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SOUTHERN COMPANY
Energy to Serve Your World℠

March 9, 1998

Docket Nos. 50-321
50-366

HL-5579

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M99394

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant
Request for Additional Information on
Extended Power Uprate License Amendment Request

Gentlemen:

By letter dated August 8, 1997, Southern Nuclear Operating Company (SNC) submitted a Technical Specification amendment request for the Edwin I. Hatch Nuclear Plant Units 1 and 2. The proposed amendment increases the authorized maximum power level for both units from the current limit of 2558 MWt to 2763 MWt.

By letter dated February 10, 1998, the NRC requested SNC to provide additional information based on the August 8th submittal. Enclosure 1 is SNC's response to the request for additional information.

The enclosure of the subject RAI was issued as proprietary due to the requested information related to Enclosure 6 of the August 8, 1997 submittal which contains the proprietary General Electric Topical Report NEDC-32749P. All responsible parties have reviewed the requested information and determined that none of the items requested are of a proprietary nature. All the responses that are provided in Enclosure 1 of this letter are considered by all responsible parties to be non-proprietary.

While preparing the responses to the subject RAI, several minor typographical errors were recognized. Also, the responses to NRC questions that resulted in changes to the August 8, submittal are noted in the applicable response. The corrected pages are provided in Enclosure 2, "Summary of Page Changes to Licensing Submittal." Enclosure 2 contains proprietary information due to the NEDC-32749P content. The General Electric affidavit is contained in Enclosure 5.

A001

As correctly stated in the subject RAI, SNC previously agreed to submit a revised non-proprietary version of NEDC-32749P. Enclosure 3, "Extended Power Uprate Safety Analysis Report for E. I. Hatch Plant Units 1 and 2," NEDO-32749, February 1998, is provided as the non-proprietary version of NEDC-32749P.

Due to the extensive nature of the RAI and responses, SNC requests a review meeting with the NRC staff reviewers be scheduled in early April. Please contact this office as soon as possible if the review meeting can be accommodated.

If you have any additional questions on this subject, please contact this office.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

TWL/eb

Enclosures:

1. Response to Request for Additional Information on Extended Power Uprate License Amendment Request
2. Summary of Page Changes to Licensing Submittal
3. "Extended Power Uprate Safety Analysis Report for E. I. Hatch Plant Units 1 & 2," NEDO-32749, DRF A13-00402 Class I, February 1998.
4. Response to Request for Additional Information on Extended Power Uprate License Amendment Request - Plant Hatch 1995 Met. Data (1 diskette)
5. GE Affidavit

cc: *Southern Nuclear Operating Company*
Mr. P. H. Wells, Nuclear Plant General Manager
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator (w/o Enclosures 3 and 4)
Mr. B. L. Holbrook, Senior Resident Inspector - Hatch (w/o Enclosures 3 and 4)

ENCLOSURE 5

Edwin I. Hatch Nuclear Plant
Request for Additional Information on
Extended Power Uprate License Amendment Request

GE Affidavit

for

Topical Report NEDC-327494P

General Electric Company

AFFIDAVIT

I, George B. Stramback, being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GE Licensing Topical Reports NEDC-32749P, *Extended Power Uprate Safety Analysis Report for Edwin I. Hatch Plant Units 1 & 2*, Class III (GE Proprietary Information), dated July 1997. This document, taken as a whole, constitutes a proprietary compilation of information, some of it also independently proprietary, prepared by the General Electric Company. The independently proprietary elements are delineated by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified by bars in the margin is classified as proprietary because it contains either detailed processes or detailed results and conclusions from these

evaluations, utilizing analytical models and methods, including computer codes, which GE has developed and obtained NRC approval. The development and / or approval of these system, component, and thermal hydraulic models and computer codes and processes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods, including justifications for not including certain analyses in applications to change the licensing basis.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 5th day of August 1997.

George B. Stramback

George B. Stramback
General Electric Company

Subscribed and sworn before me this 5th day of August 1997.

Julie A. Curts

Notary Public, State of California

JULIE A. CURTS
Commission # 1113834
Notary Public — California
Santa Clara County
My Comm. Expires Oct 20, 2000

Enclosure 1

Edwin I. Hatch Nuclear Plant
Request for Additional Information on
Extended Power Uprate License Amendment Request

ELECTRICAL SYSTEM:

NRC QUESTION 1

Enclosure 5, Summary of Plant Modifications (p. E5-1 and 2) lists nine plant design changes, and it states that most of the modifications listed are not safety-related and are not considered a commitment to the NRC.

State which changes are commitments to the NRC and provide a detailed explanation of the design changes below (listed in Enclosure 5). Discuss how these changes affect the extended power uprate analysis.

a. (Item 2) "Modify the Unit 1 and Unit 2 main generator stator water cooling system to enhance cooling capacity."

b. (Item 3) "Modify the Unit 2 main generator isophase bus cooling system to enhance cooling capacity. (Unit 1 changes are very minor)."

c. (Item 7) "Install an enhanced temperature monitoring system on the Unit 2 main transformer."

d. (Item 8) "Perform adjustments to installed plant and switchyard instrumentation as necessary...Main generator and switchyard protective devices.

SNC RESPONSE

The plant modifications considered to be NRC commitments are those requiring Technical Specifications changes. These modifications are highlighted by an asterisk in Enclosure 5 of the Extended Power Uprate License Amendment, dated August 8, 1997.

Response to NRC Question 1.a

General Electric Power Generation Engineering (GE-PGE) performed an evaluation to assess the impact of an increase in steam flow corresponding to a 113% extended flow uprate from the original design. As part of this extended power uprate, the generator rating will increase from the present rating of 1000 MVA to 1050 MVA at a 0.88 power factor. The uprate of the generator will require an upgrade of the stator water cooling system. The plant service water (heat sink) flow rate to the stator water cooling system will be increased from 1750 gpm to 1880 gpm at 95°F to remove the additional heat from the stators due to the increase in the generator rating, and the stator bar flow rate will increase to 550 gpm. The stator bar flow rate increase will be accomplished by replacing the main flow orifice and main filter. Also, the flow meter will be replaced with one having a calibration span compatible with the new flow element.

GE-PGE's evaluation of the coolers found them acceptable for extended uprate conditions. A more detailed discussion of the stator cooling evaluation is presented in the response to NRC Question 2.a.

Setpoints for the stator water cooling system alarms and interlocks will be changed to accommodate the new system operating parameters (flow, pressure and temperature) based on information provided by GE.

Response to NRC Question 1.b

GE Canada, the original equipment manufacturer for the nonsafety-related isolated phase bus duct and the isolated phase bus duct cooling equipment, performed an evaluation to assess the impact of an increase in steam flow corresponding to a 113% extended flow uprate from the original design. As part of this extended power uprate, the generator rating will increase from the present rating of 1000 MVA to 1050 MVA at a 0.88 power factor.

The SNC extended power uprate submittal stated that Unit 1 changes are very minor. The original equipment manufacturer initially recommended a minor change for Unit 1 that included increasing the duct fan speed. This was to be accomplished by changing out the fan pulley. However, subsequent to the licensing submittal, field testing has determined that the fan is currently operating at the desired speed with adequate motor horsepower. Therefore, no changes to the Unit 1 isophase bus duct cooling system are necessary to support extended power uprate.

The Unit 2 generator rerating will require an upgrade of the isolated phase bus duct cooling system. The plant service water (heat sink) flow rate to the isolated phase bus duct cooling system will be increased to 160 gpm at 95°F to remove the additional heat from the isolated phase bus duct. The cooling coils, fans, and the fan motors will be upgraded. The duct work will be modified to accommodate the larger fans and cooling coils. Instrumentation in the duct work will be replaced. To accommodate the increased plant service water flow, the plant service inlet and outlet piping to the cooling coils will be increased from 2 1/2" to 3" diameter, and an orifice in the discharge piping will be removed or resized. The inlet and outlet carbon steel piping will be upgraded to a more corrosion resistant material. Also, the aluminum bars used as connectors at the generator will be replaced with flexible copper braided connectors to improve reliability and reduce maintenance.

Response to NRC Question 1.c

The transformers are adequately rated to support power uprate operation; however a new temperature monitoring system will provide more accurate data for winding and oil temperatures during the power ascension program. Plant Operations personnel can also use the temperature monitoring system to provide more accurate data for determining transformer performance during various degrees of electrical loading against other operational influences such as environmental considerations or loss of cooling fan(s). A discussion of the analysis performed on the Unit 2 transformers follows.

Unit 2 has three single phase main generator step-up transformers each rated at 297/332.6 MVA, FOA at 55°C/65°C for a total load rating of 891/997.8 MVA.

At the extended power uprate, Unit 2 will normally generate 935 MWe (964 MVA) with an expected plant auxiliary load of 35 MWe at 0.97 power factor. The main step-up transformers will carry the normal operation load of 928 MVA (964-36 = 928 MVA) which is below the FOA ratings of 997.8 MVA for the transformers. All the above values are nominal values.

The postulated worst case loading for the Unit 2 main step-up transformers with power uprate could be as high as 991 MVA (1030-39 = 991 MVA, based on power generation of 1030 MVA and minus 39 MVA house loads on UATs). This loading is also below the FOA rating of the transformers and is based on highly unusual simultaneous events such as loss of the Thalman to Duval 500 kV line combined with the peak load demand during the winter cold temperatures.

Response to NRC Question 1.d

During the power uprate program, the main generator controls and switchyard devices will require adjustment and/or replacement of some components for proper operation of the equipment. Items 1-5 are setpoint changes for nonsafety-related, BOP equipment. Items 6-8 are changes to Technical Specifications setpoints which will not be implemented prior to NRC approval. The controls and the devices that will be impacted and would require adjustment or change are as follows:

1. Reset the underexcited reactive ampere limit (URAL) and annunciators/controls for the excitation system.

   General Electric is the original equipment manufacturer for the Plant Hatch Units 1 and 2 generator units. GE reviewed the ability of these two generators to meet the requested uprate to 1050 MVA at 0.88 power factor lagging against internal GE design criteria, as well as all applicable ANSI and IEEE standards. The review concluded that the Plant Hatch generator units are capable of meeting the proposed uprate without undergoing physical modifications to the generator, coolers, or excitation system hardware. The Units 1 and 2 generator electrical characteristics for the original design and extended power uprate conditions are provided in Table 1.d-1.

TABLE 1.d-1

GENERATOR ELECTRICAL CHARACTERISTICS

Electrical Characteristic	Original Design	Extended Power Uprate Design
KW	850,000	924,000
KVA	1,000,000	1,050,000
VOLTAGE (KV)	24	24
POWER FACTOR	0.85 LAGGING	0.88 LAGGING
FIELD AMPS	4208	4270
STATOR AMPS	24056	25259
H2 GAS PRESSURE (PSIG)	60	60
EXCITATION VOLTS	500	510
RECTIFIER OUTPUT (KW/V)	2235/500	2310/510

A new set of generator performance curves, including Excitation VEE Curves, Reactive Capability Curves, and Saturation Curves, were provided as plant specific instruction book updates. The information contained in these curves will be programmed into the turbine generator and excitation control systems during the appropriate refueling outage. These programming changes will allow operation at the new level of reactive ampere limits, annunciator setpoints, and various other settings resulting from the extended power uprate generator re-rating.

2. Reset turbine runback controls/setpoints.

The setpoints and controls associated with the stator water cooling runback logic to be changed as a result of extended power uprate are provided in Table 1.d-2.

TABLE 1.d-2

Parameter	Current Valve	EPU Valve
Bulk WTR Out Temperature Alarm (H)	76°C	77°C
Bulk WTR Out Temperature Runback (H)	81°C	82°C
Stator Bar Out Temperature Alarm (H)	81°C	82°C
Stator Bar Out Temperature Alarm (H)	76°C	82°C

3. Reset the 3000/5A GE Type BR-B Offerman Line CTs from the 1200 to the 2000 ampere tap.

   The 3000/5 A current transformers (CTs) for the Offerman line are currently tapped at 1200/5 A. During extended power uprate operations, these CTs could see currents in excess of 1200 A if 230 kV switchyard breaker 510 (between the Unit 1 generator and 230 kV Bus 1) is out for maintenance or is tripped. Therefore, the CT taps will be changed from 1200/5A to 2000/5A.

4. Reset the Offerman Line relays to account for new CT tap.

   As a result of CT tap changes for extended power uprate (Reference Item 3 above), 12 protection relays for the 230 kV Offerman Line and 2 ground detectors associated with 230 kV switchyard breaker 500 (between the Unit 1 generator and the Offerman line) and breaker 490 (between the Offerman line and 230 kV Bus 2) will be reset.

5. An IRQ-9 directional ground relay with an instantaneous range of 10 to 40 will be replaced with an identical type relay with a new range of 4 to 16 or with a microprocessor relay with fault locating capability.

   As a result of CT tap changes for power uprate (Reference Item 3 above), the existing IRQ-9 directional ground relay will be replaced with a new relay. The existing relay has a setting of 12 with an instantaneous range of 10 to 40. The replacement relay will have a setting of 7 with an instantaneous range of 4 to 16.

6. Main steam line high flow.

   The main steam line high flow setpoint will be set high enough to minimize the chance of a spurious isolation trip in the event of a single MSIV closure. Closure of one MSIV results in the full rated vessel steam flow passing through the remaining three open steamlines (133% of rated steam flow in each steamline). Therefore, the analytical limit for the high steam line flow setpoint remains the same when expressed as a percentage of rated steam flow (i.e., ≤ 140%). The absolute value of the setpoint will be increased to account for the increase in rated steam flow from the current power uprate condition to the extended power uprate condition. There is sufficient margin in the higher setpoint to provide assurance of isolation during a DBA main steamline break accident.

7. Bypass for turbine stop valve closure and turbine control bypass valve fast closure.

   The scram bypass setpoint for turbine stop valve closure and turbine control valve fast closure is based on the turbine bypass capacity and is not affected by the increase in rated reactor power. Therefore, this setpoint is conservatively maintained at the same absolute power level, and no change in the absolute setpoint is required. The relative power level at which scram bypass occurs is reduced from about 30% of the current power level to about 28% of extended uprate power.

8. APRM simulated thermal power scram.

   The bias at zero drive flow for the APRM simulated power scram has been increased from 58% to 62% of rated thermal power. This limit has been increased in response to the implementation of the MELLL rod line. This information is given in Figure 2-1 of the Plant Hatch extended power uprate safety analysis report (NEDC-32749P). The transient analyses have been performed consistent with the increased limit.

NRC QUESTION 2

Section 6.1 (p. 6-1 and 2) provides significant results of the extended power uprate analysis without any details about how the analysis was performed or what assumptions were relied upon to complete the analysis. Explain how the following results were determined, and any necessary assumptions needed to reach these results, in sufficient detail to enable the staff to evaluate your analysis method and the results.

a. "The main electrical generator stator cooling system has been enhanced to increase the MVA rating of the main generator for the extended uprate power level of 2763 MWt."

b. "With cooling system improvements, the isolated phase bus duct is adequate for both rated voltage and low voltage current." (Explain low voltage current.)

c. "The main transformer and the associated switchyard components are adequate for the uprated transformer output."

d. "System grid load flow studies...concluded that resultant switchyard voltages were adequate to perform safe shutdown from the offsite power network."

e. "Station loads under emergency operation/distribution conditions (diesel generators) are based on equipment nameplate data except for core spray, plant service water pumps, and RHR pumps, where a higher brake horse power (BHP) than required is used....the electrical supply and distribution components are adequate." (Were any power factors or diversity factors in the analysis affected?)

SNC RESPONSE

Response to NRC Question 2.a

The main electrical generator stator water cooling (SWC) system, including skid hardware and piping was reviewed for the capacity to meet the extended power uprate (EPU) conditions as defined in GE's electrical design specification. Cooling water flow, pressure and liquid cooler heat load were the three parameters of interest in reviewing the SWC system. Specifically for Plant Hatch, the data in Table 2.a-1 was used to evaluate the original system:

TABLE 2.a-1

Parameter	Original	EPU Conditions
Liquid cooler heat load (kW)	3686	4066
Winding pressure (psig)	41.14	44.7
Winding Flow (gpm)	515.4	537.0

The stator cooling system skid components are rated in classes based on their flow capacity. Actual plant operating data was gathered to ensure accuracy during the review and to verify the system was operating per original design. Three components were found to be outside the acceptable flow range for the required uprated flow. The main flow orifice, flow meter and main filter were originally rated for 525 gpm resulting in a need to change the components to 600 gpm rated replacements. The results of the review were typical because the auxiliary systems are designed with considerable margin for long-term reliability. The component changes for Plant Hatch are intended to preserve that reliability by maintaining the original design margin.

Response to NRC Question 2.b

The following detailed discussions are provided separately for each unit.

Unit 1

Unit 1 isophase bus duct sections leading to the main generator step-up transformer are rated at 25,323 A (1000 MVA, at 22.8 kV).

After the extended power uprate, Unit 1 will normally generate 915 MWe (943 MVA) with an expected plant auxiliary load of 35 MWe. These values are nominal and based on a rated voltage of 24 kV and a 0.97 power factor. The normal load current carried by the isophase bus duct sections will be 22,685 A (943 MVA). The worst case load current carried by the isophase bus duct sections will be 25,146 A (993 MVA).

At a constant MVA, the current carried by the isophase bus duct sections will be maximum when the generator output voltage will be at the minimum, 22.8 kV. Under the postulated worst case loading and minimum voltage operation of 22.8 kV, the isophase bus duct for Unit 1 will carry a maximum current of 25,146 A (993 MVA) which is below the 25,323 A (1000 MVA) rating of the isophase bus duct sections.

Unit 2

Unit 2 isophase bus duct sections leading to the main generator step-up transformers are rated at 25,322 A (1000 MVA, at 22.8 kV).

After the extended power uprate, Unit 2 will normally generate 935 MWe (964 MVA) with an expected plant auxiliary load of 35 MWe. These values are nominal and based on a rated voltage of 24 kV and a 0.97 power factor. The normal load current carried by the isophase bus duct sections will be 23,191 A (964 MVA). The worst case load current carried by the isophase bus duct sections will be 26,083 A (1030 MVA).

At a constant MVA, the current carried by the isophase bus duct sections will be maximum when the generator output voltage will be at the minimum, 22.8 kV. Under the postulated worst case loading and minimum voltage operation of 22.8 kV, the isophase bus duct for Unit 2 will carry a maximum current of 26,083 A (1030 MVA) which is higher than the 25,323 A (1000 MVA) rating of the isophase bus sections by 760 A or approximately 3%. To carry the uprated current, the rating of the isophase bus duct sections will be improved to 26,589 A (1050 MVA at 22.8 kV) which is equal to the revised MVA nameplate rating of the main generator.

The modification replaces the existing 60 hp fan cooling unit for the bus ducts with a new 125 hp fan cooling unit.

With the above modifications in place, the isophase bus duct sections will have adequate capacity to carry the maximum generator full load current at the nominal voltage of 24 kV and at the minimum voltage of 22.8 kV (low voltage current) under the postulated worst case loading conditions.

Response to NRC Question 2.c

The following detailed discussions are provided separately for each unit and include the assumption that the modifications discussed in response to NRC Question 1.d are in place.

Unit 1

The Unit 1 main generator step-up transformer is rated at 1008 MVA, FOA at 65°C.

At the uprated power levels, Unit 1 will generate 915 MWe (943 MVA) of nominal and 993 MVA of the postulated worst case power. The expected plant auxiliary loads are 35 MWe (36 MVA) nominal and 39 MVA maximum. The main step-up transformers will carry the normal operation load of 907 MVA (943-36 = 907) or the worst case load of 954 MVA (993-39 = 954). The normal and the worst case loads are below the FOA rating of 1008 MVA, therefore, the transformer is adequate for the extended power uprate operations.

The 230 kV switchyard is connected to the Unit 1 main step-up transformer by power circuit breakers and motor operated disconnect switches rated at 2500 A, except breaker 490 which is rated at 2000 A.

The postulated maximum worst case load of 2395 A (954 MVA at 230 kV) is below the 2500 A rating of the equipment except for breaker 490 which is feeding Bus No. 2. The current carried by breaker 490 would depend upon the switchyard configuration. During normal plant operation, power from Unit 1 is split between three 230 kV busses, Bus No. 1 (breaker 510), the Offerman Line, and Bus No. 2 (breaker 490). Breaker 490 would normally carry current below its 2000 A rating. The typical loads for the Offerman line are between 600 A and 1200 A with a historic low load value of 392 A, noted during April, 1995. However, if switchyard breaker 510 trips or is out for maintenance, then Plant Hatch Unit 1 output power will be split between the Offerman Line and 230 kV Bus No. 2 (breaker 490). If this condition occurs, the load of 2395 A will be split between the Offerman Line and 230 kV Bus No. 2. The Offerman Line by definition would automatically carry more load than normal due to grid configuration, thus ensuring load comparable to historical low loads are never achieved in this configuration. The breaker 490 will carry a current of 2395-600 = 1795 A which is below the 2000 A rating of the breaker, and therefore, it is adequate for the extended power uprate.

In view of the above, the 230 kV switchyard buses, breakers and switches are adequately rated for the uprated power. After the current transformers (CTs) for the Offerman line are reset to 2000/5A, IRQ-9 directional ground relay is replaced and protective relays for the Offerman Line are reset, all the devices and switchyard components will be adequately rated for the extended power uprate.

Unit 2

The three main generator step-up transformers for Unit 2 are each rated at 297/332.6 MVA, FOA at 55°C/65°C for a total load rating of 891/997.8 MVA.

At the uprated power levels, Unit 2 will generate 935 MWe (964 MVA) of nominal and 1030 MVA of postulated worst case maximum power. The expected plant auxiliary loads are 35 MWe (36 MVA) nominal and 39 MVA maximum. The main step-up transformers will carry the normal operation load of 928 MVA (964-36 = 928 MVA) or the worst case load of 991 MVA (1030-39 = 991 MVA). The normal and the worst case loads are below the FOA rating of 997.8 MVA, therefore, the transformers are adequate for the extended power uprate operations.

The 500 kV switchyard is connected to the Unit 2 main step-up transformer by power circuit breakers and motor operated disconnect switches rated at 3000 A. The postulated worst case maximum load is 1144 A (991 MVA, at 500 kV) which is below the switchyard equipment rating of 3000 A.

In view of the above, it is concluded that the 500 kV switchyard buses, breakers and switches are adequately rated for the extended power uprate operations.

Response to NRC Question 2.d

System grid load flow studies were performed using projected 1998 and 1999 peak and valley load conditions with Plant Hatch Unit(s) at 2763 MWt for scenarios outlined in the Plant Hatch Unit 2 FSAR Section 8.2.2.3.

The postulated worst case condition involves the loss of the Thalmann to Duval 500 kV line with a subsequent LOCA in either Unit. The model (see SNC response to NRC Question 15) assumes a maximum load on the grid, which is representative of the maximum one hour demand peak load during the summer season. Further, the grid is postulated to deliver the maximum guaranteed demand of 3600 MWe of power to Florida which results in the lowest voltage in the Plant Hatch switchyard.

The model shows that Unit 1 can sustain delivering up to 992 MVA (909 MWe and generating 398 MVARs) to maintain the 230 kV switchyard voltage at 101.3% per unit to assure the integrity of the offsite power system network for LOCA mitigation of Unit 2 and remain within the design capability curve of the main generator.

Similarly, the case study revealed that Unit 2 can sustain delivery up to 1013 MVA (919 MWe and generating 427 MVARs) to maintain the 230 kV switchyard voltage greater than 101.3% per unit to assure the integrity of the offsite power system network for LOCA mitigation of Unit 1 and remain within the design capability curve of the main generator.

During the postulated grid upset conditions, system dispatchers will take compensatory measures to restore the grid voltages by turning on capacitors banks etc., within minutes of the event to maintain long-term required grid voltage. This will be in accordance with their prescribed voltage schedule, which includes guaranteed minimum switchyard voltages for the Plant Hatch units.

It is concluded resultant switchyard voltages after the power uprate will be adequate to perform the safe shutdown of the units from the offsite network.

Response to NRC Question 2.e

The extended power uprate will not change the power requirements of the CS, RHR, RHRSW or PSW pumps or any other safety-related load; therefore, the power factors and diversity factors of the emergency loads will remain unchanged due to power uprate.

NRC QUESTION 3

The second paragraph of section 9.3.2. states (p.9-4):

"As part of the extended power uprate, the Station Blackout (SBO) scenario was reanalyzed assuming that suppression pool was initiated in **one hour** when the **alternate AC is assumed available**. The peak pool temperature is 167°F. **Even if SPC is not initiated until four hours**, the resulting peak pool temperature of 194°F is acceptable for containment and ECCS pump operation." (emphasis added)

This paragraph appears to indicate that the alternate AC power source, Emergency Diesel Generator (EDG) 1B, will be available to provide suppression pool cooling (SPC) 1 hour after an SBO and to limit the peak pool temperature to 167°F. However, in a May 3, 1991, response to a staff concern that EDG 1B would be overloaded unless operators shed load during an SBO, the staff was told that Georgia Power Company has completed analyses, which demonstrate that SPC was not required during the 4-hour SBO coping duration. The staff was also told that without SPC, the EDG loading could be reduced to 1390 kW, which is well within the 2850 kW continuous rating of EDG 1B. Based on these statements, the staff closed its open review item regarding overloading of the EDG during an SBO in a supplemental safety evaluation dated November 1, 1991.

a. Based on the above information, clarify whether credit is being taken for SPC availability within 1 hour or within 4 hours. If SPC is assumed available within 1 hour, provide: (1) a loading analysis of EDG-1B during an SBO; (2) copies of procedures used by operators to prevent an overload of EDG 1B during an SBO; and copies of any other information necessary for the staff to verify that EDG 1B will not be overloaded during an SBO.

b. Are there any instances where assumptions used in the extended power uprate analysis differ from assumptions made in the SBO analysis that was reviewed and accepted by the NRC staff?

SNC RESPONSE

Response to NRC Question 3.a

The staff concern documented in the NRC letter dated March 5, 1991 indicates that the SBO alternate AC power source, EDG 1B, has a continuous rating of 2850 kW, a 168 hour rating of 3250 kW and an assumed 2000 hour rating of 3100 kW (based on the documented rating of the other EDGs on site). The expected SBO load is documented as 3245 kW. The NRC concern was that the SBO loads are higher than the assumed 2000 hour rating of EDG 1B.

In the GPC response to this concern dated May 3, 1991, GPC indicated that emergency operating procedures allow the operator to load the EDGs up to the 168 hour rating of 3250 kW. GPC also stated that the operators would be provided a list of loads which could be stripped from the emergency buses, if required, to reduce EDG loading. Therefore, SPC could be initiated within 1 hour of the SBO, if required. As stated in the May 3, 1991 response, GPC also performed an analysis that indicated SPC is not required within the 4-hour coping duration of an SBO. The staff documented NRC acceptance of the GPC position in the NRC SER dated November 1, 1991. The SER indicated that without the need to initiate SPC within 4 hours the EDG loads will remain below the 2000 hour rating and found that acceptable. The SER also includes the statement by the licensee that procedural guidance will be provided for shedding loads from the emergency buses to allow loading of EDG 1B up to the 168 hour rating of 3250 kW.

The fact that the SBO analysis shows SPC is not required for 4 hours does not preclude using the EDG to power SPC within 1 hour if the diesel loading margins are met and the operator so chooses. There is no change to the safety-related loads that would be powered by EDG 1B due to extended power uprate. Therefore, the existing assumptions related to 1 hour initiation of SPC and the acceptability of the 4 hour coping period are no different than those documented in the original SBO submittals which were approved by the NRC in the SER dated November 1, 1991. Since there is no change in the EDG loading due to extended power uprate, and the original SBO analysis was approved by the NRC, it is not necessary to submit a loading analysis of EDG 1B or operating procedures for prevention of EDG overload.

Response to NRC Question 3.b

An evaluation of the SBO analyses was performed to determine the effects of increased decay heat due to extended power uprate. The reactor system and containment pressure/temperature response was reanalyzed. Because the EDG loading for an SBO did not change, the reactor/containment reanalysis for extended power uprate includes an evaluation of SPC initiation within 1 hour to ensure that, should EDG margin be available, the operator may initiate SPC. The reactor/containment reanalysis was also reviewed to ensure that initiation of SPC is not necessary within the 4 hour coping period for extended power uprate conditions.

The original SBO coping evaluation was based on analyses performed by GE for the 10 CFR 50, Appendix R program, as documented in the GPC SBO submittal dated May 3, 1991. For the original power uprate, the Appendix R program results were extrapolated for the 105% power uprate conditions and found acceptable.

The Appendix R analysis was performed using evaluation models that are conservative relative to the latest NRC approved GE evaluation models. (See the response to NRC Question 15.) Therefore, the SBO reactor/containment reanalysis for extended power uprate was performed using the more recent models. The major assumptions for the SBO event reanalysis are included in Table 3.b-1. Several SBO cases were analyzed with various assumptions as described below. The results of the various cases are included in Table 3.b-2.

Case 1 : Performed using the new model with the original SBO analysis assumptions at a power level of 2436 MWt to show the level of conservatism in the original SBO model.

Case 2 : Performed assuming the EDG loading would be below the required loading limit, and SPC could be initiated within 1 hour.

Case 3 : Performed assuming no SPC within the 4 hour coping period and no RPV depressurization.

Case 4 : Performed assuming no SPC within the 4 hour coping period with RPV depressurization when required to prevent exceeding the suppression pool HCTL.

In case 3, SPC was not initiated when the normal plant power is assumed to be available, i.e., 4 hours. The suppression pool temperature after 4 hours was determined to be 188°F. The peak pool temperature occurred approximately 7 hours after initiation of the SBO and was calculated to be 194°F. However, the suppression pool heat capacity temperature limit (HCTL) was exceeded due to RPV pressurization. The maximum pool temperature for case 3 was included in section 9.3.2 of NEDC-32749P in error. The maximum pool temperature actually occurs with case 4. The change to page 9-4 of the safety analysis report is included in Enclosure 2.

The results of the reactor system and containment pressure/temperature SBO reactor/containment reanalysis show the resulting peak pool temperatures with either the 1 hour or the 4 hour SPC initiation is acceptable for containment and ECCS performance.

An evaluation, rather than a reanalysis, of the remaining elements of the original SBO analysis was adequate to demonstrate the plant can successfully cope with the SBO event under extended power uprate conditions. The assumptions inherent in these evaluations are consistent with the assumptions made in the original SBO analysis reviewed and accepted by the NRC staff.

