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SUBJECT: Responds to 900322 request for addl info re 870817
application to amend License DPR-23 on extending OL.

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Carolina Power & Light Company

P.O. Box 1551 • Raleigh, N.C. 27602

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JUL 9 1990

A. B CUTTER
Vice President
Nuclear Services Department

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION -
EXTENSION OF OPERATING LICENSE (TAC NO. 66079)

Gentlemen:

Your letter dated March 22, 1990 transmitted a Request for Additional Information (RAI) concerning Carolina Power & Light Company's (CP&L) license amendment request dated August 17, 1987 to extend the expiration date of the H. B. Robinson Steam Electric Plant, Unit No. 2 operating license to July 31, 2010. The response to the RAI is enclosed.

This package was originally scheduled for submittal by June 25, 1990. A delay in the submittal was agreed to during a discussion with the Project Manager on June 21, 1990.

Questions regarding this matter may be referred to Mr. R. W. Prunty at (919) 546-7318.

Yours very truly,

A. B Cutter

JSK/ecc (736ECC)

Enclosure

A. B Cutter, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

Thomasine C. Valentine
Notary (Seal)

My commission expires: 1-31-95

THOMASINE C. VALENTINE
NOTARY
* * *
PUBLIC
WAKE COUNTY, N.C.

cc: Mr. S. D. Ebneter
Mr. L. Garner (NRC-HBR)
Mr. R. Lo

A001
1/1

RESPONSE TO REQUEST FOR ADDITIONAL
INFORMATION FOR THE EXTENSION OF
THE OPERATING LICENSE FOR
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

1. *Provide an assessment of the impact (10CFR100) on the Exclusion Area Boundary, Low Population Zone and the nearest population centers based on population projections through the requested extension periods.*

Response

The Exclusion Area is owned and controlled by Carolina Power and Light Company. There are no residences or agricultural activities inside the 1400 foot exclusion distance. As discussed in the Updated FSAR, the only activity not related to company operations is recreational use of the lake that extends into a portion of the Exclusion Area. As detailed in the Updated FSAR and the previous FSAR there is also a 185 Mwe fossil-fired generating unit (H. B. Robinson, Unit 1) situated within the Exclusion Area. There are no plans to alter this land use during the requested license extension; therefore, there are no changes in the assumptions that demonstrate compliance with the provisions of 10CFR100 as relates to the Exclusion Area.

The only concentration of residents within the low population zone (LPZ) are located in the town of Hartsville, S.C. approximately 5 miles ESE of the plant. Based on the most current population data (1980 census data), the distribution of the population in the LPZ (0-5 miles) surrounding the site has increased at a slower rate than projected in the original FSAR which was based on 1960 census data. Projections based on the 1980 census data indicated that the population density within the LPZ over the extended lifetime of the plant will not exceed the projected population density stated in the original FSAR. Therefore, compliance with the provisions of 10CFR100 as it relates to the LPZ is not expected to be altered.

2. *Provide a discussion of population distribution trends within 50 miles of the plant based on 1970 and 1980 census data. Include projections through the period of the proposed extensions.*

Response

The population trends in the environs of H. B. Robinson, Unit 2 are best represented by comparing the population projections presented in the original FSAR with those presented in the updated FSAR. The data for both versions are presented in Table 2.1 for ease of comparison.

TABLE 2.1

A COMPARISON OF POPULATION ESTIMATES FOR ENVIRONS OF ROBINSON, UNIT 2 (Initial FSAR Versus Updated FSAR)

Miles	1980*	1986*	2000*	2007*	2010*
0-1* (1)	502	534	557	665	686
0-1 (2)	488	537	642	680	697
0-5 (1)	13090	13940	16145	17362	17930
0-5 (2)	11124	12242	14546	15378	15501
0-10 (1)	31654	33564	38480	41202	42426
0-10 (2)	31044	34074	40221	42443	42672
0-50 (1)	729000	783813	928275	1010204	1047494
0-50 (2)	678037	736743	873075	926873	959634

(1) Population projections based on original FSAR (1960 census)
(2) Population projections based on updated FSAR (1980 census)
* Values for years not presented in the reference document were extrapolated from adjacent values.

It is important to note that the original FSAR was based on 1960 census data with updates primarily from the state of South Carolina. The updated FSAR utilized 1980 census data and incorporated trends recommended by the State of South Carolina. The difference between the original projections and the later ones is minimal. In the low population area (0-5 miles) the later data reflects a slower growth rate than was predicted originally such that the most recent projection is only 86 percent of the original projection. In the range of 5 to 10 miles updated projections slightly exceed the original projections. The overall projections (0-50 miles) indicate that the later projections are again less than the original projections by about 8 percent. These projections indicate that the population of the region does not generally exceed the original expectations and will not likely contribute to an increase in the integrated population dose.

