion date, appropriate documentation of your actions. Category B lessons learned requirements, those scheduled for implementation by January 1, 1981, will be the subject of future correspondence.

Sincerely,

H R Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

cc: See next page

Duke Power Company

Mr. William L. Porter
Duke Power Company
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806 15th Street, N.W.
Washington, D. C. 20005

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Mr. Francis Jape
U. S. Nuclear Regulatory Commission
P. O. Box 7
Seneca, South Carolina 29678

Mr. F ert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 420, 7735 Old Georgetown Road
Bethesda, Maryland 20014

Manager, LIS
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

Docket
50-269

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 25, 1978

ALL POWER REACTOR LICENSEES

Gentlemen:

This letter is being sent to all licensees authorized to operate a nuclear power reactor and to all applicants with application for a license to operate a power reactor (FSAR docketed).

The NRC has scheduled regional meetings to discuss the upgraded guard qualification and training requirements published in the Federal Register on August 23, 1978 and the guidance on this requirement contained in NUREG 0219 as well as the contingency planning requirements published in the Federal Register on March 23, 1978. An agenda, the dates and locations of the meetings, and a registration form are enclosed. Please complete the registration form and return it to Mr. Frank G. Pagano, Jr., Chief, Reactor Safeguards Development Branch, Nuclear Regulatory Commission, Washington, D.C. 20555. Hotel arrangements are the responsibility of each attendee.

For further information or comments, please contact Tom McKenna of my staff on (301) 492-7846.

Sincerely,

James R. Miller

James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

Enclosures:
1. Meeting Agenda
2. Meeting Schedule & Locations
3. Registration Form

CC: Service List

MAH

R.

ENCLOSURE 1

IMPLEMENTATION OF 10 CFR 73 APPENDICES B AND C GUARD TRAINING AND CONTINGENCY PLANNING

Meeting Agenda

Sept. 27 - Atlanta; Oct. 3 - Chicago
Oct. 5 - San Francisco; Oct. 11 - Philadelphia

TIME	SPEAKER	SUBJECT
9:00 - 9:10	J. Miller	Introduction
9:10 - 9:20	V. Stello	NRC Safeguards Responsibility
9:20 - 9:30	F. Pagano	Why We Adopted This Approach
9:30 - 9:45		Coffee Break
9:45 - 10:30	T. McKenna	The Approach
10:30 - 11:15	T. McKenna	10 CFR 73 Appendix B Guard Training
11:15 - 11:45	J. Roe	Contingency Plans
11:45 - 12:00	R. Clark	NRR Staff Reviews
12:00 - 1:30		Lunch
1:30 - 3:00		Question/Answer
3:00 - 3:30	J. Miller/ V. Stello	Closing Remarks

ENCLOSURE 2

SCHEDULES AND LOCATIONS

Region II	September 27, 1978	Stadium Hotel* 450 Capitol Ave., SE Atlanta, GA 30312 (404) 688-1900
Region III	October 3, 1978	Ramada O'Hare 6600 North Mannheim Rd. Des Plains, IL 60018 (312) 827-5131
Region IV & V	October 5, 1978	San Francisco Airport Hilton P. O. 8355 San Francisco, CA 94128 (415) 589-0770
Region I	October 11, 1978	Valley Forge Holiday Inn 260 Goddard Blvd. King of Prussia, PA 19406 (215) 265-7500

*Special rate for reservations received before September 15, 1978.

ENCLOSURE 3

Registration Form

Implementation of 10 CFR 73 Appendix B
Security Personnel Qualification Training and
Equipment Requirements by Commercial Nuclear
Power Reactors

Regional Meeting

Date ______________________________

Place ______________________________

Utility Represented ______________________________

Individuals Attending:

Name ______________________________

Title ______________________________

Office Phone ______________________________

Name ______________________________

Title ______________________________

Office Phone ______________________________

Name ______________________________

Title ______________________________

Office Phone ______________________________

Name ______________________________

Title ______________________________

Office Phone ______________________________

RETURN THIS FORM BY SEPTEMBER 22, 1978 TO:

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
ATTN: Frank G. Pagano, Jr.
Washington, D. C. 20555

Duke Power Company

50-269
50-270
50-287

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806-15th Street, NW.,
Washington, D.C. 20005

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

Docket Nos: 50-269, 50-270, 50-287,
50-302, 50-312, 50-313,
50-346

TO ALL BABCOCK & WILCOX (B&W) OPERATING PLANTS
(EXCEPT THREE MILE ISLAND, UNIT 1)

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - BAW REPORT 1564, "INTEGRATED CONTROL SYSTEM RELIABILITY ANALYSIS"

By letters dated August 31, 1979, each B&W operating plant licensee indicated a general endorsement of B&W's generic report BAW-1564, "Integrated Control System Reliability Analysis."

Our joint review of this report with Oak Ridge National Laboratory has progressed sufficiently to assure ourselves that the recommendations that the report offers with regard to potential areas of improvement in ICS reliability are reasonable. Therefore, we request that you address these recommendations and discuss your followup action plans in this matter. Response to the enclosed request for additional information should be submitted within 14 days of receipt of this letter.

As part of the continuing review of this report, additional areas may be highlighted as requiring improvement. In that event, we will provide additional requests in these specific areas as necessary.

If you cannot meet this schedule or if you require any clarification of these matters please contact Mr. Robert Capra on (301) 492-7745.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosure:
As stated

cc: J. Anderson, ORNL
Met-Ed
See attached lists

OFFICE	B&OTF	B&OTF RAC	B&OTF	B&OTF	B&OTF	B&OTF	DOR:ORB#4
SURNAME	DTHatcher:jk	RACapra	WFKane	SIsrael	TMNovak	DFRoss	RWReid
DATE	11/5/79	11/5/79	11/ /79	11/ /79	11/5/79	11/6/79	11/7/79

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Enclosure

REQUEST FOR ADDITIONAL INFORMATION
ON REPORT BAW-1564

For the following recommendations from report BAW-1564,

- Describe how you plan to implement the recommendation; and
- The schedule for the implementation; or
- Basis for not implementing the recommendation at your plant

1. ICS-Related

   a. NNI/ICS power supply reliability

      NOTE: This area is of particular importance because it is listed as the source of most of the ICS input failures. The staff is not only concerned with the "reliability" of the power supplies, but also the design philosophy of the power supply implementation. It appears that power supply failures can lead to multiple problems such as the Rancho Seco event of March, 1978.

   b. Reliability of input signal from the NI/RPS system to the ICS - specifically, the RC flow signal.

   c. ICS/BOP system tuning, particularly feedwater condensate systems and the ICS controls.

      NOTE: Although this concern is related to tuning, it appears that more basic design and/or operational problems in the feedwater (and related) system may exist.

      Therefore, include a discussion of the following items:

      (1) Any particular operational (startup, etc.) problems experienced at your plant with respect to the ICS. Reference to previously submitted information is acceptable.

