October 18, 1979

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Mr. Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
United States Nuclear Regulatory Commission
Washington, D.C. 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
DOCKET NOS. 50-325 AND 50-324
LICENSE NOS. DPR-71 AND DPR-62
FOLLOWUP TO REVIEWS REGARDING THE THREE MILE ISLAND UNIT 2 ACCIDENT

Dear Mr. Eisenhut:

On September 13, 1979, you sent a letter to all operating nuclear power plant licensees regarding followup actions resulting from the NRC Staff reviews concerning the Three Mile Island Unit 2 accident. Carolina Power & Light Company's (CP&L) response to the actions requested in your letter is addressed in the following information and is applicable to the Brunswick Steam Electric Plant, Units 1 and 2.

CP&L has been reviewing the relevant material related to short-term lessons learned from the Three Mile Island Unit 2 accident, and has concluded that the issue of NUREG-0578 implementation can best be handled within the framework of the currently existing General Electric Boiling Water Reactor Owners' Group. The Owners' Group was created specifically to address the issues raised by the accident which are of significance to BWRs.

Well defined acceptance criteria are required for many of the NUREG-0578 and other recommendations in order to ensure timely implementation. CP&L and the Owners' Group have been working to develop these criteria. The recent regional and topical clarification meetings and other discussions have been of benefit, but others may be necessary to develop adequate acceptance criteria. This could potentially impact implementation schedules due to hardware availability as well as affecting the ability to effectively utilize scheduled plant outages. It is our firm belief that some degree of flexibility in the implementation schedules should be granted for good cause shown. However, within the constraints described above and your stated position on implementation, it is our intent to meet the requirements and schedules set forth in your September 13 letter.

In response to your request that CP&L submit our commitment to meet the requirements of the NUREG-0578 report, as modified and/or supplemented by items (a) through (f) of your letter, and the implementation schedule contained in Enclosure 6 to your letter, we have prepared Enclosure 1 to this letter. In Enclosure 1, the requirements are addressed on an item-by-item basis.

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Following this submittal, it may be appropriate for us to meet with you to discuss the details and schedules for NUREG implementation for our facility. A member of our organization will be in contact with you to arrange any meetings, that may be determined to be necessary.

Your letter of September 13, 1979 also requests that we commit to comply with each of the requirements of your Enclosure 7 in accordance with the implementation schedules shown in your Enclosure 8. Enclosure 2 to this letter addresses these requirements on an item-by-item basis.

We trust this letter is responsive to your requirements at this time, and stand prepared to provide additional clarification if you so desire.

Yours very truly,

E. E. Utley
Executive Vice President
Power Supply & Customer Services

EEU/jcb

Enclosures

ENCLOSURE 1

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK UNITS 1 AND 2
COMMITMENTS TO MEET REQUIREMENTS OF NUREG-0578

This enclosure contains commitments by Carolina Power & Light Company (CP&L) addressing the requirements set forth in NUREG-0578 and items (a) through (f) of Mr. D. G. Eisenhut's letter of September 13, 1979, and the schedules set forth in Enclosure 6 of the same letter. The requirements and schedule will be addressed below on an item-by-item basis to allow ready comparison with the referenced documents.

NUREG-0578 Item 2.1.1 - Emergency Power Supply Requirement

As discussed in the General Electric document NEDO-24708, natural circulation in the BWR is strong and inherent in all off-normal modes of operation, independent of any powered system, as long as sufficient inventory is maintained. This is because even in normal operation the BWR is essentially an augmented natural circulation machine. Because the BWR operates in all modes with both liquid and steam in the reactor pressure vessel, saturation conditions are always maintained irrespective of system pressure (the BWR does not have a pressurizer). Thus there is no need for emergency power to maintain natural circulation or to keep the system pressurized.

The power-operated relief valves in BWR's are already powered by emergency power. They have no block valves.

The reactor vessel level indication instrument channels for safety system activation and control are already powered by emergency power.

Therefore, the intent of this requirement is already met at Brunswick and no modifications are planned.

NUREG-0578 Item 2.1.2 - Relief and Safety Valve Testing

The BWR design basis includes no transients or accidents in which two-phase flow or subcooled liquid flow at high pressure through relief, safety/relief, or safety valves is calculated or expected. The BWR therefore satisfies the intent of the requirement in the strictest sense. The need for performance verification, however, has been studied in a broader sense. The remainder of this discussion is intended to demonstrate that performance verification in the field fully satisfies the broad intent of the requirement.

