CP&L

Carolina Power & Light Company

HARRIS NUCLEAR PROJECT
P. O. BOX 165
NEW HILL, NC 27562

HXDE-XXX-XXX-XXX
HO-853222 (E)

August 6, 1985

Mr. Brian K. Grimes, Director
Division of Quality Assurance, Vendor and
Technical Training Center Program
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, DC 20555

INTEGRATED DESIGN INSPECTION 50-400/84-48

Dear Mr. Grimes:

At the conclusion of the IDI team's close-out inspection on July 24, 1985, the team requested additional information for examination. The requested information is submitted as enclosures to this letter and is briefly described below:

Enclosure 1 - Supplemental information for our response to item D5.7-2.

Enclosure 2 - Approved FSAR change regarding setpoint documentation for item U6.5-1.

Enclosure 3 - Approved FSAR change regarding cable tray overfill for item D2.4-2.

Enclosure 4 - Approved FSAR change regarding cesium source term for item D2.5-3.

Enclosure 5 - Approved FSAR change regarding Beta credit for washout for items D2.5-5 and D2.5-6.

Enclosure 6 - Summary of activities by Ebasco Applied Physics (items D2.5-1, D2.5-3 through D2.5-7).

Enclosure 7 - Results of QA audit of Ebasco Applied Physics.

Enclosure 8 - Lists several items which were closed in the close-out inspection. These related to circuit and relay changes. It is our understanding that implementation of these changes described in our responses may be verified by NRC Region II.

IE01 Add: IE/DQAVT/VPB Ltr Encl
1/1 1 1

Upon your consideration of the enclosed information as discussed with you during the close-out inspection exit meeting on July 24, 1985, it is our understanding that all IDI items are closed except D2.3-1, D2.7-1, D2.7-2, and D2.8-1. One of these items (D2.3-1) relates to the containment building sump for which we are trying to arrange a meeting with NRR on August 15.

We appreciate the team's efforts in working with us during the close-out inspection. We are available to provide any other information necessary to complete the closure of the remaining items.

Very truly yours,

RAWatson

R. A. Watson, Vice President
Harris Nuclear Project Department

RAW/EMH/jam

Enclosures

cc: Mr. B. C. Buckley (NRC)
Mr. G. F. Maxwell (NRC-SHNPP)
Dr. J. Nelson Grace (NRC-RII)
Mr. Joe Joyce (NRC-ICSB)
Mr. Travis Payne (KUDZU)
Mr. Daniel F. Read (CHANGE/ELP)
Wake County Public Library
Mr. Wells Eddleman
Mr. John D. Runkle
Dr. Richard D. Wilson
Mr. G. O. Bright (ASLB)
Dr. J. H. Carpenter (ASLB)
Mr. J. L. Kelley (ASLB)

ENCLOSURE 1

INSERT #1

The 480V Class 1E bus undervoltage relay settings will also be reviewed.

INSERT #2

The undervoltage relay on safety-related 480-volt Bus 1B3-SB was set at 80 volts pickup on the secondary side of the 480/120-volt potential transformer. This equates to a minimum bus voltage of 320 volts and is too low (70% of equipment-related voltage) to provide adequate protection for the equipment fed from this bus. This setting does not agree with the results of the calculation performed to determine the setting of the undervoltage alarm relay, which established a minimum voltage setpoint of 100 volts pickup for the undervoltage relay. This calculated value corresponds to a bus voltage of 400 volts (87% of rated). Also, the selection of the minimum time delay setting on the undervoltage relay may result in nuisance alarms during diesel generator loading.

INSERT #3

The undervoltage relays provided on the 480 volt Class 1E power centers are for load shedding Class 1E motors (Power Centers 1A2-SA and 1B2-SB) and for alarm (Power Centers 1A3-SA and 1B3-SB) whenever there is a loss of bus voltage. The undervoltage alarm function is not intended to alert the operator of a degraded voltage condition, but rather that power is not available to the motor control centers. Degraded voltage protection is provided by undervoltage relays located at the 6.9kV Class 1E Buses 1A-SA and 1B-SB and are set in accordance with criteria of Branch Technical Position PSB-1, "Adequacy of Station Electric Distribution System Voltages."

The criteria for the undervoltage relay settings on the 480 volt Class 1E power centers is as follows. The dropout settings on all undervoltage relays of the 480 volt Class 1E power centers will be set below the dropout settings of the 6.9kV bus primary undervoltage relays as determined by the latest voltage study, "Adequacy of Station Electric Distribution System Voltage" (Feb. 1985). The time delay setting of the undervoltage relays of all 480 volt Class 1E power center buses will be selected to ensure: a) avoiding nuisance relay operation for feeder faults by allowing the feeder breaker solid state trip device to trip first and isolate the fault from the power center bus, and b) for LOCA followed by loss of off-site power conditions, the undervoltage relay to initiate trips of the 460 volt motor breakers prior to closing of the diesel generator breaker.

Therefore, in accordance with the above, the 480 volt undervoltage relays will be set at 70V (58% of power center bus rated voltage of 480 volt) which is below the dropout setting of the 6.9kV Class 1E bus primary undervoltage relays as determined by the latest voltage study "Adequacy of Station Electric Distribution System Voltages". The time dial position will be set at 2.

5.7 MOTOR ELECTRICAL PROTECTION

In response to these concerns, the relay calculations for each of the 460 volt Class 1E motors will be updated and verified using motor specific starting times and safe stall times. The relay calculations will be incorporated in a Standard Relay Calculations and Data Sheet as required by the Harris Plant Engineering Section guidelines for relay protection, which is currently being developed. Accordingly, any missing assumptions will be identified and verified.