(2) Bases for operator intervention in place of automatic ICS action (including start-up, power operation and shutdown activities).

(3) Procedures used by the operator to perform the operation described in 1c(2) above.

(4) Additional training provided to the operator.

2. <u>Balance of Plant</u>

a. Main feedwater pump turbine drive minimum speed control - to prevent loss of main feedwater or indication of main feedwater.

b. A means to prevent or mitigate the consequences of a stuck-open main feedwater startup valve.

c. A means to prevent or mitigate the consequences of a stuck-open turbine bypass valve.

REGULATORY DOCKET FILE COPY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket Nos. 50-269, 50-270, 50-287
50-289, 50-302, 50-312
50-313, 50-346

OCT 5 1979

Robert Gill, Chairman
TMI-2 Effects Subcommittee
B&W Owner's Group
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Gill:

The NRC staff is reviewing the responses to bulletins 79-05C and 79-06C on the subject of the need for tripping of reactor coolant pumps for certain small break loss-of-coolant accidents. The conclusions reached by the PWR vendors in those responses vary to a considerable degree, and are highlighted in Table 1. To a degree these differences may be attributed to design differences. However, our present judgment is that the major differences are attributable to model differences, which are highlighted in Table 2. The various RCP trip criteria that have been proposed or are believed to be in place are presented in Table 3.

A variety of model features have been developed without the use of relevant experiments to support model justification, and no model has been demonstrated to be overall conservative by integral test comparison. There is apparent lack of agreement between vendors as to whether individual assumptions are conservative. Given these model variations, it is not surprising that the conclusions reached vary.

We have done some calculations with RELAP-4, Mod 7, and these are summarized in Table 4. These calculations, while useful to us, are not definitive, and cannot guide us as to what is a suitably conservative set of model assumptions.

Our concern is that during the pendency of traditional interactions between you and us (i.e., questions, positions, analyses, rebuttals, appeals, etc.) events are forcing a more prompt solution. I refer to the RCP trip at North Anna (a non-LOCA transient) the RCP trip at Prairie Island (a steam generator tube failure) and the Davis-Besse transient of September 27, 1979 which nearly set in motion the requirements of 79-05C.

The second half of the concern I have relates to HPI termination criteria. Table 5 lists several criteria proposed, in place, or thought to be in place. There is no apparent reason why the same set of safety considerations would lead us into more than one, uniform criterion for termination of HPI. Your assistance in achieving this desired state of uniformity is needed.

If the NRC were to adopt a Q&A approach to solving the problem separately, some candidate questions are enclosed in Table 6.

MA
4

Instead of withdrawing to the Table 6 approach, I suggest we meet soon and discuss, in an administrative sense, how the PWR regulated industry can close on these two issues. For further details regarding the issues described herein, as well as the planned meeting, Mr. Brian Sheron (301-492-7588) is available for further discussion.

Sincerely,

D. F. Ross, Jr., Director
Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation

Enclosures:
Table 1 - Conclusions Reached by PWR Vendors in Response to Bulletins 79-05C and 79-06C
Table 2 - Differences During SBLOCA with Pumps Running
Table 3 - RCP Criteria Proposed or In-Place at Plants
Table 4 - Staff Calculations for PWR Vendors
Table 5 - HPI Termination Criteria Proposed or In-Place at Plants
Table 6 - Typical Questions on RCP Trip and HPI Termination

cc: Babcock & Wilcox Company
B&W Licensees

Identical Letters sent to:
Ed Scherer, CE
Tom Anderson, W
George Liebler, CE Owner's Group
Cordell Reed, W Owner's Group
James H. Taylor, B&W

TABLE 1

CONCLUSIONS REACHED BY PWR VENDORS IN RESPONSE TO BULLETINS 79-05C AND 79-06C

	BREAK LOCATION	BREAK SIZE	MAXIMUM AVAILABLE TIME FOR PUMP TRIP	EFFECT OF CONTINUOUS PUMP OPERATION	EFFECT OF TRIPPING ONE PUMP IN EACH LOOP
B&W	RESULTS NOT SENSITIVE DUE TO HOMOGENEOUS MODELING ASSUMPTION	LIMITING BREAK SIZE ABOUT 0.02 - 0.2FT^2	~3 MINUTES (BASED ON PRELIMINARY CALCULATIONS)	ACCEPTABLE CORE COOLING	NO EVALUATION
CE*	FOUND HOT LEG BREAKS LIMITING/ SOME COLD LEG BREAKS COULD EXCEED 2200^{0}F	LIMITING BREAK SIZE ABOUT .02 - .1 FT^2	6 MINUTES AFTER TRIP + SIAS FOR EM ANALYSIS 10 MINUTES AFTER TRIP + SIAS FOR BE ANALYSIS	0.1 FT^2 BREAK IN HOT LEG LEADS TO PCT's $>$ 2200^{0}F	ACCEPTABLE CORE COOLING FOR BE ANALYSIS PROVIDED TWO PUMPS TRIPPED WITHIN 5 MINUTES AFTER BREAK
W	COLD LEG BREAKS LIMITING, NO HOT LEG BREAKS ANALYZED RESULTED IN PCT'S $>$ 2200^{0}F	LIMITING BREAK SIZE .02-.05 FT^2	10 MINUTES FOR ALL PLANT TYPES (2, 3, 4 LOOPS)	ACCEPTABLE CORE COOLING	NO EVALUATION

*CE ANALYSES PERFORMED FOR PLANTS WITH 200 PSI SIT'S. 1200 PSI HPSI PUMPS. ANALYSES CONSIDERED CONSERVATIVE WRT PLANTS WITH 600 PSI SITS AND/OR HIGHER HEAD HPSI PUMPS.

TABLE 2

MODEL DIFFERENCES DURING SBLOCA WITH PUMPS RUNNING

Item	MODEL W	CE	B&W	RELAP/MOD-7
Cold Leg Pump Discharge Pipe	Stratified Flow	Homogeneous flow	Homogeneous flow	Heterogeneous flow
Downcomer	Heterogeneous model	Model switches from homogeneous to heterogeneous model when drift velocity criteria met.	Homogeneous flow	Heterogeneous flow
Core	Heterogeneous flow	Heterogeneous flow	Homogeneous flow	Heterogeneous flow
Hot Leg Pipe	Homogeneous flow	Heterogeneous flow CE represents the hot leg with two flow paths. This representation allows counter-current flow in horizonal paths.	Homogeneous flow	Heterogeneous flow. No counter-current flow allowed
Steam Generator Hot Side Tubes	Homogeneous flow	Drift flux model - allows liquid fallback to hot leg if possible	Homogeneous flow	Heterogeneous flow no vertical slip/ fluid runback to hot leg
Steam Generator Cold Side Tubes	Homogeneous flow	Homogeneous flow	Homogeneous flow	Heterogeneous flow no vertical slip
Cold Leg Loop Seal (suction pipe)	Homogeneous flow	Homogeneous flow	Homogeneous flow	Homogeneous flow