In determining the need for special testing of BWR safety and relief valves it is essential to consider the service duty to which the primary system relief and safety valves of the BWR are exposed, and the consequences of maloperation of these valves. Relief valves are routinely used to mitigate the effects of system transients. A stuck-open valve is not an event of great significance in a BWR: in 300 reactor year of experience, 54 cases have occurred. Tables 2.1.2-1 and 2.1.2-2 summarize the experience to date. This experience,

as will be explained, clearly shows that there is no need for an extensive testing program for BWR safety and relief valves.

A. BWR Safety and Relief Valves

Table 2.1-3 of NEDO-24708 shows the complement of safety and relief valves for all domestic operating BWRs. Most BWRs have relief vavles or dual-function safety/relief valves (S/RV), the discharges of which are piped to the suppression pool. Spring safety valves discharge directly to the drywell.

B. Valve Usage

(1) Relief valves and dual-function S/RVs in BWR/2-6 (Brunswick is a BWR/4). The relief valves and dual-function S/RVs are designed to routinely mitigate the effect of system transients. Their discharges are piped to the containment suppression pool. This massive heat sink prevents significant containment heatup. Complication of a system transient by a stuck-open valve has essentially no effect on reactor vessel water level measurement or on forced or natural circulation capability. The flow through the valve is saturated steam. If the valve cannot be closed by operator action the plant can be shut down using familiar and uncomplicated procedures.

(2) Spring safety valves in BWR/2-4 (Brunswick is a BRW/4). The safety valve set-point is sufficiently higher than the relief valve set-point that the safety valves are almost never required to operate. (Table 2.1.2-3 documents the three cases in which safety valves have ever lifted in BWR operation.) Should a safety valve inadvertently lift, which has never happened in BWR operation, the effect is the same as a small steam line break inside containment. Even in this remote event, the flow through the valves will be saturated steam at all times.

C. Two-Phase Flow

Expected operating conditions and transients do not include two-phase flow through S/RVs, safety, or relief valves. However, on three occasions, circumstances combined to cause high pressure water flow down the steamlines and a steam/water mixture to flow through the valves. A summary of these events is given in Table 2.1.2-3. In these events, Electromagnetic relief valves and direct acting safety valves were actuated, discharged a steam/water mixture and reclosed, indicating that the flow media did not cause a stuck-open valve condition. These events did not lead to any concern over adequate core cooling. However, following these events, high water level trips were added to all new BWRs and retrofitted to most of the BWRs in operation. Brunswick Units 1 and 2 are equipped with this high level trip.

D. Valve Qualification

TABLE 2.1.2-1

S/RV BLOWDOWNS IN BWR OPERATION

YEAR	3-STAGE TARGET ROCK			2-STAGE TARGET ROCK		CROSBY-OKANO-DIKKERS		TOTAL S/RV BLOW-DOWNS	TOTAL S/RVs IN SERVICE	TOTAL BLOWDOWNS DIVIDED BY TOTAL VALVES IN SERVICE
	TOTAL BLOWDOWNS	STUCK OPEN FOLLOWING DEMAND	# OF VALVES IN SERVICE	TOTAL BLOWDOWNS	# OF VALVES IN SERVICE	TOTAL BLOWDOWNS	# OF VALVES IN SERVICE			
1971	2	2	14					2	14	0.15
1972	1	1	23					1	23	0.04
1973	1	1	56					1	56	0.02
1974	10	1	108					10	108	0.09
1975	7	0	127					7	127	0.06
1976	11	1	149					11	149	0.07
1977	9	4	157					9	157	0.06
1978	5	3	157	0	11	0	35	5	203	0.02
1979 to Sept.	4	1	132	0	36	0	52	4	220	0.02

NOTE: The above table does not include Dresser Safety Valves (unpiped discharge) or "Electromatic" relief valves. See Table 2.1.2-2 for information on this equipment.