TABLE 3.b-1

STATION BLACKOUT ANALYSIS ASSUMPTION

Parameter	Assumptions
1. Initial Power	2763 MWt
2. Vessel Inventory Model	SAFER
3. Containment Pressure/Temperature Evaluation Model	SHEX
4. Initial Suppression Pool Temperature	100°F
5. Drywell Heat Load	Included
6. Drywell/Wetwell Heat Sinks	Included
7. Suppression Pool Cooling Initiation Time	For the licensing submittal originally assumed 1 hour but has now also been recalculated for 4 hours.
8. RCIC Flowrate	Adjusted to match decay heat boil-off per Plant Hatch operating procedures. Vessel depressurization initiation delayed to 3.5 hours for the extended power uprate condition.

TABLE 3.b-2

RESULTS OF SBO CASES

	Case 1	Case 2	Case 3	Case 4
	Reactor power at original licensed power	SPC initiated at 1 hr	SPC initiated at 4 hrs without RPV depress	SPC initiated at 4 hrs with RPV depress
Reactor Power	2436 MWt	2763 MWt	2763 MWt	2763 MWt
SPC Initiation Time (hrs)	4 hrs	1 hr	4 hrs	4 hrs
RPV Depressurization Initiation Time	No RPV Depress.(1)	No RPV Depress.(1)	No RPV Depress.(2)	3.5 hrs(4)
Peak pool temperature at 4 hrs. (i.e., the end of the SBO coping period)	177°F	167°F	188°F (194°F(3))	206°F

Notes for Table 3.b-2:

(1) No RPV depressurization is required since the suppression pool temperature remains within the suppression pool HCTL.

(2) No RPV depressurization is assumed to be consistent with Case 2. However, the suppression pool HCTL is exceeded at 3.5 hours.

(3) The suppression pool would reach a peak pool temperature of 194°F, if no credit is taken for operator action to terminate SRV discharge to the pool once AC power is restored at four hours.

(4) Vessel depressurization required to remain within suppression pool HCTL.

NRC QUESTION 4

The last paragraph in Section 10.2.1.1 states (p. 10-3 and 4):

A review of the equipment qualification at the extended power uprate conditions identified some equipment located within the containment which is potentially affected by the higher normal radiation level. The qualification of this equipment has been reevaluated based on location specific dose calculations. These components have been found acceptable for extended power uprate conditions, although it will be necessary to decrease the qualified life of specific components because of increased radiation levels during extended power uprate conditions.

a. Provide a detailed discussion of what components are potentially affected and quantify the decrease in qualified life for potentially affected components.

b. Provide a detailed discussion of the review process used to identify potentially affected components, and provide the actual equipment qualification (EQ) analysis performed for a few of the potentially affected components to include the EQ analysis performed for the component, which has the greatest decrease in qualified life.

SNC RESPONSE

Response to NRC Question 4.a

An EQ master list is maintained which includes the Plant Hatch equipment qualified for a harsh environment. The equipment listed in the EQ Master List was evaluated to determine the effect of extended power uprate on the equipment qualification. The decrease in qualified life for the affected components is quantified in response to Question 4b.

Response to NRC Question 4.b

The environmental qualification review for extended power uprate (EPU) consisted of:

- An initial screening of all components on the EQ Master List for the effect on qualification parameters such as temperature, pressure and radiation.
- A location specific dose calculation for the equipment identified in the initial screening as potentially impacted.

- A re-evaluation of all components on the EQ Master List based on the results of the location specific dose calculation.

The greatest potential impact on equipment qualification was determined to be radiation. The initial screening assumed an 8% increase in the specified 40-year total integrated dose (TID) inside the drywell. From this initial screening, Target Rock solenoid valves model 1/2 SMS-S-02-4 and Namco limit switches models EA180 and EA740 located inside the drywell were identified as being potentially impacted. A location specific dose calculation was then prepared for the Namco limit switches and Target Rock solenoid valves inside the drywell. The following methodology was utilized for determining the location specific doses.

The total qualification dose consists of two components 1) dose due to 40 years of normal plant operation, and 2) dose due to a design basis accident. The normal operation dose inside the drywell is predominantly from the RPV, the recirculation lines, and the main steam (MS) lines. Hence, the normal dose component for extended power uprate was determined based on the contribution from the RPV, recirculation lines, and the MS lines at the location of the specific component. The doses from the MS lines included the effect of increased N-16 concentration due to the increased hydrogen injection rate expected for extended power uprate. The accident dose component was adjusted for the increased power level. The total integrated dose for these components was determined by adding the 40 years normal and accident dose components.

The calculation demonstrated the Target Rock solenoid valves inside the drywell are qualified for the 40-year TID at extended power uprate conditions. The calculation also identified five Namco limit switches with 40-year, location specific doses that will exceed the tested dose of 5.0E+07 rads. Therefore, the calculation included a determination of the revised qualified life based on radiation.

Temperature is currently the most limiting factor in determining the qualified life of the switches. The thermal qualified life remains the most limiting for three of the limit switches, however, the radiation qualified life will be the most limiting for two of the limit switches after extended power uprate. Table 4.b-1 summarizes these results.

TABLE 4.b-1

MPL	Radiation Qualified Life (yrs)		Thermal Qualified Life (yrs)	
	Current	EPU	Current	EPU
1B31-AOV F019	40.0	19.7	18.3	18.3*
1B21-AOV F037A	40.0	15.2*	18.3	18.3
1B21-AOV F037B	40.0	15.2*	18.3	18.3
2E21-AOV F037A	40.0	16.2	4.6	4.6*
2E21-AOV F037B	40.0	16.2	4.6	4.6*

*Most Limiting

The calculation also determined a revised bounding maximum dose for any equipment in the drywell. The qualified life of all EQ components in the drywell was reviewed with respect to the revised maximum bounding dose. No additional equipment was affected. The replacement intervals for affected equipment will be revised upon implementation of extended power uprate.

NRC QUESTION 5

Describe the impact of the power uprate on the diesel generator load profile during loss of offsite power (LOSP) and LOSP with loss-of-coolant accident (LOCA).

SNC RESPONSE

Extended power uprate will not change the load requirements of the CS, RHR, RHRSW or PSW pumps or any other safety-related load.

The proposed uprate will result in a slight increase in the power requirements of the Condensate Pumps, Condensate Booster Pumps, Recirculation Pump MG set motor drives, and the Isophase Bus duct coolers. These loads are connected to 4.16 kV nonsafety-related buses A, B, C and D. No safety-related busses are impacted by the change in the above loads.

Therefore, the EDG loading profiles during LOSP and LOSP/LOCA remain unchanged.

NRC QUESTION 6

Describe the impact of the power uprate on grid stability and reliability.

SNC RESPONSE

The plant stability scenarios outlined in Plant Hatch Unit 2 FSAR Section 8.2.2.3 were evaluated at the 2763 MWt extended uprate power level. The grid stability review was conducted to envelop the maximum expected power after the uprate of 930 MWe for Unit 1 and 940 MWe for Unit 2. Stability concerns are generally worse for system valley load conditions. Therefore, the valley load level was used for the review along with peak system load conditions. Previous studies have shown that close-in, three-phase faults with breaker failure are the most challenging to the stability of the Plant Hatch units and to the grid at Plant Hatch. Therefore, this stability review focused primarily on close-in, three-phase faults with breaker failure. The possible faults with breaker failure contingencies at Plant Hatch were reviewed in the study, and critical clearing times (CCT) were determined for each contingency. A review was performed of the margin between the CCT's and the existing breaker failure clearing times at Plant Hatch. The margins are sufficient, and no change is needed.

The stability of the grid was checked for all of the fault contingencies simulated. For each one, the 230 kV switchyard voltage experienced a transient dip but recovered to a value above the minimum allowed steady state voltage of 101.3% in approximately one second. Therefore, there will be no actuation of loss-of-offsite-power protection for any expected fault.

NRC QUESTION 7

Describe the impact of the power uprate on the station auxiliary electrical distribution system (i.e., changes in load, voltage, and short circuit current values).

SNC RESPONSE

The following discussions are provided separately for each unit.

Unit 1

Due to extended power uprate, there will be an increase in the power requirements of Unit 1 condensate pump motors, condensate booster pump motors, recirculation pump MG set motors and heater drain pumps. These loads are supplied from the unit auxiliary transformers (UATs), and there is no impact to the Class 1E bus voltages. These loads have been analyzed and found acceptable.

There is no required change to equipment short circuit ratings. The increase in motor loadings as a consequence of extended power uprate is within the existing nameplate rating of the affected motors, and the existing analysis is based on motor nameplate ratings.

Unit 2

Due to extended power uprate, there will be an increase in the power requirements of Unit 2 condensate pump motors, condensate booster pump motors, recirculation pump MG set motors, heater drain pumps and isophase bus cooling motor load. These loads are supplied from the unit auxiliary transformers (UATs), and there is no impact to Class 1E bus voltages. These loads have been analyzed and found acceptable.

There is no required change to equipment short circuit ratings. The increase in moto. loading as a consequence of extended power uprate is within the existing nameplate rating of the affected motors. The existing analysis is based on motor nameplate ratings. The conclusion that no analysis change is needed accounts for the upgrade of the isophase bus cooling fans from 60 to 125 hp. The upgrade is found acceptable and will have an insignificant impact on the bus voltages and short circuit levels in the system.

NRC QUESTION 8

Describe the impact of the power uprate on the degraded voltage protection scheme (i.e, degraded grid relay setpoints).

SNC RESPONSE

The degraded grid relays are located on the Class 1E buses which are supplied from the startup auxiliary transformers (SATs). There is no impact on Unit 1 voltages seen by these relays as a result of extended power uprate. The impact on Unit 2 is from the isophase bus fan motor hp increase for extended power uprate operation. The net increase is a change of about 9 amps at 4160V through the SAT configuration as discussed in response to NRC Question 7. The degraded grid relay setpoints will not change due to the decrease in bus E voltage of 0.06% and increase in buses F and G voltage of 0.02%. The change in the above voltages is due to the higher rating of the Unit 2 isophase bus cooler motor and relocation of the motor load to a different electrical bus fed from the SAT.

MATERIALS ENGINEERING:

NRC QUESTION 9

In the power uprate submittal, the licensee identified the need to address changes in the pressure-temperature limits for both Plant Hatch Unit 1 and Unit 2 based upon projected changes in the reactor pressure vessel (RPV) neutron fluence at various effective full power year (EFPY) levels. Identify the following information for the 20, 24, 28, and 32 EFPY levels (see Enclosure 4, Figure 3.4.9-1 for Plant Hatch Unit 1) and for the 32 EFPY level of Plant Hatch Unit 2.

(a) Identify the limiting material.

(b) Provide the proposed copper and nickel content of all beltline RPV welds.

(c) Provide the clad-to-base metal interface fluence, the 1/4 T fluence, and the 3/4 T fluences at the subject EFPY levels for all beltline materials.

(d) Identify and explain the use of any RPV surveillance program results which impact the determination of the limiting RPV material(s).

SNC RESPONSE

Tables 9-1 through 9-4 contain the requested information concerning the RPV beltline materials of Plant Hatch Units 1 and 2. The value of surface fluence for Unit 1 comes from the second capsule flux wire analysis results. The chemistry factor, based on percent copper and nickel content included in Regulatory Guide 1.99, Revision 2, was adjusted because of two available sets of surveillance data, as discussed in note 2 of Table 9-1. The Unit 2 surface fluence value was determined using the results of the first cycle dosimeter.

TABLE 9-1

INFORMATION ON LIMITING BELTLINE MATERIALS

EFPY Level	LIMITING MATERIAL	%Cu	%Ni	SURFACE FLUENCE (1) n/cm^2	1/4T FLUENCE (1) n/cm^2	3/4T FLUENCE (1) n/cm^2	CF	ADJUSTED CF (2)
HATCH 1:								
20 EFPY	Plate Heat C4337-1	0.17	0.62	1.11E+18	8.04E+17	4.22E+17	128	334
24 EFPY	Plate Heat C4337-1	0.17	0.62	1.39E+18	1.01E+18	5.28E+17	128	334
28 EFPY	Plate Heat C4337-1	0.17	0.62	1.66E+18	1.20E+18	6.31E+17	128	334
32 EFPY	Plate Heat C4337-1	0.17	0.62	1.94E+18	1.40E+18	7.37E+17	128	334
HATCH 2:								
32 EFPY	Weld Heat 51874, Linde 0091, Flux Lot 3458	0.18	0.50	2.17E+18	1.57E+18	8.25E+17	138	---

(1) Fluences are at peak location.

(2) Plant Hatch Unit 1 has two credible surveillance data sets. In this case, Paragraph 2.1 of Reg. Guide 1.99 Rev. 2. has guidelines for calculating adjusted reference temperature. The value of chemistry factor has been adjusted using the least squares approach and half the value of standard deviation for RT_{NDT} (8.5°F instead of 17°F) was used to calculate the margin term.

TABLE 9-2

COPPER AND NICKEL CONTENT FOR PLANT HATCH UNITS 1 AND 2 BELTLINE MATERIALS

HATCH 1 COMPONENT	WELD TYPE	HEAT OR HEAT/LOT	%Cu	%Ni	HATCH 2 COMPONENT	WELD TYPE	HEAT OR HEAT/LOT	%Cu	%Ni
PLATES:					**PLATES:**				
Lower					**Lower**				
G-4805-1		C4112-1	0.13	0.64	G-6603-1		C8553-2	0.08	0.58
G-4805-2		C4112-2	0.13	0.64	G-6603-2		C8553-1	0.08	0.58
G-4805-3		C4149-1	0.14	0.57	G-6603-3		C8571-1	0.08	0.53
Lower-Intermd.					**Lower-Intermd.**				
G-4803-7		C4337-1	0.17	0.62	G-6602-2		C8554-1	0.08	0.57
G-4804-1		C3985-2	0.13	0.58	G-6602-1		C8554-2	0.08	0.58
G-4804-2		C4114-2	0.13	0.70	G-6601-4		C8579-2	0.11	0.48
WELDS:					**WELDS:**				
Lower, Long.	1-307	13253,1092/ Flux 3791	0.27	0.74	Lower, Long.	101-842	10137, Linde 0091/ Flux 3999	0.23	0.50
Lower-Intermd. Long.	1-308	IP2809,1092/ Flux 3854	0.28	0.76	Lower-Intermd. Long.	101-834	51874, Linde 0091/ Flux 3854	0.18	0.50
		IP2815,1092/ Flux 3854	0.28	0.76					
Lower to Lower-Intermd. Girth	1-313	90099,0091/ Flux 3977 33A277,0091/ Flux 3977	0.17 0.23	1.00 1.00	Lower to Lower-Intermd. Girth	301-871	4P6052,Linde 0091/ Flux 0145	0.07	0.03

TABLE 9-3

PEAK FLUENCES FOR PLANT HATCH UNIT 1 BELTLINE MATERIALS

COMPONENT HEAT OR HEAT/LOT	20 EFPY			24 EFPY			28 EFPY			32EFPY		
	SURFACE FLUENCE n/cm^2	1/4T FLUENCE n/cm^2	3/4T FLUENCE n/cm^2	SURFACE FLUENCE n/cm^2	1/4T FLUENCE n/cm^2	3/4T FLUENCE n/cm^2	SURFACE FLUENCE n/cm^2	1/4T FLUENCE n/cm^2	3/4T FLUENCE n/cm^2	SURFACE FLUENCE n/cm^2	1/4T FLUENCE n/cm^2	3/4T FLUENCE n/cm^2
PLATES:												
Lower												
C4112-1	7.55E+17	5.15E+17	2.39E+17	9.45E+17	6.45E+17	3.00E+17	1.13E+18	7.70E+17	3.58E+17	1.32E+18	9.00E+17	4.18E+17
C4112-2	7.55E+17	5.15E+17	2.39E+17	9.45E+17	6.45E+17	3.00E+17	1.13E+18	7.70E+17	3.58E+17	1.32E+18	9.00E+17	4.18E+17
C4149-1	7.55E+17	5.15E+17	2.39E+17	9.45E+17	6.45E+17	3.00E+17	1.13E+18	7.70E+17	3.58E+17	1.32E+18	9.00E+17	4.18E+17
Lower-Intermd												
C4337-1	1.11E+18	8.04E+17	4.22E+17	1.39E+18	1.01E+18	5.28E+17	1.66E+18	1.20E+18	6.31E+17	1.94E+18	1.40E+18	7.37E+17
C3985-2	1.11E+18	8.04E+17	4.22E+17	1.39E+18	1.01E+18	5.28E+17	1.66E+18	1.20E+18	6.31E+17	1.94E+18	1.40E+18	7.37E+17
C4114-2	1.11E+18	8.04E+17	4.22E+17	1.39E+18	1.01E+18	5.28E+17	1.66E+18	1.20E+18	6.31E+17	1.94E+18	1.40E+18	7.37E+17
WELDS:												
Lower, Longitudinal												
13253,1092/ Flux 3791	7.55E+17	5.15E+17	2.39E+17	9.45E+17	6.45E+17	3.00E+17	1.13E+18	7.70E+17	3.58E+17	1.32E+18	9.00E+17	4.18E+17
Lower-Intermd, Long.												
IP2809,1092/ Flux 3854	1.11E+18	8.04E+17	4.22E+17	1.39E+18	1.01E+18	5.28E+17	1.66E+18	1.20E+18	6.31E+17	1.94E+18	1.40E+18	7.37E+17
IP2815,1092/ Flux 3854	1.11E+18	8.04E+17	4.22E+17	1.39E+18	1.01E+18	5.28E+17	1.66E+18	1.20E+18	6.31E+17	1.94E+18	1.40E+18	7.37E+17
Girth												
90099,0091/ Flux 3977	7.55E+17	5.47E+17	2.87E+17	9.45E+17	6.84E+17	3.59E+17	1.13E+18	8.17E+17	4.29E+17	1.32E+18	9.55E+17	5.02E+17
33A277,0091/ Flux 3977	7.55E+17	5.47E+17	2.87E+17	9.45E+17	6.84E+17	3.59E+17	1.13E+18	8.17E+17	4.29E+17	1.32E+18	9.55E+17	5.02E+17

TABLE 9-4

PEAK FLUENCES FOR PLANT HATCH 2 BELTLINE MATERIALS

Component Heat or Heat/Lot	Surface Fluence n/cm²	32EFPY 1/4T Fluence n/cm²	32EFPY 3/4T Fluence n/cm²
PLATES:			
Lower			
C8553-2	1.39E+18	9.47E+17	4.41E+17
C8553-1	1.39E+18	9.47E+17	4.41E+17
C8571-1	1.39E+18	9.47E+17	4.41E+17
Lower-Intermd			
C8554-1	2.17E+18	1.57E+18	8.25E+17
C8554-2	2.17E+18	1.57E+18	8.25E+17
C8579-2	2.17E+18	1.57E+18	8.25E+17
WELDS:			
Lower, Longitudinal			
10137, Linde 0091/ Flux 3999	1.39E+18	9.47E+17	4.41E+17
Lower-Intermd, Long.			
51874, Linde 0091/ Flux 3854	2.17E+18	1.57E+18	8.25E+17
Girth			
4P6052, Linde 0091/ Flux 0145	1.39E+18	1.01E+18	5.28E+17

NRC QUESTION 10

Assess the projected drop in Charpy upper shelf energy for each Plant Hatch Unit 1 and Unit 2 beltline material at end-of-license based upon the updated fluence evaluations in item 1. above. Determine and submit an evaluation to show whether all RPV materials for both units will continue to meet the requirements of 10 CFR Part 50, Appendix G or whether they will still be bound by previous equivalent margins analyses. If any RPV material does not meet these criteria, submit a new equivalent margins analysis to demonstrate that an acceptable upper shelf energy level will be maintained for that material through the end of the facility's operating license.

SNC RESPONSE

The previous power uprate evaluation showed an acceptable drop in upper shelf energy of the beltline material for Plant Hatch Units 1 and 2 using equivalent margin analysis. The equivalent margin analyses demonstrate that the 10 CFR 50, Appendix G requirements are satisfactorily met for Plant Hatch Units 1 and 2, as shown in Tables 10-1 through 10-4, for extended power uprate.

TABLE 10-1

EQUIVALENT MARGIN ANALYSIS
PLANT APPLICABILITY VERIFICATION FORM
HATCH UNIT 1 PLATE - BWR 4/MK I AT EXTENDED POWER UPRATE

BWR/3-6 PLATE

Surveillance Plate USE:

%Cu = 0.12

1st Capsule Fluence = 2.4×10^{17} n/cm^2
2nd Capsule Fluence = 4.6×10^{17} n/cm^2

Unirradiated to 1st Capsule Measured % Decrease = 4 (Charpy Curves)
Unirradiated to 2nd Capsule Measured % Decrease = -5 (Charpy Curves)

1st Capsule R.G. 1.99 Rev. 2 Predicted % Decrease = 9 (R.G. 1.99, Figure 2)
2nd Capsule R.G. 1.99 Rev. 2 Predicted % Decrease = 10 (R.G. 1.99, Figure 2)

Limiting Beltline Plate:

%Cu = 0.17

32 EFPY Fluence = 1.94×10^{18} n/cm^2

R.G. 1.99 Rev. 2 Predicted % Decrease = 18 (R.G. 1.99, Figure 2)

Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)

18 % ≤ 21%, so vessel plates are bounded by equivalent margin analysis

TABLE 10-2

**EQUIVALENT MARGIN ANALYSIS
PLANT APPLICABILITY VERIFICATION FORM
HATCH UNIT 1 WELD - BWR 4/MK I AT EXTENDED POWER UPRATE**

BWR/2-6 WELD

Surveillance Weld USE:

%Cu = 0.28

1st Capsule Fluence = 2.4×10^{17} n/cm^2
2nd Capsule Fluence = 4.6×10^{17} n/cm^2

Unirradiated to 1st or 2nd Capsule Measured % Decrease = Unknown
1st to 2nd Capsule Measured % Decrease = -16 (Charpy Curves)

1st Capsule R.G. 1.99 Rev. 2 Predicted % Decrease = 19 (R.G. 1.99, Figure 2)
2nd Capsule R.G. 1.99 Rev. 2 Predicted % Decrease = 22 (R.G. 1.99, Figure 2)

Limiting Beltline Weld USE:

%Cu = 0.28

32 EFPY Fluence = 1.94×10^{18} n/cm^2

R.G. 1.99 Rev. 2 Predicted % Decrease = 29 (R.G. 1.99, Figure 2)

Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)

29 % $\leq$ 34%, so vessel welds are bounded by equivalent margin analysis

TABLE 10-3

EQUIVALENT MARGIN ANALYSIS
PLANT APPLICABILITY VERIFICATION FORM
HATCH UNIT 2 PLATE - BWR 4/MK I AT EXTENDED POWER UPRATE

BWR/3-6 PLATE

Surveillance Plate USE:

%Cu = 0.08

Capsule Fluence = 2.3×10^{17} n/cm^2

Measured % Decrease = 0 (Charpy Curves)

R.G. 1.99 Rev. 2 Predicted % Decrease = 7 (R.G. 1.99, Figure 2)

Limiting Beltline Plate USE:

%Cu = 0.11

32 EFPY Fluence = 2.17×10^{18} n/cm^2

R.G. 1.99 Rev. 2 Predicted % Decrease = 14 (R.G. 1.99, Figure 2)

Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)

14 % ≤ 21%, so vessel plates are bounded by equivalent margin analysis

TABLE 10-4

EQUIVALENT MARGIN ANALYSIS
PLANT APPLICABILITY VERIFICATION FORM
HATCH UNIT 2 WELD - BWR 4/MK I AT EXTENDED POWER UPRATE

BWR/2-6 WELD

Surveillance Weld USE:

%Cu = 0.13

Capsule Fluence = 2.3×10^{17} n/cm^2

Measured % Decrease = 1 (Charpy Curves)

R.G. 1.99 Predicted % Decrease = 11 (R.G. 1.99, Figure 2)

Limiting Beltline Weld USE:

%Cu = 0.23

32 EFPY Fluence = 2.17×10^{18} n/cm^2

R.G. 1.99 Rev. 2 Predicted % Decrease = 26 (R.G. 1.99, Figure 2)

Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)

26 % ≤ 34%, so vessel welds are bounded by equivalent margin analysis

NRC QUESTION 11

Explain what effect, if any, the new power uprate fluence values will have on the RPV surveillance capsule withdrawal schedule for Plant Hatch Units 1 and 2. If the capsule withdrawal schedule for either facility needs to be amended, based upon this information, indicate what schedule will be pursued for the submittal of that amendment.

SNC RESPONSE

The RPV surveillance capsule withdrawal schedule for Plant Hatch units can be found in the FSAR. The schedule indicates a removal interval of 15 EFPY for the second capsule. The second surveillance capsule for Plant Hatch Unit 1 has been pulled; however, the second capsule for Unit 2 has not been pulled to-date. For Unit 2, it will not be necessary to change the removal interval due to the increased fluences associated with extended power uprate conditions.

NRC QUESTION 12

Provide a more detailed evaluation of the effects caused by the extended power uprate on the reactor internals (i.e., expand on your submitted determination of the effects).

SNC RESPONSE

Extended power uprate has only a limited effect on the reactor internals. Most of the operational parameters defining the environment for the reactor internals are unchanged for extended power uprate operation. The maximum reactor operating pressure and core flow are unchanged. The maximum recirculation drive flow is essentially unchanged for extended power uprate operation because the recirculation system is currently at the maximum flow during increased core flow operation. The downcomer and core inlet enthalpy range at the extended power uprate conditions is bounded by the enthalpy range at current power. The maximum steam generation in any single fuel bundle is unchanged for extended power uprate because the bundle thermal limits remain the same as those for the current power level.

The primary effect of extended power uprate operation is a slight increase in the reactor internal pressure differences (RIPDs). The increase in RIPDs is due to the higher two-phase flow losses caused by the increased steam generation in the core. The reactor internals have been evaluated for the higher RIPD loading at normal, upset, and faulted conditions due to extended power uprate as discussed in NEDC-32749P, Section 3.3.2. Further information on the reactor internals stress evaluation is presented in response to NRC Question 23.

The steam separators see a higher inlet quality and the steam dryers see a higher flow velocity as a result of the increased steam generation in the core. The steam dryer and separator performance evaluation is presented in NEDC-32749P, Section 3.3.5. Intergranular stress corrosion cracking and erosion/corrosion have been addressed generically for the reactor internals in Section 3.6.1 of NEDC-32523P, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR2) for extended power uprates up to 20%.

NRC QUESTION 13

Provide a more detailed evaluation of the effect caused by the extended power uprate on components exposed to single- and two-phase flow.

SNC RESPONSE

As described in the response to Question 12, the steam separators and steam dryers are the only reactor internals components that experience a change in the fluid flow conditions for extended power uprate. The steam separators see a higher inlet quality and the steam dryers see a higher flow velocity as a result of the increased steam generation in the core. Intergranular stress corrosion cracking and erosion/corrosion have been addressed generically for the reactor internals in Section 3.6.1 of NEDC-32523P, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR2) for extended power uprates up to 20%.

MECHANICAL ENGINEERING:

NRC QUESTION 14

In Section 1.1 of the reference (Enclosure 6 to the August 8, 1997, submittal), you stated that "most of GE BWR plants, including Plant Hatch, have the capability and margins for an uprate of up to 20% without major nuclear steam supply system (NSSS) hardware modifications." You also stated that the proposed power level was selected based on the expected cost of modifications to the balance-of-plant (BOP) equipment, but not because of the NSSS limitations. Provide a discussion regarding your selection of the proposed 8% (but not more, say 10%) power increase above the current power level. Why is this extended power uprate impacting the BOP equipment more than the NSSS systems? List the most critical modifications that are expected in the proposed power uprate at Plant Hatch.

SNC RESPONSE

The decision to implement the proposed extended power program at 108% of the current licensed power was based on engineering feasibility studies and Southern Company specific economic evaluations.

As stated in the submittal, most of GE BWR plants have the capability and design margins to support 20% uprates without major NSSS hardware modifications. The majority of the Balance of Plant (BOP) systems installed at Plant Hatch were originally provided with design margins to conservatively support 105% of the original power level of 2436 MWt. The original power uprate of 105%, implemented in 1995 and 1997, utilized the robust design of the BOP equipment, therefore no significant BOP modifications were necessary to accomplish the 5% uprate.

The proposed extended power uprate will accomplish the increase in electrical output primarily by the generation and supply of higher steam flow to the turbine generator. The reactor and turbine steam flows as well as the condensate and feedwater systems flows will increase approximately in proportion to the increase in reactor core thermal power.

A feasibility study was performed to assess the NSSS and BOP systems limitation for uprating both units to 103%, 105% and 110% of the current licensed power. The study concluded that the NSSS did not require major modification up to and beyond 110% of the current licensed power of 2558 MWt. However, the BOP systems were more limiting and extensive changes were required to implement an uprate in excess of 108%. A Southern Company private cost/benefit analysis was performed which considered system load demand and power supply options. These evaluations concluded that, for The Southern Company, an uprate of 108% of the current licensed power was desired. The most critical modifications necessary to achieve extended power uprate operation are the BOP modifications, items 1-6 in Enclosure 5 of the extended power uprate submittal.

The BOP modifications are based on assumed design margin which will be validated during startup testing. Unit operation after startup testing, for at least the first EPU operating cycle, may be established below the requested licensed reactor thermal power level to determine the potential seasonal performance impacts. Based on the operational cycle experience, with actual BOP margin demonstrated, power levels up to the requested licensed reactor thermal power will be established.

NRC QUESTION 15

In Section 1.2 of the reference, you stated that "continuing improvements in the analytical techniques (computer codes) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core designs have resulted in a significant increase in the difference between the calculated safety analyses results and the licensing limits." List all the new computer codes including the new versions of the codes that were used for the power uprate but were not identified in the design basis for Plant Hatch. Provide a justification for their application, a discussion of the benchmarking validation associated with these codes, and indicate if they have been reviewed and approved by the NRC.

SNC RESPONSE

The current reviewed and approved design basis for Plant Hatch is based on the original 105% power uprate analyses. Table 15-1 identifies the analysis areas where different computer codes were applied for the Plant Hatch extended power uprate compared to the previous Plant Hatch 5% power uprate. The table also provides information relating to the validation of the codes used for Plant Hatch extended power uprate.

TABLE 15-1

COMPUTER CODE DIFFERENCES BETWEEN 5% POWER UPRATE AND EXTENDED POWER UPRATE

Analysis	"Original" (5%) Power Uprate Computer Codes	Extended Power Uprate Computer Codes	Application and Validation to Extend Power Uprate
ECCS-LOCA	SAFER/GESTR-LOCA Version 02	SAFER/GESTR-LOCA Version 04V	Version 04 has been reviewed and approved by the NRC. Model description and qualification are documented in NEDC-23785PA
Transient Analysis	Loss of Feedwater - SAFE Station Blackout - SAFE/CHASTE (ECCS)	SAFER/GESTR-LOCA Version 04V SAFER/GESTR-LOCA Version 04V	SAFER/GESTR-LOCA Version 04 has been reviewed and approved by the NRC. Model description and qualification are documented in NEDC-23785PA
	Station Blackout- Simplified Pool Heatup Model (Containment)	SHEX - Long-Term Containment Response Version 04V	SHEX is a GE Q/A controlled code which has been used for many years for containment analysis and the results have been accepted by the NRC. Model description is documented in NEDE-20533. NRC approval of applicability of SHEX to containment analysis is given in a letter from A. Thadani to G. Sozzi entitled, "Use of SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," dated July 13, 1993. The application and benchmarking of SHEX for long-term containment cooling was previously addressed in the licensing submittal for Plant Hatch stretch power uprate (NEDC-32405P, Section 4.1.1). SHEX was not used for the SBO calculations for stretch power uprate because the results were extrapolated from earlier analyses

TABLE 15-1 (Cont.)