Model/Method	W	CE	B&W	EG&G Idaho
ECC Injection	No injection assumed in broken loop for cold leg breaks	No spillage assumed for hot leg breaks - no injection assumed in broken loop for cold leg breaks	~ 30% spillage of water injected in broken loop for cold leg breaks	Consistent with vendor assumptions
ECC Injection Location	Downcomer/lower plenum node (cold leg by design)	Downcomer (cold leg by design)	Cold Leg (cold leg by design)	W - upper downcom. CE - cold leg B&W - cold leg
Quench Behavior during Recovery	No carryover accounted for	No carryover accounted for	No carryover accounted for	No carryover accounted for
Steam Super-Heat Calcu-lation	Superheating considered (description proprietary)	12 axial coolant nodes in core. Superheating of each node allowed	No superheat calculated due to single control volume model of core. All core heat added to liquid phase. Separate heat-up model cal-culates super-heat but uses CRAFT mixture level.	3 axial coolant nodes in core. Superheating of each node allowed.
Core fluid quality	Thermodynamic equilibrium assumed - actual quality not calculated	Thermodynamic equilibrium assumed - actual quality not calculated	Thermodynamic equilibrium assumed - actual quality not calculated	Thermodynamic equilibrium assumed - actual quality not calculated

TABLE 3

RCP CRITERIA PROPOSED
OR
IN-PLACE AT PLANTS

NRC CRITERIA

A. Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating reactor coolant pumps.

B. Two licensed operators.

WESTINGHOUSE

A. Westinghouse proposed criteria is that reactor coolant pumps be tripped on low reactor coolant pressure of 1250 psig after verification of high head safety injection.

B. Responses from licensees are under review - most adopt NRC criteria.

COMBUSTION ENGINEERING

A. Stop two reactor coolant pumps, one in each loop after it has been verified that the rods have been inserted fully for five seconds. If this action has not been completed within the first five minutes after SIAS, all reactor coolant pumps must be stopped within 10 minutes after SIAS.

B. Responses from licensees are under review - some adopt NRC criteria.

BABCOCK & WILCOX

A. B&W proposed criteria is that reactor coolant pumps be tripped on initiation of HPI caused by low reactor coolant system pressure.

B. Licensees are currently using NRC criteria*.

*Most recent proposals for future automatic pump trip have the signal from safety injection only (not low pressure or current).

TABLE 4

STAFF CALCULATIONS FOR PWR VENDORS

PWR VENDOR	BREAK SIZE	BREAK LOCATION	RC PUMPS TRIPPED	FLUID MODEL ASSUMPTIONS
WESTINGHOUSE	4 INCH DIAMETER	COLD LEG	AT REACTOR SCRAM	HETEROGENEOUS
	4 INCH DIAMETER	COLD LEG	NOT TRIPPED	HOMOGENEOUS
				(BEFORE TRIP / AFTER TRIP)
	4 INCH DIAMETER	COLD LEG	AT 511 SECONDS (WORST CASE)	HOMOGENEOUS/HETEROGENEOUS
	4 INCH DIAMETER	COLD LEG	AT 760 SECONDS	HOMOGENEOUS/HETEROGENEOUS
	1 INCH DIAMETER	COLD LEG	AT REACTOR SCRAM	HETEROGENEOUS
	1/2 INCH DIAMETER	COLD LEG	AT REACTOR SCRAM	HETEROGENEOUS
	1 STUCK PORV	PRESSURIZER	AT REACTOR SCRAM	HETEROGENEOUS
COMBUSTION ENGINEERING	0.1 SQUARE FEET	COLD LEG	AT REACTOR SCRAM	HETEROGENEOUS
	0.02 SQUARE FEET	COLD LEG	AT REACTOR SCRAM	HETEROGENEOUS
	1 STUCK PORV	PRESSURIZER	AT REACTOR SCRAM	HETEROGENEOUS
	2 STUCK PORV'S	PRESSURIZER	AT REACTOR SCRAM	HETEROGENEOUS
BABCOCK & WILCOX	0.1 SQUARE FEET	COLD LEG	AT REACTOR SCRAM	HETEROGENEOUS
	0.07 SQUARE FEET	COLD LEG	AT REACTOR SCRAM	HETEROGENEOUS
	1 STUCK PORV	PRESSURIZER	AT REACTOR SCRAM	HETEROGENEOUS

TABLE 5

HPI TERMINATION CRITERIA
PROPOSED OR INPLACE AT PLANTS

NRC CRITERIA

Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

(1) Both low pressure injection (LPI) pumps are in operation and flowing at a rate in excess of 1000 gpm each and the situation has been stable for 20 minutes, or

(2) The HPI system has been in operation for 20 minutes*, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

BABCOCK & WILCOX

"A. The LPI System is in operation and flowing at a rate in excess of 1000 gpm in each line and the situation has been stable for 20 minutes.

or

B. All hot and cold leg temperatures are at least 50^{0} below the saturation temperature for the existing RCS pressure, the hot leg temperatures are not more than 50^{0} hotter than the secondary side saturation temperature, and the action is necessary to prevent the indicated pressurizer level from going off-scale high. If 50^{0} subcooling cannot be maintained, the HPI shall be reactivated. The degree of subcooling beyond 50^{0} and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity."

*Current staff view is that this time period should be deleted (see Table 6)

C. All licensees have adopted these criteria, for the most part.

WESTINGHOUSE - (UNDER REVIEW)

The criteria proposed by W are as follows:

"In the event of a spurious safety injection signal, the sequence of reactor trip, turbine trip and safeguards actuation will occur.

The operator must assume that the safety injection signal is non-spurious unless the following are exhibited:

a. Normal readings for containment temperature, pressure, radiation and recirculation sump level AND
b. Normal readings for auxiliary building radiation and ventilation monitoring AND
c. Normal readings for steam generator blowdown and condenser air ejector radiation AND
d. All steam generators exhibit normal pressure and water level following reactor trip and safety injection actuation (similar to a reactor trip from normal conditions).

IF all of the symptoms a through d above are met, THEN secure Safety Injection when the following are exhibited:

a. Reactor coolant pressure is greater than 2000 psig and increasing AND
b. Pressurizer water level is greater than 50% of span AND
c. Water level in at least one steam generator is in the narrow range span, or in the wide range span at a level sufficient to assure that the U-tubes are covered."

COMBUSTION ENGINEERING - (UNDER REVIEW)

"After any SIAS, operate the SIS for at least 20 minutes and until RCS hot and cold temperatures are at least 50^{o}F below the saturation temperature for the RCS pressure unless the cause of the SIAS has been verified to be an inadvertent actuation. If 50^{o} subcooling cannot be maintained after the system has been stopped, the high pressure injection system must be restarted."