TABLE 2.1.2-2

SAFETY AND ELECTROMATIC RELIEF VALVE

BLOWDOWNS IN BWR OPERATION

A. Safety Valves.

Only one event has ever occured with partially stuck open valves - the Dresden 2 event described in Table 2.1.2-3. The lifting levers which cocked the valves partially open were subsequently removed from safety valves at all plants and there have been no further occurrences. There have only been three occurrences in which safety valves have ever lifted during operation (see Table 2.1.2-3). The total number of valves in service is 76[1].

B. Electromatic Relief Valves.

There have been three occurrences of a stuck open Electromatic relief valve, two of which followed a demand. The number of valves in service is 37.[1]

[1] Some BWR's are in the process of replacing safety valves and Electromatic relief valves with Target Rock S/RV's.

TABLE 2.1.2-3

EVENTS IN WHICH TWO-PHASE FLOW OR

LIQUID PASSED THROUGH BWR SAFETY OR RELIEF VALVES

DRESDEN 2 - JUNE 5, 1970

During the course of the initial test program on Dresden 2 with the unit operating at 75% power, a spurious signal in the reactor pressure control system occurred. This spurious signal resulted in simultaneous opening of the control and the turbine bypass valves with resultant turbine trip, reactor scram, and main steamline isolation.

In response to the initial and expected water level drop, the operator switched to manual control of the feedwater system and began filling the reactor vessel at the maximum rate. Water level misinterpretation led to reactor water overflowing into the main steam lines. A pressure surge resulted in the main steam lines when relief valves were cycled. This momentarily opened one of the safety valves, resulting in a discharge directly to the containment (unpiped discharge). The fluid impinged upon the lifting levers of two other safety valves causing these safety valves to cock slightly open. The water-steam mixture from the two safety valves pressurized the primary containment to an estimated 20 psig and an estimated temperature of approximately 300°F. At no time during the event was there difficulty maintaining adequate water supply to the reactor core, and there was no question of adequate core cooling.

DRESDEN 3 - DECEMBER 8, 1971

Unit 3 was operating about 98% power on December 8, 1971, when the plant was shut down due to a reactor low water level scram. The scram resulted from a condensate/condensate booster pump trip and the subsequent trip of two reactor feed pumps on low suction pressure. Following the scram, the standby feed pump started. The vessel was overfilled and the steam lines flooded. Due to a pressure surge in the main steam lines, a safety valve lifted causing discharge directly to the containment (unpiped discharge). The containment was pressurized to approximately 20 psig. At no time during the event was there difficulty maintaining adequate water supply to the reactor core, and there was no question of adequate core cooling.

KRB (GERMANY) - JANUARY 13, 1977

The unit was operating at 100% power when a bus on two of its 200 KV lines opened. The plant was scrammed and isolated. Manual feedwater

TABLE 2.1.2-3 (cont'd)

control was initiated which resulted in flooding of the steam lines. Safety valves opened and discharged water, steam and two-phase media. The valves discharged directly to the containment (unpiped discharge). The safety valves opened and reclosed several times. Because of the unique piping arrangement (which is not present in any US-BWR), reaction forces of the discharging valves caused or contributed to a pipe rupture in two of the fourteen flanged nozzles by which the valves are connected to a U-shaped header. At no time during the event was there difficulty maintaining adequate water supply to the reactor core, and there was no question of adequate core cooling.

(1) Crosby, Dikkers, Okano and two-stage Target Rock S/RVs are tested for the expected saturated steam flow conditions. This includes life-cycle testing of 300 actuations as well as environmental qualifications including seismic, thermal, mechanical and radiation effects.

(2) Three-stage Target Rock S/RVs were subjected to restricted flow steam tests to qualify the set-point and valve opening time delay. Solenoid valves (used during power actuation) are qualified by autoclave test for the LOCA environment. Satisfactory valve operation has been demonstrated by field service.

(3) Dresser Electromatic relief valve solenoids were qualified by autoclave test for the LOCA environment. Satisfactory valve operation has been demonstrated by field service.

(4) Satisfactory operation of Dresser safety valves has been demonstrated by field service.

E. Field Experience

Since 1971 there have been 50 events in BWR plant operation wherein S/RVs have stuck open (Table 2.1.2-1). In each of these cases the reactor was depressurized, the stuck valve was repaired or replaced, and the plant was placed back into service.