Analysis	"Original" (5%) Power Uprate Computer Codes	Extended Power Uprate Computer Codes	Application and Validation to Extend Power Uprate
ATWS Evaluation	REDY	ODYN Version M10V	Application of ODYN to ATWS evaluation is documented in NEDC-24 [illegible]P Supplement 1, Volume 4. Model description and quali[illegible]on are documented in NEDO-24154. ODYN has been reviewed and approved by the NRC. Application of ODYN to ATWS is currently under review by the NRC.
Radiological Impact	CONAC Version 03- Dose Calculations	CONAC Version 04	Version 04 incorporates additional parameters to improve the instrument line break model. CONAC is a GE Q/A controlled code which encodes the NRC requirements in Regulatory Guides and SRPs for the calculation of offsite doses for the design basis accident in Chapter 15 of the FSAR. . Model description is documented in NEDO-32708. Formal approval by NRC not documented.
RPV Internals	Experiment Based Calculational Methodology for Acoustic Loads	TRACG Version 01	TRACG01 model application was initiated by NRC generic letter 94-03 on shroud cracking and the response to PRC 95-01 and PRC 96-02 on Acoustic Pressure Loads on Reactor Internal Components. This model allows for full 3-D modeling of the annulus region. Model description is documented in NEDE 32176P Rev 1 and qualification is documented in NEDE 32177P and NEDC-32725P. Formal approval by NRC not documented.
Fire Protection (Appendix R)	SAFE[illegible] Vessel Inventory)	SAFER/GESTR-LOCA Version 04V	SAFER/GESTR-LOCA Version 04 has been reviewed and approved by the NRC. Model description and qualification are documented in NEDC-23785PA
	Simplified Pool Heatup Model (Containment)	SHEX - Long-Term Containment Response Version 04V	SHEX is a GE Q/A controlled code which has been used for many years for containment analysis and the results have been accepted by the NRC. Model description is documented in NEDE-20533. NRC approval of applicability of SHEX to

TABLE 15-1 (Cont.)

Analysis	"Original" (5%) Power Uprate Computer Codes	Extended Power Uprate Computer Codes	Application and Validation to Extend Power Uprate
			containment analysis is given in a letter from A. Thadani to G. Sozzi entitled, "Use of SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," dated July 13, 1993. The application and benchmarking of SHEX for long-term containment cooling was previously addressed in the licensing submittal for Plant Hatch stretch power uprate (NEDC-32405P, Section 4.1.1). SHEX was not used for the Appendix R calculations for stretch power uprate because the results were extrapolated from earlier analyses.
Atmosphere Dispersion Factors			
Ground Level Release	Methodology of NUREG/CR-5055	Methodology of NUREG/CR-6331 (ARCON95)	NRC sponsored/generated codes. Original validation performed by NRC and the program was benchmarked against the NRC validation data. Results of the validation are maintained by Bechtel.
Stack Release	NRC Program PAVAN	NRC Program PAVAN	
Combustible Gas Control Analysis	Evaluations performed based on existing design margins in the original analysis	HYDROGEN NE309, Version 1.0	HYDROGEN is a Bechtel standard computer code which uses NRC accepted techniques for determination of hydrogen concentrations in BWRs and PWRs. Although there is no formal NRC approval, the NRC has accepted previous results obtained with HYDROGEN. For Plant Hatch extended power uprate analyses, all options utilized have been benchmarked against a code validation. The code validation has been developed and is maintained by Bechtel.

TABLE 15-1 (Cont.)

Analysis	"Original" (5%) Power Uprate Computer Codes	Extended Power Uprate Computer Codes	Application and Validation to Extend Power Uprate
Annulus Pressurization	Original analysis based on COPDA, NE599 (June 1974) and 'Stretch Power Uprate' evaluations performed based on existing design margins	COPDA, MAP164, Version G 1/9	COPDA is a Bechtel standard computer code which uses NRC accepted techniques for determination of pipe break compartment pressurization in BWRs and PWRs. The NRC has formally approved the COPDA program and the NRC has previously accepted results obtained with COPDA. For Plant Hatch extended power uprate analyses all options utilized have been benchmarked against a code validation. The code validation has been developed and is maintained by Bechtel. MAP-164, Version G 1/9 represents compilation of the code from Main Frame to PC.
PSW Flow Balance and Heat Exchanger Performance	BALANCE	BALANCE	BALANCE is a Bechtel program written specifically for flow balancing and heat exchanger performance analysis. The code uses standard engineering techniques. This Bechtel computer code has been reviewed by the NRC Region II office. The program has been validated. The program is benchmarked against Plant Hatch specific test data.
Post-LOCA Control Room and TSC Doses	LOCADOSE NE319, Version 3.0	LOCADOSE NE319, Version 4.1	LOCADOSE is a Bechtel standard computer code which uses NRC accepted techniques for determination of Control Room and TSC Doses in BWRs and PWRs. This code is also used for evaluation of Charcoal Filter requirements. Although there is no formal NRC approval, the NRC has previously accepted results obtained with LOCADOSE. For Plant Hatch extended power uprate analyses all options utilized have been benchmarked against a code validation. Code validation has been developed and is maintained by Bechtel.

TABLE 15.1 (Cont.)

Analysis	"Original" (5%) Power Uprate Computer Codes	Extended Power Uprate Computer Codes	Application and Validation to Extend Power Uprate
Drywell Equipment Qualification Doses	Doses increased proportional to the power level	SPACETRAN-II MAP-159, Version 1.0 and SHIELD-SG NE650, Version D2-10 (Package Release No. D2-11)	SPACETRAN-II and SHIELD-SG are Bechtel standard computer codes which use NRC accepted techniques for determination of Drywell Equipment Qualification Doses. Although there is no formal NRC approval, the NRC has previously accepted results obtained with SPACETRAN-II and SHIELD-SG. For Plant Hatch extended power uprate analyses all options utilized have been benchmarked against a code validation. Code validation has been developed and is maintained by Bechtel.
SGTS Filters Charcoal Requirements	LOCADOSE NE319, Version 3.0	LOCADOSE NE319, Version 3.0	LOCADOSE is a Bechtel standard computer code which uses NRC accepted techniques for determination of Control Room and TSC Doses in BWRs and PWRs. This code is also used for evaluation of Charcoal Filter requirements. Although there is no formal NRC approval, the NRC has previously accepted results obtained with LOCADOSE. For Plant Hatch extended power uprate analyses, all options utilized have been benchmarked against a code validation. Code validation has been developed and is maintained by Bechtel.
Post-LOCA Pump Diaphragm Beta Dose		BRENDA NE655, Version C1-3 and CONTDOSE MAP-142, Version 1.0	BRENDA and CONTDOSE are Bechtel standard computer codes which use NRC accepted techniques for determination of equipment doses in BWRs and PWRs. Although there is no formal NRC approval, the NRC has previously accepted results obtained with BRENDA and CONTDOSE. For Plant Hatch extended power uprate analyses, all options utilized have been benchmarked against a code validation. Code validation has been developed and is maintained by Bechtel.

TABLE 15-1 (Cont.)

Analysis	"Original" (5%) Power Uprate Computer Codes	Extended Power Uprate Computer Codes	Application and Validation to Extend Power Uprate
Compartment P/T Analysis	Evaluations performed based on existing design margins in the original analysis (PCFLUD)	PCFLUD MAP-120, Version 5.0 (Sensitivity Runs)	PCFLUD is a Bechtel standard computer code which uses NRC accepted techniques for determination of pipe break compartment temperature responses in BWRs and PWRs. The NRC has formally reviewed and accepted the PCFLUD program (NUREG-0675) and the NRC has previously accepted results obtained with PCFLUD. For Plant Hatch extended power uprate analyses all options utilized have been benchmarked against a code validation. Code validation has been developed and is maintained by Bechtel. MAP-120, Version 5.0 of PCFLUD represents compilation of the code from Main Frame to PC.
Stability and Load Flow Studies	Power Systems Simulator Electrical, Revision 21	Power Systems Simulator Electrical, Revision 23	PSS/E is a world-wide industry accepted software package used by SCS Transmission Planning Department for all steady state load flow and transient stability analyses. This software is a continually evolving product produced by Power Technologies Incorporated of Schenectady, NY. Version 23 uses the same basic algorithms as version 21 to performed core functions. New releases generally incorporate additional capabilities and features, such as graphical user interface, rather than changes to core functions. New releases have resulted in using the upward compatible features. SCS, in its extensive use of the software in day-to-day planning, has not experienced erroneous computational results due to flaws in the program.

TABLE 15-1 (Cont.)

Analysis	"Original" (5%) Power Uprate Computer Codes	Extended Power Uprate Computer Codes	Application and Validation to Extend Power Uprate
Required Offsite Voltage and Plant Load Studies	STAUX and Station Auxiliary Programs	STAUX and Station Auxiliary Programs	The STAUX computer program, developed by SCS, provides the capability to perform comprehensive station auxiliary system review including load flow, short circuit, and motor starting calculations. The computer programs have been validated and maintained by SCS. The program was revised to provide additional flexibility for detail modeling and reporting. The computational algorithms or methods remain unchanged. Hence, the computational results are essentially unchanged.
BOP Heat Balances	Evaluations performed using SYNTHA-II v. May '92, serial no. SCS 04042792.	Evaluations performed using SYNTHA-II v. April '96, serial no. SCS W3082896.	SYNTHA is an SCS standard computer code used for thermal modeling. The versions used for the original and extended uprate studies are revised versions of the same code. The verification calculation for validating the SYNTHA code for SCS use, is based on the Plant Hatch model.

NRC QUESTION 16

In regard to Section 2.5.1 of the reference, provide an evaluation of the control rod drive mechanism with regard to the maximum stress and fatigue usage and the allowable code limits, for the critical components evaluated as a result of the extended power uprate. If the acceptance criteria in the evaluation are based on codes that are different from the code of record, justify and reconcile the differences.

SNC RESPONSE

Section 2.5.1 states that the CRD mechanism structural and functional integrity are acceptable for at least 1250 psig. This pressure is above the extended power uprate operating pressure and the high pressure scram setpoint, including hydrostatic head. Also included is information on the design Codes and Editions for Units 1 and 2. Extended power uprate will not increase reactor operating pressure, therefore the previous power uprate stress and fatigue usage analyses remain valid.

The limiting component of the CRD mechanism is the indicator tube which has a calculated stress of 20,790 psi, the allowable stress is 26,060 psi. The maximum stress on this component results from a maximum CRD internal hydraulic pressure of 1750 psig with no other event having a significant impact on the total load.

The cyclic operation of the CRD was conservatively evaluated in accordance with ASME Code, N-415.1 which was the code of record for both Unit 1 and Unit 2. The analysis was performed based on the loads from scram with a leaking scram discharge valve and a failed scram buffer. The limiting component was the CRD main flange. The fatigue usage factor, calculated based on NB-3222.4(E) is 0.15, which is less than the allowable limit of 1.0. Considering the original power uprate increase in vessel bottom head pressure, all requirements of N-415.1 are satisfied, thereby satisfying the peak stress intensity limits governed by fatigue.

The maximum vessel pressure will not increase as a result of extended power uprate. Therefore, deformation of components was not specifically calculated for extended power uprate conditions. The CRD mechanism has been subjected to intensive testing at 1250 psi, which is higher than the maximum extended power uprate pressure. Based on the demonstrated performance of the mechanism at these high pressures, it was concluded that deformation resulting from the original power uprate pressure increase is of no significant consequence.

NRC QUESTION 17

On page 3-1 of the reference, it is stated that the safety relief valves (SRVs) operate in the safety (spring) mode only and that the overpressure analysis assumes that the SRVs all have an assumed opening tolerance at least 3% above the nominal setpoints with one of the SRVs completely out of service. Table 5-1 further indicates that the assumed setpoints include a +3% tolerance. Since the SRVs are the Target Rock 2-stage design, which has a history of upward setpoint drift significantly greater than +3% in the safety mode (including some SRVs found to be effectively stuck closed during testing), the assumption of a 3% opening tolerance appears to be nonconservative. Therefore, provide the analysis results for more representative Target Rock SRV setpoint performance, or additional information to demonstrate that the 3% tolerance is conservative.

SNC RESPONSE

The maximum operation dome pressure is not increased for extended power uprate, therefore the SRV setpoints and analytical limits are not changed. There are no changes in the currently approved SRV setpoint values or tolerance for extended power uprate. The extended power uprate analysis utilizes the currently approved analytical setpoint tolerance values (i.e. 3% above the nominal setpoints). Current plant procedures address the testing, analysis, and reporting requirements that must be complied with in the event that SRVs are found to exceed the allowed +/-3% tolerance. These requirements are also unchanged for extended power uprate.

SNC views the potential problem of tolerance on the SRV opening setpoint as a generic issue separate from extended power uprate. The currently approved licensing methodology as well as current reporting requirements will be adhered to until further generic guidance and requirements are provided. It should be noted that the NRC recently issued a safety evaluation and Technical Specifications (TS) change approving the current 1150 psig nominal SRV setpoint with $\pm$ 3% drift. The NRC safety evaluation was issued on March 21, 1997, and approved Unit 1 TS Amendment 204 and Unit 2 TS Amendment 145. Conservatisms in the safety analysis are documented in the SNC licensing submittal for the SRV setpoint change and in the NRC's SER.

NRC QUESTION 18

In reference to Section 3.3.2, the load combinations for the current licensing basis of the reactor vessel and internals are the reactor internal pressure difference, Main Steam Line and Recirculation Line break LOCA loads, seismic and fuel lift loads. Provide an explanation of why the asymmetric pressurization loads and the thrust jet loads that are increased for the power uprate were not included in the load combinations for evaluation of the reactor vessel and internal components.

SNC RESPONSE

The asymmetric pressurization loads and the thrust jet loads are associated with the asymmetrical loads on the vessel, attached piping, and biological shield wall from a postulated pipe break in the annulus between the reactor vessel and biological shield wall. It has been determined that these loads will increase slightly for the extended power uprate condition (reference paragraph 4.1.2.3 of NEDC-32749P). However, these loads are not part of the current licensing basis for Plant Hatch reactor internals. Extended power uprate analyses were performed consistent with the current plant licensing basis. This approach is in agreement with the extended power uprate generic guidelines (ELTR-1), NEDC-32424P-1, dated February 1995.

NRC QUESTION 19

In reference to Table 3-1, provide the maximum calculated stress at the critical locations of the reactor internal components and the allowable code limits, at Plant Hatch Unit 2.

SNC RESPONSE

Table 19-1 provides the comparable stress values for Plant Hatch Unit 2:

TABLE 19-1
COMPONENT STRESS VALUES

Reactor Component	Maximum Stress(Location & Type)	Emergency		Faulted	
		Stress Result (ksi)	Allowable Stress (ksi)	Stress Result (ksi)	Allowable Stress (ksi)
Shroud Repair Unit 2	Tie Rod*	22.0 (17.4)	44.36	45.3 (47.3)	59.2
Shroud Unit 2	Lower Spring Interface*	16.0 (7.3)	41.67	17.72 (7.95)	55.56
Jetpump Diffuser Unit 1 & 2	Diffuser Base*	34.8 (34.9)	38.0	35.4 (34.9)	50.70
Access Hole Cover Unit 2	Bolt**	57.1 (18.0)	107.7	57.1 (18.0)	159.0

* Primary membrane stress.
** Primary membrane + bending stress.
( ) Stress result at current power (2558 MWt) in parenthesis.

NRC QUESTION 20

Section 3.3.2 states that no specific ASME code edition was applicable to the reactor internals during the time of Plant Hatch design and construction. Identify the Code used for evaluation of reactor internals considering the extended power uprate, and clarify whether it differs from the Code of record. If different, provide a justification.

SNC RESPONSE

For the extended power uprate evaluation of the more recent reactor repair modifications (for example shroud repair and access hole cover), the same code of record as the original repair modification analysis was used. These were the 1989 or 1986 ASME Code Edition depending on the project time frame. For the extended power uprate evaluation of original construction components, the same code criteria (for example allowable stresses) originally used in GE design records were maintained with the exception of the Control Rod Guide Tube collapse evaluation which required the use of a more recent criteria based on an ASME 1983 Edition.

NRC QUESTION 21

In reference to Section 3.3.3, you concluded that flow induced vibration is expected to remain acceptable during the extended power uprate. This conclusion was based on reactor internals measurements at Fitzpatrick, the prototype plant, and on GE Nuclear Energy BWR operating experience. Provide a detailed description on the measurements at Fitzpatrick and the GE operating data. Also, provide the technical basis of how these experience data can be used for determining the potential for flow induced vibration at Plant Hatch under the extended power uprate condition.

SNC RESPONSE

During extended power uprate, the components in the upper zone of the reactor, such as the steam separators and dryer, are most affected by the increased steam flow. Components in the core region and components such as the core spray line are affected by increased core flow and not by increased steam flow. Components in the annulus region such as the jet pump are significantly affected by the recirculation pump drive flow and not the steam flow. Hence, only the steam separator and dryer are significantly affected by extended power uprate conditions. The flow induced vibrations of all other reactor internals depend primarily on the core flow and the recirculation pump drive flow. Since extended power uprate conditions do not require any increase in core flow, and very little increase in the drive flow, little increase in flow induced vibrations on the components in the annulus and core regions would be expected as a result of extended power uprate.

Fitzpatrick Unit 1 is the designated BWR prototype for Plant Hatch Units 1 and 2. The Fitzpatrick reactor internals were extensively instrumented during the startup testing of the plant for purposes of vibration monitoring to confirm the structural integrity of major components in the reactor with respect to flow induced vibration. There were a total of 51 sensors installed in the reactor. The sensors consisted of 28 strain gages, 3 accelerometers, 4 pressure gages and 16 displacement gages. Measurements were made at the following locations: shroud, riser top, jet pump elbow, diffuser top, jet pump riser braces, fuel channels and shroud head and separator upper bolt guide ring. Extensive vibration measurements were made over a period of two years covering a wide range of operational conditions from pre-operational (without fuel), precritical (with fuel but not critical) and power operational tests. The power operational tests were conducted at 50%, 75%, and 100% rod line conditions at various core flows. Using this data, extrapolations were conservatively carried out to project responses at 115% rod line. The predicted responses were compared to the GE allowables corresponding to 10,000 psi peak stress intensity (acceptance criteria) to determine the acceptability of the vibration level. The maximum response obtained was for the jet pump riser braces, and it corresponded to 41.9% of the acceptance criteria. A value of less than 100% of the acceptance criteria for the response implies that no fatigue usage is accumulated by the component. The calculations for extended power uprate conditions thus indicate a substantial safety margin for the critical reactor internals.

Plant Hatch and Fitzpatrick belong to the BWR/4 family with the 218 size. There are other plants in the BWR/4 family with the 251 size which are currently operating at 105% uprated power conditions. Since the core flow, steam flow, and original power rating of these plants are much higher than those of Plant Hatch, and their measured reactor internals vibrations are well within limits, conclusions reached based on the Fitzpatrick plant measurements are valid.

NRC QUESTION 22

On page 3-4, Sections 3.3.2 of the reference, you stated that a review of the loads affected by the higher power determined components for which loading has not changed significantly, and these components were considered acceptable for extended power uprate without further evaluation. What is the criterion used to define an insignificant change in loading? What was the limiting component?

SNC RESPONSE

All the reactor components affected by load changes due to extended power uprate were reviewed. The criteria for not performing a detailed structural re-evaluation for some components was the high structural margin on record in prior analyses in conjunction with the small change in RIPD load for that component.

The limiting component for this criteria was the jet pump. The current jet pump subassemblies stress evaluations were reviewed for the effects of changes to RIPD loading and the recirculation line break LOCA loads. Based on RIPD load increases of < 7%, detailed stress re-evaluation for RIPD loading was not required due to the high structural margin on record in prior analyses in conjunction with this small change in RIPD loads. For example, the jet pump riser pipe stress, which was only at 59% of allowable for the evaluation done for the currently licensed power level, and the RIPD load increase would result in a stress increase of < 7%.

NRC QUESTION 23

In reference to Sections 3.3 and 3.3.4, provide a detailed description of the analysis of the reactor vessel and internal components, including methodology, assumptions, and load combinations used for the evaluation with regard to the stresses and fatigue usage for the extended power uprate. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes used in this application.

SNC RESPONSE

The evaluation of the reactor vessel and component integrity uses Section III, Class 1 subsections NB-3222 and NB-3223, of the ASME boiler and pressure vessel code. The Unit 1 feedwater nozzle, control rod drive nozzle, and vessel shell were reanalyzed for extended power uprated conditions. The feedwater nozzle evaluation uses the 1974 ASME Code edition with addenda to and including Summer of 1976. The control rod drive nozzle and vessel shell evaluations use the 1965 ASME Code edition with addenda to and including Winter 1966. The Unit 2 closure vessel shell, closure region bolts, feedwater nozzle, and basin seal skirt were also reanalyzed for extended power uprate conditions. The closure vessel shell, closure region bolts, and basin seal skirt evaluations use the 1968 ASME Code edition with addenda to and including Summer 1970. The feedwater nozzle evaluation used the 1971 ASME Code edition with addenda to and including Summer 1973. The analytical computer codes used were the same as those used for the previous power uprate analysis.

A technique was used to conservatively scale the original stresses to account for pressure, temperature, and flow increases due to extended power uprate conditions. Scaling factors are determined by the ratio of the new to the original values of pressure, temperature, and flow. The scaling factors have been applied to the component or principle stresses in a manner that insures obtaining conservative results.

Changes in design, normal and upset, and emergency and faulted conditions were evaluated where applicable. The evaluated primary stresses were compared to S_m for membrane, and 1.5 S_m for local membrane or membrane plus bending. Primary plus secondary stresses were compared to an allowable of 3 S_m. The fatigue usage allowable is 1.0. For the closure region bolts, the average stress was compared to 2 S_m and the maximum stress was compared to 3 S_m.

In addition to the above stress and fatigue evaluation, a fracture mechanics analysis was performed for the Unit 1 and Unit 2 feedwater nozzles in accordance with the ASME Code Section XI and NUREG-0619.

In general, structural evaluation for a component consists of determining the stress value at the peak stress location for power uprate loads in conjunction with a comparison of stress results to Plant Hatch FSAR criteria. Table 23-1 summarizes these criteria:

TABLE 23-1

PLANT HATCH FSAR LOADING AND ALLOWABLE STRESS BASIS

	Loading		**Stress Allowable**		**Buckling**
Category	**Combination 1**	**Combination 2**	**P_m**	**$P_m + P_b$**	**Criteria**
Upset	Upset RIPD	Normal RIPD plus OBE	1.0 S_m	1.5 S_m	$\leq 0.40\ \sigma_{cr}$
Emergency	Normal RIPD plus DBE	LOCA	1.5 S_m	2.25 S_m	$\leq 0.60\ \sigma_{cr}$
Faulted	LOCA plus DBE	---	2.0 S_m	3.0 S_m	$\leq 0.80\ \sigma_{cr}$

OBE and DBE are the plant operating and design base earthquake loadings. S_m is the allowable stress basis for the component material as specified by the ASME code, and σ_{cr} is the collapse stress value as calculated by elastic analysis.

In addition to the extended power uprate heat balance, RIPD load evaluations are conservatively based on a full core of GE13 fuel which bounds a mixed core load of GE13 with other fuel types. The core flow, feedwater temperature, and power level that result in the maximum loads are also used as initial conditions. The following paragraphs characterize stress determination methods depending on the nature of the prior analysis record for the component involved.

If component structural evaluations were originally based on simple calculations, specific revised stress values were determined for power uprate loads. If component structural evaluations were originally based on detailed stress analyses, such as finite element modeling, the extended power uprate loads were compared with the load combinations for the prior analysis. If the loads for extended power uprate were found to be less than or equal to the previous analysis loads, the stress determination was concluded by documenting that the stress remains equal to or less than the previous analysis.

If extended power uprate loads exceeded the prior analysis load basis for detailed analyses, a revised stress was conservatively estimated from the previously calculated stress by scaling the ratio of extended power uprate load values to comparable load values in the prior analysis. For the shroud and shroud repair, it was necessary to perform a revised detailed stress evaluation to determine acceptable stress values.

NRC QUESTION 24

In reference to Section 3.3.4, provide the maximum calculated stresses for the limiting components that were reanalyzed for the extended power uprate. If the calculated stress of a component is greater than 3 S_m, provide a sample calculation to demonstrate how the component is considered acceptable.

SNC RESPONSE

According to the ASME Code, structural adequacy is met if the maximum primary plus secondary stress intensity at a location on a component is less than 3 S_m of the material. If the 3 S_m limit is not met, plastic behavior is assumed and the simplified elastic-plastic analysis of ASME Code Subarticle NB-3228.3, can be used to determine structural adequacy. Specifically, Code criteria are as follows:

1. The calculated range of primary plus secondary membrane plus bending stress intensity excluding thermal bending stresses ($P+Q-Q_b$) shall be below 3 S_m for all conditions where P+Q exceed 3 S_m. Tables 24-1 and 24-2 provide a summary of limiting component stress intensities for Plant Hatch Unit 1 and Unit 2 respectively. Tables 24-3 and 24-4 provide, as an example, a detailed calculation of the stress intensities for the Plant Hatch Unit 1 feedwater nozzle.

2. The type of stress necessary to evaluate alternating stress, S_a, is primary plus secondary plus peak (P+Q+F). The value of S_a used for entering the design fatigue curve is multiplied by the appropriate factor K_e. Table 24-5 provides as an example, a detailed calculation of the stress intensities for the Plant Hatch Unit 1 feedwater nozzle.

3. The component meets the thermal ratcheting requirements, as calculated below in the discussion titled "Thermal Ratcheting Check."

4. The material temperature does not exceed the maximum temperature permitted for the material and the material shall have a specified minimum yield strength to specified minimum tensile strength ratio of less than 0.80. This is determined below in the paragraph titled "Table Temperature and Strength Ratio Check."

TABLE 24-1
SUMMARY OF EXTENDED POWER UPRATE STRESS AND FATIGUE RESULTS FOR HATCH 1

Component	Limiting Location	Stretch Power Uprate P+Q (ksi)	Extended Power Uprate P+Q (ksi)	Extended Power Uprate P+Q Allowable (ksi)
Feedwater Nozzle	Element 229	74.8 (1) 19.0 (2)	74.8 (1) 19.0 (2)	55.7
Control Rod Drive Nozzle	Shell Cut 5, Top Surface	57.2	57.2	69.9
Vessel Shell	Inside Shell Surface	45.5	45.5	80.1

(1) This stress intensity exceeds the P+Q allowable of 3 S_m.
(2) Excluding thermal bending, this stress intensity is less than the P+Q allowable of 3 S_m.

TABLE 24-2
SUMMARY OF EXTENDED POWER UPRATE STRESS AND FATIGUE RESULTS FOR HATCH 2

Component	Limiting Location	Stretch Power Uprate P+Q (ksi)	Extended Power Uprate P+Q (ksi)	Extended Power Uprate P+Q Allowable (ksi)
Closure Vessel Shell	Cut "v"	43.6	43.6	80.1
Closure Region Bolts	Bolts	100.3	100.3	129.9
Feedwater Nozzle	Element 367	132.3 (1) 36.7 (2)	132.3 (1) 36.7 (2)	69.9
Basin Seal Skirt	Flange-Cylinder Junction	74.5 (1) 45.9 (2)	74.5 (1) 45.9 (2)	58.7

(1) This stress intensity exceeds the P+Q allowable of 3 S_m.
(2) Excluding thermal bending, this stress intensity is less than the P+Q allowable of 3 S_m.