TABLE 6

TYPICAL QUESTIONS ON RCP TRIP
AND
HPI TERMINATION

1. We believe that if it is <u>necessary</u> to trip the RCP for small break LOCA, then it should be done automatically. Further, such automatic action should be from circuits of high quality usually associated with the reactor protection system. This would provide a high likelihood that the action would take place when needed, and at the same time would reduce the likelihood of unneeded trips. Since you have indicated that, for some breaks, RCP trip is necessary, it follows that RCP automatic trip circuits should be used. For this reason it is requested that you comply with the following:

   a. Identify an alternate or coincident signal (to SIAS) that could be used to initiate RCP trip.

   b. Support the choice of this alternate signal by showing that it will conservatively protect the core during the most limiting LOCA (in the critical break range) and that it will not be actuated for the most limiting non-LOCA event.

   c. Provide an auotmatic trip system that meets IEEE-279 requirements; particularly with respect to meeting single failure for tripping all the pumps and with minimum likelihood for spurious actuation. Submit a proposed Technical Specification to include limiting conditions of operation and surveillance requirements

   d. Because of recent non-LOCA events which actuated the HPI, the staff believes that simple HPI termination or throttling criteria tied to subcooling of the fluid in the core and hot and cold legs would be appropriate for hypothesized events, regardless of pressure in the primary system. Our examination of pressurizer level traces obtained during recent transients indicates that this parameter should be considered

as a coincident instruction to the operator. A simple RCS pressure criteria is not useful during SG tube failure, SLB, overcooling transients, especially if one must also consider potential pressure vessel integrity problems.

Provide arguments why subcooling should not be used or should be tempered by other indications.

2. Adding a protective feature such as RCP trip must have some overall effect on the risk characteristic of the reactor. For the general class of LOCAs, it seems reasonable that:

   a. for large LOCA there is no effect;

   b. for most small break LOCAs the pumps are expected to be powered (i.e., low likelihood of LOOP) and continued RCP running would offset some degraded ECCS conditions; and,

   c. for a small range of LOCAs continued pump running may be counterproductive.

   However, for non-LOCA events, especially transients, it seems reasonable to postulate that spurious RCP trip increases risk.

   Indicate whether adding the RCP auto trip feature poses a significant shift in the risk characteristic.

3. When RCPs are tripped due to a SI signal, there is (for some designs) no pressurizer spray. This, in combination with continuous HPI operation, may lead to PORV operation. Thus a transient (such as overcooling) may progress to a degree not heretofore contemplated. Indicate the effect of no pressurizer spray on pressure control during the set of transients. How can auxiliary spray be supplied, if needed, in the presence of an SI signal?

4. It seems like auto RCP trip, if based on a pressure signal in whole or in part, should be based on the lowest possible pressure. Presumably this trip pressure would be below the SI setpoint, but above the steam generator safety valve

setpoint. Please comment on the parameters needed to generate a RCP trip signal, and how they might respond to non-LOCA events.

5. HPI termination based on adequate subcooling would avoid undesirable formation of voids in the RCS. If, however, the pressurizer were low in level it would be desirable to continue adding coolant. Thus low pressurizer level would be a parameter that one would logically monitor before terminating HPI. If the steam generators have <u>all</u> boiled dry (from arbitrary causes) then the only available heat removal path will be through the PORV. This is generally an ICC* situation. In this instance the PORV will be opened and the parameters of subcooling, pressurizer level, and RCS pressure probably have no further influence on the courses of action.

   Thus we believe that subcooling should be a permissive before reducing HPI flow that may have been automatically actuated. Common sense says that the makeup systems should continue in operation until pressurizer level is restored. (See question 1-d).

   Please consider why these parameters are not adequate for all situations except ICC.

6. Provide verification for the individual modeling assumptions used in your analyses of SBLOCAs with the pumps on; in particular the flow regime modeling. What verification is proposed for integral model assessment?

   Alternatively, explain why your present model is appropriately conservative to bound the analysis uncertainties or propose the additional conservatisms needed to assure the uncertainties are adequately bounded.

7. Recent experience has shown that many non-LOCA events will exhibit front-end

*Inadequate Core Cooling

symptoms of a SBLOCA. In particular, the events at North Anna 1 and Prairie Island indicate that loss of the RCPs can increase the difficulty in bringing the plant to a hot standby condition. The staff therefore feels that many of the events analyzed in Chapter 15 of SARs need to be reanalyzed to account for RCP trip in the event of SI initiation (i.e., SG tube rupture, secondary side overcooling events, etc.). Provide alternative criteria for SBLOCA pump trip which significantly decrease the probability of requiring the trip for non-LOCA events, or provide reanalyses showing the acceptability of affected Chapter 15 events assuming pump trip.

Docket
50-287

MAR 22 1979

REGULATORY DOCKET FILE COPY

All Power Reactor Licensees

Gentlemen:

This letter is being sent to all licensees or applicants authorized to operate or construct a nuclear power reactor.

In May 1978, the NRR staff held a meeting with the nuclear power reactor industry to discuss the implementation of the physical security requirements and other topics of interest to licensees in this area. Since the May 1978 meeting, additional requirements as well as Commission decisions regarding the implementation of the nuclear power reactor physical security program and other proposed rules and regulations have been issued.

To maintain a forum for the continuation of dialogue between the staff and industry in order to provide a more uniform understanding of the requirements of 10 CFR Part 73, the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, has scheduled a meeting on June 11-12, 1979 at the Hilton Inn, 1901 University Boulevard, N.E., Albuquerque, New Mexico. This meeting was scheduled following discussions with utilities and industry groups who indicated that a general meeting should be held following the guard training workshops which will be concluded in early May.

Individuals attending the meeting will be responsible for making their own travel and lodging accommodations for the meeting. To assist the attendees in lodging accommodations, the Hilton Inn will hold a block of rooms for meeting attendees only until May 28, 1979. Experience has shown that attendees may not be able to obtain accommodations at the Hilton Inn after this cut-off date. Rates for single rooms range from \$30 to \$39, and from \$38 to \$47 for doubles.

Pre-Registration will be held in the Garden Room of the Hilton Inn on Sunday, June 10, 1979 from 7:00 p.m. to 9:00 p.m. Registration will also be available from 7:00 a.m. to 11:00 a.m. on Monday, June 11, 1979 at the entrance to the International Room.

The meeting will consist of 1 full day of presentations by the staffs of the NRC and Los Alamos Scientific Laboratories with question and answer periods for the licensees and applicants, and 1 day of topical discussion on the development and implementation of the guard training plans.

A tentative agenda is enclosed.

Please indicate on the enclosed registration form those individuals who will be attending the meeting and return the form to the address below by May 28, 1979:

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
ATTN: Frank G. Pagano, Jr.
Washington, DC 20555

Please provide all the information requested on the registration form.