Although a stuck-open S/RV is of no significant safety concern in the BWR, programs are underway to reduce the frequency of such events. From Table 2.1.2-1 it is seen that the total number of S/RV blowdowns has steadily decreased since the mid-70s. The improvement in the number of S/RV blowdowns as a factor of number of S/RVs in service has been even more dramatic.

From Table 2.1.2-2 it is seen that experience with spring safety valves and Electromatic relief valves has always been good; there have only been four blowdowns.

F. Summary

(1) BWR S/RVs are routinely tested for the only expected mode of operation (saturated steam), both by in-place functional tests and by frequent usage in mitigating plant transients;

(2) There is no design-basis transient or acident which requires safety, relief, or dual function S/RVs to pass two-phase or liquid flow at high pressure;

(3) Inadvertnet passage of two-phase flow is not likely where high pressure feedwater and injection systems are tripped by high vessel water level.

(4) In the three events wherein BWR S/RVs did pass two-phase flow, the valves reclosed;

(5) Spring safety valves are almost never required to open; in the even less likely event that one should stick open, the effect is identical to that of a small steam line break. There is no concern for adequate core cooling.

(6) Electromatic relief valves and dual-function S/RVs are frequently called on to operate, and dual-function S/RVs occasionally stick open. The consequences of a stuck-open valve are minimal and reactor shutdown is uncomplicated, as proven by numerous field occurrences. In some BWRs the procedures for responding to a stuck-open relief valve includes the opening of additional relief valves. Improvement programs are reducing the frequency of such events.

Therefore, no special performance testing of safety/relief valves is planned for Brunswick Units 1 and 2.

NUREG-0578 Item 2.1.3.a - Direct Indication of Valve Position

Flow sensing devices will be provided for all safety relief valves and relief valves. The schedule for implementation contained in Enclosure 6 to Mr. Eisenhut's letter will be met pending availability of hardware needed to complete the installation of the system.

NUREG-0578 Item 2.1.3.b - Instrumentation for Inadequate Core Cooling

Additional hardware to identify inadequate core cooling on BWR's has not been determined to be necessary at this time. Present procedures identify the diverse methods of determining inadequate core cooling, using existing instrumentation. The results of analysis being performed in response to 2.1.9 will be factored into procedures as required, after the analysis is complete and in accordance with the implementation schedule contained in Mr. Eisenhut's letter of September 13, 1979.

Because the BWR operates in all modes with both liquid and steam in the reactor pressure vessel, saturation conditions are always maintained irrespective of system pressure. Thus there is no need for a subcooling meter in the BWR.

NUREG-0578 Item 2.1.4 - Diverse Containment Isolation

There is diversity in the parameters sensed for the initiation of BWR containment isolation. Following an isolation, deliberate operator action is required to open valves as discussed in our responses to I&E Bulletin 79-08 dated April 23 and August 3, 1979. However, CP&L is continuing to review this issue and will:

A. Review all systems penetrating primary containment to identify all essential systems. The basis of such classification shall be documented and supplied to the NRC.

B. All systems not identified as essential will be reviewed. If automatic isolation is not provided, justification for not isolating will be presented to the NRC.

C. Review and modify isolation control systems and administrative controls, as appropriate, such that no isolation valve will open when the isolation logic is reset. Any valves that will automatically open when the isolation logic is reset, will have the isolation logic changed to prevent the valves from opening when reset. Administrative controls to prevent valves from reopening will be implemented by 1/1/80; logic modifications will be implemented by 1/1/81.

NUREG-0578 Item 2.1.5.a - Dedicated H_2 Control Penetration

Carolina Power & Light Company will investigate the capability to employ external hydrogen recombiners through either dedicated penetration or other single-failure proof designs and report the status of this capability by January 1, 1980. If modifications are required to provide this capability, a description of the modification and an implementation schedule will also be provided. In any event, the required capability will be provided by January 1, 1981.

NUREG-0578 Item 2.1.5.c - Recombiners

Brunswick Units 1 and 2 do not currently employ hydrogen recombiners for control of hydrogen gas in the primary containment resulting from zirc-water reaction or other sources. The primary containments of the units operate with an inerted atmosphere, thus eliminating the need for recombiners under present regulations regarding hydrogen gas releases following a loss-of-coolant accident.