3. *Provide a quantification of the radiological impacts upon the general population based on the impact of the estimated calculated off-site doses. Include a discussion on the impact of the estimated dose commitments for 40 years of operation. How do off-site dose calculations for actual effluent releases compare with 10CFR50, Appendix I, objectives?*

Response

H. B. Robinson, Unit 2 operations to date have maintained effluent releases well within the guidelines of 10CFR50, Appendix I. Table 3.1 demonstrates the results of effluents in terms of the maximum exposed member of the public for the last four years along with a presentation of the 10CFR50 guidelines.

TABLE 3.1

Recent Effluent Doses to a Maximum Exposed Member of the Public

Pathway	10CFR50 Appendix I (mrem/yr.)	Doses Calculated by LADTAP & GASPARS (mrem/yr.)			
Gaseous		1986	1987	1988	1989
Total Body	5	.126	.230	.261	.0349
Thyroid	15	.8998	2.00	.349	.0345
Liquid					
Total Body	3	.0663	.110	.0304	.0295
Thyroid	10	.00438	.00926	.0113	.0078

The integrated population doses expressed in person-rem are presented in Table 3.2. The estimate for the year 2010 is calculated based on the average of the integrated doses for the years 1986, 1987, 1988, and 1989 and multiplied by the estimated population increases presented in the Robinson Updated FSAR and extrapolated for the year 2010.

TABLE 3.2
INTEGRATED POPULATION DOSE
50 Mile radius
(Person-Rem)

YEAR	ORGAN	GASEOUS	LIQUID	TOTAL
1986	Thyroid Total Body	.029 .042	.059 .107	.088 .149
1987	Thyroid Total Body	.548 .086	.137 .218	.685 .304
1988	Thyroid Total Body	.132 .107	.184 .205	.316 .312
1989	Thyroid Total Body	.008 .008	.134 .155	.142 .163
2010	Thyroid Total Body	.258 .081	.182 .242	.440 .323

In summary, the extension of Unit 2 operations through the year 2010 will extend the dose commitment over its environs by 0.3 person-rem per year of extension assuming similar operations to the average of the past four years. No credit is taken for future technologies which may be utilized to further reduce radiological effluents. Estimates of doses to the maximum exposed member of the public continue to remain a small fraction of the 10CFR50, Appendix I objectives. This is expected to continue for the remaining life of the plant.

4. *For the uranium fuel cycle, provide a statement regarding the environmental impact of the longer production run for the fuel cycle and any net annual effects per Table S-3 in 10CFR51.51. Also state any impacts of the 18-month fuel cycle versus the 12-month cycles used for the FES and FSAR.*

Response

The requested increase in the duration of the Operating License for Robinson Unit 2 is approximately three years, three months. This additional period of operation would involve roughly two core reloads based on a refueling frequency of 18 months. The percentage increase in the uranium fuel requirements for the lifetime of the unit is small, particularly when the decreased fuel requirements associated with implementation of higher enrichment, higher burnup fuel management are considered.

The Robinson Plant has not experienced a significant increase in offsite radiation exposure or a significant increase in the amount of effluents released offsite due to transition from 12 month to 16 month fuel cycles. Offsite releases are monitored and reported in the Semi-Annual Radioactive Effluent Release Report as required by Technical Specification.

The Robinson Plant was originally fueled with core loadings containing a maximum enrichment of 3.1 weight percent U-235. Reload cores were initially limited to a maximum enrichment of 3.50 weight percent U-235. Subsequent license amendments approved the use of reload fuel with enrichments up to 3.90 weight percent U-235. On February 9, 1990, the NRC issued Amendment No. 125 which further increased the maximum allowable fuel enrichment for core reloads to 4.2 weight percent U-235. In a Safety Evaluation Report dated January 7, 1988, the NRC accepted use of ANF fuel to extended burnup. The increase in the allowable fuel enrichment and allowable fuel burnups facilitated the implementation of 16 to 17 month fuel cycles rather than the 12 month fuel cycles previously employed. In issuing Amendment 125, which supports the use of longer fuel cycles, the NRC determined that, based on the environmental assessment, the issuance would not have a significant effect on the quality of the human environment.

The impact on 10CFR51.51, Table S-3 and 10CFR51.52, Table S-4 associated with higher fuel burnup and correspondingly longer operating cycles have been extensively addressed by the Atomic Industrial Forum (AIF). In a study prepared for the National Environmental Studies Project (NESP) of the AIF, it was concluded that "the current values in Tables S-3 and S-4, and the generic analyses of environmental dose commitments performed by the NRC Staff, are applicable to fuel burnups up to 60,000 (MWD/MT)[1]." This conservatively envelopes the anticipated operational range of current and anticipated future average core burnups for the Robinson Plant.

Additional margin to the values contained in Tables S-3 and S-4 lies in the fact that these tables were developed based on the anticipated fuel requirements of a 1,000 MWe reactor. Since the reactor at the Robinson

Plant is rated at 730 MWe net, the corresponding fuel requirements are lower and, thus, the environmental impact of the uranium fuel cycle is more modest.

The conclusions of this study are validated by assessments of the NRC relating to extended enrichment and burnup (2,3).