For further information, contact Dale A. Kers on (301) 492-7846.

Sincerely,

James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

cc: Service List

MEETING AGENDA

June 10, 1979

7:00 p.m. - 9:00 p.m.	PRE-REGISTRATION - GARDEN ROOM

June 11, 1979

7:00 a.m. - 11:00 a.m.	REGISTRATION
8:30 a.m. - 8:40 a.m.	OPENING REMARKS
8:40 a.m. - 9:00 a.m.	NRC SAFEGUARDS PROGRAM OVERVIEW
9:00 a.m. - 9:20 a.m.	COMMISSION ACTIONS ON TWO-MAN RULE, COMPARTMENTALIZATION AND SEARCHING OF EMPLOYEES
9:20 a.m. - 9:45 a.m.	VITAL AREA ANALYSIS UPDATE
9:45 a.m. - 10:15 a.m.	BREAK
10:15 a.m. - 11:00 a.m.	PERSONNEL CLEARANCES AND REWRITE OF ANSI N18.17
11:00 a.m. - 11:20 a.m.	STATUS OF PART 73.55 IMPLEMENTATION
11:20 a.m. - 12:00 p.m.	CONTINGENCY AND GUARD TRAINING PLANS REVIEW PROCEDURES
12:00 p.m. - 1:30 p.m.	LUNCH
1:30 p.m. - 1:45 p.m.	PROPOSED LEGISLATION FOR PROTECTING LWR PHYSICAL SECURITY PLANS
1:45 p.m. - 2:20 p.m.	LITIGATING PHYSICAL SECURITY - THE DIABLO CANYON EXPERIENCE
2:20 p.m. - 2:30 p.m.	UPDATE ON CURRENT RULEMAKING
2:30 p.m. - 2:40 p.m.	INTERNATIONAL SAFEGUARDS (IAEA)
2:40 p.m. - 3:00 p.m.	GUARD TRAINING WORKSHOP SUMMARY
3:00 p.m. - 3:30 p.m.	BREAK
3:30 p.m. - 4:00 p.m.	I&E EXPERIENCE WITH APPROVED PHYSICAL SECURITY PROGRAMS
4:00 p.m. - (open)	LICENSEE QUESTIONS
	CLOSING REMARKS AND INTRODUCTION TO TOPICAL DISCUSSIONS

June 12, 1979

IN PROVIDING FOR INTERACTION BETWEEN THE NRC STAFF AND LICENSEE IN THE DEVELOPMENT AND SUBSEQUENT IMPLEMENTATION OF THE 10 CFR PART 73, APPENDIX B, REQUIRED TRAINING AND QUALIFICATION PLANS ON AN INDIVIDUAL BASIS, NRR WILL PROVIDE TECHNICAL ASSISTANCE TO ANSWER QUESTIONS IN THE FOLLOWING AREAS:

- o APPENDIX B REQUIREMENTS
- o JOB ANALYSIS
- o TEST DEVELOPMENT AND PLANNING
- o TEST DEVELOPMENT
- o TEST VALIDATION
- o TRAINING AND QUALIFICATION PLAN FORMAT AND ASSEMBLY

ATTENDEES CAN ATTEND ALL OR ONLY THOSE TOPICAL DISCUSSIONS WHICH THE INDIVIDUAL HAS AN INTEREST IN OR QUESTIONS CONCERNING THE DEVELOPMENT OF THEIR FACILITIES PROGRAM.

8:30 a.m. - 9:00 a.m.	OPENING REMARKS
9:00 a.m. - 10:00 a.m.	TOPICAL DISCUSSIONS
10:00 a.m. - 10:30 a.m.	BREAK
10:30 a.m. - 12:00 p.m.	TOPICAL DISCUSSIONS
12:00 p.m. - 1:30 p.m.	LUNCH
1:30 p.m. - (open)	TOPICAL DISCUSSIONS

MEETING REGISTRATION FORM

Implementation of 10 CFR Part 73 Requirements For Physical Protection of Licensed Activities In Nuclear Power Reactors Against Industrial Sabotage

June 11 - 12, 1979

Name/Title ______________________________

Organization ______________________________

Address ______________________________

Phone ______________________________

Name/Title ______________________________

Organization ______________________________

Address ______________________________

Phone ______________________________

Name/Title ______________________________

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cc:
Mr. William L. Porter
Duke Power Company
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806 15th Street, N.W.
Washington, D. C. 20005

U. S. Nuclear Regulatory Commission
Region II
Office of Inspection and Enforcement
ATTN: Mr. Francis Jape
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 420, 7735 Old Georgetown Road
Bethesda, Maryland 20014

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201 South Spring Street
Walhalla, South Carolina 29691

FOR THE RECORD:

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THE FILES

[7590-01-M]

DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Request for public comment on NUREG-0531 "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants" February 1979.

SUMMARY: On September 14, 1978, the Nuclear Regulatory Commission established a new Pipe Crack Study Group. The Group was to evaluate recent pipe and safe end cracking experience relative to previous staff conclusions and recommendations. The NRC seeks public comment on the report which summarizes the Group's review and conclusions.

DATES: The public comment period expires May 15, 1979.

FOR FURTHER INFORMATION CONTACT:

Darrell G. Eisenhut, Deputy Director for Operating Reactors, Division of Operating Reactors, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. (Phone: 301-492-7221)

SUPPLEMENTARY INFORMATION: In 1975, a Pipe Cracking Study Group was established by the United States Nuclear Regulatory Commission (USNRC) to review intergranular stress-corrosion cracking (IGSCC) in Boiling Water Reactors (BWRs). The Group reported its findings concerning stress-corrosion cracking in by-pass lines and core spray piping of austenitic stainless steel in a report, *Technical Report—Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants* (NUREG-75/067).

During 1978, IGSCC was reported for the first time in large-diameter piping in a BWR. This discovery, together with questions concerning the capability of ultrasonic detection methods to detect small cracks, led to the formation of a new Pipe Crack Study Group (PCSG) by USNRC on September 14, 1978.

The charter of the new PCSG was to specifically address the five following questions:

"1. The significance of the cracks discovered in large-diameter pipes relative to the conclusions and recommendations set forth in the referenced report (NUREG-75/067) and its implementation document, NUREG-0313;

2. Resolution of the concerns raised over the ability to use ultrasonic techniques to detect cracks in austenitic stainless steel;

3. The significance of cracks found in large-diameter sensitized safe ends and any recommendations regarding the current NRC program for dealing with this matter;

4. The potential for stress corrosion cracking in PWRs;

5. Examine the significance of cracking in the Inconel safe ends that has been experienced at the Duane Arnold Operating Facility, and develop any recommendations regarding NRC actions taken or to be taken."