NUREG-0578 Item 2.1.6.a - Systems Integrity for High Radioactivity

Carolina Power & Light Company will define those systems which could contain highly radioactive fluids following a hypothetical accident resulting in severe core damage, and investigate the leaktightness of these systems. Maintenance will be performed to minimize any leakage observed during this investigation, and a program will be developed which will be designed to maintain the designated systems as leakage-free as practicable. It is the intent of CP&L that the above work will be accomplished prior to January 1, 1980, and the results of the leakage determinations will be reported to NRC by the same date.

NUREG-0578 Item 2.1.6.b - Plant Shielding Review

A plant shielding review for those systems defined in response to NUREG-0578 Item 2.1.6.a will be performed to investigate the adequacy of existing plant shielding under conditions of a hypothetical accident resulting in severe core

damage, using the criteria set forth in the regional meetings related to the NUREG implementation. Corrective actions, if any are required, to provide for adequate access to vital areas and protection of safety equipment will be defined and reviewed with the NRC prior to implementation of any modifications. It is the intent of CP&L that the shielding review be completed by January 1, 1980, and the definition of corrective actions and review with the Commission be held such that implementation of any plant modifications can be accomplished by January 1, 1981.

NUREG-0578 Items 2.1.7.a and 2.1.7.b - Auto Initiation of Auxiliary Feed and Auxiliary Feed Flow Indication

These items require specific actions related to PWRs and therefore are not applicable to Brunswick Units 1 and 2.

NUREG-0578 Item 2.1.8.a - Post Accident Sampling

CP&L will review the Brunswick Units 1 and 2 sampling procedures and sampling capability for highly radioactive fluid samples to determine if plant design or analysis capability changes are required to meet the NUREG item. It is the intent of CP&L that the implementation schedule specified in Enclosure 6 for this requirement will be adhered to.

NUREG-0578 Item 2.1.8.b - High Range Radiation Monitors

Carolina Power & Light Company will implement the requirements of position 2.1.8.b, items 1, 2, and 3, consistent with commercial availability of equipment.

Procedures will be developed to estimate noble gas and radio-iodine release rates if the existing effluent instrumentation goes off scale.

It is CP&L's intent to meet the implementation schedule contained in Enclosure 6 to Mr. Eisenhut's letter of September 13, 1979.

NUREG-0578 Item 2.1.8.c - Improved Iodine Instrumentation

Carolina Power & Light Company will implement the requirements of position 2.1.8.c.

Procedures will be developed to accurately determine in-plant iodine concentrations.

It is CP&L's intent to meet the implementation schedule contained in Enclosure 6 to Mr. Eisenhut's letter of September 13, 1979.

NUREG-0578 Requirement 2.1.9: Analysis of Design and Off-Normal Transients and Accidents

The specific requirements and schedules are being developed in a continuing series of meetings between BWR Owners' Group, of which CP&L is a member, and the NRC Bulletins and Orders Task Force.

The implementation of emergency procedures and retraining will be done on a schedule consistent with those established with the Bulletins and Orders Task Force.

ACRS Items - Containment Pressure, Water Level and Hydrogen Monitors

Carolina Power & Light Company concurs with the ACRS recommendations for additional instrumentation for the following parameters:

A. Containment water level monitoring

B. Containment pressure monitoring

C. Containment hydrogen monitoring

For practical reasons, it is not desirable to monitor suppression pool water level all the way to the bottom of the suppression pool. This is because an instrument tap at the very bottom could become obstructed by sludge and small debris. CP&L believes the water level monitoring down to the elevation of the lowest ECCS pump suction is more practical and fully satisfies the intent of the requirement.

It is our current interpretation that the hydrogen monitoring requirement is associated with ECCS performance and core degradation, rather than with containment atmospheric control.

Therefore, CP&L intends to implement containment pressure, water level, and hydrogen monitoring which will be designed and installed to meet Engineered Safety System criteria.

The lowest suppression pool water level monitored will be at or below the elevation of the lowest ECCS pump suction.

It is our intent to meet the implementation schedule contained in Enclosure 6 to Mr. Eisenhut's letter of September 13, 1979.

H. R. Denton Item - RCS Venting

Domestic BWRs are provided with a number of power operated safety grade relief valves which can be manually operated from the control room to vent the reactor pressure vessel. The point of connection of the vent lines from the vessel to these valves is such that accumulation of gases above that point in the vessel will not affect natural accumulation of gases of the reactor core.