Based on previous environmental analyses associated with the increased fuel enrichment license amendments and the preceding discussion, it can be concluded the use of higher fuel burnup, longer operating cycles, and the proposed increased duration of the Operating Licenses do not alter the conclusions of 10CFR51.51, Table S-3; 10CFR51.52, Table S-4, the Final Environmental Statement; or the Final Safety Analysis Report.

REFERENCES:

1. Atomic Industrial Forum, Inc., "The Environmental Consequences of Higher Fuel Burnup," AIF/NSEP-032, June 1985.

2. Notice of Environmental Assessment and Finding of No Significant Impact for Extended Burnup Fuel Used in Commercial LWRs, Federal Register (53FR6040), February 29, 1988.

3. NRC Assessment of the Environmental Effects of Transportation Resulting from Extended Fuel Enrichment and Irradiation, Federal Register (53FR30355), August 11, 1988.

5. *Provide a discussion on how HBR2 intends to meet the requirements of 10CFR51.52, paragraph (a) or (b), and Table S-4.*

<u>Response</u>

See response to question 4.

6. *Describe any dose goals you may have for HBR-2 annual doses through the requested extension period, the bases for these dose goals (e.g., input from each plant department, historical doses), and CP&L's time frame for meeting these goals. Provide dose goals for both outage and non-outage years. Describe the HBR-2 "track record" for meeting dose goals in the past, the accuracy of these past dose goals, and how HBR intends to establish and enforce realistic dose goals in the future.*

Response

During the last five years (1985-1989). Robinson did not meet it's challenging person-rem goals. These goals were based on an expected scope of work, allowing for little or no contingency, and in every case the outage work expanded resulting in the plant exceeding it's goals.

The Company's plan for improving the collective dose at Robinson and the other plants is described in the Company Dose Reduction Program. The charter for this program is attached. As indicated in this charter, one of the key elements of this program is to set challenging goals that will result in our nuclear plants being perceived as "premier" (one of the best) performers compared to other nuclear plants.

CP&L is presently in the process of setting future challenging person-rem goals that will achieve our overall goal of becoming one of the best nuclear utilities. These dose goals will be at or better than the goals that INPO is presently setting as industry goals for 1995. We would expect to continue to set dose goals that are challenging and are at or better than the respective industry averages for the future, out to, and including, the license extension period. This is expressed in the operational goal established by HBR2 Plant Management: that collective radiation exposure has been maintained below the three-year rolling industry average beginning in 1992.

CAROLINA POWER & LIGHT COMPANY

DOSE REDUCTION PROGRAM CHARTER

Recommended By:

H.R.Banks
Manager
Quality Assurance

A.B.Cutter
Vice President
Nuclear Services

C.R.Dietz
Manager
Robinson Nuclear Project

A.M.Lucas
Manager
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R.B.Richey
Manager
Harris Nuclear Project

R.B.Starkey
Vice President
Brunswick Nuclear Project

Approved By:

R.A.Watson
Senior Vice President
Nuclear Generation

CAROLINA POWER & LIGHT COMPANY
DOSE REDUCTION PROGRAM CHARTER

Purpose

The purpose of the Company Dose Reduction Program is to identify and implement dose reduction actions and programs that will ensure that the Company is recognized as a premier nuclear utility. The Dose Reduction Program will consist of proactive initiatives that the nuclear plants and their support departments will implement to lower the Company's collective dose.

Organization

The Company Dose Reduction Program will be managed and directed by two formally recognized groups and implemented by the line management of the nuclear plants and nuclear support departments. The groups are the Dose Reduction Steering Committee and the Dose Reduction Committee. The Dose Reduction Committee reports to the Dose Reduction Steering Committee. Specific actions to reduce dose will be assigned to and are the responsibility of the nuclear plant and nuclear support line management organizations in the Company.

Membership

The Dose Reduction Steering Committee shall be Chaired by the Senior Vice President of Nuclear Generation and include Vice Presidents and/or Managers of the following Departments:

Brunswick Project

Harris Project

Robinson Project

Nuclear Services

Nuclear Engineering

Quality Assurance

The Manager - Health Physics & Chemistry Section shall be a non-voting member and shall furnish staff support to the Steering Committee.

The Dose Reduction Committee shall be Chaired by the Manager of the Health Physics and Chemistry Section and include representatives from the following Departments:

Brunswick Plant - E&RC Manager

Harris Plant - E&RC Manager

Robinson Plant - E&RC Manager

Nuclear Services - Nuclear Fuels representative
- Health Physics & Chemistry representative

Nuclear Engineering - Representative

Additional participation shall be provided, when requested, by named representatives from Nuclear Plant Support Section and Technical Services Department (Materials).

The E&RC Managers represent their respective plant management and the views of their plant management.

Responsibilities

The Dose Reduction Steering Committee is responsible for:

Assisting Senior Management in setting Company ALARA expectations.

Setting long range (5 year) collective person-rem dose goals for the Company and the nuclear plants.

Setting management standards for dose reduction.

Reviewing the nuclear plant's annual collective person-rem goals for consistency with long range goals.

Approving the Company Dose Reduction Action Plan.

Reviewing Company ALARA Program audits and assessments and taking the appropriate corrective actions.