Table 24-3
P+Q
(Element 229)

Stress Cycle Pair	ΔΣRR	ΔΣZZ	ΔΣTT	ΔτRZ	PRIN Σ1 (1)	PRIN Σ2 (1)	PRIN Σ3 (2)	RANGE P+Q (3)	$3 S_m$ (4) ALLOWABLE
TR30.128 LFP.145	125	-70366	-74628	890	136	-70377	-74628	74765	5570
TR.127 TR30.128	0	63210	66814	-764	63219	-9	66814	66824	55700
TR30.128 HSB.145	0	-62383	-66061	776	10	-62393	-66061	66071	55700
ZEROLOAD HSB.145	1048	-53883	-59141	1347	1081	-53916	-59141	60222	55700
LFP9.03 HSB.145	-125	-53356	-56846	878	-111	-53370	-56846	56735	55700
HSB.145 SD113.63	-948	48973	58573	-989	48992	-968	58573	59540	55700
HSB.145 HSB4.03	0	45625	52567	-1120	45652	-27	52567	52595	55700
SVB1.02 HSB4.03	134	4095	7999	-696	4214	16	7999	7983	55700
ZEROLOAD LFP.145	1173	-61866	-67709	1461	1207	-61900	-67709	68916	55700

(1) $\Sigma 1, \Sigma 2 = (\Sigma RR+\Sigma ZZ)/2 \pm [(\Sigma RR-\Sigma ZZ)^2/4 + (\tau RZ)^2]^{1/2}$
(2) $\Sigma 3 = \Sigma TT$
(3) P+Q = maximum ABS[(Σ1-Σ2), (Σ2-Σ3), (Σ3-Σ1)]
(4) At 552 °F

Table 24-4
P+Q
(Element 229)

Stress Cycle Pair	$\Delta\Sigma RR$	$\Delta\Sigma ZZ$	$\Delta\Sigma TT$	$\Delta\tau RZ$	PRIN $\Sigma 1$ (1)	PRIN $\Sigma 2$ (1)	PRIN $\Sigma 3$ (2)	RANGE $P+Q-Q_b$ (3)	$3 S_m$ (4) ALLOWABLE
TR30.128 LFP.145	125	-386	-11893	890	795	-1056	-11893	12688	55700
TR.127 TR30.128	0	235	10195	-764	891	-656	10195	10851	55700
TR30.128 HSB.145	0	-235	-10233	776	668	-903	-10233	10900	55700
ZEROLOAD HSB.145	1048	-4475	-16340	1347	1359	-4786	-16340	17700	55700
LFP9.03 HSB.145	-125	-82	-7268	878	775	-982	-7268	8043	55700
HSB.145 SD113.63	-948	1630	15390	-989	1966	-1284	15390	16674	55700
HSB.145 HSB4.03	0	234	8100	-1120	1243	-1009	8100	9109	55700
SVB1.02 HSB4.03	134	54	250	-696	791	-603	250	1393	55700
ZEROLOAD LFP.145	1173	1494	-17490	87	1516	1151	-17490	19006	55700

(1) $\Sigma 1, \Sigma 2 = (\Sigma RR+\Sigma ZZ)/2 \pm [(\Sigma RR-\Sigma ZZ)^2/4 + (\tau RZ)^2]^{1/2}$
(2) $\Sigma 3 = \Sigma TT$
(3) $P+Q-Q_b$ = maximum ABS[$(\Sigma 1-\Sigma 2)$, $(\Sigma 2-\Sigma 3)$, $(\Sigma 3-\Sigma 1)$]
(4) At 552 °F

Table 24-5
FATIGUE USAGE
(Element 193)

Stress Cycle Pair	Range S_n [1] (psi)	$P+Q-Q_b$ (psi)	$3 S_m$ Allowable (psi)	K_e [2]	S_p (psi)	S_a [3] (psi)	N_{actual}	$N_{allowable}$ [4]	Incremental Ui	Summed Ui
TR30.128 LFP.145	66329	6858	55700	1.38	113072	90006	192	787	0.244	0.244
TR.127 TR30.128	58253	5925	55700	1.09	100229	63038	200	2099	0.095	0.339
TR30.128 HSB.145	57551	5868	55700	1.07	99133	60909	44	2342	0.019	0.358
ZEROLOAD HSB.145	57100	14343	55700	1.05	93239	56418	200	2989	0.067	0.425
LFP9.03 HSB.145	52078	3568	55700	1.0	89444	51531	144	3988	0.036	0.461
HSB.145 SD113.63	52647	12749	55700	1.0	87413	50361	198	4291	0.046	0.507
HSB.145 HSB4.03	47443	4953	55700	1.0	80981	46655	2014	5440	0.370	0.877
SVB1.02 HSB4.03	9542	548	55700	1.0	16962	9772	56711	8493337	0.007	0.884

(1) Sn = P+Q
(2) K_e = 1+ {[(1-n)/(n)(m-1)] [(Sn/3Sm)-1]}, where m=3.0, n=0.2
(3) $S_a = (1/2)(E_c/E_a)(K_e S_p)$, where E_c = 3.00E+07 psi, E_a = 2.60E+07 psi
(4) $N = Ni(Nj/Ni)^{[\log(Si/Sa)]/[\log(Si/Sj)]}$: Interpolation formula for UTS ≤ 80 ksi with S-N curve Fig. I-9.1 of ASME Code Appendix I

Thermal Ratcheting Check

Per paragraph NB-3228.3(d), the thermal stress ratchet check of paragraph NB-3222.5 must be performed. Compliance is demonstrated in the calculation below in that the largest cyclic range of thermal stress of 74.8 ksi (mechanical stress conservatively included, occurring at element 229) is less than the limiting value of maximum cyclic stress to prevent cyclic growth in diameter. The comparison is 74.8 ksi to 85.0 ksi; therefore no thermal ratchet effect will be experienced at that location.

Maximum extended power uprate general membrane stress due to pressure:

$$\sigma_{\theta M,old} = p_{old} * R/t ,$$

where p_{old} is the original maximum normal operating vessel pressure. Therefore,

$$\begin{aligned}\sigma_{\theta M,new} &= \sigma_{\theta M,old} * (p_{new} / p_{old}) \\ &= 8704 \text{ psi} * (1048 \text{ psig} / 1000 \text{ psig}) \\ &= 9122 \text{ psi}\end{aligned}$$

$$\begin{aligned}x_{new} &= \sigma_{\theta M,new} / (1.5\ S_{m,new}) \\ &= 9122 \text{ psi} / (1.5 * 18567 \text{ psi}) \\ &= 0.328\end{aligned}$$

where $S_{m,new}$ is evaluated at 552°F. For $0 < x < 0.5$, the

$$\begin{aligned}y' &= 1/x \quad \text{and} \quad y' = S_n / (1.5\ S_m) \\ S_n &= 1.5\ S_m / x \\ S_n &= 1.5 * 18567 \text{ psi} / 0.3275 \\ &= 85.0 \text{ ksi}\end{aligned}$$

Table Temperature and Strength Ratio Checks

According to Paragraph NB-3228.3(e), the temperature used in the analysis should not exceed those of the table of Paragraph NB-3228.3. The maximum temperature in the table is 700°F for carbon steel, and the maximum analysis temperature is 566°F for any stress cycle for the nozzle safe end. Since 566°F < 700°F, this requirement is met.

According to Paragraph NB-3228.3(f), the ratio of the material minimum specified yield strength ($S_{y,ms}$) to the minimum specified ultimate strength ($S_{u,ms}$) shall be less than 0.80. From Reference 9, $S_{y,ms}$ = 36.0 ksi and $S_{u,ms}$ = 70.0 ksi for SA-105 Grade II carbon steel, which results in a ratio of 0.51. Since 0.51 < 0.80, this requirement is met.

Thermal Ratcheting Check

Per paragraph NB-3228.3(d), the thermal stress ratchet check of paragraph NB-3222.5 must be performed. Compliance is demonstrated in the calculation below in that the largest cyclic range of thermal stress of 74.8 ksi (mechanical stress conservatively included, occurring at element 229) is less than the limiting value of maximum cyclic stress to prevent cyclic growth in diameter. The comparison is 74.8 ksi to 85.0 ksi; therefore no thermal ratchet effect will be experienced at that location.

Maximum extended power uprate general membrane stress due to pressure:

$$\sigma_{\theta M,old} = p_{old} * R/t ,$$

where p_{old} is the original maximum normal operating vessel pressure. Therefore,

$$\begin{aligned} \sigma_{\theta M,new} &= \sigma_{\theta M,old} * (p_{new} / p_{old}) \\ &= 8704 \text{ psi} * (1048 \text{ psig} / 1000 \text{ psig}) \\ &= 9122 \text{ psi} \end{aligned}$$

$$\begin{aligned} x_{new} &= \sigma_{\theta M,new} / (1.5\ S_{m,new}) \\ &= 9122 \text{ psi} / (1.5 * 18567 \text{ psi}) \\ &= 0.328 \end{aligned}$$

where $S_{m,new}$ is evaluated at 552°F. For $0 < x < 0.5$, the

$$\begin{aligned} y' &= 1/x \quad \text{and} \quad y' = S_n / (1.5\ S_m) \\ S_n &= 1.5\ S_m / x \\ S_n &= 1.5 * 18567 \text{ psi} / 0.3275 \\ &= 85.0 \text{ ksi} \end{aligned}$$

Table Temperature and Strength Ratio Checks

According to Paragraph NB-3228.3(e), the temperature used in the analysis should not exceed those of the table of Paragraph NB-3228.3. The maximum temperature in the table is 700°F for carbon steel, and the maximum analysis temperature is 566°F for any stress cycle for the nozzle safe end. Since 566°F < 700°F, this requirement is met.

According to Paragraph NB-3228.3(f), the ratio of the material minimum specified yield strength ($S_{y,ms}$) to the minimum specified ultimate strength ($S_{u,ms}$) shall be less than 0.80. From Reference 9, $S_{y,ms}$ = 36.0 ksi and $S_{u,ms}$ = 70.0 ksi for SA-105 Grade II carbon steel, which results in a ratio of 0.51. Since 0.51 < 0.80, this requirement is met.

NRC QUESTION 25

Cumulative fatigue usage factors (CUF) for the limiting components of the reactor vessel were provided in Table 3-5 of the reference for both Plant Hatch Units 1 and 2. Provide a discussion of how the CUFs listed in the table are the same for the current and the extended power uprate condition.

SNC RESPONSE

A scaling technique was used to determine primary plus secondary plus peak (P+Q+F) stresses to calculate CUF for extended power uprate, and the same technique was used for the original power uprate. Scaling factors were determined by the ratio of the new to the original values of pressure, temperature, and flow. The scaling factors were applied to the component stresses or principle stresses in such a manner to obtain conservative results. Current condition in Table 3-5 represents a 2733 MWt power uprate condition. The extended power uprate condition corresponds to 2763 MWt. As shown in Table 25-1 below, the pressures and temperatures specified for analysis of the limiting components are the same for both the original power uprate and the extended power uprate conditions. This table is applicable to Regions A, B and C of the RPV. Pressures and temperatures remained the same for both uprated conditions. Conservative scaling factors were used in the original power uprate evaluation to determine the impact on stresses due to changes in flow. Therefore, the effects of increased flow for the extended power uprate was bounded by that determined for the previous power uprate.

The scaling factors for pressure, temperature, and flow used to determine P+Q+F stresses in the extended power uprate analysis were the same as those used in the original power uprate analysis; thus, the CUF for limiting components remained the same.

TABLE 25-1

Original 2436 MWt		Original Power Uprate 2733 MWt		Extended Power Uprate 2763 MWt	
Pressure (psig)	Temperature °F	Pressure (psig)	Temperature °F	Pressure (psig)	Temperature °F
1180	573	1228	573	1228	573
1125	561	1173	566	1173	566
875	530	923	537	923	537
665	500	713	507	713	507
930	538	978	544	978	544

NRC QUESTION 26

Table 3-5 indicates that the CUF for the reactor vessel feedwater nozzle for Plant Hatch Unit 1 was evaluated at selected locations that did not include the nozzle blend radius area. Provide a detailed discussion of the analysis performed including the methodology and assumptions to demonstrate the structural integrity of the feedwater nozzle blend radius. Also, discuss how the power uprate will affect the feedwater sparger with a welded thermal sleeve used at Plant Hatch.

SNC RESPONSE

The value of CUF for the feedwater nozzle reported in Table 3-5 occurs in the safe end location where system cyclic fatigue is highest and rapid cycling is lowest (essentially zero). The scaling technique discussed in response to NRC Question 25 was used to calculate the stresses for determining the CUF. This technique is based on appropriately applying the ratio of the new to the original pressure, temperature, and flow in the determination of stress.

Along the nozzle blend radius of the feedwater nozzle, system cycling is present, and the effects of rapid cycling are greatest. In the analysis of rapid cycling, increased core flow (ICF) and final feedwater temperature reduction (FFWTR) were considered. Corrosion rates at the seal interfaces were assumed to be - 5.75E-4 inch per year (seal radial gap), 4.7E-4 inch per year (seal axial gap), and 4.4E-4 inch per year (safe end) for the triple sleeve double seal design. By computation, leakage and stress calculations and an evaluation of seal refurbishment intervals were performed.

The analysis of the triple sleeve double seal design indicated that in 15 years of hot operating service, the fatigue usage will exceed the 1.0 allowable along the nozzle blend radius. However, if a seal refurbishment strategy is used, a usage factor < 1.0 for a 40 year design life can be achieved with a seal refurbishment interval of 12 years. The nozzle region is currently examined to the requirements of NUREG-0619. A fracture mechanics analysis (based on NUREG-0619 as amended by NRC Generic Letter 81-11) was performed to support the extended power uprate conditions. Therefore, should a fatigue crack initiate at the nozzle blend radius, the current feedwater nozzle inspection program is adequate to detect such a crack before the Code required safety margins are compromised. This approach of assuring structural integrity and required safety margins is consistent with the concepts used in the recently introduced non-mandatory Appendix-L of ASME Code Section XI.

The effect of rapid cycling along the safe end and the nozzle blend radius is minor for the welded thermal sleeve design. Therefore, no evaluation was necessary at the nozzle blend radius because the highest CUF was at the safe end location.

NRC QUESTION 27

In Table 3-5, the CUF for the feedwater nozzle at Plant Hatch Unit 2 was obtained using the cycle counting approach for the original power condition. Provide a discussion regarding why the extended power uprated conditions were not used, and why the cycle counting approach was used only for the feedwater nozzle but not for other components.

SNC RESPONSE

Using a design basis approach and considering uprated power, the end of life CUF of the feedwater nozzle for Plant Hatch Unit 2 was greater than 1.0; thus, a more realistic CUF was calculated by combining the fatigue usage factor based on actual operational data with a design basis fatigue usage factor calculated for uprated power conditions. Using actual plant data and a cycle counting approach for pre-uprate conditions eliminated some of the conservatism in using thermal cycle diagram data in the design basis analysis. The end of life CUF was then determined to be 0.93. That is, credit was taken for actual operating thermal cycle counting information to estimate the actual fatigue usage up to the time of uprated conditions. From September [illegible] until end of life, the uprated fatigue usage factors were determined using the design basis approach. As noted in Table 3-5, the cycle counting approach was also used in the original power uprate analysis.

Since the feedwater nozzle was the only component to exceed a usage factor of 1.0, only the feedwater nozzle was evaluated with the more realistic cycle counting method. All other components were evaluated using the conservative design basis approach.

NRC QUESTION 28

Provide a comparison of the calculated CUFs for the limiting reactor vessel and piping components to the CUFs resulting from the actual loading cycles (cycle counting) based on the data recorded during plant operation

SNC RESPONSE

The Plant Hatch Unit 2 feedwater nozzle was the only reactor vessel component in which the actual plant data was used to determine CUF. Table 28-1 below provides a comparison of the actual operating cycle counting CUF and the same period design basis CUF. The most conservative approach would be to apply an uprated power scaling for the entire life of the plant. In doing so, the CUF would be 1.117.

TABLE 28-1

Duration	Cycle Counting Incremental U_i	Design Basis Incremental U_i (1)
For first 5.92 Years	0.165 (2)	0.139 (3)
Design Basis Data, August 1, 1985 to December 31, 1985	0.010	0.010
January 1, 1986 to December 31, 1993	0.047 (4)	0.188 (3)
Original Design Basis Data from January 1, 1994 to August 31, 1995	0.039	0.039
Power Uprate Design Basis from September 1, 1995 to August 31, 1998 and extended power uprate from September 1, 1998 to August 31, 2019 (5)	0.670	0.670
Total Fatigue Usage Factor	0.930 (6)	1.046

(1) Uprated power scaling only applied from September 1, 1995 through end of life.
(2) Actual plant data was updated for the modified feedwater nozzle design based on the modified stresses.
(3) Design Basis Data.
(4) Actual Plant Data.
(5) Assumes original power uprate ends August 31, 1998, and extended power uprate begins September 1, 1998.
(6) Same as value in Table 3-5 of NEDC-32749P.

NRC QUESTION 29

In reference to Section 3.6, provide a discussion of the methodology and assumptions used for evaluating the reactor coolant piping systems for the power uprate. Also, provide the calculated maximum stress, critical locations, allowable stress limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, justify and reconcile the differences.

SNC RESPONSE

The methodology used in the evaluation of the reactor coolant piping system is as follows:

Existing design basis documents, such as design specifications and piping stress reports, were reviewed to determine the design and analytical basis for reactor coolant piping (RCP) systems. The extended power uprate parameters of RCP systems were compared with the existing analytical basis to determine increases in temperature, pressure, and flow due to extended power uprate conditions. The original Code of record, Code allowable, and analytical techniques were used. No new assumptions were introduced.

ASME B&PV Code, Section III, Subsection NB-3600, Code equations 9, 10, 12, 13 and 14 were reviewed to determine the equations impacted by temperature, pressure, and flow increases due to extended power uprate conditions on the ASME Code Class 1 piping systems.

General Electric performed a parametric study for the RCP systems to determine the percent increases in applicable Code stresses, displacements, CUFs, and pipe interface component loads (including supports) as a function of percentage increase in pressure, temperature, and flow due to extended power uprate conditions. The percent increases were applied to the highest calculated stresses, displacements, and the CUF at applicable RCP piping system node points to conservatively determine the maximum extended power uprate calculated stresses, displacements and usage factors. This approach is conservative since extended power uprate does not affect all dynamic loads; e.g., seismic loads are not affected by extended power uprate.

Piping interfaces with RPV nozzles, anchors, struts, penetrations, flanges, pumps, and valves were evaluated in a similar manner. The effect of extended power uprate conditions on thermal and vibration displacement limits was also evaluated, and GE concluded that thermal displacements are acceptable for extended power uprate conditions.

The results of these evaluations demonstrate that the requirements of ASME Code, Section III, Subsection NB-3600, are satisfied for both the main steam and recirculation piping systems at the extended power uprate conditions. Interface loads on system components, which have increased due to extended power uprate conditions, do not exceed component acceptance criteria. Table 29-1 provides a summary of the limiting stress ratios from the RCP systems evaluation for extended power uprate conditions.

Table 29-1

HATCH REACTOR COOLANT PIPING SYSTEMS
EXTENDED POWER UPRATE SUMMARY OF HIGHEST STRESS RATIOS
(Calculated to Allowable)

UNIT	SYSTEM	EQ. 8	EQ.9 N/U	EQ. 9 EMER	EQ. 9 FAULTED	EQ. 10	EQ.12	EQ. 13	EQ. 14
1	M'.IN STEAM		0.73	0.54	0.53	1.35	0.93	0.67	0.64
1	SRVDLs	0.32	0.99	0.79	0.70	1.32	0.94		
1	HPCI		0.58	N/A*	0.38	0.66			
2	MAIN STEAM		0.52	0.41	0.33	1.08	0.81	0.61	0.28
2	SRVDLs	0.20	0.82	0.65	0.52	1.09	0.77		
2	TURBINE BYPASS	0.29	0.29	0.38	0.16	0.36			

*Note: N/A - Not Available

NRC QUESTION 30

Discuss the operability of safety-related mechanical components (i.e., valves and pumps) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Provide a list of safety-related valves affected by the power uprate, their functions and operating conditions (including pressure, temperature, differential pressure, flow rate, ambient pressure and stroke times) at the current (100% power design basis) and the power uprate conditions. Identify mechanical components for which operability at the uprated power level could not be confirmed. Also, provide a discussion of how the Plant Hatch Motor-Operated-Valve and Air-Operated-Valve programs have been updated to reflect the extended power uprate condition.

SNC RESPONSE

Industry AOV and MOV operator sizing equations used to calculate the required thrust and torque depend on the differential pressure across the valves, line pressure, and static (packing) loads. Extended power uprate will not increase any system operating pressure, reactor pressure, or the SRV setpoints. Extended power uprate will not change RHR, CS, or recirculation pump discharge pressure permissives. Valve stroke times and pump head curves are not affected. The differential and line pressures used to determine valve/operator thrust and torque requirements are not affected by extended power uprate conditions. Therefore, it is not necessary to revise the Plant Hatch MOV and AOV programs due to extended power uprate.

The valve radiation qualification was evaluated and it was determined the valves are qualified for the increased radiation dose due to extended power uprate conditions. Containment isolation valves were designed for containment design accident pressure. The design pressure for the containment is not affected by extended power uprate.

Extended power uprate operation requirements were evaluated for each of the NSSS systems. Each evaluation shows there is no impact on the system performance capabilities due to extended power uprate. Table 30-1 lists the impact of extended power uprate on the NSSS and components.

TABLE 30-1

SUMMARY OF SYSTEMS AND COMPONENTS EVALUATED FOR EXTENDED POWER UPRATE

System	*Component or Function*	*Parameter*	*Impact of Extended Power Uprate*	*Consequence/Result*
NMS	**System**	Pressure and Temperature	**Unchanged**	No Hardware Change Required
	SRM	Range	**Unchanged**	No Hardware or Setpoint Change Required
	IRM	Range	**Reduced**	Adjust Overlap
	IRM	Sensor Lifetime	**Shortened by 8 % max.**	Adequate for Power Uprate
	LPRM	Sensor Lifetime	**Shortened by 8 % max.**	Adequate for Power Uprate
	LPRM	Calibration	**Changed**	Adjustment Required
	APRM	Calibration	**Changed**	Adjustment Required
	TIP hardware	Radiation Exposure	**Increased**	Negligible Effect
RRS	**Pump**	Speed to Maintain Core Flow	**Must Be Increased**	Calculated 15 rpm increase to maintain 100% core flow. Within design range.
	Pump	Head, Flow and Horsepower	**Increased with Speed**	Within Design Range
	Motor	Speed and Power Output	**Increased with Speed**	Within Design Range
	M-G Set	Speed, Power and Frequency	**Increased with Speed**	Within Design Range
	System	Pressure and Temperature	**Insignificant increase**	Within Design Range
CRD	**System**	Pressure and Temperature	**Unchanged**	No Hardware Change Required
	Scram Performance	Driving Force on Control Rod	**Unchanged**	No Effect on Scram Time
	Insertion/Withdrawal	Available Driving Pressure	**Unchanged**	
RHR	**System**	Pressure and Temperature	**Unchanged**	No Hardware Change Required
	LPCI Mode	Flow and Response Time	**Decay Heat Increased**	Meets Design Requirements
	Heat Removal	Heat Removal Rate	**Heat Load Increased**	Adequate for Power Uprate
	Shutdown Cooling Mode	Cooldown Time	**Increased Slightly**	Adequate for Power Uprate
	Pump	NPSH Margin	**Changed by Containment T and P**	Sufficient Margin for Power Uprate NPSH input to new strainer design
SLC	**System**	Pressure and Temperature	**Unchanged**	No Hardware Change Required

TABLE 30-1 (Cont.)

System	*Component or Function*	*Parameter*	*Impact of Extended Power Uprate*	*Consequence/Result*
	Boron Solution	Concentration of Boron	**No Change Required**	
	Injection	Rate and Time	**No Change Required**	
	Pump	Discharge Pressure at constant flow rate	**No pressure increase**	
RWCU	**System**	Pressure and Temperature	**Unchanged**	No Hardware Change Required
	System	Water Purity	**Fe and Conductivity Increased**	Within Plant Limits
	Filter/Demineralizers	Inlet Temperature	**Unchanged**	
HPCI	**System**	Pressure and Temperature	**Unchanged**	No Hardware or Setpoint Change Required
	Injection	Required Rate and Time	**Unchanged**	
	Pump	Discharge Head	**Unchanged**	
	Pump & Turbine	Max. Speed	**Unchanged**	
	Turbine	Steam Flow Rate	**Unchanged**	
	Turbine	Overspeed Trip	**Unchanged**	
	MOVs	Operating Pressure	**Unchanged**	
CS	**System**	Pressure and Temperature	**Unchanged**	No Hardware or Setpoint Change Required
	System	Flow Rate and Response Time	**Post-LOCA Pool Temp. Increased**	Adequate for Power Uprate
	Pump	NPSH Margin	**Suppression Pool T & P Increase**	Sufficient Margin for Power Uprate NPSH input to new strainer design
	Pump Seals	Design Temperature Limit	**Suppression Pool Temperature Increase**	SP Maximum Temperature does not exceed design temperature of 212 °F
	Break Detection	Delta P Setpoint	**Small Change in Expected Delta P**	No Setpoint Change Required
RCIC	**System**	Pressure and Temperature	**Unchanged**	No Hardware or Setpoint Change Required
	Injection	Required Rate and Time	**Unchanged**	
	Pump	Discharge Head	**Unchanged**	
	Pump & Turbine	Max. Speed	**Unchanged**	
	Turbine	Steam Flow Rate	**Unchanged**	

TABLE 30-1 (Cont.)

System	*Component or Function*	*Parameter*	*Impact of Extended Power Uprate*	*Consequence/Result*
	Turbine	Overspeed Trip	**Unchanged**	
	MOVs	Operating Pressure	**Unchanged**	
SRVs	**System**	Pressure and Temperature	**Unchanged**	No Hardware or Setpoint Change Required
	Safety Valve	Spring Setpoints	**Unchanged**	
MSIVs	**Valve**	Pressure and Temperature	**Unchanged**	No Hardware or Setpoint Change Required
	Valve	Fatigue Usage	**Insignificant**	Ample margin exists
	Valve	Closing Time	**Speed Control Valves Will Adjust**	Margin is Adequate
	Valves	Pressure Drop	**Exceeds Design**	No Impact on Valve Performance.
OGS	**Recombiner**	Inlet H_2 from Core Radiolysis	**Increased by 8%**	Design Margin is Adequate
	Recombiner	Outlet Temperature	**Increased by 32 °F**	Margin is Adequate
SGTS	**Charcoal**	Weight of Adsorber	Increase in Required Weight	Existing Margin (Charcoal Weight Available) Adequate per Calcs Provided by SNC

NRC QUESTION 31

In reference to Section 3.12, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate. State the Code and edition used for the power uprate evaluation of BOP piping, in-line components (valves and nozzles, etc.), and pipe supports including anchorages. List the limiting BOP piping systems and components with respect to the maximum stresses and corresponding allowables as a result of the extended power uprate. If the analytical computer codes used in the evaluation are different from those used in the original design-basis analysis, identify the new codes used in this application.

SNC RESPONSE

The Code and Edition used for the extended power uprate evaluations of BOP piping, in-line components (pumps, valves, nozzles, penetrations, etc.), and pipe supports, including anchorages, are the same as the original Code of record. Code allowables and analytical techniques are the same. No new assumptions were introduced.

The evaluation of the BOP piping and supports was performed in a manner similar to the evaluation of RCP piping systems and supports. The response to NRC Question 29 provides a detailed discussion of this analysis method. For BOP piping systems, the percentage increase in the extended power uprate operating conditions remains the same or less than that conservatively analyzed for the 105 % power uprate, hence no evaluations are required.

NRC QUESTION 32

Referring to Sections 3.6 and 4.1.2, provide the evaluation of piping systems attached to the torus shell, vent penetrations, pumps, and valves, that may be affected by the LOCA dynamic loads (pool swell, condensation oscillation, and chugging) and the projected increase in the pool temperature as a result of the extended power uprate.

SNC RESPONSE

Evaluations of containment LOCA dynamic loads were performed as part of the extended power uprate containment review. These evaluations used the results of the LOCA containment pressure and temperature analyses performed at uprate conditions. The evaluations confirmed that the LOCA containment conditions with power uprate are bounded by either the test conditions or analysis conditions used to define the original LOCA hydrodynamic loads for Plant Hatch Units 1 and 2 in the Plant Unique Analyses Reports, submitted to the NRC by letter dated November 21, 1989. Therefore the evaluations demonstrated that expected LOCA dynamic loads with extended power uprate are bounded by the current LOCA dynamic load definition. This is reflected in Section 4.1.2.1 of the Plant Hatch extended power uprate licensing report (NEDC-32749P). Therefore, LOCA dynamic design loads (such as pool swell, condensation oscillation and chugging loads) on the torus shell, vent penetrations, and pumps and valves are not exceeded due to extended power uprate conditions.

The Plant Hatch Units 1 and 2 calculated peak bulk suppression pool temperatures which are discussed in Section 4.1.1.1 and are tabulated in Table 4-1 of NEDC-32749P. This tabulation shows that extended power uprate results in a peak pool temperature increase of 6°F (from 202°F to 208°F). The peak pool temperatures are well below the wetwell structural design temperatures of 281°F for Unit 1 and 340°F for Unit 2. Therefore, the peak bulk pool temperature with extended power uprate is acceptable from structural design standpoints.

Detailed evaluation results of LOCA loads and bulk pool temperatures for engineered safety systems are provided in Section 4 of GE report NEDC-32749P. For the torus attached piping the maximum torus design temperatures of 210°F for Unit 1 and 209°F for Unit 2 are acceptable from all piping design standpoints.

NRC QUESTION 33

Discuss the potential for flow-induced vibration in the heat exchangers for the condensate and feedwater system, following the extended power uprate at Plant Hatch.

SNC RESPONSE

Main Condenser

Condenser tube vibration is a function of steam velocity. For a fixed exhaust area, such as the condenser, steam velocity is a function of steam volumetric flow. With extended power uprate, the volumetric flow approaches or exceeds the design volumetric flow. The results of an evaluation specific to tube vibration indicated additional condenser staking along the top and sides of the tube bundles is required for both units. The staking modification will be implemented on each unit prior to extended power uprate operation.

Feedwater Heaters

The condensate and feedwater system feedwater heaters were evaluated to ensure adequate function at maximum extended power uprate conditions. The tube and shell-side design pressures and temperatures did not change, and operating pressures at extended uprate are less than design values. Tube and shell-side flow velocities were calculated at extended power uprate conditions and compared to manufacture's recommendations and recognized industry guidelines such as Heat Exchanger Institute, Inc. (HEI) standards. The calculated velocities did not exceed industry and manufacture's guidelines, with the exception of the Unit 1 8th stage, Unit 2 6th stage, and Unit 2 8th stage heater drain inlet nozzles. The mass velocities for these nozzles were greater than the recommended HEI standard by 1% for the Unit 1 8th stage nozzle, 1.4% for the Unit 2 6th stage nozzle, and 17% for the Unit 2 8th stage nozzle. The greater nozzle velocities could result in additional nozzle wear, however, stainless steel liners are installed in feedwater drain inlet nozzles to prevent excessive wear. The feedwater heater performance will be monitored during the extended uprate power ascension program.

Moisture Separator Reheaters

As a retrofit project independent of extended power uprate, the moisture separator reheaters (MSRs) will be modified. New low and high pressure tube bundles, as well as new high performance moisture separator chevron assemblies will be installed. The objective of this retrofit is to improve the reheater terminal temperature differences, improve the moisture effectiveness of the moisture separator section, minimize the pressure drop across the vessels, and improve the reliability of the MSRs at extended uprate conditions. Vibrational design considerations were included in the design of the tube bundles and chevron assemblies. The performance of the MSRs will be monitored during the extended uprate power ascension program.

NRC QUESTION 34

In reference to Section 7.4, provide the evaluation of the feedwater heater for the power uprate with regard to vibration, stress and fatigue usage.

SNC RESPONSE

The feedwater heaters were evaluated at maximum extended power uprate conditions to ensure that adequate shell overpressure protection, heater thermal performance, and heater tube and shell-side design conditions were not adversely affected. The tube and shell-side design pressures and temperatures did not change, and operating pressures at extended power uprate are less than design values.

The major effect of extended uprate on the feedwater heaters is increased velocities through the feedwater heater tubes and increased steam flow through the shell-side of the heater. The tube-side velocities were calculated at the higher extended uprate flow and determined to be less than the Heat Exchanger Institute, Inc. (HEI) recommended 10 feet-per-second. On the shell-side, only the inlet extractions to the 5th and 10th stage heaters of Unit 1 and the 4th and 12th stage heaters of Unit 2 showed an increase in piping velocities. These velocities were still within industry standards.

The tube-side and shell-side inlet nozzle velocities were also evaluated and determined to meet recommended HEI standards with the exception of the Unit 1 8th stage, Unit 2 6th stage, and Unit 2 8th stage heater drain inlet nozzles. The mass velocities for these nozzles were greater than the recommended HEI standard by 1% for the Unit 1 8th stage nozzle, 1.4% for the Unit 2 6th stage nozzle, and 17% for the Unit 2 8th stage nozzle. The greater nozzle velocities could result in additional nozzle wear, however, stainless steel liners are installed in the feedwater drain inlet nozzles to prevent excessive wear. Tube-side and shell-side pressure drops were calculated and meet HEI standards.