INSERT #1

~~Concerning the 480 volt Class 1E bus undervoltage relay setting, the latest voltage study, "Adequacy of Station Electric Distribution System Voltage" (Feb, 1985), has determined that the relay dropout settings of 100 volt for two of the 480 volt Class 1E power centers were acceptable and the relay dropout settings for the remaining two 480 volt Class 1E power centers will be revised to 110 volt. The drawings for the subject settings will be revised to reflect the correct bus undervoltage relay settings and the time delay settings on the undervoltage relays of all Class 1E power center buses will be reviewed to ensure avoiding nuisance relay operation under transient undervoltage conditions.~~ The Harris Plant Engineering Section will develop a design guideline and procedures for relay protection.

The Startup Organization will revise Procedure 1/2-9000-E-05 to require the bus voltage to be recorded when measuring the motor running current under both the loaded and unloaded conditions. In addition, the appropriate process will be established to require the startup organization, upon request from the Harris Plant Engineering Section, to perform additional tests on large 460 volt and 6.6KV motors for which specific data cannot be obtained from the vendor.

5.8 CABLE DESIGN AND ANALYSIS

The IDI team reviewed cable sizes to ensure cables were properly sized for both normal and overload conditions. Power and control cables were reviewed for both ampacity and voltage drop considerations. The IDI team identified certain cases where cables were not adequately sized for the worst case voltage conditions.

Deficiency 5.8-1 stated that when the dc system is at minimum voltage, the power cable feeding the dc motor operated valves is not adequate. Corrective action has been taken by revising the cable size.

The basis of Deficiency 5.8-2 was the possibility of simultaneous operation of multiple relays in the auxiliary relay panels. Under worst case conditions, several relays can operate concurrently, which might result in a high inrush current (approximately 125A) through the cable feeding the panel. Due to this high inrush current, the voltage at the relay coils may not be adequate to operate the relay. The Ebasco electrical group has just completed an evaluation of all ac and dc control loops. The objective of this effort was to ensure that all associated cables were properly sized. As a result of this evaluation, in conjunction with D5.8-2, the subject cable size has been revised.

D5.7-2 (DEFICIENCY) 480V BUS UNDERVOLTAGE ALARM

DESCRIPTION

INSERT #2

~~According to the project load study of 1982, the minimum allowable steady state voltage of the 480V Class 1E Power Center Bus 1B3-SB (supplying MCC loads only) was 428V (i.e. 90% of motor rated voltage or 414V plus 3% for cable voltage drop of MCC motors to the PC bus). The calculated bus undervoltage relay settings are: drop out of 100V (correspondong to 400V bus voltage) and a time dial position 4.~~

~~However the relay settings for all 480V Power Center (PC) buses are incorrectly entered in the relay setting drawings as 80V and time dial position 1 instead of the above calculated settings.~~

RESPONSE

There are total of four 480V Class 1E Power Center Buses and all are affected.

INSERT #3

The latest project voltage study, "Adequacy of Station Electric Distribution System Voltages" for compliance with BTP PSB-1 of February, 1985 which was not available at the time of the IDI inspection, revises the minimum allowable steady state voltage for PC buses 1A3-SA and 1B3-SB (both supplying MCC loads only) to 444V and that for PC buses 1A2-SA and 1B2-SB (supplying motor loads only) to 424V. Considering that the preferred drop out setting of the bus undervoltage relay is just below the bus minimum allowable steady state voltage, bus PT ratio (480/120V), and the relay available taps (60,70,80,100&110V - not continuously adjustable) the relay dropout setting of 100V was acceptable. However, in view of the revised values for the minimum allowable steady state voltages for the 480V Class 1E PC buses per the voltage study of 1985, the dropout settings of the undervoltage relays for buses 1A3-SA and 1B3-SB will be revised to 110V. The relay dropout settings of 100V for PC buses 1A2-SA and 1B3-SB are considered acceptable.

The time delay settings on the undervoltage relays of all Class 1E PC buses will be reviewed to ensure avoiding nuisance relay operation under any transient undervoltage conditions.

The relay setting drawings for all 480V PC buses will be revised to indicate the correct bus undervoltage relay settings.

The Harris Plant Engineering Section is developing design guidelines for relay protection including that for all 480V PC buses undervoltage relays.

ENCLOSURE 2

Routed 7/25/85

EMH

FSAR AMENDMENT
REVIEW APPROVAL FORM
(RAF)

RAF-HMES

(IDI Questions U6.5-1) EBFC-827 F-302 ITEM NO. 393
DUE DATE ASAP

SUBJECT: Reg Guide 1.105 Instrument Setpoints

FSAR/ER SECTION TO BE AMENDED: 1.8 RG 1.105

CPSL REVIEW RESPONSIBILITY/PRIMARY SIGNATURE REVIEWER:

M F Thompson

RECOMMENDED CHANGE: (Attach marked-up FSAR/ER page's) for clarity)

per attached EBFC-827 F-302

REASON FOR REVISION:

per attached EBFC-827 F-302

Initial Reviewers: RWP - 7/25/85

Initial Reviewers: ~~LSR~~ EMH EMH 7/25/85

MFT- 7/25/85
Primary Signature Reviewer/Date

(8600PRO/cfr)

(ADVANCE COPY - IDI)

EBASCO SERVICES INCORPORATED

EBASCO

Two World Trade Center, New York, N.Y. 10048

JUL 24 1985

EB-FC-827
File No.: 11.Q.D.6

Mr L I Loflin, Manager
Engineering - Harris Plant
Carolina Power & Light Company
P O Box 101
New Hill, North Carolina 27562

Dear Mr Loflin:

Subject: SHEARON HARRIS NUCLEAR POWER PLANT
FSAR CHANGE NOTICE NO. F-302

Attached for your use and approval is FSAR Change Notice No. F-302. This change notice revises Page 1.8-136 in order to clarify the program in place for determining safety-related, non-technical specification, BOP Setpoints as per IDI Item U6.5-1.