The Dose Reduction Committee is responsible for:

Assisting the Dose Reduction Steering Committee in setting long range (5 year) collective person-rem dose goals for the Company and the nuclear plants.

Identifying candidate methods to achieve dose reduction.

Recommending a dose reduction action plan including cost-effective methodology for attaining the dose reduction goals.

Providing assistance to line management in implementing elements of the dose reduction action plan.

Reviewing Company ALARA Program audits and assessments and recommending the corrective actions that are supported by the nuclear plant management.

The nuclear plant and nuclear support departments line management are responsible for:

Setting annual plant collective person-rem goals that achieve the long range dose reduction goals.

Assisting the Dose Reduction Committee identify candidate methods to achieve dose reduction.

Recommending proactive, cost-effective dose reduction actions for attaining the dose reduction goals.

Implementing the Dose Reduction Action Plan

Meetings

The Dose Reduction Steering Committee will meet at least quarterly and an agenda and minutes from each meeting will be recorded.

The Dose Reduction Committee will meet, at least quarterly, prior to the quarterly Steering Committee meeting and as often as necessary to carrying out its responsibilities. An agenda and minutes from each meeting will be recorded.

7.a. Although Robinson's annual collective dose of 209 R/yr in 1989 was below the PWR industry average, Robinson's annual collective exposures for most of the plant life have been well above the industry average for PWRs. Describe how CP&L plans to maintain the 1989 level of success and reduce the HBR-2 annual collective dose over the next few years to levels which will be more comparable to the industry average during the period of extension. Describe any changes/improvements that may have already been made to reduce the annual doses at HBR-2 and their effectiveness in reducing annual doses.

Response

Robinson has completed several modifications and made procedure changes that will reduce future doses. The Resistance Temperature Detection (RTD) Bypass system has been removed eliminating one of the largest sources of exposure in the reactor coolant pump bays. The reactor thermocouples are now inserted from below the reactor via the thimbles. This will result in less dose since it eliminates work on the reactor head. The plant shutdown procedures have been changed to better ensure controlled "crud" burst and proper cleanup. This reduces the out-of-core exposure rates on the reactor coolant system piping. Robinson has also been operating on the "elevated" lithium chemistry program. This will also reduce exposure rates on the out-of-core piping. Other changes which have effected lower radiation doses include: (1) live load valve packing, (2) lowered RCS leak rate, (3) reduced contaminated square footage, (4) creation of the Plant Management ALARA Review Committee, and (5) assignment of a "Rad Budget" to plant work groups. All of these changes and changes as a result of the Company's Dose Reduction Program will result in less dose during the remaining life of the plant.

7.b. Describe Robinson's radioactive "source term" relative to other plants of the same vintage. What plans (short-range and long-range) does CP&L have to reduce this source term (e.g., system chemical decon, cobalt material replacement during the time of extension).

Response

Based on comparisons with other Westinghouse plants, Robinson's exposure rates are average or lower than average. The Dose Reduction Program will result in specific "source term", exposure rate reduction actions, e.g., "crud" control, cobalt elimination, etc. that will reduce future doses including the license extension period. These specifics and other actions as a result of the Dose Reduction Program will be available for NRC inspection.

8. *Detailed and accurate ALARA job preplanning plays an important role in minimizing job time and the resultant occupational doses. Describe how ALARA job preplanning, along with coordination and cooperation among plant management, ensures that pertinent jobs receive adequate ALARA reviews well in advance of the actual outage. Describe how these ALARA reviews, coupled with accurate man-hour job estimates, can contribute to lowering annual doses during the period of the extension.*

Response

CP&L recognizes that between 75 and 90 percent of the dose comes from jobs during outages. We also recognize that we must improve our outage and job planning in order to decrease the dose. As previously mentioned, the Company is embarking on a Dose Reduction Program. This effort is above and beyond the present plant ALARA program and any changes that the plant is specifically making to improve it's ALARA program. As a part of this Dose Reduction Program, outage and job dose reduction planning will be evaluated and specific actions to improve dose will be undertaken. As previously mentioned, these specifics and other actions as a result of the Dose Reduction Program will be available for NRC inspection.

9. *Provide a comparison between actual radwaste shipments in recent years and the information provided in FES section 11.1.7, and estimate the impact for the extension periods.*

Response

The FES for the H. B. Robinson Nuclear Project, Section 11.1.7, discusses the anticipated volumes of solid radioactive waste generation associated with the operation of the facility. The original projections estimated that approximately 500, 55 gallon drums of wet solid wastes (spent demineralizer resins, filter sludges, and evaporator bottoms) would be generated and shipped annually. In addition, approximately 450, 55 gallon drums of dry solid wastes (ventilation air filters, contaminated clothing, paper and miscellaneous other items) would be generated and shipped annually. This equates to a total of 950, 55 gallon drums or approximately 7125 ft^3. Currently, the facility no longer generates liquid concentrates and has switched to an in line demineralization system for radwaste processing in an effort to minimize solid radwaste generation.