The existing shell and tube-side relief valves have sufficient capacity and since the design pressures remained the same, no relief set point changes are required. The feedwater heaters should not be subjected to temperature and pressure transients outside those considered in the original design by the heater manufacturer. It was determined that extended uprate would not have any adverse affect on the feedwater heaters. Feedwater heater performance will be monitored during the extended uprate power ascension program.

NRC QUESTION 35

Do you project modifications to piping or equipment supports for the extended power uprate? If any, provide examples of pipe supports requiring modification and discuss the nature of these modifications.

SNC RESPONSE

There are no piping or equipment support modifications required due to extended power uprate. The response to NRC Question 29 provides a detailed discussion of the analysis method.

INSTRUMENTATION AND CONTROLS:

NRC QUESTION 36

For power uprate, the plant-specific setpoint methodology has been used to determine instrument setpoints. Therefore, this methodology should be referenced in the Bases section of the TS.

SCS RESPONSE

The plant-specific setpoint methodology for extended power uprate is consistent with the current licensing basis for Plant Hatch and was not uniquely applied for extended power uprate. The plant-specific methodology was approved by the NRC under the Plant Hatch Analog Transmitter Trip System Program documented as follows:

1. Unit 1 - Submittals are documented in GPC letters dated September 5, 1984, and July 24, 1985, with NRC approval in Technical Specifications Amendment 121, dated January 17, 1986.

2. Unit 2 - Submittals are documented in GPC letters dated February 23, 1983; January 23, 1984; and June 14, 1984, with NRC approval in Technical Specifications Amendment 39 dated July 13, 1984.

The setpoint methodology, although plant-specific, is similar to the setpoint methodology given in GE topical report NEDC-31336, "Instrument Setpoint Methodology," October 1986.

Southern Nuclear does not plan to include the setpoint methodology within the Technical Specifications Bases. The contents of the Plant Hatch Technical Specifications Bases are in accordance with the standard Technical Specifications given in NUREG 1433.

RADIATION PROTECTION:

NRC QUESTION 37

Section 9.2, Design Basis Accidents, of Enclosure 6 to the Extended Power Uprate Safety Analysis Report for the Plant Hatch Units 1 and 2, states that plant-specific radiological consequence analyses were performed at extended power uprate conditions for selected postulated accidents.

Provide major parameters and assumptions used in the reevaluation of the radiological consequences complete with dose calculations performed for the site boundaries (Exclusion Area Boundary and Low Population zone) and for the control room operators resulting from (1) the LOCA, (2) the Fuel Handling Accident (Refueling Accident), and (3) the Control Rod Drop Accident.

SNC RESPONSE

Major parameters and assumptions utilized for the purposes of the revised radiological consequences as a result of LOCA, Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA) were consistent to the extent possible with previous analyses. Major parameters and assumptions used in the analyses are provided below:

LOCA

The detailed post-LOCA dose methodology and results of the analysis for extended power uprate were provided to the NRC in letter, "Edwin Hatch Nuclear Plant Revised LOCA Doses," dated April 17, 1997. The methodology and assumptions specified above are consistent with and derived from this letter.

The duration of the accident is considered to be 30 days, and 100% of the core noble gases and 25% of the core iodines are assumed to be instantaneously released and uniformly distributed in the containment at the onset of the accident. Other design input parameters used for the purposes of the Main Control Room (MCR), Technical Support Center (TSC), site boundary, and low population zone (LPZ) dose analyses are listed in Table 37-1. The assumptions utilized for the purposes of the analysis are consistent with Regulatory Guide 1.3, Standard Review Plans (SRP) 6.4, 15.6.5 and SRP 15.6.5, Appendix A. The sum of the dose contributions from the main stack, MSIV leakage, and the reactor building releases is used to determine the total MCR, TSC, site boundary, and LPZ doses. For LOCA MSIV leakage dose contribution, credit was taken for the holdup and deposition of activity in the main steam line and main condenser flow paths prior to release of activity from the condenser to the environment. Key assumptions for LOCA MSIV leakage dose analyses are listed in Table 37-2. Separate analyses were performed for Unit 1 and Unit 2 using the bounding post-LOCA doses (determined to result from a Unit 1 LOCA). The bounding LOCA doses are included in Table 9-3 of the Extended Power Uprate Safety Analysis Report (GE Nuclear Energy Topical Report NEDC-32749P).

TABLE 37-1

LOCA ANALYSIS PARAMETERS

PARAMETER	VALUE
Containment	
Containment Volume	2.6E5 ft^3
Containment Leakage Rate	1.2% per day
Containment Bypass Leakage	0.9% of 1.2% per day
Iodine Form Elemental Organic Particulate	 91% 4% 5%
Containment Filter Efficiency	95% iodine, 0% for noble gases
Control Room	
Control Room Envelope Volume	93,500 ft^3
Control Room Occupancy Factors	
0 - 8 hours	1.0
8 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4
MCR Breathing Rate (0 - 30 days)	3.47E-4 m^3/sec
Filtered Intake Rate	400 cfm
Unfiltered Intake Rate	0 cfm
MCR Recirculation Rate	2100 cfm
Filter Efficiency	95% iodines, 0% noble gases
Technical Support Center	
Technical Support Center Volume	1.56E4 ft^3
Technical Support Center Occupancy Factors	
0 - 8 hours	1.0
8 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4
TSC Breathing Rate (0 - 30 days)	3.47E-4 m^3/sec
Filtered Intake Rate	500 cfm
Unfiltered Intake Rate	0 cfm
Filtered Recirculation Rate	500 cfm
Intake/Recirculation Filter Iodine Efficiency	99%
Site Boundary/LPZ	
Breathing Rates	
0-8 hr	3.47E-4 m^3/sec
8-24 hr	1.75E-4 m^3/sec
24h - 30d	2.32E-4 m^3/sec

TABLE 37-2

KEY ASSUMPTIONS FOR LOCA MSIV LEAKAGE DOSE ANALYSIS

Reactor Power (MWt)	2818
Reactor Fission Product Inventory (Ci/MWt)	Revised GE Generic Source Term
Initial Inventory Fractions in Containment Atmosphere (%) Noble gases Iodines	 100 25
Primary Containment Leak Rate Excludes MSIV Leakage (%/day)	1.2
Total Primary Containment Free Volume (ft^3)	269,910 (Unit 1) 267,066 (Unit 2)
MSIV Leakage Rate (scfh, Total of 4 Lines) MSIV Leakage Rate (%/day, Total of 4 Lines)	250 0.814 (Unit 1), 0.812 (Unit 2)
Effective Condenser Free Volume (ft^3)	83,180
Condenser Leak Rate to Environment (%/day)	6.83 (Unit 1), 6.84 (Unit 2)
Control Room Intake (m^3/sec) Filtered Unfiltered	 0.190 0
Control Room Intake Filter Effic. (%)	95
Control Room Recirculation Rate (m^3/sec)	0.991 (Unit 1), 0.990 (Unit 2)
Control Room Recirculation Filter Efficiency (%)	95
Control Structure Ventilated Free Volume (m^3)	2,648
Control Room Free Volume (m^3)	2,648
Technical Support Center Intake (m^3/sec) Filtered Unfiltered	 0.236 0
Technical Support Center Intake Filter Effic. (%)	99
Technical Support Center Recirculation Rate (m^3/sec)	0.236
Technical Support Center Recirculation Filter Effic. (%)	99
Technical Support Center Free Volume (m^3)	442
Atmospheric Dispersion Factors	Conservatively assumed same as for Main Control Room Intake

The current Unit 1 analysis assumes 46 standard cubic feet per hour (scfh), or 11.5 scfh per valve of MSIV leakage into the Reactor Building. This leakage is included in the 1.2% per day leak from primary containment and is treated as a filtered release. This remains the design basis for Unit 1. However, for the purposes of the dose analysis for extended power uprate, it was conservatively assumed that the MSIV leakage rate for Unit 1 was equal to the Unit 2 Technical Specifications value of 250 scfh. Releases to the environment are via the turbine building as assumed in the Unit 2 dose model. Also, no credit was taken for decay or holdup in the Reactor Building or for the operation of the drywell spray system.

The post-LOCA plant-specific doses are based on new atmospheric dispersion factors (χ/Qs) and the GE generic source term for a power level of 2818 MWt (2763 MWt plus a margin of 2%). The computer model ARCON95 was used to develop new χ/Qs at the MCR and TSC intakes for releases from the reactor building, turbine building, and elevated release from the main stack. The revised post-LOCA site boundary and low population zone dose analysis was based on existing Unit 2 χ/Q values, which are more conservative than existing Unit 1 χ/Q values. The doses were calculated using International Commission on Radiological Protection-30 (ICRP) conversion factors.

Fuel Handling (Refueling) Accident (FHA)

The offsite dose calculations for the FHA under extended power uprate conditions were performed with the GE Computer Code CONAC04A. The key parameters assumed for the dose analysis are provided in Table 37-3. The only differences in the parameters from the original power uprate dose analysis are the increase in thermal power level (2818 MWt) and the use of GE generic fission product inventory. Results of the offsite dose analysis for the FHA are presented in Table 37-5 and are identified as negligible in the Safety Analysis Report for Extended Power Uprate (GE Nuclear Energy Topical Report NEDC-32749P) since doses are less than 10% of the applicable regulatory limits.

Control Rod Drop Accident (CRDA)

The offsite dose calculations for the CRDA under extended power uprate conditions were performed with the GE Computer Code CONAC04A. The key parameters assumed for the purposes of the dose analysis are provided in Table 37-4. The only differences in the parameters from the original power uprate dose analysis are the increase in thermal power level (2818 MWt) and the use of GE generic fission product inventory. Results of the offsite dose analysis for the CRDA are presented in Table 37-6 and are identified as negligible in the Safety Analysis Report for Extended Power Uprate (GE Nuclear Energy Topical Report NEDC-32749P) since the doses are less than 10% of the applicable regulatory limits.

While preparing the response to NRC questions for extended power uprate, a discrepancy was identified between the calculations for design of the TSC HVAC system and the FSAR. The design calculations are based on outside makeup air of 250 cfm, whereas Unit 2 FSAR Section 9.4.9 specifies a makeup air rate of 500 cfm. The actual flow has been verified to be below the FSAR value but higher than the assumed value of the design calculation. Since higher makeup air to the TSC provides more conservative TSC doses, the dose analysis for the TSC was revised to be consistent with the FSAR. Table 37-7 provides a summary of the revised TSC doses following a LOCA. The change to Page 9-8 (Table 9-3) of the licensing submittal is provided in Enclosure 2. The total TSC doses following a LOCA are still well below Regulatory Limits.

TABLE 37-3

KEY ASSUMPTIONS FOR FHA OFFSITE DOSE ANALYSIS

Reactor Power (MWt)	2818
Decay Time after Shutdown (hr)	24
Number of Damaged Fuel Rods	125
Number of Fuel Rods per Bundle	62
Number of Fuel Bundles in Core	560
Fuel Rod Plenum Activity Fractions (%)	
Noble gases (except Kr-85)	10
Kr-85	30
Iodines	10
Radial Power Peaking Factor	1.5
Decontamination Factor in Water	
Noble gases	1
Iodines	100
Secondary Containment Drawdown	
Time to Establish -1/4" wg (sec)	100
Building Leak Rate During Drawdown (%/day)	360
Time Period for Release of Activity from Reactor Building (hr)	<2
SGTS Iodine Filter Efficiency (%)	95
Atmospheric Dispersion Factor (sec/m3)	
Ground Level:	
0 - 100 sec	4.1E-04
Stack Release:	
100 sec - 2 hrs	1.7E-06
Thyroid Inhalation DCF (rem/Ci)	
I-131	1.08E+6
I-132	6.44E+3
I-133	1.80E+5
I-134	1.07E+3
I-135	3.13E+4

TABLE 37-4

KEY ASSUMPTIONS FOR CRDA OFFSITE DOSE ANALYSIS

Reactor Power (MWt)	2818
Number of Fuel Rods Damaged Number with Cladding Failure Number with Melting	770 763 7
Number of Fuel Rods per Bundle	62
Number of Fuel Bundles in Core	560
Radial Power Peaking Factor	1.5
Decay Time after Full Power Operation Assumed for Fuel Rod Activity	0
Activity Release from Damaged Rods (%) Rods with Cladding Failure: Noble gases Iodines Rods with Melting: Noble gases Iodines	 10 10 100 50
Fraction of Released Activity Transported to Condenser (%) Noble gases Iodines	 100 10
Fraction of Iodine Washed Out in Condenser (%)	50
Leak Rate from Condenser (%/day)	0.5
Duration of Release from Condenser (hrs)	24
Holdup in Turbine Building	No
Release Height (m)	0
Atmospheric Dispersion Factor (sec/m3) Ground Level 0 - 2 hrs 2 - 8 hrs 8 - 24 hrs	 3.1E-04 1.7E-04 2.3E-05
Thyroid Inhalation DCF (rem/Ci) I-131 I-132 I-133 I-134 I-135	 1.08E+6 6.44E+3 1.80E+5 1.07E+3 3.13E+4

TABLE 37-5

FHA RADIOLOGICAL CONSEQUENCES FOR EXTENDED POWER UPRATE

LOCATION	DOSE (REM) RATED (2664 MWt)	EXTENDED POWER UPRATE (2818 MWt)	LIMIT
Exclusion Area (2 Hrs)			
Whole Body	0.0084	0.0083	≤6
Thyroid	0.30	0.30	≤75
Low Population Zone (30 Days)			
Whole Body	0.0084	0.0083	≤6
Thyroid	0.30	0.30	≤75

TABLE 37-6

CRDA RADIOLOGICAL CONSEQUENCES FOR EXTENDED POWER UPRATE

LOCATION	DOSE (REM) RATED (2664 MWt)	EXTENDED POWER UPRATE (2818 MWt)	LIMIT
Exclusion Area (2 Hrs)			
Whole Body	0.015	0.021	≤6
Thyroid	0.80	0.86	≤75
Low Population Zone (30 Days)			
Whole Body	0.026	0.035	≤6
Thyroid	2.30	2.4	≤75

TABLE 37-7

SUMMARY OF TECHNICAL SUPPORT CENTER DOSES

	TSC Thyroid (rem)	TSC Beta Skin (rem)	TSC Whole Body (rem)
Reactor Bldg. Release	9.15	1.21	0.257
Main Stack Release	0.258	0.148	6.09E-3
MSIV Leakage	8.1	4.1	0.15
Total Dose	17.5	5.46	0.412
Regulatory Limits	30.0	30.0	5.0

NRC QUESTION 38

Provide an electronic copy of the meteorological data used in the revised post-LOCA dose assessment for Plant Hatch. The hourly data should be provided in the format specified in Appendix A to Section 2.7, "Meteorology and Air Quality," of the August 1997 draft NUREG-1555, "Environmental Standard Review Plan," as attached. If the data are in a compressed format, the capability to decompress the data should be provided with the data. In addition, summary joint wind speed, wind direction, and atmospheric stability frequency distributions of the data should be provided.

SNC RESPONSE

The Plant Hatch onsite meteorological data for the year 1995 is provided in ASCII on a diskette in Enclosure 4. The data is provided in the format specified in Appendix A to Section 2.7, "Meteorology and Air Quality" of the August 1997 draft of NUREG-1555, "Environmental Standard Review Plan."

Tables 38-1 through 38-16 provide the joint frequency distributions of wind speed and direction 10m versus delta temperature 60-10m, and wind speed and direction 100m versus delta temperature 100-10m, respectively.

TABLE 38-1

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1[illegible]5

Stability Class: A

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	7	108	5	0	0	0	120
NNE	11	41	0	0	0	0	52
NE	9	144	4	0	0	0	157
ENE	16	114	6	0	0	0	136
E	10	64	0	0	0	0	74
ESE	16	64	1	0	0	0	81
SE	13	59	5	0	0	0	77
SSE	11	50	6	0	0	0	67
S	5	57	4	0	0	0	66
SSW	7	79	9	0	0	0	95
SW	17	76	5	0	0	0	98
WSW	6	82	7	0	0	0	95
W	11	70	11	0	0	0	92
WNW	10	110	13	0	0	0	133
NW	20	112	12	1	0	0	145
NNW	7	63	9	1	0	0	80
Total	176	1293	97	2	0	0	1568

Periods of Calm (Hours): 0
Variable Direction: 746
Hours of Missing Data: 1

TABLE 38-2

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: B

Hours at Each Wind Speed and Direction

Wind Direction	Wind Speed (MPH)						
	1-3	4-7	8-12	13-18	19-24	>24	Total
N	10	16	2	0	0	0	28
NNE	6	10	0	0	0	0	16
NE	11	29	0	0	0	0	40
ENE	10	27	0	0	0	0	37
E	10	8	0	0	0	0	18
ESE	4	7	0	0	0	0	11
SE	8	10	2	0	0	0	20
SSE	10	15	2	0	0	0	27
S	4	11	0	0	0	0	15
SSW	9	20	4	0	0	0	33
SW	5	15	0	0	0	0	20
WSW	8	15	0	0	0	0	23
W	2	15	0	0	0	0	17
WNW	3	18	6	0	0	0	27
NW	10	22	1	0	0	0	33
NNW	4	15	1	0	0	0	20
Total	114	253	18	0	0	0	385

Periods of Calm (Hours): 0
Variable Direction: 264
Hours of Missing Data: 1

TABLE 38-3

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: C

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	6	18	3	0	0	0	27
NNE	7	6	0	0	0	0	13
NE	11	28	0	0	0	0	39
ENE	12	19	0	0	0	0	31
E	10	11	0	0	0	0	21
ESE	7	5	0	0	0	0	12
SE	8	10	0	0	0	0	18
SSE	8	8	0	0	0	0	16
S	15	12	0	0	0	0	27
SSW	10	16	1	0	0	0	27
SW	18	19	1	0	0	0	38
WSW	5	9	0	0	0	0	14
W	9	15	2	0	0	0	26
WNW	11	21	4	0	0	0	36
NW	10	19	2	0	0	0	31
NNW	16	12	0	0	0	0	28
Total	163	228	13	0	0	0	404

Periods of Calm (Hours): 0
Variable Direction: 306
Hours of Missing Data: 1

TABLE 38-4

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: D

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	49	78	3	0	0	0	130
NNE	48	55	1	0	0	0	104
NE	95	133	3	0	0	0	231
ENE	74	89	2	0	0	0	165
E	64	33	0	0	0	0	97
ESE	46	36	0	0	0	0	82
SE	44	49	5	0	0	0	98
SSE	41	31	1	0	0	0	73
S	44	37	9	0	0	0	90
SSW	48	49	7	0	0	0	104
SW	45	49	2	0	0	0	96
WSW	39	29	1	0	0	0	69
W	46	40	6	0	0	0	92
WNW	42	50	17	0	0	0	109
NW	42	67	6	0	0	0	115
NNW	35	81	6	0	0	0	122
Total	802	906	69	0	0	0	1777

Periods of Calm (Hours): 0
Variable Direction: 1358
Hours of Missing Data: 1

TABLE 38-5

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: E

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	73	49	0	0	0	0	122
NNE	70	18	0	0	0	0	88
NE	135	78	0	0	0	0	213
ENE	147	82	2	0	0	0	231
E	102	29	0	0	0	0	131
ESE	105	33	1	0	0	0	139
SE	126	61	5	0	0	0	192
SSE	99	55	6	0	0	0	160
S	111	72	14	1	0	0	198
SSW	141	78	9	0	0	0	228
SW	150	39	1	0	0	0	190
WSW	106	50	0	0	0	0	156
W	81	45	2	0	0	0	128
WNW	102	66	5	0	0	0	173
NW	99	64	1	0	0	0	164
NNW	75	51	1	0	0	0	127
Total	1722	870	47	1	0	0	2640

Periods of Calm (Hours): 0
Variable Direction: 2161
Hours of Missing Data: 1

TABLE 38-6

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: F

Hours at Each Wind Speed and Direction

Wind Direction	Wind Speed (MPH) 1-3	4-7	8-12	13-18	19-24	>24	Total
N	44	8	0	0	0	0	52
NNE	34	1	0	0	0	0	35
NE	63	3	0	0	0	0	66
ENE	80	3	0	0	0	0	83
E	56	0	0	0	0	0	56
ESE	57	0	0	0	0	0	57
SE	44	3	0	0	0	0	47
SSE	29	1	0	0	0	0	30
S	64	4	0	0	0	0	68
SSW	67	3	0	0	0	0	70
SW	115	5	0	0	0	0	120
WSW	100	1	0	0	0	0	101
W	99	1	0	0	0	0	100
WNW	56	3	0	0	0	0	59
NW	53	2	0	0	0	0	55
NNW	35	8	0	0	0	0	43
Total	996	46	0	0	0	0	1042

Periods of Calm (Hours): 0
Variable Direction: 862
Hours of Missing Data: 1

TABLE 38-7

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: G

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	17	0	0	0	0	0	17
NNE	30	0	0	0	0	0	30
NE	37	0	0	0	0	0	37
ENE	28	0	0	0	0	0	28
E	28	1	0	0	0	0	29
ESE	31	0	0	0	0	0	31
SE	34	1	0	0	0	0	35
SSE	35	0	0	0	0	0	35
S	90	1	0	0	0	0	91
SSW	123	0	0	0	0	0	123
SW	153	0	0	0	0	0	153
WSW	133	0	0	0	0	0	133
W	100	0	0	0	0	0	100
WNW	45	0	0	0	0	0	45
NW	34	0	0	0	0	0	34
NNW	22	0	0	0	0	0	22
Total	940	3	0	0	0	0	943

Periods of Calm (Hours): 0
Variable Direction: 564
Hours of Missing Data: 1

TABLE 38-8

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 10m VS DELTA TEMPERATURE 60-10m
January 1, 1995 Through December 31, 1995

Stability Class: All

Hours at Each Wind Speed and Direction

Wind Direction	Wind Speed (MPH)						
	1-3	4-7	8-12	13-18	19-24	>24	Total
N	206	277	13	0	0	0	496
NNE	206	131	1	0	0	0	338
NE	361	415	7	0	0	0	783
ENE	367	334	10	0	0	0	711
E	280	146	0	0	0	0	426
ESE	266	145	2	0	0	0	413
SE	277	193	17	0	0	0	487
SSE	233	160	15	0	0	0	408
S	333	194	27	1	0	0	555
SSW	405	245	30	0	0	0	680
SW	503	203	9	0	0	0	715
WSW	397	186	8	0	0	0	591
W	348	186	21	0	0	0	555
WNW	269	268	45	0	0	0	582
NW	268	286	22	1	0	0	577
NNW	194	230	17	1	0	0	442
Total	4913	3599	244	3	0	0	8759

Periods of Calm (Hours): 0
Variable Direction: 6261
Hours of Missing Data: 1

TABLE 38-9

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 100m VS DELTA TEMPERATURE 100-10m
January 1, 1995 Through December 31, 1995

Stability Class: A

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	0	3	2	2	0	0	7
NNE	0	5	7	0	0	0	12
NE	0	6	38	8	0	0	52
ENE	0	12	35	18	0	0	65
E	0	9	15	12	1	0	37
ESE	0	4	17	3	1	0	25
SE	1	8	7	2	0	0	18
SSE	1	6	6	2	1	0	16
S	0	2	5	3	0	0	10
SSW	0	4	9	5	0	0	18
SW	0	6	7	13	1	0	27
WSW	0	0	19	16	7	0	42
W	0	1	11	19	4	0	35
WNW	0	3	14	14	2	1	34
NW	0	5	12	6	3	2	28
NNW	0	3	2	0	0	0	5
Total	2	77	206	123	20	3	431

Periods of Calm (Hours): 0
Variable Direction: 16
Hours of Missing Data: 2

TABLE 38-10

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 100m VS DELTA TEMPERATURE 100-10m
January 1, 1995 Through December 31, 1995

Stability Class: B

Hours at Each Wind Speed and Direction

Wind Direction	Wind Speed (MPH)						
	1-3	4-7	8-12	13-18	19-24	>24	Total
N	1	12	12	3	0	0	28
NNE	0	3	14	0	0	0	17
NE	0	8	31	18	0	0	57
ENE	0	11	22	9	2	0	44
E	2	12	11	5	0	0	30
ESE	1	7	6	1	1	0	16
SE	0	5	8	0	1	0	14
SSE	1	4	2	3	0	0	10
S	0	10	10	3	0	0	23
SSW	1	4	12	5	1	0	23
SW	0	6	10	4	1	0	21
WSW	1	5	10	4	6	0	26
W	0	6	14	11	8	0	39
WNW	0	4	17	15	1	0	37
NW	0	7	17	7	2	0	33
NNW	0	6	4	2	0	0	12
Total	7	110	200	90	23	0	430

Periods of Calm (Hours): 0
Variable Direction: 25
Hours of Missing Data: 2

TABLE 38-11

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 100m VS DELTA TEMPERATURE 100-10m
January 1, 1995 Through December 31, 1995

Stability Class: C

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	1	16	13	2	0	0	32
NNE	3	6	4	2	0	0	15
NE	1	15	14	17	0	0	47
ENE	1	11	24	11	0	0	47
E	3	12	8	4	0	0	27
ESE	1	11	7	7	0	0	26
SE	0	4	12	2	0	0	18
SSE	1	5	7	1	1	0	15
S	0	5	9	5	0	0	19
SSW	0	7	9	5	2	0	23
SW	2	4	10	10	4	0	30
WSW	0	10	16	6	3	0	35
W	2	8	20	8	6	0	44
WNW	1	8	28	11	1	3	52
NW	0	10	21	8	2	1	42
NNW	0	13	9	3	0	0	25
Total	16	145	211	102	19	4	497

Periods of Calm (Hours): 0
Variable Direction: 47
Hours of Missing Data: 2

TABLE 38-12

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 100m VS DELTA TEMPERATURE 100-10m
January 1, 1995 Through December 31, 1995

Stability Class: D

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	11	43	35	24	1	0	114
NNE	15	30	53	16	0	0	114
NE	15	64	126	80	8	0	293
ENE	10	66	96	49	3	1	225
E	9	46	59	16	0	0	130
ESE	6	32	30	31	3	0	102
SE	11	31	38	10	2	0	92
SSE	8	32	39	10	2	1	92
S	10	45	36	12	3	1	107
SSW	15	42	51	36	12	1	157
SW	7	35	46	34	14	0	136
WSW	12	33	31	27	4	1	108
W	11	48	58	36	21	3	177
WNW	14	50	77	45	17	9	212
NW	7	64	40	39	10	3	163
NNW	7	41	41	21	3	0	113
Total	168	702	856	486	103	20	2335

Periods of Calm (Hours): 0
Variable Direction: 315
Hours of Missing Data: 2

TABLE 38-13

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 100m VS DELTA TEMPERATURE 100-10m
January 1, 1995 Through December 31, 1995

Stability Class: E

Hours at Each Wind Speed and Direction

Wind Direction	Wind Speed (MPH)						
	1-3	4-7	8-12	13-18	19-24	>24	Total
N	8	17	42	46	0	0	113
NNE	3	27	32	26	0	0	88
NE	6	33	88	204	20	0	351
ENE	8	21	88	89	12	1	219
E	4	24	95	83	8	1	215
ESE	5	30	95	88	13	0	231
SE	7	72	105	26	3	0	213
SSE	5	22	83	87	7	3	207
S	5	19	66	95	21	6	212
SSW	9	17	77	114	41	2	260
SW	7	19	69	102	13	0	210
WSW	2	8	61	65	18	0	154
W	3	23	55	82	22	4	189
WNW	5	21	48	106	12	2	194
NW	5	16	58	100	8	0	187
NNW	5	17	43	54	1	0	120
Total	87	386	1105	1367	199	19	3163

Periods of Calm (Hours): 0
Variable Direction: 87
Hours of Missing Data: 2

TABLE 38-14

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 100m VS DELTA TEMPERATURE 100-10m
January 1, 1995 Through December 31, 1995

Stability Class: F

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	3	3	14	24	2	1	47
NNE	2	4	16	13	0	0	35
NE	4	7	20	28	1	0	60
ENE	1	4	24	60	0	0	89
E	2	10	25	43	0	0	80
ESE	0	14	23	25	9	0	71
SE	6	31	33	11	0	0	81
SSE	4	9	23	16	2	0	54
S	4	7	18	39	2	0	70
SSW	5	7	28	31	5	0	76
SW	1	4	30	36	8	0	79
WSW	4	6	19	58	21	0	108
W	1	13	22	85	18	0	139
WNW	0	10	12	45	6	0	73
NW	3	8	16	37	1	0	65
NNW	2	7	20	21	1	0	51
Total	42	144	343	572	76	1	1178

Periods of Calm (Hours): 0
Variable Direction: 17
Hours of Missing Data: 2

TABLE 38-15

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 100m VS DELTA TEMPERATURE 100-10m
January 1, 1995 Through December 31, 1995

Stability Class: G

Hours at Each Wind Speed and Direction

	Wind Speed (MPH)						
Wind Direction	1-3	4-7	8-12	13-18	19-24	>24	Total
N	3	9	9	5	1	0	27
NNE	3	3	9	1	0	0	16
NE	4	7	10	4	2	0	27
ENE	5	8	7	16	4	0	40
E	1	5	19	21	5	0	51
ESE	3	8	15	17	1	0	44
SE	6	29	17	4	1	0	57
SSE	2	8	12	12	0	0	34
S	2	4	15	17	7	0	45
SSW	0	12	27	20	8	0	67
SW	2	7	30	13	5	0	57
WSW	7	12	21	16	2	0	58
W	1	11	24	17	3	0	56
WNW	1	14	22	10	2	0	49
NW	4	19	20	11	1	0	55
NNW	2	14	11	13	1	0	41
Total	46	170	268	197	43	0	724

Periods of Calm (Hours): 0
Variable Direction: 16
Hours of Missing Data: 2

TABLE 38-16

PLANT HATCH JOINT FREQUENCY TABLES OF WIND SPEED AND WIND DIRECTION 100m VS DELTA TEMPERATURE 100-10m
January 1, 1995 Through December 31, 1995

Stability Class: All
Hours at Each Wind Speed and Direction

Wind Direction	Wind Speed (MPH)						
	1-3	4-7	8-12	13-18	19-24	>24	Total
N	27	103	127	106	4	1	368
NNE	26	78	135	58	0	0	297
NE	30	140	327	359	31	0	887
ENE	25	133	296	252	21	2	729
E	21	118	232	184	14	1	570
ESE	16	106	193	172	28	0	515
SE	31	180	220	55	7	0	493
SSE	22	86	172	131	13	4	428
S	21	92	159	174	33	7	486
SSW	30	93	213	216	69	3	624
SW	19	81	202	212	46	0	560
WSW	26	74	177	192	61	1	531
W	18	110	204	258	82	7	679
WNW	21	110	218	246	41	15	651
NW	19	129	184	208	27	6	573
NNW	16	101	130	114	6	0	367
Total	368	1734	3189	2937	483	47	8758

Periods of Calm (Hours): 0
Variable Direction: 523
Hours of Missing Data: 2

NRC QUESTION 39

Provide a description and diagram of the meteorological system used to collect the data used in the assessment, including a layout of the meteorological system with respect to topographic features, vegetation, buildings, and other structures. The heights of and distances between the various features and meteorological tower should be shown. A description of instrument placement and orientation on the tower, the period of measurement, temporal and spacial representativeness, and quality assurance measures should also be provided. Provide a description of any site-specific phenomena that would aid in better understanding the representativeness of the meteorological data.