It is recommended that this change notice be incorporated into the next FSAR Amendment. Please advise us of your comments and/or approval of the subject change notice. In accordance with Ebasco's procedures, this change notice will not be considered in effect until formal CP&L approval is received.

If there are any questions, please advise.

Very truly yours,

A C Anderson/WKa

A C Anderson
Project Manager

DMR/vhr
Attachment

cc: All with Attachment

L I Loflin
M Thompson
E Harris
D McCarthy
N J Chiangi
J L Willis
A T Parker
M Shannon (2)

2907/4-77

RECEIVED
JUL 24 1985
D. E. CONNELLY

SAR CHANGE REQUEST

CHANGE NO. F-302

TO W Malec
PROJECT LICENSING ENGINEER

FROM W Pehush
LEAD DISCIPLINE ENGINEER

CLIENT Carolina Power & Light Company

PROJECT Shearon Harris NPP

REQUESTED CHANGE:

☐ PSAR ☒ FSAR Section(s) 1.8 Page(s) 1.8-136

Recommended change and reasons: (Note – Attach marked-up copy of affected SAR pages)

Clarification of actual program in place for determining safety non-tech spec. BOP setpoints. This is a follow up to the IDI Unresolved Item U6.5-1 which is now closed.

Approval W J Pehush Date 7-24-85
LEAD DISCIPLINE ENGINEER

Approval W J Pehush Date 7-24-85
DISCIPLINE SUPERVISOR

LICENSING RECOMMENDATIONS:

☒ Submit with next SAR amendment

☐ Reject proposed change

☐ Hold for FSAR preparation

☐ Other

Notify NRC ☐ Yes ☒ No

Comments

Approval John F. Garibaldi Date 7-24-85
PROJECT LICENSING ENGINEER

DISPOSITION:

Letter to client EB-FC-827

Client letter

☐ Implement

☐ Do not implement

Comments

cc: Project Engineer

TO BE RETAINED IN PLE FILES

d) The accuracy of all setpoints is equal to or better than the accuracy assumed in the safety analysis. Instrument intervals are chosen for the design conditions in which they are installed in order not to anneal, stress relieve, or work harden to the extent that they will not maintain the required accuracy. Design verification is included as part of the equipment qualification program as recommended in Regulatory Guide 1.89.

e) Instruments important to safety have securing devices on the setpoint adjustment mechanism and/or are under administrative control. The securing device is designed such that during securing or releasing it will not alter the setpoint.

f) Documentation of assumptions used in selecting setpoint values and minimum margins, drift rates and test intervals is contained in the Technical Specifications. Chapter 15 also contains documentation of assumptions used in selecting setpoint values.

g) Safety related setpoints not covered by technical specification have sufficient documentation to support the setpoint value, tolerance, and margin to system process limits.

ENCLOSURE 3

FSAR AMENDMENT
REVIEW APPROVAL FORM
(RAF)

ITEM NO. HPES-400
DUE DATE 8/2/85

SUBJECT: FSAR CHANGE NOTICE F-304

FSAR/~~ER~~ SECTION TO BE AMENDED: 9.5.1

CP&L REVIEW RESPONSIBILITY/PRIMARY SIGNATURE REVIEWER:
M. F. THOMPSON

RECOMMENDED CHANGE: (Attach marked-up FSAR/ER page(s) for clarity)

ATTACHED

REASON FOR REVISION:

IDI ITEM D2.4-2

Initial Reviewers: [signature] 8-5-85
JFP- [signature] 8/2/85

Initial Reviewers: ______________
____

[signature] 8/5/85
Primary Signature Reviewer/Date
M.F. THOMPSON, JR.

(8600PRO/cfr)

3907/4-77

SAR CHANGE REQUEST

CHANGE NO. F-304

TO D. CONNELLY
PROJECT LICENSING ENGINEER

FROM M. A. SERBANESCU
LEAD DISCIPLINE ENGINEER

CLIENT CAROLINA POWER & LIGHT CO. PROJECT SHEARON HARRIS

REQUESTED CHANGE:

☐ PSAR ☒ FSAR Section(s) 9.5.1 Pages 9.5.1-50

Recommended change and reasons: (Note – Attach marked-up copy of affected SAR pages) I.D.I Finding D.2.4-2 CABLE TRAY OVERFILL. See Insert 1, Attached.

Approval J. GRANDE / MAS.- Date 7-31-85
LEAD DISCIPLINE ENGINEER

Approval Margareta A Serbanescu. Date 7-31-85
DISCIPLINE SUPERVISOR

LICENSING RECOMMENDATIONS:

☒ Submit with next SAR amendment

☐ Reject proposed change

☐ Hold for FSAR preparation

☐ Other

Notify NRC ☐ Yes ☒ No

Comments

Approval John F. Garibaldi Date 8-1-85
PROJECT LICENSING ENGINEER

DISPOSITION:

Letter to client EB-FC-834

Client letter

☐ Implement

☐ Do not implement

Comments

cc: Project Engineer

TO BE RETAINED IN PLE FILES

combustible materials which were representative of heat values of specific materials grouped within the class. These are:

Ordinary Combustibles	8,000 Btu/lb.
Combustible or Flammable Liquids	20,000 Btu/lb.(108,000 Btu/gal.)
Diesel Fuel Oil (No. 2)	140,000 Btu/gal.
Charcoal	10,000 Btu/lb.