During the past five years of operation the facility has made tremendous progress in reduction of radwaste volume. While the facility's solid radwaste volume had at one time been greater than projected by the FES (partially due to solid wastes generated as part of the steam generator replacement outage), it has been reduced well below this value and has also been well below the industry average as defined by the Institute for Nuclear Power Operations (INPO). Table 9.1 provides the total solid radwaste volume in cubic feet produced per year for the past five years of operation along with the industry average for PWR's for the past five years. As shown in this table, the H. B. Robinson Nuclear Project has been well below the industry average for the past three years. Based on the volume produced to date this year, it is expected that 1990's radwaste volume will also be well below the industry average. CP&L is continuing to actively pursue additional volume reduction techniques such as incineration of dry active wastes and resin oxidation. In addition, CP&L is also pursuing source term reduction techniques such as cobalt elimination which, in addition to volume reduction, will likely produce a reduction in radwaste activity.

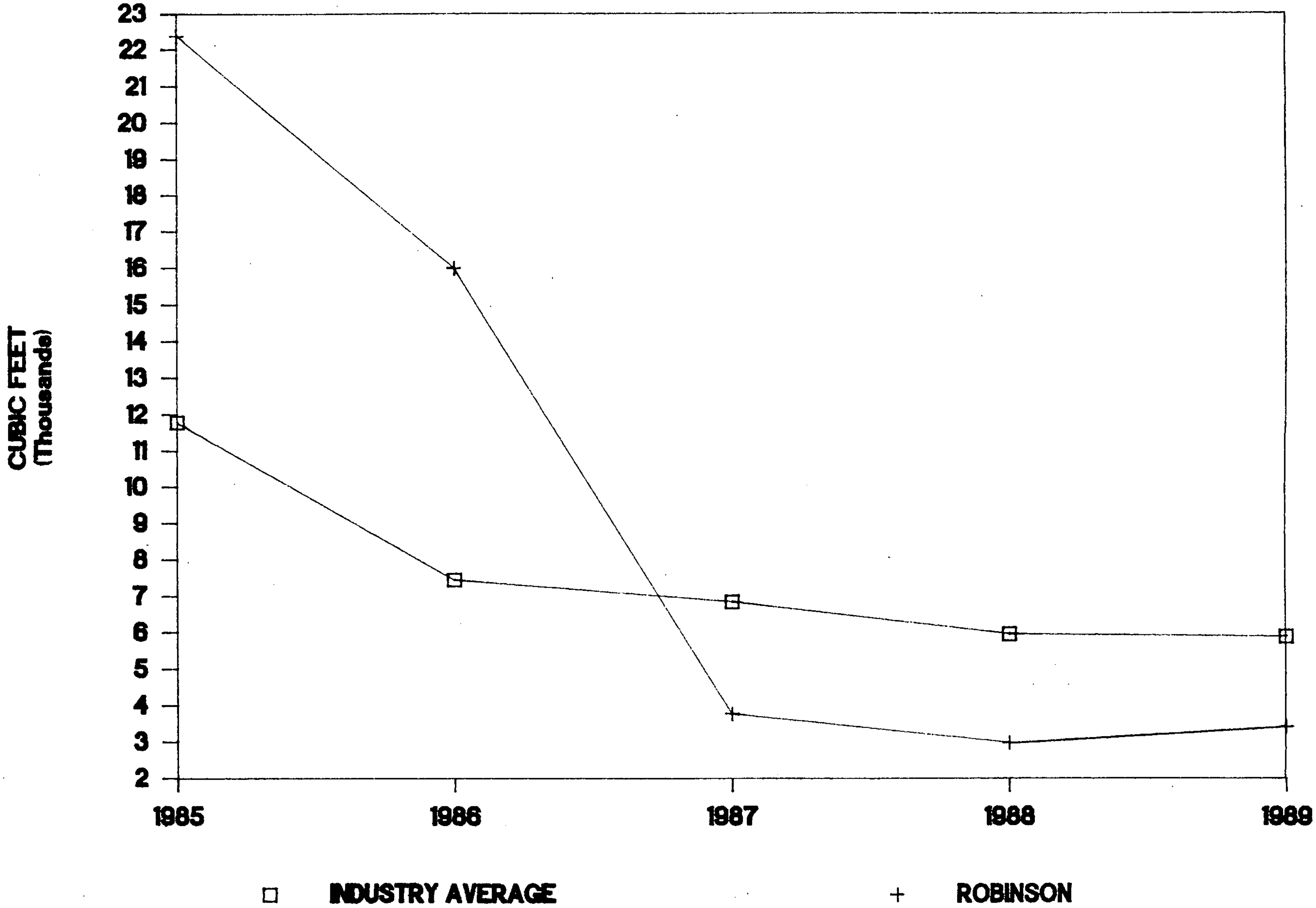
Based on this information, it is expected that annual average waste volumes will continue to remain below the original projections of the FES and that solid radwaste production of the facility will continue to be reduced as new technologies become available. The expected solid radwaste generation for the years of requested license extension will be a small fraction of the total generated over the facilities lifetime. This volume is not considered to be significant and will not alter the conclusions of the FSAR or the FES.