SNC RESPONSE

The meteorological system used to collect the data for the extended power uprate dose assessment is the Plant Hatch primary meteorological tower. The primary meteorological tower is located approximately 3/4 of a mile SSW of the power block within the plant boundaries. The tower stands in an open area where cooling tower draft is unlikely due to the area's prevailing winds. The primary meteorological tower was designed and built to meet the requirements of proposed Revision 1 of Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants" (September 20, 1980) and Revision 1 of NUREG-0654, "Criteria for Preparation and Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."

The primary meteorological tower is instrumented at 10, 60 and 100 meters (above tower base) to characterize the conditions for diffusion estimates of radiological releases at different levels. Wind speed, direction, and direction variability instrumentation is installed at all three elevations. Dew point and ambient temperature instrumentation is located at the 10 meter level. Instrumentation is also provided to measure vertical temperature difference between 100 and 10 meters and 60 and 10 meters. Meteorological data are normally reduced to 15 minute averages centered on the hour. These hourly averages provide the information from which monthly, seasonal, and annual summaries can be prepared.

The instrumentation on the primary meteorological tower goes through a full calibration semi-annually per ANSI Standard 2.5, "Standard for Determining Meteorological Information at Nuclear Power Sites, 1984." The meteorological program at Plant Hatch includes surveillance requirements to assure a high quality of data. No significant quality assurance or calibration problems have been identified regarding the Plant Hatch meteorological tower during the last 5 years of operation.

The topography of the Plant Hatch site, including the locations of the Primary Meteorological tower and plant structures, is shown in the Unit 2 FSAR Figure 2.3-13.

There are 15 ft tall pine trees in the area near the tower. However, they are approximately 150 ft or more from the tower to prevent any adverse impact.

The temporal and spacial representativeness of the meteorological data is discussed in our response to NRC Question 41.

NRC QUESTION 40

Provide a list of the inputs used in the ARCON runs and, for each input, justify the appropriateness of its use. Figures may be helpful in describing why a release is postulated to occur from a particular location, and the distances, directions, heights, dimensions, and relationships related to the postulated sources, receptors, and paths of transport. What is the basis for the stability class categorization used in the ARCON computer runs?

SNC RESPONSE

Inputs used in the ARCON runs with justifications for their use are provided in Table 40-1.

TABLE 40-1

ARCON95 MODEL

Input to ARCON95 Model	Justification for Use
Meteorological Data One year (1995) of continuous hourly on-site meteorological data (wind speed, wind direction from the 10 meter and 100 meter levels, and stability class) provided by PLG Inc. was used as input to the ARCON95 model to calculate the χ/Q values.	Justification for one year of meteorological data to be representative of long term site conditions is provided in response to NRC Question 41.
Plant Data Plant data (e.g. building dimensions, stack location, stack height, and distances and elevations of the release points to the MCR and TSC intakes) were used as input to the ARCON95 model.	Plant data used for the purposes of the analysis are based on actual as-built conditions and have been derived from controlled plant drawings.
Effective Wind Directions The wind direction window sector size of $\pm$ 45° was utilized for the MCR and TSC χ/Q calculations. To properly simulate the impact of releases, the direction of some of the sources from the intake was adjusted.	In accordance with the guidelines provided by NUREG-CR6331.

Releases, which contribute to doses resulting from a postulated LOCA at the Edwin I. Hatch Nuclear Plant are:

- Unfiltered Reactor Building Ground Level Release (before Standby Gas Treatment System develops design negative pressure in the Reactor Building)
- Filtered Stack Release (via Standby Gas Treatment System)
- Reactor Building Bypass Leakage (Unfiltered ground level release)
- MSIV Leakage from the Turbine Building (Unit 2 only)

Based upon these contributions, potential release points were identified. Values of χ/Q were calculated for each of these postulated release points. The basis for selection of the release paths are as follows:

Reactor Building Release

Following a LOCA, the secondary containment is isolated, normal ventilation trips, and the supply and exhaust paths for the normal ventilation systems are isolated. The standby gas treatment system (SGTS) is initiated following a LOCA and establishes the design negative pressure in the secondary containment within 2 minutes. All access doors to the reactor building (RB) are airlock doors, and thus, leakage through the doors is not considered to be credible. The RB wall is a minimum of two feet thick concrete; therefore, leakage through the RB wall is also not considered credible. Intake louvers and the exhaust path to the RB vent for the RB ventilation system were also identified as the potential leakage paths. Of all the potential leakage paths identified, the RB vent provides the most conservative χ/Q values. Hence, the RB vents (RV1 and RV2 as shown on Figure 40-1) are considered as the release path from the RB.

Releases from RB vents were assumed to travel directly towards the MCR air intake following the slant range path calculated by the ARCON95 model. The assumption is conservative because it results in a shorter distance between release point and the air intake, than if the release was assumed to follow the building outline. Based upon the cavity height, these releases were treated as ground-level sources.

Turbine Building (MSIV Leakage Path) Release - Unit 2

For Unit 2, MSIV leakage via the main steam lines and main steam drain lines is directed to the main condenser where holdup and iodine plateout would occur. Subsequently, release to the environment would be via leakage through the low pressure turbine seals. This leakage would be into the turbine building (TB). The identified paths for release from the TB are as follows:

- Railroad doors (T1 and T2)
- TB ventilation system exhaust connected to the RB vent (RV1 and RV2)
- TB ventilation system supply air louvers (SL1 and SL2)

The Turbine Building railroad doors (T1 and T2 on Figure 40-1) are located on the east side of the building near each corner (approximately five feet from the building's corners) on the opposite side of the building from the MCR air intake location. It was assumed that the leakage through the doors would travel over the turbine building and into the wakes at the lee side of the building. This assumption yields a shorter distance between the release points and the air intakes compared to traveling around the perimeter of the building and therefore, it is conservative.

The TB supply louvers (SL1 and SL2 on Figure 40-1) are located at approximately the same elevation as the MCR air intake. Therefore, the shortest path for releases from the louvers to the air intakes would be over the TB roof, following the roof contour, and downward towards the MCR air intake. Following this path, horizontal and vertical distances between the louvers and the MCR air intake were accounted for as a part of the total distance between the louvers and the intakes.

Values of χ/Q for TB releases via the railroad doors, RB vent (as discussed in the reactor building release section), and TB ventilation system supply air louver were determined for both units. The RB building vent path provided the most conservative χ/Q values which were utilized for the dose analysis.

Separate χ/Q values were determined for the different release points from the RB and the TB to the TSC air intake. However, the MCR intake χ/Q values for the RB vent path were verified to provide the most conservative values and were utilized for the purpose of TSC dose analysis.

<u>Main Stack Release</u>

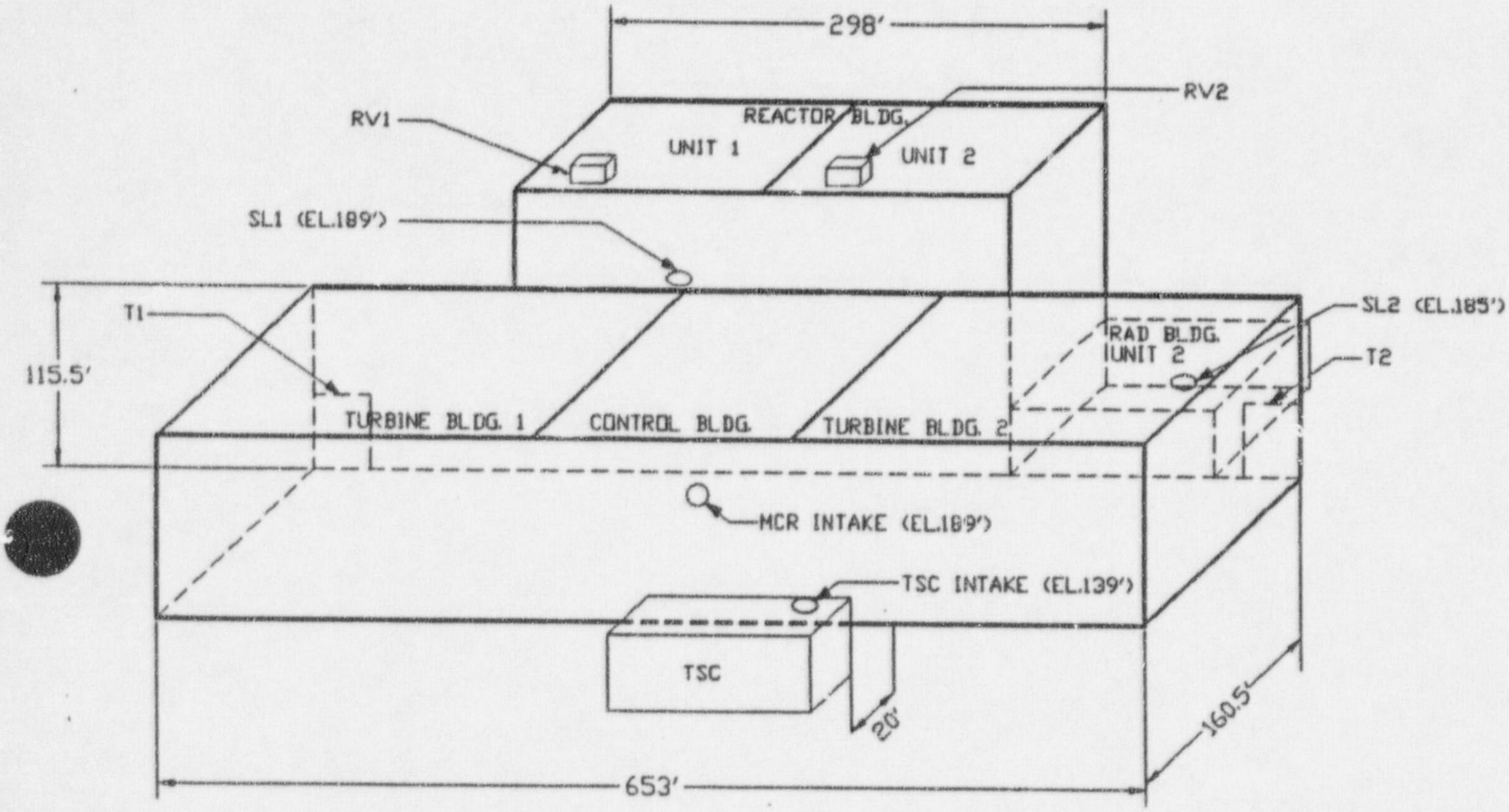
As discussed in NUREG/CR-6331, a conservative approach for estimating the concentrations downwind of short stacks and roof-top vents is to assume that a release takes place at the ground level unless the release point is 2.5 times the building height. Since the main stack has a height greater than 2.5 times the height of the reactor building, releases from the main stack were treated as elevated releases by the ARCON95 model. Elevated stack releases were assumed to be transported directly towards the MCR and TSC air intakes.

A detailed description of the postulated release paths in relation to the postulated release sources, receptors, and paths of transport for extended power uprate is provided in SNC letter "Edwin I. Hatch Nuclear Plant Revised Post-LOCA Doses," dated April 17, 1997.

The meteorological data used by the ARCON program consist of the following hourly records: lower wind direction, lower wind speed, stability class, upper wind direction, and upper wind speed. The classification of atmospheric stability was based on NRC's delta T ($\Delta T/\Delta z$) method as described in Regulatory Guide 1.23. At the Plant Hatch site, meteorological data were collected at 10 m, 60 m and 100 m levels. For the ARCON runs, the stability class categorization was based on temperature change with height between 10 m and 60 m levels.

Figure 40-1

Location of Potential Release



<u>LEGEND</u>

T1, T2 Turbine Building Railroad Doors
RV1, RV2 Reactor Building Vents
SL1, SL2 Turbine Building Ventilation System Intake
Ground Level Elevation 129'

<u>NOTES</u>

1. Drawing Not to Scale
2. Elevations Shown are Approximate

NRC QUESTION 41

The control room habitability assessment performed uses 1 year of meteorological data. Provide justification that the chosen year adequately represents long-term (e.g., 30 years) and site area (e.g., free from local obstructions such as trees or plant structures) conditions, and adequate measures to assure high data quality were taken.

SNC RESPONSE

The calendar year 1995 onsite meteorological data was chosen to be used for the control room habitability study because it was the most recent year completed at the time of the study and a typically representative year. Over the past 10 or more years, the average composite data recovery for 10 m wind speed, wind direction, and the delta temperature 60-10m has been better than 97%. In particular, over the last six years, the data recovery for each year has been over 99%.

Table 41-1 shows the last 10 years of joint frequency distributions summarized by stability class in percent including the 10 year average, total hours of valid data, and percent data recovery for the year. As is evident from the table, the data are quite consistent each year. As expected, there are some normal year-to-year climatic variations. These account for years with occurrences such as higher than normal precipitation and therefore more cloud cover and less unstable hours, or conversely, several dry years with abundant sunshine producing a higher percentage of unstable hours. For the year 1995, all stability classes (A-G) were within two percent of the 10 year average.

The wind directions have also been very consistent over the years of meteorological tower operation. The peak direction is typically from the northeast at all levels on the meteorological tower with secondary peaks from the southwest to the west. Figures 41-1 through 41-6 show wind roses for two periods, 1995 and 1994-96, and for each level (10, 60, 100m) on the Plant Hatch meteorological tower.

The meteorological tower sits in an open area about 1200 m south-southwest of any of the plant structures. There are 15 ft tall pine trees in the area near the tower but they are an adequate distance (150 ft or more) from the tower. See the response to NRC Question 39 for more information about the tower location and any nearby obstructions.

All of the meteorological data collected at Plant Hatch since 1970 has been processed by PLG, Incorporated. A quality assurance review of the data was performed by a meteorologist using PLG's Meteorological Projects Policies and Procedures manual during the licensing process. Data from the site meteorological tower were compared with data from local National Weather Service sites and found to be representative of the area. Each subsequent year has been compared with data from previous years and found to be representative of the Plant Hatch site area.

TABLE 41-1

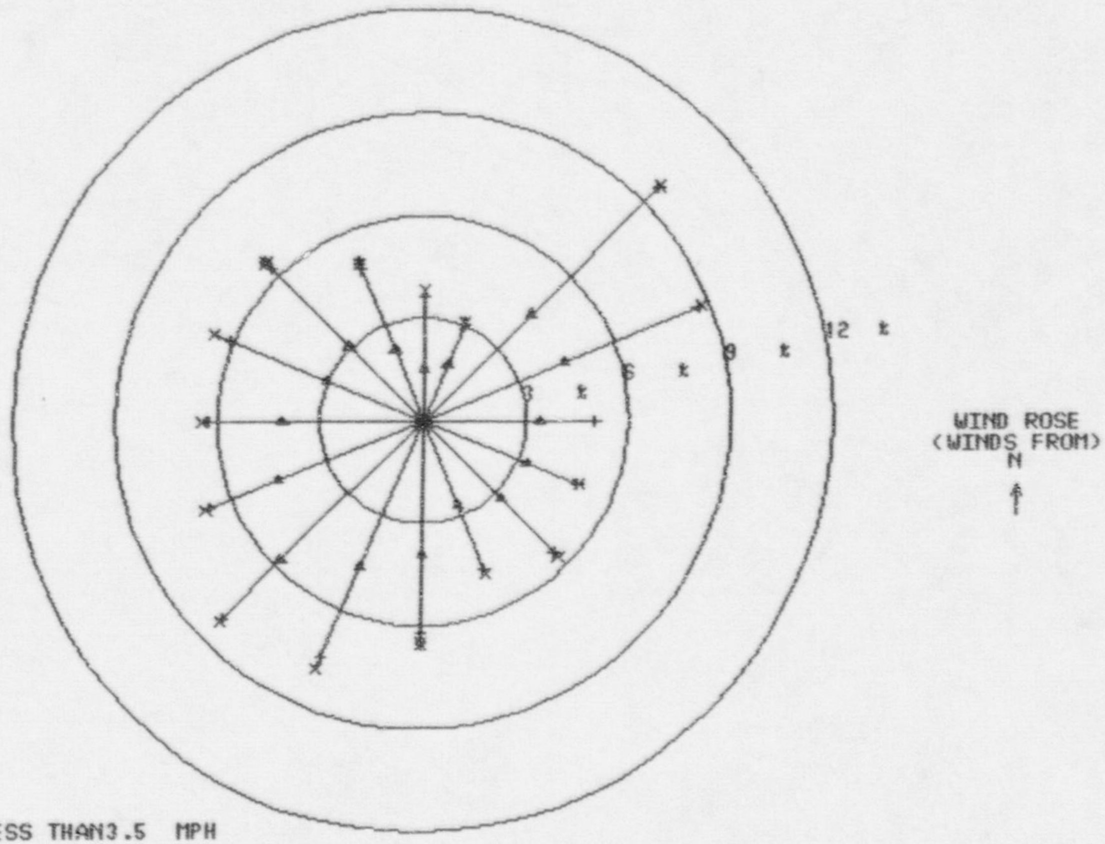
PLANT HATCH STABILITY CLASSIFICATION

Stability Group	Percent Stability										10-Year Average
	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	
A	15.7	12.2	11.8	16.3	14.3	17.8	12.7	18.0	17.9	20.4	15.7
B	5.0	4.0	4.0	4.7	5.1	4.3	5.1	4.7	4.4	3.6	4.5
C	4.0	3.1	3.3	3.9	4.6	4.1	5.2	4.4	4.6	3.5	4.1
D	16.8	18.4	22.0	19.1	24.7	23.6	22.7	23.5	20.3	18.0	20.9
E	31.6	31.8	33.1	29.3	31.0	28.7	28.3	28.1	30.1	28.4	30.0
F	12.3	15.3	12.8	12.8	9.8	14.0	10.4	11.7	11.9	12.5	12.4
G	14.6	15.2	13.0	13.9	10.5	9.8	12.0	10.9	10.8	13.6	12.4
Total Valid Hours	8162	8541	7886	8604	8692	8753	8695	8712	8759	8709	8551
Composite* Data Recovery	93.2	97.2	90.0	98.2	99.2	99.6	99.3	99.5	100.0	99.1	97.5

*Wind speed and direction 10 m and delta temperature 60-10 m.

Figure 41-1

Plant Hatch 10m Wind Rose
1995



WIND SPEED LESS THAN3.5 MPH
WIND SPEED LESS THAN7.5 MPH
WIND SPEED LESS THAN12.5 MPH
WIND SPEED GREATER THAN 12.5 MPH

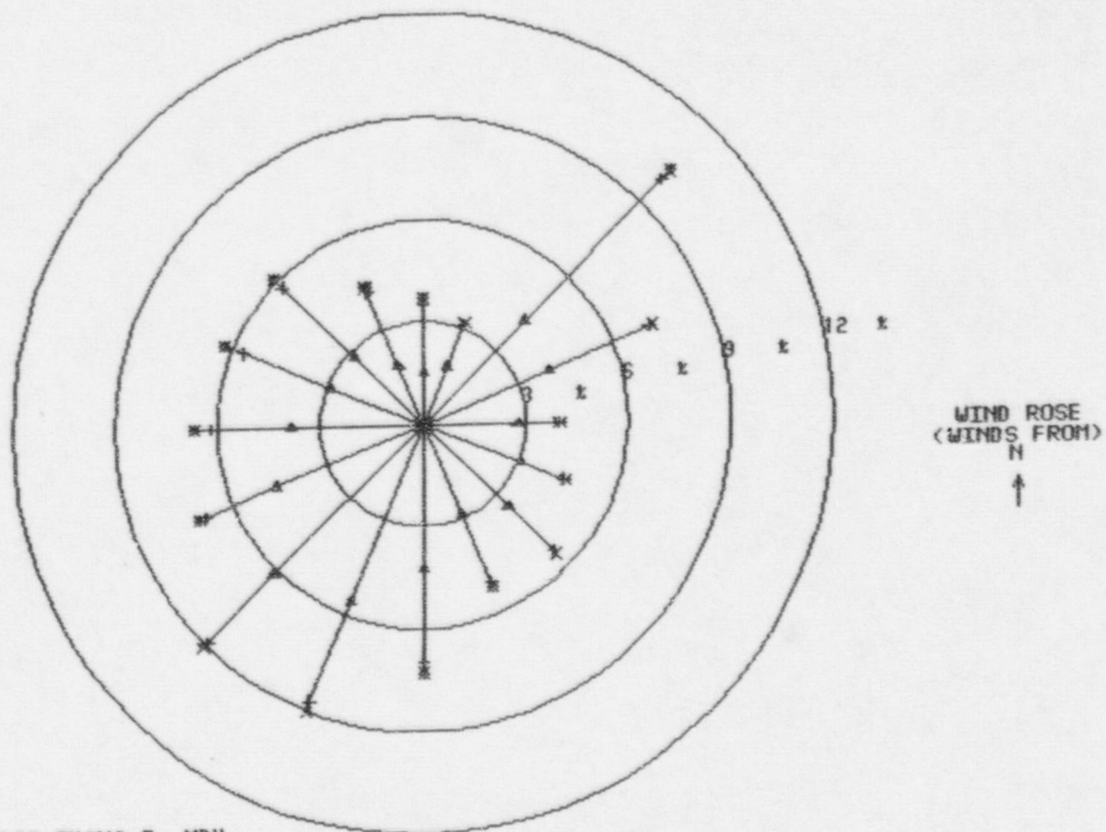
SITE: PLANT HATCH

0.0 PERCENT CALMS
(NOT INCLUDED IN PLOT)

01/22/98 15:39

Figure 41-2

**Plant Hatch 10m Wind Rose
1994 through 1996**



▲ WIND SPEED LESS THAN3.5 MPH
+ WIND SPEED LESS THAN7.5 MPH
× WIND SPEED LESS THAN12.5 MPH
◆ WIND SPEED GREATER THAN 12.5 MPH

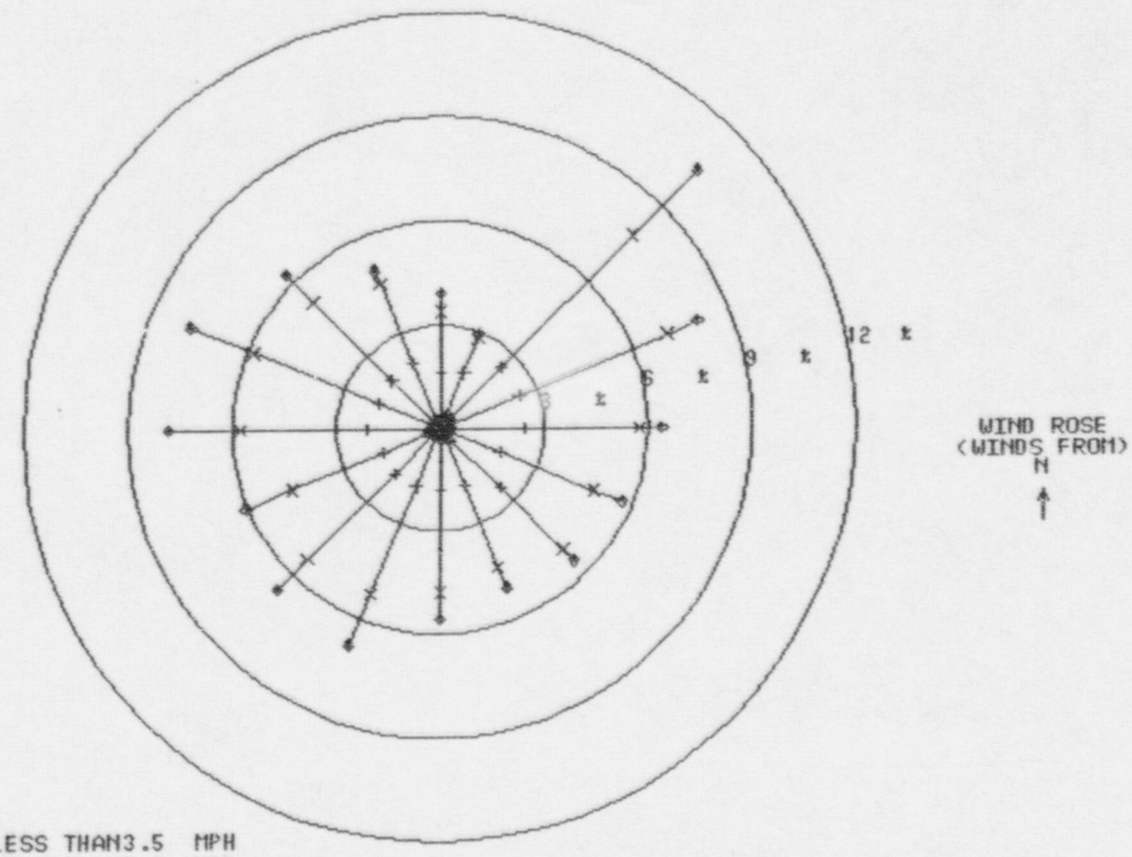
SITE: PLANT HATCH

0.4 PERCENT CALMS
(NOT INCLUDED IN PLOT)

01/22/98 15:36

Figure 41-3

Plant Hatch 60m Wind Rose
1995



▲WIND SPEED LESS THAN3.5 MPH
+WIND SPEED LESS THAN7.5 MPH
×WIND SPEED LESS THAN12.5 MPH
◆WIND SPEED GREATER THAN 12.5 MPH

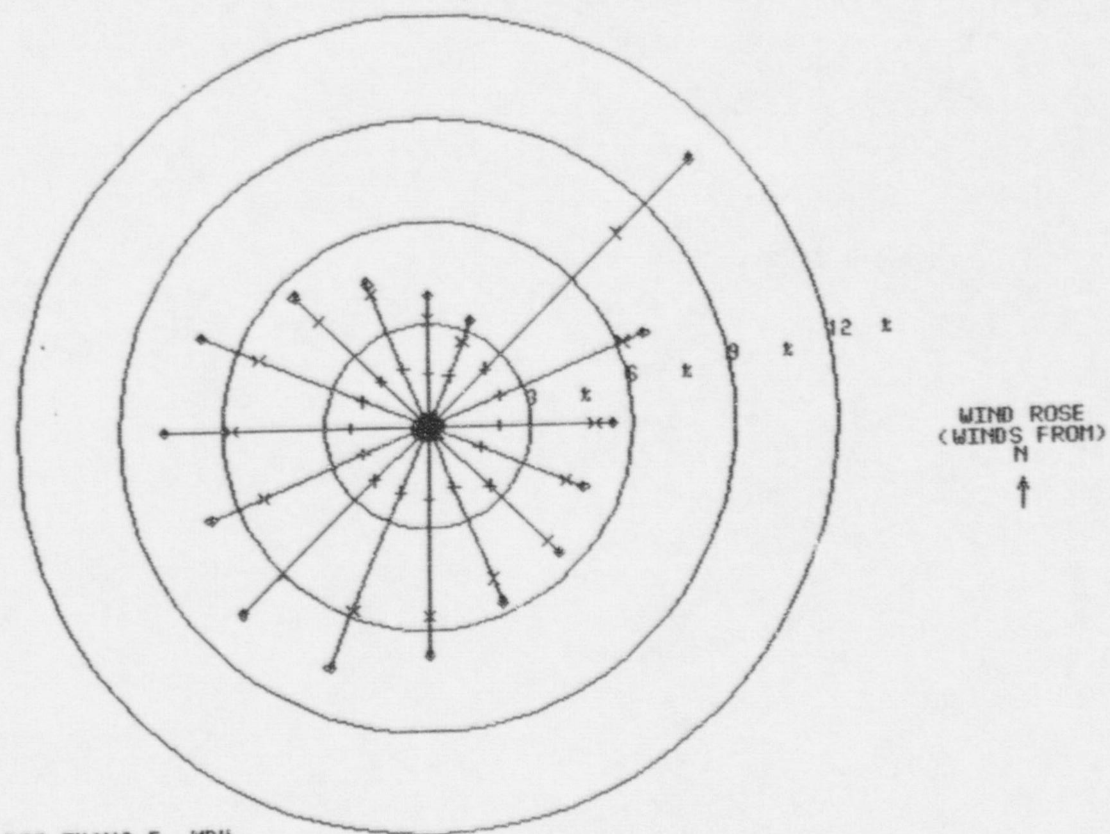
SITE: PLANT HATCH

0.0 PERCENT CALMS
(NOT INCLUDED IN PLOT)

01/22/98 15:40

Figure 41-4

Plant Hatch 60m Wind Rose
1994 through 1996



▲WIND SPEED LESS THAN3.5 MPH
+WIND SPEED LESS THAN7.5 MPH
×WIND SPEED LESS THAN12.5 MPH
◆WIND SPEED GREATER THAN 12.5 MPH