Combustible loading for minor amounts of grease, integral with equipment, not exceeding one lb. each, was not inventoried since it does not create a significant fire hazard.

Using manufacturer's data on specific cable construction used in SHNPP and the Btu content of the insulation materials, Btu values were derived for each running foot (RF) of 24 in. wide, 4-inch deep, 40 percent loaded, power, control and instrumentation cable trays.

These are:

Power	200,000 Btu/RF
Control	170,000 Btu/RF
Instrumentation	155,000 Btu/RF

20

These values are adjusted proportionally for trays of different width or cable tray loading (fill depth). Maximum allowable fill is assumed and is based on plant design criteria of 30 percent for power trays and 60 percent for control and instrumentation trays. The running foot (RF) value reflected in the fire hazard analysis represents cable trays of various widths and depths and should only be used as reference to determine the linear feet of tray.

The combustible loading for all cables routed in conduit, cast concrete trenches, or contained within metallic cabinets or consoles was not inventoried since they do not create a fire hazard.

Based on the above, three fire areas were identified where the Btu content exceeded 240,000 Btu per square foot, assuming all electrical trays filled to maximum capacity. These fire areas were Cable Spreading Rooms 1A, 1B, and the Auxiliary Control (Panel) Room. In order to verify a more accurate combustible load in these fire areas, as opposed to the critical maximum allowable load, the actual cable tray fill as indicated in the Cable and Conduit List was used to calculate average actual tray fill for each cable tray within each of these three fire areas. The percent fill was then used to calculate the combustible load in these three fire areas and represents the actual cable tray fill percentage, including approximately 5 percent additional for potential future use. The resultant combustible loads for these three fire areas are shown in Appendix 9.5A.

In addition to the combustibles normally present in an area, transient combustibles which might realistically be introduced into areas as a part of planned operation are considered, as detailed in fire hazard analysis for each

INSERT 1 (Attached)

INSERT 1

The cable insulation combustible loading will be REVISED TO reflect the LATEST INFORMATION ~~verified prior to core load~~ after cable routing is complete. The Cable and Conduit List will be used to determine actual cable tray fill. In certain instances, actual cable tray fill may be permitted to exceed plant design criteria. Since this situation is not expected to occur for the entire length of a cable tray contained in a fire area or zone, the actual average fill for that cable tray is not expected to exceed the values assumed in the combustible loading calculations. If the actual average fill for a cable tray exceeds plant design criteria, the actual average fill for all cable trays in the respective fire area or zone will be calculated. Should the overall actual average cable tray fill for the fire area or zone exceed plant design criteria, the combustible loading calculation and Fire Hazards Analysis will be revised accordingly.

ENCLOSURE 4

FSAR AMENDMENT
REVIEW APPROVAL FORM
(RAF)

ITEM NO. HPES-401
DUE DATE 8/2/85

SUBJECT: FSAR CHANGE NOTICE F-303

FSAR/ER SECTION TO BE AMENDED: 12.2.1, 12.3.2

CP&L REVIEW RESPONSIBILITY/PRIMARY SIGNATURE REVIEWER:
M.F. THOMPSON

RECOMMENDED CHANGE: (Attach marked-up FSAR/ER page(s) for clarity)

ATTACHED

REASON FOR REVISION:

IDI ITEM D2.5-3

Initial Reviewers: [signature] 8/2/85
[initials]

Initial Reviewers: ______________
____

[signature] 8/5/85
Primary Signature Reviewer/Date
M.F. THOMPSON, JR.

(8600PRO/cfr)

2807/4-77

JUL 24 1985

D. E. CONNELLY

SAR CHANGE REQUEST

CHANGE NO. F-303

TO D E Connelly (PROJECT LICENSING ENGINEER) FROM V. Chitnis (LEAD DISCIPLINE ENGINEER)

CLIENT C P & L PROJECT SHEARON HARRIS

REQUESTED CHANGE:

☐ PSAR ☒ FSAR Section(s) 12.2.1.12 Page(s) 12.2.1-4, 12.2.1-32

Recommended change and reasons: (Note – Attach marked-up copy of affected SAR pages)

As stated in F-299 the source strength changed. Revised Tables indicate updated source strengths. Page 12.2.1-32 (Table 12.2.1-28) is deleted along with its reference on page 12.2.1-4.

Approval V. Chitnis (LEAD DISCIPLINE ENGINEER) Date 7/22/85

Approval S. Othm (DISCIPLINE SUPERVISOR) Date 7/23/85

LICENSING RECOMMENDATIONS:

☐ Submit with next SAR amendment ☐ Reject proposed change

☐ Hold for FSAR preparation ☐ Other

Notify NRC ☐ Yes ☐ No

Comments The deletion of Table 12.2.1-28 address IDI item 2.5-3 regarding the "additional 20% cesium source term"

Approval (PROJECT LICENSING ENGINEER) Date

DISPOSITION:

Letter to client EB-FL-835 8/1/85

Client letter

☐ Implement

☐ Do not implement

Comments

cc: Project Engineer

TO BE RETAINED IN PLE FILES

time. The activity concentrations on both demineralizers reflect annual resin replacement.

12.2.1.11 Reactor Startup Neutron Sources

The primary and secondary reactor startup neutron sources (see Section 4.2) remain in the reactor for the duration of their useful life and thus are not considered as a separate radiation shielding problem.

Only the primary reactor startup neutron source is initially radioactive and is adequately shielded when shipped to, and handled at the site.