TABLE 9.1
RADWASTE VOLUMES (ft^3)

	H.B. Robinson	Ind. Avg.	FES Projection
1985	22386	11795	7125
1986	15998	7451	7125
1987	3772	6851	7125
1988	2974	5968	7125
1989	3426	5898	7125

H.B. ROBINSON RADWASTE DATA

COMPARISON AGAINST INDUSTRY AVERAGE



10. *Provide the current National Pollutant Discharge Elimination System (NPDES) permit number and dates of issuance and expiration.*

<u>Response</u>

The HBR2 NPDES permit number is SC0002925. The permit was issued on October 26, 1983, and became effective on December 1, 1983. The permit expired on midnight, November 30, 1988.

CP&L made timely application for renewal on May 25, 1988 and operation under the provisions of the expired permit are continuing as allowed by law until the permitting agency acts upon our application.

11. *Identify any potential impact that prolonged plant operation may have on properties with historical, architectural, or archeological significance.*

<u>Response</u>

Per an inquiry to the SC Department of Archives and History, no impacts on properties with historical, architectural, or archeological significance are expected due to prolonged plant operation.

12. *Assess the impact of the proposed extension on the reactor vessel, mechanical equipment, electrical equipment, and plant structures.*

Response

A. MECHANICAL EQUIPMENT

Most of the following information summarizes material previously provided in the HBR Unit 2 Updated Final Safety Analysis Report (UFSAR) or other referenced submittals.

A.1 Reactor Coolant System

HBR Unit 2 was purchased from Westinghouse Electric Corporation as a "Turnkey" project. Accordingly, the reactor coolant system, which is a 3-loop PWR, was specified, designed, and constructed per applicable Westinghouse requirements. Chapter 5 of the UFSAR presents a substantial amount of information on the reactor coolant system and its major equipment.

A.1.1 Reactor Vessel

The reactor vessel was manufactured by Combustion Engineering to the requirements of Westinghouse Equipment Specifications 676367 and 676244. These specifications required that the vessel be designed for a 40-year life in accordance with the ASME Code, Section III. Applicable pressure and thermal transients and the number of occurrences are included in the specifications.

Radiation induced embrittlement of the HBR Unit 2 reactor vessel has been a past concern but is now substantially resolved in accordance with current criteria. In Reference 1, CP&L submitted information to the NRC demonstrating that the "... vessel will exhibit acceptable embrittlement characteristics for more than 95 Effective Full Power Years (EFPY) with the present fluence profile". The EFPY is a technical approach to accounting for the shutdown time, and industry experience has shown that calendar time is always greater than the EFPY (i.e., nominally one calendar year equals 0.8 EFPY).

In evaluating CP&L's submittal, the NRC concurred in Reference 2 with CP&L's calculation of the reactor vessel pressure/temperature limits, based upon the reference temperature using the material surveillance capsule data. These pressure and temperature limits are used to develop reactor vessel heatup and cooldown curves which are contained within the plant Technical Specifications, specifically 3.1.2.

An extension of the operating license of 3.3 calendar years would increase the neutron fluence to the controlling location in the reactor vessel by about 7.3% to 2.125×10^{19}

n/cm^2. Using the Regulatory Guide 1.99, Revision 2, shelf energy decrease figure (Figure 2), this change is so small (about 2%) that, in essence, it is hidden within the lines on the figure. While there will be a calculational effect on the heatup and cooldown curves, the change is insignificant and the reactor vessel can be safely operated.

The subject of pressurized thermal shock is governed by 10CFR50.61. In Reference 3, the NRC found that the HBR reactor vessel satisfied the screening criteria through the present end of the license. Using the parameters in Reference 3, Enclosure 2, the NRC's (BNL's) calculated fluence (increased by the additional exposure discussed above) and Equation 1 for RT_{PTS}, the calculated value at the end of the requested license extension would be about 283°F which is less than the allowable 300°F screening criteria specified by 10CFR50.61. Thus, pressurized thermal shock considerations demonstrate that the reactor vessel is acceptable.

A.1.2 Steam Generators (SGs)

The original steam generators were provided in accordance with Westinghouse Specification 676397. The specification required that the SGs be designed and fabricated per the requirements of the ASME Code, Section III. The specification does not contain an explicit identification of the design life. However, a comparison of the SG transient occurrences, as contained within the specification, to those contained within the reactor vessel specification, indicates that the design life was 40 years.

The tube bundle portion of the steam generators was replaced in 1984. This project was approved by the NRC as indicated in Reference 4.

The replacement SG tube bundles were designed and fabricated per the requirements of Westinghouse Specification 955479. This specification identifies a design life of 40 years, contains pressure and temperature transients essentially identical to those in the original specification, and invoked the requirements of the ASME Code, Section III.

As the boundary between the primary system and the feedwater/steam system, the steam generator tubing is required by the plant Technical Specifications to be periodically inspected with a report of the results provided to the NRC.

A.1.3 Reactor Coolant Pumps (RCPs)

The RCPs were provided per the requirements of Westinghouse Specification 676429. This specification identifies the design life as 40 years, contains the applicable pressure and temperature transients, and invokes the requirements of the ASME Code, Section III.