SITE: PLANT HATCH

0.0 PERCENT CALMS
(NOT INCLUDED IN PLOT)

01/22/98 15:38

Figure 41-5

Plant Hatch 100m Wind Rose
1995

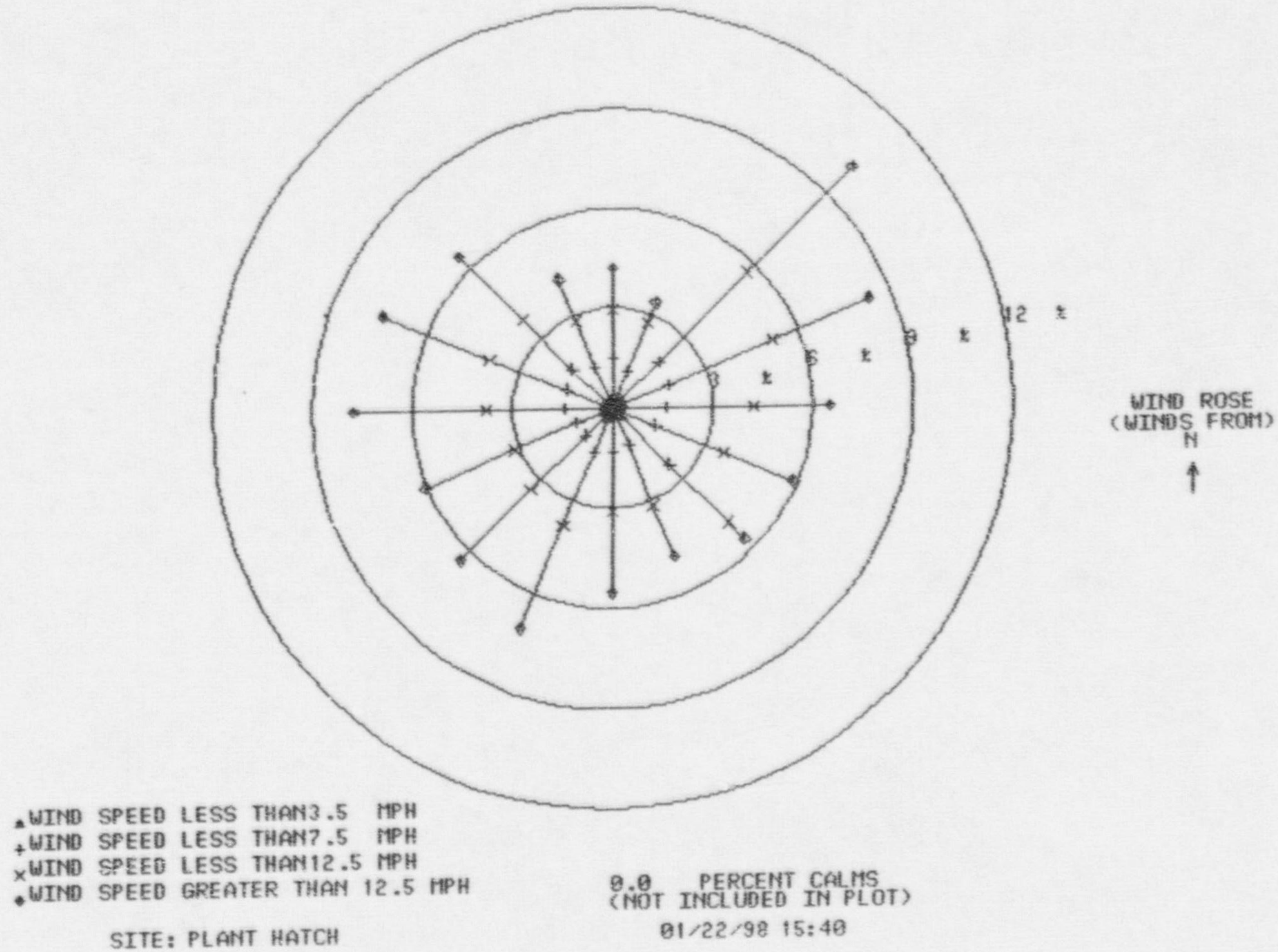
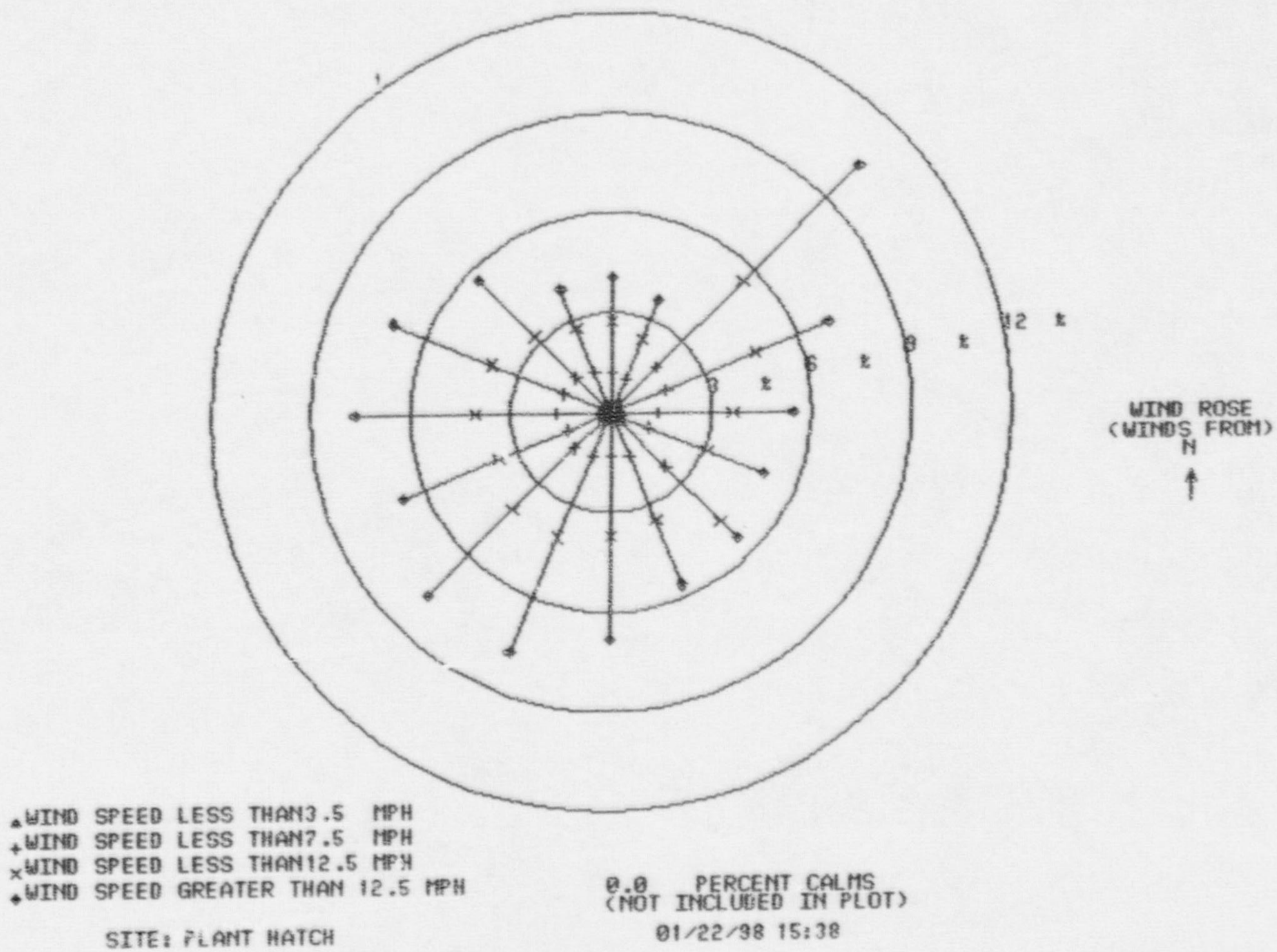


Figure 41-6

Plant Hatch 100m Wind Rose
1994 through 1996



NRC QUESTION 42

It is noted that PAVAN was used for the elevated release calculations (page E-1), yet it is also implied that ARCON was used (page E-5). Which one was used?

SNC RESPONSE

The existing dose analysis for a power level of 2558 MWt used the NRC program PAVAN, as discussed on page E-1 of the SNC letter "Edwin I. Hatch Revised Post-LOCA Doses," dated April 17, 1997. However, in the revised dose analysis for extended power uprate, the χ/Q values for main stack releases to the intakes of the MCR and TSC are determined based on the ARCON95 Model as stated on page E-5.

NRC QUESTION 43

Provide examples of operator actions potentially sensitive to power uprate and address whether the power uprate will have any effect on operator reliability or performance. Identify operator actions that would necessitate reduced response times associated with a power uprate? Please specify the expected response times before the power uprate and the reduced response times. What have simulator observations shown relative to operator response times for operator actions that are potentially sensitive to power uprate. Please state why reduced operator response times are needed. Please state whether reduced time available to the operator due to the power uprate will significantly affect the operator's ability to complete manual actions in the times required.

SNC RESPONSE

Operator actions will remain unchanged with the implementation of extended power uprate. Transients such as circulating water pump or feedwater pump trips do not introduce additional plant operator manual actions that affect response times and thus affect plant transient behavior. The operator response time due to the BOP changes are unchanged for plant transients such as a loss of feedwater or loss of circulation water pump event. The NSSS systems will not be significantly affected, therefore, the operator will not experience a significant difference in required response time due to these minor changes.

Plant operation at the extended power uprate conditions will not introduce any additional transients that will impact limits, or change operator performance in a transient condition. Emergency Operating Procedure Flow Chart actions will not be modified as a result of extended power uprate. No discernible reduction in effectiveness of operator response has been identified based on simulator observations and over 4 years of operation at the original power uprate level of 2558 MWt. Section 10.5.2.4 of NEDC-32749P provided a summary of the operator action analysis performed to evaluate the impact of extended power uprate on the Plant Hatch PRA.

NRC QUESTION 44

Discuss any changes the power uprate will have on control room instruments, alarms, and displays. Are zone markings on meters changed (e.g., normal range, marginal range, and out-of-tolerance range)?

SNC RESPONSE

The impact on control room instrumentation and controls is minimal. No changes to front panel indicators or controls are required. New programmable controls for the recirculation pump runback are being added. The ranges for the main steam line flow transmitters that interface with the ATTS trip units are changing; however there is no change to the ranges of the steam flow transmitters which send signals to the control panel indicators. Water level system initiation and trip points are not changed with extended power uprate. Absolute values for these variables will remain the same. There is no change to the steam saturation conditions; therefore, there are no density corrections required for the level instrument calibration values.

NRC QUESTION 45

Discuss any changes the power uprate will have on the Safety Parameter Display System (SPDS).

SNC RESPONSE

The analog and digital signal I/O lists for the SPDS and the Emergency Parameter Display System (ERDS) were reviewed to determine the impacts from extended power uprate. The digital inputs are not affected. Extended power uprate changes (setpoints and calibration) made to devices will be automatically reflected in SPDS. That is, the computer effectively receives a signal from a contact closure of field devices. Proposed design changes for extended power uprate were reviewed against the analog input lists for ERDS. The ERDS computer instrument range scaling was adequate.

NRC QUESTION 46

Describe any changes the power uprate will have on the operator training program and the plant simulator. Provide a copy of the post-modification test report (or test abstracts) to document and support the effectiveness of simulator changes as required by ANSI/ANS 3.5-1985, Section 5.4.1.

Specifically, please propose a license condition and/or commitments that address the following:

(a) Provide classroom and simulator training on the power uprate modification.

(b) Complete simulator changes that are consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be revalidated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." Simulator revalidation will include comparison of individual simulated systems and components and simulated integrated plant steady state and transient performance with reference plant responses using similar startup test procedures.

(c) Complete control room and plant process computer system changes as a result of the power uprate.

(d) Modify training and plant simulator relative to issues and discrepancies identified during the startup testing program.

SNC RESPONSE

Response to NRC Question 46.a

The Operator Training section will develop a lesson plan on plant changes as a result of extended power uprate. The lesson plan will be presented to all licensed/certified shift eligible plant personnel prior to plant startup (post outage) and to ALL licensed/certified personnel during the following segment of re-qualification training. Plant changes that are implemented in the simulator will be covered in the simulator during the first day of simulator training of each segment. Additionally, individual lesson plans will be flagged for revision during the next revision process. All modification information will be provided to other training staff sections for inclusion in their continuing training lesson plans.

Response to NRC Question 46.b

To compare the simulator performance with plant operation after implementation of the extended power uprate modifications, applicable tests that affect the feedwater system, including integrated plant response, will be conducted following plant startup. Expected system response utilizing currently evaluated steady state/transient performance criteria will be implemented and tested prior to plant startup. Post startup plant data will be collected allowing any necessary adjustments to simulator model performance.

Response to NRC Question 46.c

The Process Computer System was evaluated for impacts due to extended power uprate. Proposed design changes for extended power uprate were reviewed against the analog input lists to verify process computer instrument range scaling. The process computer digital and analog inputs require no modification due to extended power uprate. Constants input into the process computer for feedwater flow, rated thermal power, nominal reactor pressure, and rated electrical output will be modified with the implementation of extended power uprate to establish a baseline for uprate operation.

Response to NRC Question 46.d

During extended power uprate power ascension testing, plant data will be collected and compared with the simulator performance. Discrepancy reports will be written as necessary to document any additional modifications to the simulator computer programs/models.

MISCELLANEOUS:

NRC QUESTION 47

Based on ELTR1, it is the staff's understanding that the licensing report NEDC-32749P provided the summary and conclusions of the plant-specific engineering and safety evaluations that were conducted in support of the proposed power uprate. Please list all the plant-specific engineering and safety evaluations that were used to generate NEDC-32749P in the following format:

Title	Approval date	Author	Reviewer
EQ	xx/xx/xx	GE or SC	GE or SC

If the engineering reports were both authored and reviewed by General Electric (GE) only, then please indicate how you have assured yourselves on adequacy of plant-specific input assumptions and acceptability of results.

SCS RESPONSE

Table 47-1 is a matrix of the plant specific engineering and safety evaluations performed for extended power uprate. The evaluations were performed and verified prior to the licensing report submittal. The plant specific technical review and approval of all engineering and safety evaluation results were performed by a team from Southern Company, Bechtel Power and General Electric during the week of June 23 to 27, 1997 (represented in the column titled "Review & Approval of Eng. Evaluation"). Though the analysis results were complete at the time of the review, they were not formally documented in a final report. Publishing the final reports documenting the various analyses is in progress

TABLE 47-1

PLANT SPECIFIC ENGINEERING AND SAFETY EVALUATION FOR EXTEND POWER UPRATE

Title of GE Evaluation	Review & Approval Of Eng. Evaluation	Verified Draft Report Date	Author	Reviewer
Reactor Heat Balance	6/23/97 thru 6/27/97	10/24/97	GE	SC & Bechtel
Reactor Power/Flow Map	6/23/97 thru 6/27/97	10/24/97	GE	SC & Bechtel
ECCS Performance Evaluation	6/23/97 thru 6/27/97	1024/97	GE	SC & Bechtel
Containment Evaluations	6/23/97 thru 6/27/97	10/31/97	GE	SC & Bechtel
Transient Analysis	6/23/97 thru 6/27/97	11/7/97	GE	SC & Bechtel
Loss of Feedwater Event and Station Blackout Event	6/23/97 thru 6/27/97	11/7/97	GE	SC & Bechtel
Thermal-Hydraulic Stability Evaluation	6/23/97 thru 6/27/97	10/24/97	GE	SC & Bechtel
Anticipated Transients Without Scram Analysis	6/23/97 thru 6/27/97	11/21/97	GE	SC & Bechtel
Radiological Impact Analysis	6/23/97 thru 6/27/97	12/8/97	GE	SC & Bechtel
Revised Impact on Vessel Fracture Toughness Evaluation	6/23/97 thru 6/27/97	11/7/97	GE	SC & Bechtel

TABLE 47-1 (Cont.)

Title of GE Evaluation	Review & Approval Of Eng. Evaluation	Verified Draft Report Date	Author	Reviewer
Plant Hatch Unit 1 Reactor Pressure Vessel Stress Report Reconciliation	6/23/97 thru 6/27/97	12/8/97	GE	SC & Bechtel
Plant Hatch 2 Reactor Pressure Vessel Stress Report Reconciliation	6/23/97 thru 6/27/97	12/8/97	GE	SC & Bechtel
Reactor Internals Evaluation	6/23/97 thru 6/27/97	10/31/97	GE	SC & Bechtel
Plant Piping System and Supports Evaluation	6/23/97 thru 6/27/97	12/8/97	GE	SC & Bechtel
Nuclear Steam Supply System and Associated Systems Evaluation	6/23/97 thru 6/27/97	12/8/97	GE	SC & Bechtel
Reactor Core and Fuel Performance Evaluation	6/23/97 thru 6/27/97	11/7/98	GE	SC & Bechtel
Instrumentation & Control Evaluation	6/23/97 thru 6/27/97	10/24/97	GE	SC & Bechtel
Appendix R Fire Events Evaluation	6/23/97 thru 6/27/97	11/21/97	GE	SC & Bechtel

REACTOR SYSTEMS:

NRC QUESTION 48

The submittal stated that Unit 2, Cycle 14 (Reload 13) was used as the reference core for the extended power thermal limits evaluation and that the cycle-specific safety limit minimum critical power ratio safety limit (SLMCPR) will be evaluated for each reload as required by the NRC-approved topical report GESTAR. The submittal also stated that according to the current review, the thermal margins can be maintained. Several inconsistent statements in the submittal and related questions are presented below.

(a) The safety analysis report (SAR) stated that the SLMCPR is provided in Table 9-1 for the extended power uprate. However, Enclosure 1 states that the SLMCPR was assumed. Clarify how the SLMCPR was established. If the SLMCPR was evaluated using reference Cycle 14, Unit 2 core, explain how the reference core was selected to bound the proposed 13% power uprate core. Discuss the core parameters used to conclude that Reload 13 core (Cycle 14) will provide bounding results.

(b) Enclosure 6 of the submittal states that revised fuel loading pattern, larger batch sizes and potentially new fuel design may be used to provide additional operating flexibility and maintain fuel cycle length. Consequently the current design basis safety analysis report is based on a cycle reference core that may be atypical, and the proposed 13% core has not yet been yet defined. Explain how the current thermal limit evaluation and transient analysis serves as plant specific safety evaluation for the 13% power uprate. The NRC is not obligated to review the reload analysis and the current evaluation would have to serve as design basis analysis equivalent to the objectives of the Final Safety Analysis Report (FSAR) in the initial core evaluation.

(c) For the transient delta CPR, indicate if the experimental data range used to develop the critical heat flux correlation includes the anticipated transient peak pressure?

(d) Section 2.2.1 states that the required operating limit MCPR safety limit is not expected to change significantly based on Table 3-1 of Reference 2. However, the referenced Table 3-1 in the generic extended power uprate topical report does not address the thermal limits. Clarify this referencing discrepancy.

(e) The submittal also states that Section 5.7.2 of the generic topical report covers the MAPLHGR and MLHGR. However, the sections referred to are not in the extended power uprate topical report. Again, clarify if the stretch power or the extended power uprate generic reports are being referenced.

<u>SNC RESPONSE</u>

Response to NRC Question 48.a

The purpose of the SLMCPR evaluation was not to provide bounding results, but to determine the effects of extended power uprate on a representative core. The SLMCPR used in the extended power uprate analyses was assumed to be the same as the safety limit documented for Plant Hatch-2 Cycle 14. As part of the upcoming Cycle 15 reload analysis, a cycle specific safety limit for the core at extended power uprate (8% above current and 13% above original licensed power) was calculated and verified. The results show that the SLMCPR assumed in the uprate analyses is conservative by 0.01. Therefore, it is concluded that the results of the power uprate analyses, while not necessarily bounding, are representative of what can be expected from the cycle specific reload licensing calculations which will be performed on the uprated core.

Response to NRC Question 48.b

The purpose of the transient analysis in section 9.1 is to demonstrate the impact of extended power uprate on system response and thermal margin. For this purpose the same core loading was used to obtain a direct comparison between Plant Hatch 2 Cycle 14 reload licensing analysis and the extended power uprate operating conditions, i.e., to calculate the effect of the increased power level only.

The results showed that the impact on thermal margin is small (i.e. 0.01). Since this impact is small, the results can also be used to confirm that the turbine trip without bypass is the limiting transient event. At the time of the actual reload licensing of the uprated core (8% above current, 13% above original), the limiting transients will be re-calculated and operating limits determined for plant operation such that the SLMCPR for the core is protected.

Response to NRC Question 48.c

The critical heat flux (CHF) correlation application range does not include the anticipated transient peak pressure. However, all pressurization transients for which ΔCPR is calculated fall within the CHF application range.

Response to NRC Question 48.d

The reference number is incorrect in Section 2.2.1 of Topical Report NEDC-32749P. The reference should be to Table 3-1 in Reference 1, the generic guideline for extended power uprate (NEDC-32424P), not Reference 2. A change to page 2-2 (section 2.2.1) of the licensing submittal is provided in Enclosure 2.

Response to NRC Question 48.e

Both reference numbers are incorrect in Section 2.2.1 of Topical Report NEDC-32749P. The references should be to Reference 1, the generic guideline for extended power uprate (NEDC-32424P), not Reference 2. Section 5.7.2 discusses the impact on thermal limits including MAPLHGR and MLHGR and the options that can be used to minimize the impact. The fuel thermal-mechanical limits for Plant Hatch at extended power uprate conditions will be confirmed to be within the fuel design criteria as part of the core reload analysis.

The references in NEDC-32749P are incorrect in the following sections:

1. In Section 2.2.2 the reference should be to Reference 1, not Reference 2. The change to page 2-2 of the licensing submittal is provided in Enclosure 2.

2. The reference in the first paragraph in Section 2.3.1 should refer to Reference 1, not Reference 2. The change to page 2-3 of the licensing submittal is provided in Enclosure 2.

3. The reference in the first paragraph in Section 9.3.1 should refer to Reference 2, not Reference 1. The change to page 7-2 of the licensing submittal is provided in Enclosure 2.

NRC QUESTION 49

Provide a list of the computer codes used to analyze the normal low frequency operational transients, the abnormal operational occurrences and the design-basis accidents. Indicate the codes used to provide the initiating inputs for the Chapter 15 DBA analysis.

SNC RESPONSE

Infrequent transients are not specifically analyzed for the Plant Hatch extended power uprate as they are not Chapter 15 events. Events such turbine generator trip with bypass failure, and loss of feedwater flow with failure of HPCI are conservatively included with anticipated operational occurrences (AOOs) events. The following codes were used for the AOO analyses:

1.) ODYN - system response for pressurization events

2.) PANACEA - initial conditions, Rod Withdrawal Error, and slow cold water transients

3.) TASC and ISCOR - fuel thermal margin evaluations for ODYN cases

4.) SAFER - loss of FW flow

The following codes were used for design basis accidents:

1.) LAMB,

2.) SCAT

3.) SAFER - LOCA-ECCS

The following codes were used for the radiological evaluation:

1.) CONAC (FHA, CRDA, MSLBA -- offsite),

2.) MSIV (LOCA -- offsite & control room)

Codes which supplied initiating thermal-hydraulic input conditions are:

1.) ISCOR - heat balance and core steady-state thermal hydraulics

2.) PANACEA - initial core conditions

NRC QUESTION 50

The peak pressure for the overpressure transient was indicated to be 1325 psi. Expand on the following issues regarding the overpressure analysis.

(a) In analyzing the overpressure conditions, were the pressure drops across the inlet main steam piping to the safety relief valve (SRV) and the exit piping considered? High pressure drop in the inlet or outlet piping will result in lower steam flow and may increase the peak pressure after the initiation of the SRVs. Considering that the 110% design pressure margin is only 50 psid, the effect of the reduced steam flow should be included in the analysis. The extended power uprate topical report discusses this issue in the overprotection guidelines.

(b) The SRVs have design pressure of 1250 or 1375 psig depending on the plant. Specify the SRV design pressure for Plant Hatch Units 1 and 2.

(c) The submittal did not address what the vessel bottom head pressure would be during overpressure transient. Indicate if the pressure at the bottom vessel head is (1325 psig + 40 psid) 1365 psig? Also, specify what the bottom head pressure would be during over pressure transient if you take into account the SRV's inlet/exhaust piping pressure losses.

(d) The FSAR states that the Control Rod Drive (CRD) mechanism was acceptable for bottom head pressure of 1250 psig and Section 2.5.1 of the submittal (page 2.4) states that a pressure of 1250 psig is the limiting pressure for the bottom head pressure. Did you consider the effect of peak pressure on the CRD mechanism?

SNC RESPONSE

Response to NRC Question 50.a

The GE ODYN transient analysis model considers the dynamic pressure loss between the main steamline and the SRVs. This model includes a flow-squared loss coefficient for the pipe from the main steamline to the inlet section of the valve. The pressure drop in the SRV outlet piping is irrelevant since the flow through SRVs is choked at the valve throat, and the down stream pressure loss in the exhaust piping does not affect the flow rate.

Response to NRC Question 50.b

Plant Hatch Unit 1 and Unit 2 SRVs are Target Rock Model 7567F safety/relief valves with a design pressure of 1250 psi in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1968 edition and addenda through Summer 1970.

Response to NRC Question 50.c

The peak vessel pressure calculated by the GE transient code, ODYN, (discussed in Section 3.2 of Topical Report NEDC-32749P for Plant Hatch extended power uprate) represents the vessel bottom pressure during the transient. The overpressure transient analysis performed for the extended power uprate submittal results in a peak vessel bottom pressure of 1347 psig which meets the 1375 psig vessel pressure criteria. The overpressure analysis is recalculated on a plant-specific basis for each reload to ensure continued compliance with the 1375 psig limit. Refer to the response to NRC Question 50a for the consideration of SRV's inlet/exhaust piping pressure losses.

Response to NRC Question 50.d

The components of the CRD mechanism, which form part of the primary pressure boundary, have been designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The design pressure of the reactor coolant pressure boundary remains at 1250 psig. The ASME code allowable peak pressure is 1375 psig (110% of design value), which is the acceptance limit for pressurization events. At extended power uprate conditions, a higher peak RPV pressure results, but remains below the 1375 psig ASME limit. Therefore, the effect of the peak RPV pressure on the CRD mechanism was considered.

NRC QUESTION 51

The submittal addressed the ability of the recirculation pumps to provide the 3% increase in core flow, but did not address whether the 105% core flow can be achieved taking into account the increase in void fraction due to the higher power as well as the expected higher void fraction in the periphery zone if the high powered bundles are placed in the periphery zone as proposed.

SCS RESPONSE

At the current rated recirculation pump speed, the core flow rate can be expected to decrease by about 1%. Plant Hatch can maintain the same core flow rate with the calculated higher pressure loss by increasing the speed of the recirculation pumps by approximately 15 RPM, if the operator chooses to do so. At 105% core flow, the calculated pump, pump motor, and MG Set power requirements remain well within the individual equipment ratings. The MG Set drive motor amperage is calculated to be within its rated full load amperage limit at the design m-ratio of the jet pumps at 105% core flow. The recent cleaning of the jet pump throats at both units reduced the load on the drive motors and is expected to provide ample margin to the full load amperage limit of the drive motor.

NRC QUESTION 52

The submittal included proposed changes to the Technical Specifications. However, the submittal did not provide any matrix or plan indicating which sections of the FSAR will be superseded by current extended power uprate reanalysis. Provide a list or matrix that identifies which subsections of the FSAR will be superseded and identify the corresponding sections of the current submittal. The actual updating of the FSAR will be governed by the current regulations; however, a new license at the uprated power will be issued and the effected FSAR subsection should be documented.

SCS RESPONSE

Matrices of the Unit 1 and Unit 2 FSAR changes are provided in Tables 52-1 and 52-2 respectively. These tables are the result of concurrent design and licensing reviews for potential FSAR changes as a result of extended power uprate. These tables do not represent a final review or approval of the affected FSAR sections. Therefore, the information in these preliminary tables does not represent Southern Nuclear's submittal of extended power uprate FSAR changes. As stated in NRC Question 52, the actual update of the FSAR will be governed by the current regulations.