These power operated relief valves satisfy the intent of the NRC position. Information regarding the design, qualification, power source, etc., of these valves has been provided in the Brunswick Safety Analysis Report.

CP&L's position is that the requirement of single failure criteria for prevention of inadvertent actuation of these valves, and the requirement (stated in the October 11 topical meeting) that power be removed during normal operation, are not applicable to BWRs. These valves serve an important function in mitigating the effects of transients and in many plants provide

ASME code overpressure protection. Therefore, the addition of a second "block" valve to the vent lines could result in a less safe design and in some cases a violation of the code. Also, inadvertent opening of relief valve in a BWR is a design basis event and is a controllable transient (this is discussed in our position on NUREG-0578, item 2.1.2).

For the Brunswick primary system there are eleven relief valves on each unit, located on the main steam line and discharing to the suppression pool. All valves are remotely operable from the control room and are safety grade. The vessel head is also capable of being vented from the control room by the opening of two remote manual vent valves. These valves are qualified for the environment inside the drywell resulting from a LOCA. The line is two inches in diameter and discharges to the radwaste system. A third means of venting the vessel upper head exists through a two inch line which runs to main steam line "A". Venting of the primary system can also be achieved through the steam driven RCIC and HPCI pumps, which are operable from the control room and discharge to the suppression pool.

In the October 11, 1979, topical meeting on this subject, three procedural questions were raised:

A. Where to vent to (suppression pool vs. containment);

B. When to vent;

C. When not to vent.

Under most circumstances, there would be no choice as to where to vent to or when to vent, since the relief valves (as part of the Automatic Depressurization System), HPCI, and RCIC will function automatically in their design modes to ensure adequate core cooling, and these will provide continuous venting to the suppression pool. The current assessment is that it would not be desirable to interfere with emergency core cooling functions in order to prevent venting, but the matter will be studied further.

The result of a break in the safety/relief valve discharge line, or any of the other systems enumerated above, would be the same as a small steam line break. A complete steam line break is part of the plants' design basis, and smaller-size breaks have been shown to be of lesser severity. A number of reactor system blowdowns due to stuck-open relief valves (also equivalent to a small steam line break) have confirmed this in practice (see our position on Requirement 2.1.2). Thus, no new analyses to show conformance with 10CFR50.46 are required.

Because the relief valves, HPCI, and RCIC will vent the reactor continuously, and because containment hydrogen calculations in normal safety analysis calculations assume continuous venting, no special analyses are required to demonstrate "that the direct venting of noncondensible gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment."

Therefore, CP&L believes that adequate reactor coolant system venting is provided by the existing plant design and that no new 10CFR50.46 conformance

calculations or containment combustible gas concentration calculations are required, since systems in the plant's original design and covered by the original design bases are used; CP&L will, however, provide plant procedures to govern the operator's use of the relief valves for venting the reactor pressure vessel.

It is our intent to meet the implementation schedule contained in Enclosure 6 to Mr. Eisenhut's letter of September 13, 1979.

NUREG-0578 Item 2.2.1.a - Shift Supervisor Responsibilities

Carolina Power & Light Company agrees with the intent of the staff's position. However, in order to remove any ambiguity from the meaning of the term "accident situation" in item 2.b of the staff's position in Appendix A of NUREG-0578, the entire sentence will be interpreted as follows: The shift supervisor (or equivalent, such as the supervising control operator in some plants), until properly relieved, shall remain in the control room at all times whenever a site or general emergency has been declared to direct the activities of control room operators.

The staff's position will be implemented as stated and subject to the interpretation of item 2.b as discussed above in accordance with the implementation schedule contained in Enclosure 6 to Mr. Eisenhut's letter of September 13, 1979.

NUREG-0578 Item 2.2.1.b - Shift Technical Advisor

Carolina Power & Light Company will implement the requirement for a Shift Technical Advisor on the schedule outlined in Enclosure 6 to Mr. Eisenhut's letter of September 13. The shift technical advisor will be on duty by January 1, 1980, and training will be achieved by January 1, 1981. There have been numerous discussions between the NRC staff and licensees on the details of implementing this position. Carolina Power & Light Company is in the process of assessing these discussions and will submit our position on implementation by November 16, 1979.