A.1.4 Pressurizer (PZR)

The pressurizer was provided per the requirements of Westinghouse Specification 676360. This specification invokes the requirements of the ASME Code, Section III, and identifies applicable pressure and temperature requirements. The specification does not explicitly identify the design life. However, a comparison of the pressurizer transient occurrences to those contained in the reactor vessel specification, clearly indicates that the design life basis was 40 years.

A.1.5 Other Mechanical Equipment

Inservice inspection and surveillance of equipment important to safety is addressed in Section 4 of the plant Technical Specifications. Basically, this invokes an ASME Section XI program, as required by 10CFR50.55a(g) on components categorized as ASME Code Class 1, 2, and 3. This testing and inspection program is utilized by CP&L to continuously assure that components are capable of operating when needed and are capable of performing their intended function.

REFERENCES

1. CP&L to NRC, Response to Generic Letter 88-11, NLS-88-225, December 22, 1988.

2. NRC (R. H. Lo) to CP&L (L. W. Eury), Pressure-Temperature Limits Relating to Generic Letter 88-11, March 9, 1990.

3. NRC (K. Eccleston) to CP&L (E. Utley), Fast Neutron Fluence for the Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", June 19, 1987.

4. Safety Evaluation Report related to steam generator repair work at H. B. Robinson Steam Electric Plant Unit No. 2, NUREG-1004, November 1983.

B. ASSESSMENT OF IMPACT ON ELECTRICAL EQUIPMENT OF THE PROPOSED EXTENSION OF H. B. ROBINSON UNIT 2 OPERATING LICENSE TO 2010

An Environmental Qualification (EQ) program based on the requirements of 10CFR50.49 is in place at the Robinson Plant and has been reviewed and audited by the NRC. Deficiencies identified by these audits have been or are in the process of being corrected.

The EQ program has reviewed the service environments of safety related electrical equipment. For each item falling within the scope of the EQ program a qualified life has been established based on available test data and engineering evaluation of this test data and the specific service parameters for that item. The Robinson EQ program monitors the operation and performance of EQ equipment. EQ equipment with less than a 40 year qualified life is replaced under the EQ program prior to expiration of the qualified life.

The design life for H. B. Robinson electrical equipment is 40 years. The continued operation of non-EQ electrical equipment at the Robinson Plant (safety and non-safety) is monitored and assured through the preventive and corrective functions of the plant maintenance program. Periodic testing and inspection has been established in plant procedures to assure satisfactory operation for those components deemed necessary through operational experience and/or vendor recommendations.

The 40 year design basis of H. B. Robinson Unit 2 electrical equipment in conjunction with the functions of the EQ program and the maintenance program to assure continued operation and function of plant electrical equipment, assure that this equipment can adequately function for a 40 year operating life.

C. STRUCTURAL

Seven unique structures at the Robinson Nuclear Plant are classified and designed as Seismic Class I. Six of these structures, the Containment, the Spent Fuel Pit, the Control Room, the Diesel Generator Rooms, the Intake Structure, and the Auxiliary Building are all reinforced concrete structures with some steel forms. The remaining Class I structure is a specific section in the Turbine Generator Building and it is constructed of structural steel.

All of these structures were reviewed and found acceptable by the NRC at the time of licensing. The structures were designed to resist various combinations of dead loads, live loads, environmental loads, including those due to external phenomena such as wind, tornadoes, and earthquakes (RNP is designed to withstand a .1 g operational basis earthquake and a .2 g safe shutdown earthquake), as well as loads generated by design accidents, including pressure, temperature, and pipe rupture effects. The prestressed concrete containment was designed in accordance with ASME Code Section III and American Concrete Institute Standard ACI-318-63. The NRC found the design, materials, construction methods, and quality assurance utilized for all of these structures to be acceptable for satisfying relevant requirements as discussed in Section 3.1 of the updated FSAR.

Industrial experience with materials such as concrete and steel establishes that a service life of well in excess of forty years can be anticipated.

In addition to the industrial experience which provides verifiable evidence that steel and concrete are durable materials that maintain their original design characteristics over years, CP&L has established certain surveillance practices that provide periodic information for the structural integrity of the Seismic Class I containment building.

Prior to initial plant operations, an integrated leak rate test (ILRT) was required to be performed on the containment at the original design pressure (P_p) of 42 psig and a test pressure (P_t) of 21 psig to establish the respective measured leak rates L_m (42) and L_m (21). The minimum test temperature was to be 50°F.

The Plant Technical Specifications (Section 4.4.1.1.g) further requires that two ILRT's be scheduled at approximately equal intervals between the major shutdowns for inservice inspection conducted at ten-year intervals. In addition, an ILRT shall be performed at the end of the ten-year interval which may coincide with the inservice inspection shutdown period. These tests shall be performed at an initial pressure at or above 21 psig (50% of design pressure). The first ILRT shall be performed at 21 psig and 42 psig.