The PCSG limited the scope of the study to BWR and PWR piping and safe ends attached to the reactor pressure vessel. The PCSG reviewed existing information—either that contained in written records or that collected through meetings in this country and in foreign countries. The specific areas considered are presented in the chapters of this report:

- BWR Cracking Experience and Corrective Actions
- PWR Cracking Experience and Corrective Actions
- Metallurgy Associated with Pipe Cracking
- Reactor Coolant Chemistry
- Pipe Configuration and Stress Levels
- Duane Arnold Safe-End Cracking
- Methods of Detecting Cracks
- Significance of Cracks
- Recent Development Relevant to Control and Detection of IGSCC

The review of these topics in the context of changes occurring since the preparation of NUREG-75/067 led to the preparation of specific conclusions and recommendations relevant to the current status of IGSCC, the significance of the problem, and the reliability of detection and measures available to correct or minimize IGSCC in existing and future plants. These conclusions and recommendations are presented in the newly issued PCSG report.

The NRC staff will review the Study Group report and its conclusions/recommendations and the public comments received during this comment period. Following this review, the staff will decide what further actions, if any, are required for the licensing and operation of reactors.

Requests for a single copy of the report should be made in writing to U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

Comments on this report should be sent to the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Duputy Director, Division of Operating Reactors. The comment period expires May 15, 1979. Copies of all comments received will be available for examination in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C.

Dated at Bethesda, Md., this 6th day of March, 1979.

For the Nuclear Regulatory Commission.

VICTOR STELLO, Jr.,
Director, Division of Operating Reactors, Office of Nuclear Reactor Regulation.

[FR Doc. 79-7705 Filed 3-12-79; 8:45 am]

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Docket

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 15, 1979

REGULATORY DOCKET FILE COPY

50-270

ALL POWER REACTOR LICENSEES

Gentlemen:

On September 14, 1978, the Nuclear Regulatory Commission established a new Pipe Crack Study Group which was to evaluate recent pipe and safe end cracking experience relative to previous staff conclusions and recommendations. The bases for establishing the new Study Group were (1) the discovery of cracks in the inner surface of large-diameter austenitic stainless steel piping (recirculation lines) in a BWR and (2) questions concerning the capability of ultrasonic detection methods to detect small cracks.

The new PCSG reviewed existing information that either was contained in written records or had been collected through meetings in this country and in foreign countries. The review was in the context of changes occurring since the preparation by the original Pipe Cracking Study Group of NUREG-75/067 "Technical Report--Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants". The conclusions and recommendations of the new Pipe Crack Study Group are presented in the enclosed "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants", NUREG-0531. This report is for your information and comment. Also enclosed is a copy of a related Federal Register Notice.

The NRC staff will review the Study Group report and its conclusions/recommendations and any comments received about the report. Following this review, the staff will decide what further actions, if any, are required for the licensing and operation of reactors.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosures:
1. NUREG-0531
2. Notice

FOR THE RECORD:

As of, 10 July 79, TERA Corp., has not placed this document in the Document Control System and has not returned the second "Docket File" copy. Consequently, this "only copy" is being filed without an accession number and under the docket number for the first unit.

THE FILES

Duke Power Company

cc:
Mr. William L. Porter
Duke Power Company
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806 15th Street, N.W.
Washington, D. C. 20005

U. S. Nuclear Regulatory Commission
Region II
Office of Inspection and Enforcement
ATTN: Mr. Francis Jape
P. O. Box 85
Seneca, South Carolina 29678

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 420, 7735 Old Georgetown Road
Bethesda, Maryland 20014

Manager, LIS
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

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Docket
50-269

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 9, 1979

ALL POWER REACTOR LICENSEES

GENTLEMEN:

The document "Interservice Procedures for Instructional Systems Development," TRADOC Pamphlet 350-30, that was referenced in the NRC report, "Nuclear Security Personnel for Power Plants," NUREG-0219, is now available, free, in microfiche form from NRC. You may obtain a microfiche copy of TRADOC 350-30 by writing to:

Distribution Services Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Sincerely,

William J. Besaw

William J. Besaw, Director
Division of Technical Information
and Document Control, ADM

cc: Service List

Duke Power Company

50-269
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cc:
Mr. William L. Porter
Duke Power Company
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806 15th Street, N.W.
Washington, D. C. 20005

U. S. Nuclear Regulatory Commission
Region II
Office of Inspection and Enforcement
ATTN: Mr. Francis Jape
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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Nuclear Power Generation Division
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50-269

W. O. Parker, Jr.
Duke Power Company
422 S. Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, has contracted with Mountain West Research Incorporated to undertake a series of post-licensing studies of nuclear power stations. These studies will analyze the community and regional socioeconomic impacts which occurred during the construction and post-construction periods of each station. Oconee Nuclear Plant Units 1 and 3 has been included in a sample of fourteen stations to be studied.

This contract is designed to provide information and tools of analysis which will improve and facilitate the treatment of socioeconomic impacts in the NRC licensing of nuclear power plants. It is our belief that this study will place the type, magnitude, and significance of socioeconomic impacts from nuclear power plant siting into a more useful perspective. It should, thereby, be useful to communities located near the proposed site for future nuclear power plants as well as the NRC in meeting its responsibilities with regard to the National Environmental Policy Act (NEPA).

These studies will involve information gathering in the communities surrounding each site. Only a limited amount of information, primarily historical in nature, will be sought directly from the utilities. Dr. James Chalmers of Mountain West Research will contact you within the next several weeks to arrange a briefing on the objectives and design of this study as well as the type of information that will be sought from you.

It is hoped that you can be of some assistance to the successful performance of this study.

TTA4
ccp

OFFICE ➤						
SURNAME ➤						
DATE ➤						

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 – 626-624

If you have any questions concerning this study, please call B. J. Youngblood, Chief, Cost-Benefit Analysis Branch, Division of Site Safety and Environmental Analysis, on 301-492-7593.

Sincerely,

Original signed by R. C. DeYoung

Richard C. DeYoung, Director
Division of Site Safety and
Environmental Analysis
Office of Nuclear Reactor Regulation

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UNITED STATES
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REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

NOV 9 1978

In Reply Refer To:
RII:SCEwald

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Duke Power Company
ATTN: Mr. L. C. Dail, Vice President
Design Engineering
Post Office Box 33189
Charlotte, North Carolina 28242

Gentlemen:

The U.S. Nuclear Regulatory Commission will conduct a one-day seminar in Atlanta on December 5, 1978, to discuss new Model Radiological Effluent Technical Specifications for Nuclear Power Plants. Representatives from the Office of Nuclear Reactor Regulation will make presentations on background, development, and implementation of the effluent technical specifications and will discuss the major changes from existing specifications. A question and answer period will follow the prepared presentations.

All utilities which hold operating licenses or have applied for licenses to operate nuclear power reactors are invited to attend. We encourage you to send those members of your staff most closely involved in your radiological effluent program. Information on time, location, accommodations, and an attendance registration form are enclosed. A copy of the agenda is also enclosed.