TABLE 52-1

UNIT 1 FSAR REVISIONS MATRIX FOR EXTENDED POWER UPRATE
(PRELIMINARY)

SECTION	TITLE	DESCRIPTION OF REVISION
§ 1.1	PROJECT IDENTIFICATION	Change core thermal power level
§1.1.1.1	Applicant Licensee	Change current licensed power level in Paragraph H
Table 1.11-1	Topical Reports Submitted to the NRC In Support Of HNP-1 Design/Operation	Add 56 "Plant Hatch Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDC-32720P, dated March, 1997 Add 57 "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, dated July 1997
§2.3.5	Accident Diffusion Estimate for Main Control Room and Technical Support Center	Add this section and include in the Table of Contents
Page 2.3-8	REFERENCES	Add Reference 9 Letter HL-5333, dated April 17, 1997, Southern Nuclear Company to NRC, "Revised Post-LOCA Doses", Docket Nos. 50-321 and 50-366 Add Reference 10 NUREG/CR-6331, "Atmospheric Relative Concentration in Building Wakes," May 1997
Table 2.3-7	Values x/Q at the control room air intake and TSC intake	Add this section and include in the Table of Contents
§3.3.5.2.1	Accident Definition	Revise to identify analyses performed for extended power uprate
Page 3.3-24	REFERENCES	Change Reference 7 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
Table 3.3-1	MAXIMUM DIFFERENTIAL PRESSURE ACROSS REACTOR ASSEMBLY INTERNALS	Replace results with the analysis results for extended power uprate
§3.8.4	SAFETY EVALUATION	Revise to identify analyses performed for extended power uprate

TABLE 52-1 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Page 3.8-10	REFERENCES	Change Reference 3 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
§4.2.4.1.1.1	Fracture Toughness	Revise last paragraph to show applicability of NEDO-32205 for extended power uprate or refer to new analyses
§4.2.4.1.1.1.2.1	Irradiation Effects on Core Beltline	Revise to show evaluation at extended power uprate
§4.2.5	SAFETY EVALUATION	Revise to show evaluation at extended power uprate
Table 4.3-1	REACTOR RECIRCULATION SYSTEM DESIGN DATA	Revise for extended power uprate.
§4.4.6	SAFETY EVALUATION	Revise to show SRV performance evaluation at extended power uprate
§4.4.7	OPERATION EVALUATION	Revise to show SRV operation evaluation at extended power uprate
Page 4.4.11	REFERENCES	Change Reference 2 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
Page 4.7-8	REFERENCES	Change Reference 1 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
Figure 4.7-1	RCIC SYSTEM PIPING & INSTRUMENTATION DIAGRAM	Revise to show condition for extended power uprate
Figure 4.7-3	RCIC SYSTEM PROCESS DIAGRAM	Revise to show condition for extended power uprate
Page 4.11-5	REFERENCES	Change Reference 3 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
§5.2.4.6.2	Analyses of Pressure Transients Due to Severance of a Recirculation Line	Revise Paragraph A to show extended power uprate conditions Revise Paragraph C to show extended power uprate conditions
§5.2.4.9	Control of Combustible Gas Concentrations in Containment Following a LOCA	Add a statement that the impact of extended power uprate is shown in §5.2.4.10
§5.2.4.10	Combustible Gas Control Analysis for Extended Power Uprate (2763 MWt)	Add this section and include in the Table of Contents

TABLE 52-1 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Page 5.2-69	REFERENCES	Change Reference 13 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
Table 5.2-1	PRIMARY CONTAINMENT SYSTEM PRINCIPAL DESIGN PARAMETERS AND CHARACTERISTICS	Revise to show extended power uprate conditions
Figure 5.2-21 (SHEET 4 of 5)	CONTAINMENT PRESSURE RESPONSE TO THE DESIGN BASIS LOCA	Replace with results for extended power uprate
Figure 5.2-21 (SHEET 5 of 5)	CONTAINMENT TEMPERATURE RESPONSE TO THE DESIGN BASIS LOCA	Replace with results for extended power uprate
Figure 6.4-1	HIGH PRESSURE COOLANT INJECTION SYSTEM PROCESS DIAGRAM	Revise for extended power uprate conditions
Figure 6.4-2	CORE SPRAY SYSTEM PROCESS DIAGRAM	Revise for extended power uprate conditions
§6.5.3	INTEGRATED OPERATION OF THE CSCS	Revise to show evaluation for extended power uprate
§6.5.3.3	Single-Failure Considerations	Revise to be consistent with the power uprate LOCA analysis
Page 6.5-28	REFERENCES	Change Reference 10 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
Table 6.5-1	ECCS PERFORMANCE RESULTS	Replace with results at extended power uprate conditions
Table 6.5-2	SINGLE-FAILURE ASSESSMENT	Revise systems remaining to be consistent with the power uprate LOCA analysis.
Table 6.5-3	OPERATIONAL PARAMETERS FOR LOCA ANALYSIS	Replace with parameters for extended power uprate
Table 6.5-4	PLANT ECCS PARAMETERS	Replace with parameters for extended power uprate
Table 6.5-5	SUMMARY OF LOCA ANALYSIS RESULTS	Replace with results for extended power uprate

TABLE 52-1 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
§7.2.3.6	REACTOR PROTECTION SYSTEM - Scram Functions and Bases for Trip Settings	Revise Turbine Stop Valve Closure first stage pressure cutoff setpoint in Paragraph K
§7.2.3.9	REACTOR PROTECTION SYSTEM - Instrumentation	Revise Turbine Stop Valve Closure first stage pressure cutoff setpoint in Paragraph J
Table 7.2-1	REACTOR PROTECTION SYSTEM SCRAM SETTINGS	Revise APRM Thermai Power Trip Setting
Table 7.3-2	PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM ISOLATION SETPOINTS	Revise main steam line high flow trip setting
§7.11.2	POWER GENERATION DESIGN BASES	Revise steam bypass capacity for extended power uprate conditions
§7.11.4	POWER GENERATION EVALUATION	Revise steam bypass capacity for extended power uprate conditions
Figure 7.16-2	WORST-CASE ACCIDENT PROFILE FOR EQUIPMENT LOCATED IN CONTAINMENT	Revise for extended power uprate conditions
§7.22.4	RWM OPERATION	Revise instrument setting power level
§8.4	Standby AC Power	Revise LOCA analysis response times for Core Spray and LPCI
Table 8.4-4	SEQUENCE FOR AUTOMATICALLY CONNECTING EMERGENCY AC LOADS ON LOCA	Revise LOCA analysis response times for Core Spray and LPCI
§10.4.3	FUEL POOL COOLING AND CLEANUP SYSTEM - DESCRIPTION	Revise last paragraph to show evaluations at extended power uprate conditions
§10.4.3.1	LOCAL FUEL BUNDLE THERMAL HYDRAULICS	Revise to show evaluations at extended power uprate conditions
§10.5.3	REACTOR BUILDING CLOSED COOLING WATER SYSTEM - DESCRIPTION	Revise to show evaluations at extended power uprate conditions

TABLE 52-1 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
§11.1	SUMMARY DESCRIPTION	Revise condenser capacity as percent of design flow for extended power uprate
§11.3.2	POWER GENERATION DESIGN BASES	Revise condenser capacity as percent of design flow for extended power uprate
§11.5.2	POWER GENERATION DESIGN BASES	Revise turbine bypass capacity as percent of design flow for extended power uprate
§12.7.3	DESCRIPTION	Revise to show evaluations at extended power uprate conditions
§14	PLANT SAFETY ANALYSIS	Revise to show analyses at extended power uprate conditions
Table A.2-2	CODES FOR COMPONENTS AND SYSTEMS WHICH COMPRISE THE REACTOR COOLANT PRESSURE BOUNDARY (SHEET 3 OF 5)	Revise recirculation piping to show evaluation at extended power uprate
§A.3.1	PIPING DESIGN	Revise to show analyses at extended power uprate
§A.3.4	SUPPORTS	Revise to show analyses at extended power uprate
Page A.3-7	REFERENCES	Change Reference 1 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July 1997 Delete Reference 2
Table C.2-4	BUCKLING STABILITY LIMIT	Revise to be consistent with the extended power uprate analyses
Table C.3-1	REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT (Sheet 5 of 26)	Revise calculated loads for extended power uprate conditions
§KA.2	PLANT UNIQUE ANALYSIS REPORT	In second paragraph; change 2558 MWt to 2763 MWt
Page KA-4	REFERENCES	Change Reference 7 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July 1997.
§M.4.1	OPERATING CONDITIONS	Revise to show evaluation at extended power uprate conditions Change the cited report to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July 1997.
Table N.4-1	HIGH-ENERGY LINES	Revise the service temperatures and pressures for extended power uprate conditions

TABLE 52-1 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
§N.5.1.3	Analysis of Shutdown Capability	Revise footnote (a) to show thermal power of 2763 MWt
§N.5.2	FEEDWATER LINE BREAK	Revise feedwater temperature for extended power uprate conditions

TABLE 52-2

UNIT 2 FSAR REVISIONS MATRIX FOR EXTENDED POWER UPRATE
(PRELIMINARY)

SECTION	TITLE	DESCRIPTION
§ 1.1.1	LICENSE REQUESTED	Change core thermal power level
§1.1.6	POWER OUTPUT	Change operating thermal power level, gross electrical output and net electrical output
§1.2.3	NUCLEAR SYSTEMS (Units 1 and 2)	Revise to show power of 2763 MWt
§1.2.4.2	TURBINE BYPASS SYSTEM	Revise bypass capacity percentage
Table 1.2-2	Original and Uprated Reactor Operating Conditions	Revise to show extended power uprate conditions
Figure 1.2-3	Reactor System Heat Balance (Unit 2)	Revise to show extended power uprate conditions
Figure 1.2-4	Reactor System Heat Balance (Unit 1)	Revise to show extended power uprate conditions
Figure 1.2-7	Rated Turbine Generator Heat Balance (Unit 2)	Revise to show extended power uprate conditions
Figure 1.2-8	Valve Wide Open Heat Balance (Unit 2)	Revise to show extended power uprate conditions
Figure 1.2-9	Rated Turbine Generator Heat Balance (Unit 1)	Revise to show extended power uprate conditions
§1.3.1	COMPARISON WITH SIMILAR FACILITY DESIGNS	Revise to show extended power uprate power level
Table 1.3-2	Significant Changes (HNP-2 Only)	Add "Significant Changes" due to extended power uprate
Table 1.6-1	General Electric Company Reports	Add "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997
§2.3.6	Accident Diffusion Estimate for Main Control Room and Technical Support Center	Add this section and include in the Table of Contents
Table 2.3-15	Values x/Q at the control room air intake and TSC intake	Add this Table and include in the Table of Contents

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
§3.8B.5	POWER UPRATE OPERATION	Change 2558 MWt to 2763 MWt. Change 45.5 psig to 46.9 psig
Page 3.8B-4	REFERENCES	Change Reference 8 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
Page 3.9-61	REFERENCES	Change Reference 7 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
Table 3.9-2	MAIN STEAM LINE PIPING SYSTEM (CLASS 1 PIPE)	Change footnote (a) to refer to evaluations at extended power uprate
Table 3.9-6	RECIRCULATION PIPING SYSTEM (CLASS 1 PIPE)	Change footnote (a) to refer to evaluations at extended power uprate
Table 3.11-4	RADIATION ENVIRONMENTAL CONDITIONS	Revise to show extended power uprate conditions
Figure 3.11-2	WORST-CASE ACCIDENT PROFILE FOR EQUIPMENT LOCATED IN CONTAINMENT	Revise to show extended power uprate conditions
§4.2.2.3.2.2	Effects of Initial Reactor Power and Core Flow	Replace the fourth paragraph with a discussion of the analysis performed at 2763 MWt
§4.2.3.4.3	Safety Evaluation	Replace "... SLCS at 2558 MWt ..." with " ... SLCS at 2763 MWt ..."
Page 4.2-104	REFERENCES	Change Reference 48 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
Table 4.2-20	Maximum Differential Pressure Across the Reactor Assembly Internals	Replace with the analysis results for extended power uprate
§4.4.4.4	Core Thermal Response	Replace "2558 MWt" with "2763 MWt"
Figure 5.1-1	REACTOR COOLANT SYSTEM FLOW DIAGRAM	Replace tabulated temperatures and enthalpies with the appropriate values for operation at 2763 MWt
§5.2.2	Overpressure Protection	Revise assumptions to state "Ten safety relief valves open at the nominal setpoint in Table 5.2-4 plus at least 3% drift tolerance"
§5.2.4.2	Compliance With 10CFR50, Appendix G, Fracture Toughness Requirements	Revise to show evaluation at extended power uprate

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
§5.2.4.2.2.1	Effects of Irradiation	Revise to show evaluation at extended power uprate
Page 5.2-63	REFERENCES	Change Reference 18 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997. Add Reference 20 "Hatch Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDC-32720P, March, 1997.
Table 5.2-7	BELTLINE ART VALUES FOR HATCH 2 BASED ON UPRATED POWER	Revise to show evaluations at extended power uprate (Section 5.2.4.2.2.1)
Figure 5.2-4	ADJUSTED REFERENCE TEMPERATURE FOR LIMITING BELTLINE MATERIALS	Revise to show evaluations at extended power uprate (Section 5.2.4.2.2.1)
§5.4.6.4	Safety Evaluation	Change operating pressure from ~1035 psig to ~1038 psig Revise equations for cumulative fatigue usage factors
§5.5.7.4	Safety Evaluation	In last paragraph, change peak suppression pool temperature from 202 °F to 208 °F.
Page 5.5-59	REFERENCES	Change Reference 6 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2", NEDC-32749P, July, 1997.
Table 5.5-1	REACTOR RECIRCULATION SYSTEM DESIGN CHARACTERISTICS	Revise for extended power uprate conditions
Figure 5.5-8	REACTOR CORE ISOLATION COOLING SYSTEM PROCESS FLOW DIAGRAM	Revise to show condition for 2763 MWt
§6.2.1.4.2	Recirculation Line Break - Short-Term Response	Revise first paragraph to show analyses at extended power uprate Revise Section C to show analysis results at extended power uprate
§6.2.1.4.5	Small Breaks	Revise Section D to show analysis at extended power uprate
§6.2.5	COMBUSTIBLE GAS CONTROL SYSTEM	Revise to be consistent with modified system initiating conditions
Page 6.2-117	REFERENCES	Change Reference 27 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2", NEDC-32749P, July, 1997.

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Table 6.2-1	PRIMARY SYSTEM AND CONTAINMENT DESIGN DATA AND CRITERIA	Change Initial Conditions on Sheet 1 of 5 from 102% of 2558 MWt to 102% of 2763 MWt Add analysis conditions for extended power uprate
Table 6.2-8	LOSS-OF-COOLANT ACCIDENT PRIMARY CONTAINMENT RESPONSE SUMMARY	Revise to show analysis results for extended power uprate
Table 6.2-15	COMBUSTIBLE GAS CONTROL SYSTEM CHRONOLOGY OF EVENTS FOLLOWING LOCA	Revise to be consistent with revised Section 6.2.5
Figure 6.2-27	LONG TERM RECIRCULATION BREAK CALCULATED CONTAINMENT PRESSURE RESPONSE	Replace with results for extended power uprate
Figure 6.2-28	LONG TERM RECIRCULATION BREAK CALCULATED DRYWELL TEMPERATURE RESPONSE	Replace with results for extended power uprate
Figure 6.2-29	LONG TERM RECIRCULATION BREAK CALCULATED SUPPRESSION POOL TEMPERATURE RESPONSE	Replace with results for extended power uprate
Figure 6.2-45	TIME DEPENDENT H2 CONCENTRATION WITH CGCS (LONG TERM)	Replace with results for extended power uprate
Figure 6.2-46	TIME DEPENDENT CONTAINMENT HYDROGEN CONCENTRATION WITHOUT COMBUSTIBLE GAS CONTROL	Replace with results for extended power uprate
Figure 6.2-47	TIME DEPENDENT H2 CONCENTRATION WITH CGCS (SHORT TERM)	Replace with results for extended power uprate

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Figure 6.2-48	CORE AND SUPPRESSION POOL RADIOLYTIC GENERATION	Replace with results for extended power uprate
Figure 6.2-49	CONTAINMENT H2 CONCENTRATION VS TIME 0-PERCENT RELATIVE HUMIDITY	-- DELETE --
Figure 6.2-50	CONTAINMENT H2 CONCENTRATION VS TIME 0-PERCENT RELATIVE HUMIDITY	-- DELETE --
Figure 6.2-65	CONTAINMENT PRESSURE RESPONSE TO THE DESIGN BASIS LOCA - OPERATION AT 102% OF RATED POWER	Replace with results for extended power uprate
Figure 6.2-66	CONTAINMENT TEMPERATURE RESPONSE TO THE DESIGN BASIS LOCA - OPERATION AT 102% OF RATED POWER	Replace with results for extended power uprate
§6.3.2.14	Net Positive Suction Head	Replace with conditions and calculated results for extended power uprate
§6.3.3	PERFORMANCE EVALUATION	Revise to show analysis at extended power uprate
Page 6.3-43	REFERENCES	Change Reference 6 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2", NEDC-32749P, July, 1997.
Table 6.3-1	ECCS PERFORMANCE RESULTS	Replace results with results at extended power uprate conditions
Table 6.3-2	SINGLE-FAILURE ASSESSMENT	Revise systems remaining to be consistent with the power uprate LOCA analysis.
Table 6.3-6	OPERATIONAL PARAMETERS FOR LOCA ANALYSIS	Replace with parameters for extended power uprate
Table 6.3-7	PLANT ECCS PARAMETERS	Replace with parameters for extended power uprate
Table 6.3-8	SUMMARY OF LOCA ANALYSIS RESULTS	Replace with results for extended power uprate
Figure 6.3-1	HPCI SYSTEM P&ID	Revise for extended power uprate conditions

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Figure 6.3-2	HPCI SYSTEM PROCESS FLOW DIAGRAM	Revise for extended power uprate conditions
§6A.4	SUBCOMPARTMENT PRESSURIZATION ANALYSIS WITHIN SACRIFICAL SHIELD AND THE DRYWELL AREA	Remove last sentence referring to analysis at 2558 MWt
§6A.4.2	Effect of Power Uprate	Add new section for evaluation of subcompartment pressurization with extended power uprate conditions (and add to Table of Contents)
§6A.4.3	Drywell Area	Revise for extended power uprate
Page 6A-100	REFERENCES	Add reference to AP Mass & Energy and Drywell & Torus Temperature Profiles
Table 6A-4	BLOWDOWN MASS, ENERGY RELEASE RATE, AND BREAK AREAS FOR LINES IN DRYWELL	Revise for extended power uprate
Table 6A-5(a)	BLOWDOWN MASS, ENERGY RELEASE RATE, AND BREAK AREAS FOR RECIRCULATION OUTLET AND INLET, FEEDWATER AND HEAD SPRAY LINES	Revise for extended power uprate
Table 6A-5(b)	BLOWDOWN MASS AND ENERGY RELEASE RATE FOR AP LOADS EVALUATION FOR HATCH 2 - EXTENDED POWER UPRATE	New Table for extended power uprate
Figure 6A-50	MASS FLUX VS TIME: 10-in CORE SPRAY	Revise for extended power uprate
Figure 6A-51	MASS FLUX VS TIME: RHR SUCTION	Revise for extended power uprate
Figure 6A-52	MASS FLUX VS TIME: RHR DISCHARGE	Revise for extended power uprate

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Figure 6A-53	MASS FLUX VS TIME: MSLB UPPER, LOWER DRYWELL	Revise for extended power uprate
Figure 6A-54	MASS FLUX VS TIME: MSCD	Revise for extended power uprate
Figure 6A-55	MASS FLUX VS TIME: RCIC STEAM	Revise for extended power uprate
Figure 6A-56	MASS FLUX VS TIME: RWCU	Revise for extended power uprate
Figure 6A-57	MASS FLUX VS TIME: HPCI STEAM	Revise for extended power uprate
§7.2.2	REACTOR PROTECTION SYSTEM - Description	Revise Turbine Stop Valve Closure first stage pressure cutoff setpoint
§7.2.2.4	BYPASSES AND INTERLOCKS	Revise turbine control valve fast closure and turbine stop valve closure bypass value
§7.3.1.3.1	Conformance to General Functional Requirements	Revise response times for Core Spray and for LPCI
Table 7.3-9	PRIMARY CONTAINMENT AND RPV ISOLATION CONTROL SYSTEM ISOLATION SETPOINTS	Revise main steam line high flow trip setting
Table 7.6-7	AVERAGE POWER RANGE MONITOR SYSTEM TRIPS	Revise APRM Upscale Nominal Setpoints
Table 7.6-8	ROD BLOCK MONITOR SYSTEM TRIPS	Revise RBM Upscale and Downscale Setpoints
§8.3.1.1.3	Standby AC Power	Revise LOCA analysis response times for Core Spray and LPCI
Table 8.3-3	SEQUENCE FOR AUTOMATICALLY CONNECTING EMERGENCY AC LOAD ON LOCA/LOSP	Revise LOCA analysis response times for Core Spray and LPCI
§8.4.2	SBO COPING EVALUATION	Revise first paragraph to refer to evaluations at 2763 MWt

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Page 8.4-9	REFERENCES	Change Reference 3 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
Page 8.4-10	BIBLIOGRAPHY	Add Reference 7 for extended power uprate
§9.1.3.2.2	FUEL POOL COOLING AND CLEANUP SYSTEM - Heat Removal Capacity	Revise last paragraph of the subsection to show evaluations at extended power uprate
§9.1.3.4	FUEL POOL COOLING AND CLEANUP SYSTEM - Safety Evaluation	Revise to show evaluations at extended power uprate
Table 9.2-2	PSW SYSTEM COMPONENT REQUIREMENTS	Revise to show normal operating condition flow rates for extended power uprate
§9.2.5.4	ULTIMATE HEAT SINK - Safety Evaluation	Revise Item A to discuss analyses at 2763 MWt
§10.1	SUMMARY DESCRIPTION	Revise steam bypass capacity as percent of design flow for extended power uprate
§10.2.1	DESIGN BASES AND OBJECTIVE	Revise steam bypass capacity as percent of design flow for extended power uprate
§10.2.3.2	Turbine Wheel Cracks and Missile Prevention	Revise to discuss evaluation at extended power uprate conditions
Page 10.2-14	REFERENCES	Change Reference 6 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997.
Table 10.2-1	TURBINE GENERATOR DESIGN CONDITIONS	Revise to show conditions at extended power uprate
§10.4.1.1	MAIN CONDENSER - Design Bases	Revise paragraph B to show bypass flow fraction at extended power uprate
§10.4.1.1	TURBINE BYPASS SYSTEM - Design Bases	Revise first paragraph to show bypass flow fraction at extended power uprate
§11.1.3	TRITIUM	Revise first paragraph on page 11.1-11 to show evaluation for extended power uprate
§11.1.6.1	Main Steam and Condensate Systems	Revise to show evaluation for extended power uprate
Table 11.1-4	COOLANT ACTIVATION PRODUCTS IN REACTOR WATER AND STEAM	Revise to show activation for extended power uprate

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Table 11.1-6	MAJOR RADIOISOTOPES IN STEAM AT REACTOR NOZZLE	Revise to show activation for extended power uprate
Table 11.1-7	MAJOR ISOTOPES IN MAIN CONDENSER	Revise to show activation for extended power uprate
§11.2.4.1.1	Estimate of Radionuclides Released	Revise last paragraph to show evaluation at extended power uprate
Table 11.2-3	GALE INPUT DATA (BWR GALE Code Input Data File)	Replace with table of values for extended power uprate
Table 11.2-4	EXPECTED ANNUAL RELEASES (BWR GALE Code Output Data File)	Replace with table of values for extended power uprate
Table 11.2-5	MAXIMUM INDIVIDUAL DOSES FROM LIQUID EFFLUENTS (LADTAP OUTPUT FILE)	Replace with table of values for extended power uprate
§11.3.4.1.2	Estimate of Radionuclides Expected to be Released	Revise to show evaluations for extended power uprate
§11.3.4.1.5	Estimated Doses	Revise to show evaluations for extended power uprate
Table 11.3-6	GALE CODE INPUT DATA (Units 1 and 2)	Replace with table of values for extended power uprate
Table 11.3-9	MAXIMUM DOSES FROM GASEOUS EFFLUENTS (Unit 1 and Unit 2)	Replace with table of values for extended power uprate
Table 12.2-1	PARAMETERS FOR CALCULATING AIRBORNE RADIOACTIVITY CONCENTRATIONS	Replace with table of values for extended power uprate
Table 12.2-2	PEAK AIRBORNE CONCENTRATIONS IN THE DIFFERENT REGIONS OF THE PLANT	Replace with table of values for extended power uprate
§12.4.3	SITE BOUNDARY AND PLANT AREA DOSE RATES	Show evaluations at extended power uprate

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Table 12.4-1	ESTIMATED ANNUAL GAMMA DOSE TO PLANT PERSONNEL AND EXPECTED NUMBER OF MANHOURS OF OCCUPANCY PER YEAR	Show evaluations at extended power uprate
Table 12.4-3	ESTIMATED ANNUAL INHALATION DOSES TO PLANT PERSONNEL DUE TO AIRBORNE RADIOACTIVITY	Show evaluations at extended power uprate
§15.1	ACCIDENT ANALYSIS - GENERAL	Revise third paragraph to show licensed power level of 2763 MWt Revise discussion of increase in rated thermal power (RTP) on page 15.1-3 to demonstrate that Unit 2 can operate at an RTP of 2763 MWt
§15.1.1.2	Generator Load Rejection with No Bypass	Modify starting conditions in §15.1.1.2.1 to show power level of 2763 MWt Revise results in §15.1.1.2.4.3 to show results for 2763 MWt
§15.1.1.2	Generator Load Rejection with No Bypass, with Flux Scram and No RPT	Revise the last paragraph to discuss evaluation at 2763 MWt
§15.1.2	TURBINE TRIP	Modify starting conditions in §15.1.2.1.1 to show power level of 2763 MWt
§15.1.7	EXCESS COOLANT INVENTORY	Modify starting conditions in §15.1.7.1.1 to show power level of 2763 MWt
§15.1.8	LOSS OF A FEEDWATER HEATER	Modify starting conditions in §15.1.8.1.1 to show power level of 2763 MWt
§15.1.11	CONTINUOUS CONTROL ROD WITHDRAWAL DURING POWER RANGE OPERATION	Modify last paragraph (Discussion of ARTS statistical limits) to show power level of 2763 MWt
§15.1.25	RECIRCULATION FLOW CONTROL FAILURE WITH INCREASING FLOW	Modify last paragraph (Discussion of applicability of ARTS limits) to show power level of 2763 MWt
§15.1.39	LOSS-OF-COOLANT ACCIDENT (RADIOLOGICAL CONSEQUENCES)	Revise §15.1.39.4.1.2 for reactor building activity to add discussion of MSIV leakage Revise §15.1.39.4.2.2 to refer to Table 15.1-52
Page 15.1-177	REFERENCES	Change Reference 32 to "Extended Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," NEDC-32749P, July, 1997. Add Reference 42: Letter HL-5333, "Revised Post-LOCA Doses," H.L. Sumner, Jr. to NRC, April 17, 1997

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Table 15.1-28	MCR POST-ACCIDENT DOSES TO AN OPERATOR	Revise to show values for analysis at extended power uprate
Table 15.1-28(a)	TECHNICAL SUPPORT CENTER POST-LOCA DOSE TO OCCUPANTS	Revise to show values for analysis at extended power uprate
Table 15.1-29	CONTROL ROD DROP ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES	Revise to show values for analysis at extended power uprate
Table 15.1-36	LOSS-OF-COOLANT ACCIDENT (NRC ANALYSIS) RADIOLOGICAL EFFECTS)	Revise to show values for analysis at extended power uprate
Table 15.1-37	LOSS-OF-COOLANT ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES	Revise to show values for analysis at extended power uprate
Table 15.1-41	MAIN STEAM LINE BREAK ACCIDENT (NRC ANALYSIS) RADIOLOGICAL EFFECTS)	Revise to show values for analysis at extended power uprate
Table 15.1-42	MAIN STEAM LINE BREAK ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES	Revise to show values for analysis at extended power uprate
Table 15.1-47	FUEL HANDLING ACCIDENT (NRC ANALYSIS) RADIOLOGICAL EFFECTS)	Revise to show values for analysis at extended power uprate
Table 15.1-48	FUEL HANDLING ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES	Revise to show values for analysis at extended power uprate

TABLE 52-2 (Cont.)

SECTION	TITLE	DESCRIPTION OF REVISION
Table 15.1-52	LOSS-OF-COOLANT ACCIDENT RADIOLOGICAL EFFECTS (2763+2% MWt)	Add new table for extended power uprate analysis
§15A.4.1.1	High Energy Lines Identified	Revise service temperatures for extended power uprate
§15A.4.1.2	Moderate Energy Lines Identified	Revise footnote (d) for extended power uprate conditions
§15A.5.2	FEEDWATER LINE BREAK	Revise feedwater temperature for extended power uprate conditions
§15A.5.6.4	Sampling Lines	Revise RWC System pressure in paragraph A Revise feedwater temperature and pressure in paragraph C
§15A.A.2	ADDITIONAL HIGH-ENERGY LINES ANALYZED	Revise footnote (a) for applicability to extended power uprate conditions

NRC QUESTION 53

Provide a list of all the codes used to perform the reanalysis and indicate if the particular code was approved for the specific application. Respond to the following requests which pertain to the codes used in the power uprate.

(a) Review the approving SER for each code and state whether your application of the code comply with any limitations, restrictions or conditions specified in the approving SER. Demonstrate that your applications of the computer codes in the reanalysis conforms with all assumptions and restrictions given by the corresponding approving SER.

(b) In addition, review the approving SERs for the extended power uprate generic reports and indicate if you complied with all restrictions stated in the approving SER.

SNC RESPONSE 53

The restrictions and conditions applicable to the GE Nuclear Energy core and fuel design are documented in GESTAR II, NEDE-24011-P-A-13, Revision 13, "General Electric Standard Application for Reactor Fuel," August 1996 and GESTAR II, NEDE-24011-P-A-13-US, Revision 13, "General Electric Standard Application for Reactor Fuel (Supplement for United States)," August 1996. The safety evaluations performed in support of the Plant Hatch extended power uprate used GE's standard analysis codes and application methodologies as referenced in the GESTAR II documentation.

A list of codes used to perform the extended power uprate analyses is provided in Table 53-1. Most of the codes and their applications were previously used on the Plant Hatch original power uprate which was approved by the NRC in the fall of 1995. The few instances where analysis codes and applications have been upgraded from the previous original power uprate evaluation have been addressed in the response to NRC Question 15. The application of these codes to the extended power uprate analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The application of the codes also complies with the SERs for the extended power uprate programs. The safety analyses performed in support of the Plant Hatch extended power uprate licensing amendment submittal used GE's standard analysis codes and application methodologies. The GE analysis codes and methodologies have been developed and approved for application to a wide range of GE BWR plant types and operating conditions. The extended power uprate operating and accident conditions analyzed for Plant Hatch are within the range of analysis experience for GE BWRs and are within the allowed range of code and methodology application.

Table 53-1

Codes Used for Extended Power Uprate Analyses

GE Analysis	Computer Codes/Standard
Reactor Heat Balances	ISCOR Version 09V
Power-Flow Map	BILBO Version 04V
ECCS-LOCA	SAFER/GESTR-LOCA Version 04V (System Response) LAMB Version 08 (Core Flow) SCAT Version 02 (Time of Boiling Transition)
Containment Evaluations	M3CPT Version 05V (Short-Term Containment Response) SHEX Version 04V (Long-Term Containment Response)
Transient Analysis	PANAC Version 10V (RWE & LFWH) ODYN Version M10V (Pressurization Events) TASC Version 03V (Thermal Margin) SAFER/GESTR-LOCA Version 04V (Loss of Feedwater & Station Blackout - System Response) SHEX Version 04V (Station Blackout - Long-Term Containment Response)
ATWS Evaluation	ODYN Version M10V Application to ATWS Evaluation in NEDC 24154P Supplement 1 Vol 4
Radiological Impact	DORTG Version 01V (Vessel Flux) CONAC Version 04 (FHA, CRDA,MSLBA) MSIV Version 1.2a (LOCA-Offsite, Control Room & TSC)
Reactor Pressure Vessel Evaluation	FWNOZ Version 01 - (Feedwater Nozzle Fatigue Due to Rapid Cycling) ANSYS Version 4.4A (Feedwater Nozzle Fracture Mechanics Analysis) KRACKO Version 01 (Crack Growth Calculation Model)
RPV Internals	LAMB Version 07 (RIPDs) TRACG Version 01 (Acoustic Loads) PIPST Version 01 (Core Plate Buckling) COSMOS/M, Version 1.75 (3-D FEA Program)
Fire Protection (Appendix R)	SAFER/GESTR-LOCA Version 04V (System Response) SHEX Version 04V (Long-Term Containment Response)

NRC QUESTION 54

The submittal stated that the reactor can be cooled to 212°F in 36 hours, assuming failure of one RHR system. Clarify if the reactor can be brought to ≤ 200°F in 36 hours, assuming single system failure (see draft Reg. Guide 1.139 and TS).

SNC RESPONSE

Plant Hatch is not licensed to the requirements of draft Reg. Guide 1.139 for shutdown cooling analysis. For extended power uprate, a shutdown cooling analysis, based on

1 RHR loop in service and 95°F RHR service water temperature, was performed. The results of this analysis show that the reactor coolant temperature can be reduced from the proposed operating pressure and temperature at 2763 MWt (1050 psia, 546°F) to 200°F within 12 hours.

NRC QUESTION 55

The net positive suction head (NPSH) for Plant Hatch Unit 1 took credit for containment post-LOCA pressure of 9.7 psig. Specify the approving SER and the value of NPSH margin.

SNC RESPONSE

The licensing basis for Unit 1 does not require full conformance with the requirements of NRC Safety Guide 1. The Plant Hatch Unit 1 FSAR (page 14.4-15) states that a minimum margin between containment pressure and the pressure required for adequate NPSH is approximately 5 psi for the LPCI pumps and over 6 psi for the core spray pumps. The section also states that the core spray pumps have adequate NPSH at atmospheric pressure; however, there is a time period when 1 psig (of the 5 psig minimum available) of containment pressure is needed for LPCI to have adequate NPSH. The containment overpressure credit is documented in SNC's response, dated May 28, 1997, to the NRC request for additional information on Bulletin 96-03. Consequently, it is established that the containment backpressure was reviewed and allowed as defined in the current Unit 1 licensing basis.

ENCLOSURE 3

"Extended Power Uprate Safety Analysis Report for
E. I. Hatch Plant Units 1 & 2,"
NEDO-32749, DRF A13-00402 Class I,
February 1998