In addition to the above tests, another surveillance test performed on the Reactor Containment Building (RCB) is the tendon surveillance test which involves removing and inspecting test specimens from an embedded location after five years and 25 years with the latter tendon yet to be removed.

In a similar manner, RCB penetrations, certain containment isolation valves, and double gasket seals, except for the personnel air lock, are tested at each refueling. This requirement is in accordance with 10CFR50 Appendix J. There are 46 containment penetration sleeves for pipes in the RCB. In the case of the pipes carrying hot fluids, the pipe is insulated, and cooling is provided to maintain the concrete temperature adjoining the embedded sleeve below specified limits. Although no official structural surveillance program is currently in place at Robinson, no unusual occurrences (i.e. flaking, chipping, spalling, or cracking) have been visually observed at the junction of the pipe sleeves and the concrete. It is concluded that the integrity of the containment pipe penetrations (as well as electrical and duct penetrations) is maintained per original design and can be projected to remain stable throughout the current licensed period and beyond.

In summary, the results of these original tests and the ongoing surveillance program plus the industrial experiences support the fact that the RCB reinforced concrete exterior structure and the interior structures such as walls, compartments, floors, slabs, beams, columns, and interior steel frames are adequate to meet the design requirements. They will maintain satisfactory structural integrity throughout the proposed licensing extension period.

The RCB is prestressed; all the remaining Class I seismic structures are not prestressed. Although these structures are not observed and monitored as closely as the RCB, they will remain structurally stable as well.

Therefore, based on the above considerations, CP&L concludes that the plant structures will maintain design integrity during the proposed extension of the operating license.

13. *Provide a listing of all FES or FSAR sections in which less than 40 years of operation was assumed; provide an assessment of the impact of the extensions on conclusions found in the sections identified. For example, 30 years was used as the expected plant life in section 5.1.1 of the FES.*

Response

a. FES Section 2.2

Section 2.2, Regional Demography provides population data estimates based on a February 1964 study and 1970 census data.

Assessment

The effect of updated population trends is discussed in the response to Question 2.

b. FES Section 5.1.1

Section 5.1.1 notes that "land committed to plant buildings is not available for other uses during the expected 30 year life of the plant." Section 5.1.3 "Conclusion" states that ". . . continued operation of Unit 2 will result in acceptable impacts on area land use."

Assessment

The Conclusions of Section 5.1.3 would not be changed by extended operation of HBR2 since there is no change in land use required by continued operation.

It should be noted that CP&L has an existing ISFSI license for eight dry fuel storage modules at the site. CP&L is currently planning to request an amendment to allow construction of up to 175 more modules as a contingency against interruptions in planned fuel shipments to the Shearon Harris Plant in North Carolina. The ISFSI license expires in 2006 and has been granted under the provision of 10CFR72; a separate Safety Analysis Report and Environmental Report have been submitted on Docket 72-3.

c. FES Section 8

Section 8, "The Need For Generating Capacity," concludes that ". . . continued operation of Unit 2 is required to meet the energy requirements and peak power demands of the CP&L service area."

This conclusion is based in part on an assumption of a plant life of at least 30 years, as well as other data including Federal Power Commission projections of electric power requirements to 1990.

Assessment

The conclusion is valid for extended operation. Although the specific data analyzed is outdated, power requirements have continued to increase and CP&L has added additional generating capacity to the system.

d. FES Section 10.2

Section 10.2 discusses commitment of land for production of electrical energy for at least the next 30 years and concludes that the benefits outweigh the short-term uses of the environment in the vicinity.

Assessment

The conclusion remains valid since major environmental impacts resulting from the construction and operation of the plant have already been incurred and impacts from 39 months of incremental operation would be insignificant.

e. FES Section 10.3

Section 10.3 discusses the irreversible and irretrievable commitment of resources, and concludes that these are appropriate for the benefits gained. This section discusses consumption of uranium and diesel fuel during the 30 year lifetime of the plant.

Assessment

The conclusion remains valid. Increasing the operating life of the plant will increase the amount of uranium and diesel fuel consumed. The increase in diesel fuel consumed is minuscule when measured against world wide consumption of diesel fuel. Possible increases in the use of uranium may be offset by advances in fuel management, such as longer cycles (see Response to Question 4). Since uranium fuel is a major cost, CP&L is economically driven to minimize fuel usage. Economic motivation and the scarcity of competing uses for uranium indicate that the FES conclusion remains unchanged.

f. FES Section 10.5

Section 10.5.1 assumes a useful plant life of 30 years. Section 10.5.4 concludes that the "benefits of continued operation of Unit 2 substantially outweigh the costs."

Assessment

The conclusion remains valid for the extension of the operating license. Major environmental effects of construction and operation have already been incurred and will not substantially increase. Further, HBR2 operating costs are relatively lower than would likely be realized from a newly constructed facility. The benefits of extended operation would be substantial continued electric generation with only a small increase in costs.

g. UFSAR Section 2.1.3

Section 2.1.3 discusses population distribution

Assessment

Extended plant operation would have minimal impact on conclusions drawn from data presented in UFSAR Section 2.1.3. See the response to Question 2.