If you have questions beyond the scope of the agenda, please submit them with your registration forms so that the speakers will be able to provide answers to those questions that apply to the entire audience. We will attempt to answer any questions pertaining to radiological effluents and environmental monitoring; however, if the problem is only specific to your case and not of general interest to the entire audience, we will communicate with you directly, rather than at this seminar.

Sincerely,

James P. O'Reilly

James P. O'Reilly
Director

50-369/370
269/270/287

Enclosures:
Seminar Information Sheet
Registration Form
Proposed Agenda

cc w/enclosures:
Mr. D. G. Beam, Project Manager
Catawba Nuclear Station
Post Office Box 223
Clover, South Carolina 29710

Mr. J. T. Moore, Project Manager
Cherokee Nuclear Station
Post Office Box 422
Gaffney, South Carolina 29340

SEMINAR INFORMATION SHEET

SUBJECT

Model Radiological Effluent Technical Specifications and Environmental Monitoring

DATE

December 5, 1978

TIME

8:00 a.m. to 5:00 p.m.

LOCATION

Stadium Hotel, Braves Room
450 Capitol Avenue, S.E.
Atlanta, Georgia 30312

TRANSPORTATION

Limousine service to and from the Atlanta airport is available.

ACCOMMODATIONS

A number of rooms have been set aside for seminar participants at the Stadium Hotel for the night of December 4, 1978. Rates for seminar participants are $20.00 per night (single), and $26.00 per night (double), plus 4% sales tax. Reservations are the responsibility of the attendee(s) and should be made no later than November 22, 1978.

LUNCHEON

A buffet luncheon has been arranged for participants at a charge of $6.00 per person (includes tax and gratuity). In addition, coffee will be served during morning and afternoon breaks at a charge of $1.00 per person. Money for lunch and coffee will be collected during registration on December 5.

NRC CONTACT

S. C. Ewald
404/221-4181

SEMINAR REGISTRATION

Please submit the following information for each representative planning to attend:

NAME

TITLE

REPRESENTING

Do you plan to take advantage of:

LUNCHEON ($6.00)	Yes:_____	No:_____
COFFEE SERVICE ($1.00)	Yes:_____	No:_____

To assist us in making preparations for the seminar, please return all registration forms by November 27, 1978, to:

S. C. Ewald
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street
Suite 3100
Atlanta, Georgia 30303

AGENDA FOR REGIONAL SEMINAR ON RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

I. INTRODUCTION AND BACKGROUND

- A. Standard Technical Specifications
- B. Appendix A vs. Appendix B
- C. Implementation of Appropriate Regulations
    1. 10 CFR Part 20
    2. 10 CFR Part 50
    3. 40 CFR Part 190

II. SCHEDULE FOR IMPLEMENTATION

- A. OL Applications
- B. Operating Reactors

III. INTERFACES WITH NRR

IV. DEVELOPMENT OF STANDARD TECHNICAL SPECIFICATIONS

- A. NRC Task Force
- B. AIF Task Force
- C. Approval by R^3C
- D. Deviations from Standard Technical Specifications

V. METHOD OF IMPLEMENTATION

- A. Amendment to Operating License
- B. Amendment to FSAR

VI. MAJOR CHANGES FROM PRESENT TECHNICAL SPECIFICATIONS

- A. 10 CFR Part 50, Appendix I
- B. 40 CFR Part 190
- C. Maintenance and Use of Equipment
- D. Solidification - Process Control Program
- E. Special Reports (40 CFR Part 141)
- F. Environmental Monitoring

VII. DISCUSSION OF NUREG-0133

VIII. DISCUSSION OF TECHNICAL SPECIFICATIONS

- A. LCO
- B. Action
- C. Surveillance

IX. QUESTION AND ANSWER ~~PER~~IOD

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Docket Nos. 50-269
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and 50-287

OCTOBER 5 1978

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

By letter dated August 11, 1978, we issued a license amendment for the Oconee Nuclear Station relating to the completion of facility modifications for fire protection. Our Safety Evaluation, supporting this action, in Table 3.1, Item 3.1.17 Interim Measures, states that spare dedicated cables will be provided for use with the low pressure injection pumps, low pressure service water (LPSW) pumps and the high pressure injection pumps. Your letter of June 19, 1978, regarding cold shutdown, had previously informed us that dedicated cables will be supplied for the LPI and HPI pumps but that the low pressure service pumps are separated such that a single fire would not affect all pumps. You stated that the piping for the pumps is interconnected so pumps can supply water to any of the three Oconee units to bring them from a hot to cold shutdown condition. Our review has found that the LPSW system piping arrangement is acceptable in terms of accomplishing cold shutdown and that there is no requirement for spare dedicated cables for the motors of the LPSW pumps. This letter supplements our August 11, 1978 Safety Evaluation and thereby deletes the requirement to provide spare dedicated cables for the LPSW system pump motors.

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

cc: see next page

*SEE PREVIOUS YELLOW FOR CONCURRENCE

OFFICE	ORB#4	PSYB	ORB#4			
SURNAME	*MFairtile:ar	*GLainas	RReid			
DATE	9/28/78	9/28/78	10/5/78			

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Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

By letter dated August 11, 1978, we issued a license amendment for the Oconee Nuclear Station relating to the completion of facility modifications for fire protection. Our Safety Evaluation, supporting this action, in Table 3.1, Item 3.1.17 Interim Measures, states that spare dedicated cables will be provided for use with the low pressure injection pumps, low pressure service water pumps and the high pressure injection pumps. Your letter of June 19, 1978, regarding cold shutdown, had previously informed us that dedicated cables will be supplied for the LPI and HPI pumps but that the low pressure service pumps are separated such that a single fire would not affect all pumps. You stated that the piping for the pumps is interconnected so pumps can supply water to any of the three Oconee units to bring them from a hot to cold shutdown condition. Our review has found that the LPSW system piping arrangement is acceptable in terms of accomplishing cold shutdown and that there is no requirement for spare dedicated cables for the motors of the LPSW pumps. Therefore, amend Item 3.1.17 of Table 3.1 on Page 3-4 of our August 11, 1978 Safety Evaluation to read: "Spare dedicated cables will be provided for use with the low pressure injection pumps and the high pressure injection pumps (4.10)." In addition, amend Paragraphs4.10 on Page 4-11 by deleting (2) in its entirety and renumbering (3) as (2).

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

cc: See next page

OFFICE ➤	ORB#4:DOR	C-PSB:DOR	C-ORB#4:DOR			
SURNAME ➤	MFairtile:rf	GLainas	RReid			
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Duke Power Company

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
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SEPTEMBER 2 6 1978

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

By letter dated April 21, 1978, you informed us of an error in your ECCS Small Break LOCA evaluation model analyses. We authorized continued operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, based in part on credit to be taken for prompt operator actions. You were issued an Order dated April 26, 1978, to implement procedures and training necessary to assure that the required operator actions could be initiated within five minutes and completed within ten additional minutes. In addition, you were directed to submit reanalyses along with proposals for long-term modifications to eliminate the need for prompt operator actions including those from inside and from outside the control room.