12.2.1.12 Post TMI Shielding Review Accident Sources

The source terms used in the estimation of radiation doses are based on those supplied by Westinghouse for this FSAR shown in Table 12.2.1-23 and those required by the NRC (Regulatory Guides 1.3, 1.4, 1.7, and SRP 15.6.5).

From the total core inventory, 100 percent of noble gases, 50 percent of halogens, and one percent of the other solids are used to determine the source terms for the reactor coolant. These are tabulated according to energy groups at various time intervals in Table 12.2.1-25. In addition, the source terms have also been determined which take into account the dilution by refueling
3 water storage tank volume and are shown in Table 12.2.1-26.

The gaseous source terms for gamma radiation within the containment atmosphere are based on 100 percent of the core noble gas inventory and 25 percent of the core halogen inventory. These are tabulated in Table 12.2.1-27.

For the long term radiation sources, 20 percent Cesium inventory is added. These are shown in Table 12.2.1-28 at time one year after release of activity.

It must be emphasized that 100 percent noble gas core inventory is included in source terms for liquid systems as well as for containment air. Hence, the calculated dose rates are extremely conservative.

INSERT A

3.1 years. Activity decay is taken for the minimum waiting period of three days following shutdown.

c) The shielding design of the fuel pools is based upon the loadings described in Section 9.1.

Shielding for the Spent Fuel Pool Cooling and Cleanup System is based upon source terms derived from normal system operation as described in Sections 9.1.3 and 11.1 in conjunction with approximately two-thirds of a core
15 stored in the pool and fuel clad defects in the fuel rods which generate one percent of rated core thermal power. Refer to Sections 9.1.1 and 9.1.2 for a discussion on the capacity of the fuel pools.

12.3.2.14 Control Room

For the purpose of designing control room shielding, the radioactivity releases from the maximum loss-of-coolant accident (LOCA) are controlling. The two sources considered in designing the shielding were:

a) Direct gamma radiation from the containment atmosphere and from radioactivity collected on emergency filters.

TID-14844 (Reference 12.3.2-8) source terms, release fractions and plate-out fractions are considered. A ~~zero containment leak rate is assumed for the duration of the accident. A uniform distribution of radioactivity within the Containment is assumed. No credit is taken for iodine removal from the atmosphere by sprays.~~ Credit for post-accident decay is considered.
INSERT B

The direct whole body dose to control room personnel following a LOCA is computed to be less than 3.0 rem in 90 days. Credit for 8.5 ft. of concrete (4.5 ft. for thecontainment shield wall and 4 ft. for the control room shielding) was taken. A minimum shield thickness of 6 ft. separates the emergency filters from the Control Room.

In addition to shielding from external exposures, the Control Room is designed to operate in an isolated mode under accident conditions in order to minimize the quantity of airborne radioactivity which enters the Control Room and thereby ensure compliance with GDC 19 of 10CFR50 Appendix A. A detailed description of the system design is provided in Section 9.4.1. Chapter 15 includes an evaluation of the exposures to control room personnel following the design basis accident.

b) Direct gamma radiation from radiation leakage external to containment

In addition to direct shine exposure from airborne radioactivity in the Containment following a design basis loss-of-coolant accident, control room personnel may receive a small exposure due to external exposure to the passage of the gaseous plume which could result from containment leakage. Assuming the TID-14844 source terms, an integrated containment leak rate of 0.1 percent per day for the first 24 hours post LOCA and 0.05 percent per day for the remainder of the accident and five percentile meteorology, the integrated exposure (30 days) to control room personnel from this source of exposure will not exceed 0.1 rem.

INSERT A

12.2.1.12 Post TMI Shielding Review Accident Sources

The source terms used in the estimation of radiation doses are based on the Westinghouse Radiation Analysis Manual (WRAM-Reference 12.2.1-4). The plant specific core design parameters are stated in WRAM Table 5-1. Noble gas and halogen inventories are given in WRAM Table 5-9; while solid fission products from the spent fuel fission products inventory at shutdown are given in WRAM Table 5-29. The values are converted to a total core inventory basis. In addition fission products not mentioned in WRAM but which are described in Reference 12.2.1-5 have been included in the evaluation. Decay daughter dose contributions for all fission products have been included in the inventory.

The model assumes removal by plate-out using a mechanistic model considering elemental, particulate and organic fractions of solid fission products. Plate-out source strengths in containment are given in Table 12.2.1-25. The model also assumes mechanistic spray removal and dilution of the released 50 percent halogen core inventory and 1 percent solid fission products inventory source terms with the combined volumes of the Reactor coolant, accumulators and the refueling water storage tank. The resulting liquid activity as a function of time is given in Table 12.2.1-26. (Note to CP&L - This Table was revised via F-299, EB-FC-825).

The gaseous source terms for gamma radiation within the containment atmosphere are based on 100 percent of the core noble gas inventory, 50 percent of the core halogen inventory, and 1 percent solid fission products inventory. The assumptions of removal by plate-out and spray are as stated above. These values are tabulated in Table 12.2.1-27.

INSERT B

12.3.2.14 Control Room

The model assumes removal by plate-out using a mechanistic model considering elemental, particulate and organic fractions of halogens as well as the particulate fractions of solid fission products. Plate-out source strengths in containment are given in Table 12.2.1-25. Credit is taken for iodine removal from the atmosphere by sprays. Credit for post-accident decay is considered.

References: Section 12.2

12.2.1-5 Kolar Michael J. and Olson Nolan C., "Calculation of Accident Doses to Equipment inside Containment of Power Reactors," Nuclear Technology, Vol. 36, November 1977, pages 56-64.