NUREG-0578 Item 2.2.1.c - Shift Turnover Procedures

The shift turnover procedures are currently being reviewed to ensure that the oncoming shift reviews all critical parameters, allowable limits and technical specification limits for alarming parameters, and a proper alignment check of operating equipment. All revisions which are necessary as a result of this review as well as an administrative program for evaluating the effectiveness of the shift turnover procedures will be implemented by January 1, 1980.

NUREG-0578 Item 2.2.2.a - Control Room Access

Prior to January 1, 1980, an administrative program will be implemented to limit access to the control room to those individuals responsible for the direct operation of the nuclear power plant. This procedure will describe the authority and responsibility of the person charged with limiting access to the control room and the conditions under which access is to be limited.

In addition, procedures which identify the line of command in the control room will be reviewed and revised as necessary to ensure a clear line of command exists for operation of the plant in emergency conditions.

NUREG-0578 Item 2.2.2.b - On Site Technical Support Center

A temporary on-site Technical Support Center will be established prior to January 1, 1980. Drawings which pertain to the as-built conditions and layout of structures, systems, and components will be readily available for use in this center. A voice communications link between this center and the control room as well as between the NRC and the Technical Support Center will be installed.

The requirements for a permanent Technical Support Center are being reviewed in preparation for establishing that center. The requirements for upgrading the Technical Support Center by January 1, 1981 are acknowledged; an implementation schedule will be developed consistent with the January 1, 1981 date.

NUREG-0578 Item 2.2.2.c - On Site Operations Support Center

The location of the Operational Support Center will be identified prior to January 1, 1980. This center will be separate from, but have the capability of establishing voice communications with, the control room. The site emergency plan will be revised to reflect the existence of this center.

POOR ORIGINAL

ENCLOSURE II
ENCLOSURE 2

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
COMMITMENT TO MEET NEAR TERM EMERGENCY PREPAREDNESS IMPROVEMENTS

This enclosure contains commitments by Carolina Power & Light Company (CP&L) addressing the requirements and schedules set forth in Enclosures 7 and 8 of Mr. D. G. Eisenhut's letter of September 13, 1979. The items will be addressed below on an item-by-item basis to allow ready comparisons with the referenced documents.

Item 1 - Upgrade Emergency Plans

A review of plant emergency plans is underway to determine if changes are needed to bring the plans into compliance with Regulatory Guide 1.101. Any changes identified will be included in the emergency plans by June 30, 1980.

Item 2 - Implement Short Term Actions from NUREG-0578

CP&L's commitments in this area are addressed under NUREG-0578 Items 2.1.8(a), 2.1.8(b) and 2.1.8(c) in Enclosure I.

Item 3 - Establish Emergency Operations Center

(a) An Emergency Operations Center and an alternate center for Federal, State, and local officials will be designated by June 30, 1980 and communications provided between the plant and the centers.

(b) Plans are being developed for the Technical Support Center. It is our intent to submit the plans to the NRC by January of 1980 and to have the Technical Support Center operational by January 1, 1981. A temporary Technical Support Center will be established by January 1, 1980. See response to NUREG-0578 Item 2.2.2.b in Enclosure I for more specific commitments.

Item 4 - Improve Offsite Monitoring Capabilities

A review is underway to determine any additional offsite monitoring capability necessary including the placement of additional TLD materials in the environment. Offsite monitoring capability will be improved prior to June 30, 1980.

Item 5 - Adequacy of State/Local Plans

The NRC was in the process of reviewing the Emergency Plan prepared by the State of North Carolina when the accident occurred at the Three Mile Island plant. The State subsequently removed the plan from the review process in order to incorporate experience gained from the accident. The State plan has been drafted and is again starting the review process. We will continue to work with the State on development of the plan.

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Item 6 - Conduct Test Exercises

The Company emergency plan for the Brunswick plant is tested on an annual basis. The State of North Carolina conducted a test of its plan in April of 1979 at the Brunswick Plant. Their mobile laboratory was dispatched to the scene along with personnel. The Governor participated in the test. Another test of the North Carolina plan is scheduled prior to January 1, 1980. The Brunswick plan will be tested prior to June 30, 1980. Joint tests of the Federal, State and local plans along with the Company plan for the Brunswick facility will be scheduled within five years.