Your proposal for long-term modifications relies on operator actions. It appears that "prompt" operator action is still required.

The staff position regarding allowable operator actions for which credit may be taken following a Condition III event (small LOCA), is as follows:

1. No credit for operator actions required within ten minutes of the initiating event will be allowed.

2. Credit may be taken for operator action from within the control room after at least ten minutes if adequate information is available and displayed, and if the operator has sufficient time to correctly analyze the situation, recognize any error or malfunction and to take a corrective action. The allowed operator actions are limited to "simple" actions which are characterized by the "pushing of a button" or the "flipping of a switch." Balancing of ECCS flows by adjusting valve positions is <u>not</u> considered a "simple" operator action.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

3. Credit may be taken for more complex operator actions which may be assumed to begin after at least ten minutes. The credit assumed in the ECCS analysis should consider a reasonable time delay after initiation of the operator action for the system to respond to assure that design flows have been achieved. However, the credit for these complex actions cannot be assumed in less than 20 minutes after the accident.

If the emergency core cooling system analysis for your plant, submitted July 14, 1978, does not meet the position stated above, you should either: (1) provide a new analysis which demonstrates that the modification you have proposed will meet the staff position with respect to allowable credit for operator action or (2) propose a new modification and an analysis. If your July 14 submittal does meet one of the above positions, state the position you meet and the basis for your statement.

In your consideration of the above, you should be aware that we have approved use of the simplified FOAM code as proposed by Babcock & Wilcox in their letter dated August 11, 1978.

It is requested that you inform us by telephone, confirmed by letter, within seven days of receipt of this letter which of the options you plan to pursue. This will allow us to determine whether or not to continue our review of the design of your proposed modification. Your response should include your schedule for submittal of the required analysis and/or modification.

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

cc: See next page

MBF C-

OFFICE	ORB#4:DOR	ORB#4:DOR				
SURNAME	MFairtile:rf	RReid				
DATE	9/25/78	9/26/78				

Duke Power Company

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

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DeBevoise & Liberman
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806-15th Street, NW.,
Washington, D.C. 20005

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Docket 50-269

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 12, 1978

ALL POWER REACTOR LICENSEES

Gentlemen:

This letter is being sent to all licensees authorized to operate a nuclear power reactor and to all applicants with application for a license to operate a power reactor (FSAR docketed).

The NRC recently notified you that it had scheduled a series of meetings to discuss implementation of upgraded guard qualification, training and contingency planning requirements. The Region I meeting scheduled to be held on October 11, 1978 has been rescheduled to October 13, 1978. A revised schedule is enclosed.

For further information or comments, please contact Tom McKenna of my staff on (301) 492-7846.

Sincerely,

Jack W Roe
for
James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

Enclosure:
Meeting Schedule & Locations

cc: Service List

MA4

REVISED SCHEDULE

9:30 to 3:30

Region II	September 27, 1978	Stadium Hotel* 450 Capitol Ave., SE Atlanta, GA 30312 (404) 688-1900
Region III	October 3, 1978	Ramada O'Hare 6600 North Mannheim Rd. Des Plains, IL 60018 (312) 827-5131
Region IV & V	October 5, 1978	San Francisco Airport Hilton P. O. 8355 San Francisco, CA 94128 (415) 589-0770
Region I	October 13, 1978	Valley Forge Holiday Inn 260 Goddard Blvd. King of Prussia, PA 19406 (215) 265-7500

*Special rate for reservations received before September 15, 1978.

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
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806-15th Street, NW.,
Washington, D.C. 20005

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AUG 17 1978

Docket Nos. 50-269/270/287

Duke Power Company
ATTN: Mr. William O. Parker
Vice President-Steam Production
Power Building
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

This is in reply to your letter of July 17, 1978, in which you submitted a revised "Oconee Nuclear Station Requalification Program for NRC Licensed Personnel." Specifically, the revised program:

1. clarifies the definition of "Backup Licensee" and

2. relieves such Backup Licensees from mandatory participation in the requalification lectures preceeding the annual examination segments.

Based on our review, we have concluded that the revision does not materially decrease the effectiveness of the program. The revised program continues to meet the requirements of Section 50.54(i-1) of 10 CFR Part 50 and Appendix A of 10 CFR Part 55.

Sincerely,

Paul F. Collins, Chief
Operator Licensing Branch
Division of Project Management

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Continued on next page.

MA 4

Mr. Troy B. Conner
Conner & Knotts
1747 Pennsylvania Avenue, N.W.
Washington, D. C. 20006

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SURNAME	JJHolman:jmb	M. Fairtile	PFCollins			
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August 15, 1978

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Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President - Steam Production
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

By letter dated May 16, 1978, Duke Power Company submitted a proposed program for the installation of a limited number of tube sleeves in the Oconee Station once-through steam generator tubes. The purpose of these sleeves is to reduce dynamic stress in the region of previously indicated tube abnormalities.

We have previously reviewed the document submitted with your May 16 letter, entitled "Technical Justification for Installing and Testing Tube Sleeves in Once-Through Steam Generator - October 3, 1977 (Rev. 1 February 17, 1978)" on the Three Mile Island Unit No. 2 docket. The results of our review are provided in the Safety Evaluation Report, Supplement 2, for TMI-2, dated February 1978. Based on the above review we have no objection to your proposed installation and test program. A copy of page 5-4 of the TMI-2 Safety Evaluation Report is enclosed.

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosure:
Page 5-4 of TMI-2 SER

cc w/enclosure: See next page

MB7

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SURNAME	MFairtile:rm		RReid		
DATE	X8/8/78	8/9/78	8/15/78		

Duke Power Company

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806-15th Street, NW.,
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Tube sleeving test program

Both Babcock & Wilcox and Combustion Engineering have designed test sleeves for insertion in the upper end of a number of steam generator tubes, with appropriate instrumentation, to assess their effect in stiffening the tubes to improve their capability to resist flow-induced vibration, the most likely cause of tube damage found in some other Babcock & Wilcox steam generators.

Safety evaluations have been submitted by the applicant for both designs, including descriptions of the sleeves, locations in the steam generator, method of installation, instrumentation, bench test results, stress analyses, and an evaluation of the effects of the test equipment on the safety of the plant.

Based on a review of the information supplied by the applicant and the similarity of the sleeves to installations in some Combustion Engineering steam generators, we conclude that neither the structural integrity nor the operational characteristics of safety-related equipment involved will be affected by the test program modifications, nor will the consequences of an accident be increased as a result thereof, and therefore that these modifications are acceptable.

After completion of all the above programs we will determine what modifications, if any, we will require to be made to Three Mile Island Unit 2.