EBASCO SERVICES INCORPORATED

BY ______ DATE ______ SHEET ______ OF ______

CHKD. BY ______ DATE ______ OFS NO. ______ DEPT. NO. ______

CLIENT ______

PROJECT note: This replaces the Am. No. 3 Table 12.2.1-25

SUBJECT ______

TABLE 12.2.1-25

CONTAINMENT PLATE-OUT SOURCE STRENGTHS

Energy Group	Source Strengths at Time after Release (γs/cm²-sec)						
MeV/γ	1 Hour	1 Day	3 Days	7 Days	14 Days	1 Month	1 Year
0.00 – 0.106	2.51(+7)	3.16(+7)	2.31(+7)	1.51(+7)	9.03(+6)	3.84(+6)	7.05(+5) *
0.106 – 0.440	8.05(+8)	5.62(+8)	4.65(+8)	3.27(+8)	1.81(+8)	4.97(+7)	6.32(+5)
0.440 – 0.865	3.92(+9)	7.97(+8)	2.42(+8)	9.68(+7)	6.97(+7)	4.59(+7)	5.65(+6)
0.865 – 1.332	1.82(+9)	1.76(+8)	5.95(+7)	4.51(+7)	3.92(+7)	3.00(+7)	1.65(+5)
1.332 – 1.720	1.53(+8)	1.26(+8)	8.65(+7)	4.81(+7)	1.93(+7)	2.54(+6)	6.35(+4)
1.720 – 2.210	4.81(+8)	3.24(+7)	4.86(+5)	6.51(+4)	5.46(+3)	9.97(+1)	2.95(-9)
2.210 – 2.754	3.65(+7)	3.57(+6)	2.12(+5)	3.65(+4)	2.02(+3)	2.69(+0)	– 0 – –
2.754 – 3.930	4.60(+6)	– 0 –	– 0 –	– 0 –	– 0 –	– 0 –	– 0 – –

* Power of ten

Note: Numbers in parentheses indicate powers of ten

12.2.1-29

EBASCO SERVICES INCORPORATED

BY ________ DATE ________ SHEET ____ OF ____

CHKD. BY ________ DATE ________ OFS NO. ________ DEPT. NO. ________

CLIENT ________

PROJECT ________

SUBJECT ________

TABLE 12.2.1 – 27
CONTAINMENT AIRBORNE ACTIVITY

Energy Group	Source Strengths at Time after Release (γs/cc-sec)							
Mev/γ	Zero	1 Hour	1 Day	3 Days	7 Days	14 Days	1 Month	1 Year
0.00 – 0.106	4.43(+7)	3.95(+7)	3.24(+7)	2.45(+7)	1.42(+7)	5.65(+6)	7.11(+5)	2.01(-1)*
0.106 – 0.440	2.33(+8)	5.57(+7)	4.78(+6)	1.13(+6)	5.70(+5)	2.60(+5)	6.27(+4)	1.94(-1)
0.440 – 0.865	2.97(+8)	1.71(+7)	1.08(+6)	2.44(+5)	5.05(+4)	2.42(+4)	7.08(+3)	1.24(+3)
0.865 – 1.332	1.49(+8)	1.06(+7)	1.91(+5)	1.47(+4)	5.73(+2)	2.14(+0)	7.95(-3)	1.51(-4)
1.332 – 1.720	3.95(+7)	7.49(+6)	2.95(+4)	3.51(+3)	1.44(+2)	5.49(-1)	---	---
1.720 – 2.210	6.43(+7)	1.08(+7)	6.92(+4)	2.92(+2)	3.19(-2)	1.65(-3)	---	---
2.210 – 2.754	2.29(+7)	1.21(+7)	3.76(+4)	2.89(+1)	1.29(-2)	6.51(-4)	--	--
2.754 – 3.930	1.34(+7)	1.28(+4)	----	----	----	----	----	---

* Power of ten

Note: Numbers in parenthese indicate powers of ten

TABLE 12.2.1-28

ACTIVITIES AFTER ONE YEAR INCLUDING

20% CESIUM INVENTORY

Energy Group Mev/γ	Source Strength Undiluted Coolant	(γs/cc-sec). Diluted Coolant
0.2 - 0.4	5.62 (+6)	8.37 (+5)
0.4 - 0.9	9.94 (+8)	1.49 (+8)
0.9 - 1.35	1.12 (+7)	1.67 (+6)
1.35 - 1.8	1.21 (+7)	1.80 (+6)
1.8 - 2.2	4.25 (+5)	6.31 (+4)

3

Note: Numbers in parentheses indicate power of ten.

Delete

ENCLOSURE 5

FSAR AMENDMENT
REVIEW APPROVAL FORM
(RAF)

ITEM NO. HPES-402
DUE DATE

SUBJECT: FSAR CHANGE NOTICE F-299

FSAR/ER SECTION TO BE AMENDED: 3.11A, 12.2

CP&L REVIEW RESPONSIBILITY/PRIMARY SIGNATURE REVIEWER:

M.F THOMPSON

RECOMMENDED CHANGE: (Attach marked-up FSAR/ER page(s) for clarity)

ATTACHED

REASON FOR REVISION:

IDI ITEM D 2.5-5

Initial Reviewers: 8/2/85

Initial Reviewers:

8/5/85
Primary Signature Reviewer/Date
M.F. THOMPSON, JR.

(8600PRO/cfr)

14XFT-x-x-x

EBASCO SERVICES INCORPORATED

Two World Trade Center, New York, N Y 10048

EBASCO

JUL 19 1985

EB-FC-825
File No.: 11.Q.D.6

Mr L I Loflin, Manager
Engineering - Harris Plant
Carolina Power & Light Company
P O Box 101
New Hill, North Carolina 27562

Dear Mr Loflin:

Subject: SHEARON HARRIS NUCLEAR POWER PLANT
FSAR CHANGE NOTICE NO. 299

Attached for your use and approval is FSAR Change Notice No. F-299. This change notice revises pages 3.11A-5, 7 and Table 12.2.1-26 as a result of using a mechanistic model for the removal of activity by plate-out and sprays.

It is recommended that this change notice be incorporated into the next FSAR Amendment. Please advise us of your comments and/or approval of the subject change notice. In accordance with Ebasco's procedures, this change notice will not be considered in effect until formal CP&L approval is received.

If you have any questions, please advise.

Very truly yours,

AC Anderson/wta

A C Anderson
Project Manager

DMR:maa
Attachment

cc: All with Attachment

L I Loflin
M Thompson
E Harris
D McCarthy
N J Chiangi
J L Willis
A T Parker
M Shannon (2)

REC'D JUL 25 1985

2807/4-77

SAR CHANGE REQUEST

CHANGE NO. F-597

TO D E Connelly (PROJECT LICENSING ENGINEER) FROM V. Chitnis (LEAD DISCIPLINE ENGINEER)

CLIENT C P & L PROJECT SHEARON HARRIS

REQUESTED CHANGE:

☐ PSAR ☒ FSAR Section(s) APPENDIX 3.11A, 12.2.1 Page(s) 3.11A-5, -7, 12.2.1-30

Recommended change and reasons: (Note – Attach marked-up copy of affected SAR pages) Pages enclosed

As a result of using mechanistic model for the removal of activity by plate-out and sprays, assumptio 1.4 on page 3.11A-5, (5) on page 3.11A-7, and Table 12.2 on page 12.2.1-30 require revisions.

Approval V. Chitnis (LEAD DISCIPLINE ENGINEER) Date 7/10/85

Approval Stephen [illegible] (DISCIPLINE SUPERVISOR) Date 7/10/85

LICENSING RECOMMENDATIONS:

☑ Submit with next SAR amendment ☐ Reject proposed change

☐ Hold for FSAR preparation ☐ Other

Notify NRC ☐ Yes ☑ No

Comments

Approval Dean E Connelly (PROJECT LICENSING ENGINEER) Date 7-19-85

DISPOSITION:

Letter to client EB-FD-825 Client letter

☐ Implement

☐ Do not implement

Comments

cc: Project Engineer

TO BE RETAINED IN PLE FILES

APPENDIX 3.11A

NUREG-0588 COMPARISON

CATEGORY II

Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
can maintain its required functional operability if its surface temperature reaches the calculated value or (ii) requalification testing be performed with appropriate margins, or (iii) qualified physical protection be provided to assure that the surface temperature will not exceed the actual qualification temperature.	
1.3 Effects of Chemical Spray	
The effects of caustic spray should be addressed for the equipment qualification. The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system. If the chemical composition of the caustic spray can be affected by equipment malfunctions, the most severe caustic spray environment that results from a single failure in the spray system should be assumed. See SRP Section 6.5.2 (NUREG-75/087), paragraph II, item (e) for caustic spray solution guidelines	1.3 The most severe containment spray environment (boron concentration and pH level) is used for environmental qualification. The actual (calculated) spray environment bounds any postulated single failure.
1.4 Radiation Conditions Inside and Outside Containment	
The radiation environment for qualification of equipment should be based on the normally expected radiation environment over the equipment qualified life, plus that associated with the most severe design basis accident (DBA) during or following which that equipment must remain functional. It should be assumed that the DBA related environmental conditions occur at the end of the equipment qualified life.	1.4 For qualification purposes, reductions in air dose due to spray washout and plateout are [redacted] used in calculating the post-accident radiation environments. Therefore, radiation doses used in qualification are maximum total integrated dose calculated over the equipment qualified life, plus that associated with the most severe design basis accident.

16

APPENDIX 3.11A

NUREG-0588 COMPARISON

CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971	Shearon Harris Nuclear Power Plant Program
the drywell. The assumption of uniform distribution of activity throughout the containment at time zero is not appropriate.	the steam generators and the reactor vessel and since the Containment Spray and/or the containment ventilation and filtration systems provide mixing for the containment atmosphere, a determination was made to assume a uniform distribution of activity throughout the containment.
(4) Effects of ESF systems, such as containment sprays and containment ventilation and filtration systems, which act to remove airborne activity and redistribute activity within containment, should be calculated using the same assumptions used in the calculation of offsite dose. See SRP Section 15.6.5 (NUREG-75/087) and the related sections referenced in the Appendices to that section.	(4) Credit for the removal of airborne activity by ESF systems has been taken. In addition, the distribution of activity is taken into account as described in (3) above and by (5) below.
16 (5) Natural deposition (i.e., plate-out) of airborne activity should be determined using a mechanistic model and best estimates for the model parameters. The assumption of 50 percent instantaneous plate-out of the iodine released from the core should not be made. Removal of iodine from surfaces by steam condensate flow or washoff by the containment spray may be assumed if such effects can be justified and quantified by analysis or experiment.	INSERT A (5) ~~The SHNPP model assumes zero removal for plate-out; however, the containment sump source terms are developed by assuming dilution of 50 percent of the core inventory of halogens and 1 percent of other nuclides with the combined volumes of the Reactor Coolant, Accumulators, Boron Injection Surge Tanks and the Refueling Water Storage Tank. The resulting initial sump (diluted coolant) activity is given on FSAR Table 12.2.1-26.~~
(6) For unshielded equipment located in the containment, the gamma dose and dose rate should be equal to the dose and dose rate at the centerpoint of the containment plus the contribution from location dependent sources such as the sump water and plate-out, unless it can be shown by analyses that location and shielding of the equipment reduces the dose and dose rate.	(6) The gamma dose and dose rate used in qualification for equipment located inside containment is calculated for various zones utilizing distance and shielding credits. Refer to FSAR Appendix 3.11B for applicable doses in various zones.

(5) The SHNPP model assumes removal by plate-out using a mechanistic model considering elemental, particulate, and organic fractions of halogens as well as the particulate fractions of solid fission products.

The SHNPP model also assumes mechanistic spray removal and dilution of the 50 percent halogen core inventory and 1 percent solid fission products inventory source terms with the combined volumes of the Reactor Coolant, Accumulators, and the Refueling Water Storage Tank. The resulting sump activity as a function of time is given on FSAR Table 12.2.1-26.

INSERT A

TABLE 12.2.1-26

SUMP ACTIVITY

Energy Group	Source Strengths at Time after Release (γs/cc-sec)						
MeV/γ	1 Hour	1 Day	3 Days	7 Days	14 Days	1 Month	1 Year
0.00 - 0.106	4.12(+7)	5.43(+7)	3.89(+7)	2.51(+7)	1.52(+7)	6.78(+6)	1.65(+6)*
0.106 - 0.440	1.21(+9)	8.58(+8)	7.06(+8)	4.95(+8)	2.74(+8)	7.72(+7)	1.55(+6)
0.440 - 0.865	5.81(+9)	1.23(+9)	3.91(+8)	1.69(+8)	1.26(+8)	8.51(+7)	1.29(+7)
0.865 - 1.332	2.67(+9)	1.95(+8)	2.12(+7)	3.37(+6)	1.61(+6)	9.42(+5)	1.24(+5)
1.332 - 1.720	1.65(+8)	4.29(+7)	1.48(+7)	2.70(+6)	4.43(+5)	1.86(+5)	1.31(+5)
1.720 - 2.210	7.10(+8)	4.86(+7)	8.38(+5)	1.23(+5)	1.04(+4)	1.90(+2)	----
2.210 - 2.754	5.42(+7)	5.49(+6)	3.95(+5)	6.98(+4)	3.85(+3)	5.13	----
2.754 - 3.930	6.83(+6)	----	----	----	----	----	----

* Power of ten

ENCLOSURE 6

The following documents activities completed by Applied Physics in response to the IDI, as well as the completion of or commitment to activities that were outstanding at the time of the reinspection exit interview:

1) SHIELDING CALCULATIONS

All the calculations noted in the audit that had deficiencies have been marked "Superseded" and have been replaced by a new set of calculations. These new calculations were inspected by the IDI team. As a result of the new calculations, the integrated dose to equipment increases in certain areas. Although this increase may not affect equipment qualification since most equipment is qualified to the highest "envelope" dose, Applied Physics has issued a revised document for Equipment Qualification Doses, and the Project EQ team is conducting a formal review of equipment in the affected zones.

Applied Physics is also conducting a review of other calculations that might have been affected by the superseded calculations. This review is ongoing, but so far no affected calculation has been identified.

The revised calculations in some respects use a different methodology than reported in the FSAR. FSAR Change Notices have been issued to revise the applicable sections.

At the time of the reinspection exit interview certain items pertaining to the calculations were considered "open" and these are addressed below:

a. Deficiency 2.5-1. New calculation 040 for the Volume Control Tank referenced the FSAR for source terms. Revision 1 of this calculation has now been issued with the FSAR referenced as a design input deleted, and a reference to the latest Westinghouse source term document added as the design input.

b. Deficiency 2.5-3 New calculation 041 for the TM1 Shielding Design Review was improperly revised by crossing out portions after it had been signed by the verifier. A formal revision 1 to this calculation has been issued, noting previously revised pages and deleting any FSAR references as the design input.

c. Deficiency 2.5-4 through 2.5-7. New calculations for Equipment Qualification had not been completed in that Beta daughter contribution had not yet been added and revised EQ Dose Maps had not been issued. The Beta daughter contribution has now been calculated and revised EQ Dose Maps have been issued to project personnel as SHNPP sketches. Affected calculations have been revised to reflect this as well as deletion of FSAR references as design inputs. The assessment of electrical equipment qualification based on the revised Dose Maps is an ongoing effort.

2) WESTINGHOUSE SOURCE TERMS

The inspection of the Volume Control Tank (VCT) calculation indicated that in 1979 Westinghouse issued a revised source term manual. We have redone the VCT calculation using the revised manual; however, there exist other calculations performed prior to December 1979 that used the older manual for data input. Applied Physics will review all shielding calculations performed prior to 12/79 to see if they used Westinghouse source terms. If the old Westinghouse source terms were used, the impact of the revised manual will be assessed and noted in the calculation. Where the source terms have changed, the calculation will be revised.

3) REVIEW OF PAST CALCULATIONS

Additional Applied Physics calculations have been reviewed to determine if the deficiencies noted in the IDI were typical and to assess the departmental design verification effort. This review was done in two parts; first, a formal design re-verification of selected calculations; secondly, an informal review of a larger number of calculations. The review was conducted under the direction of the Chief Engineer of Applied Physics by various engineers in the department.

Ten Shearon Harris calculations were selected to cover a cross section of typical Applied Physics work. These included 5 shielding calculations. 3 Thermal-Hydraulic (dynamic fluid loads, heat transfer) and 2 Applied Mechanics (special stress analyses). A preponderance of shielding calculations was chosen because that was the area noted in the IDI. The calculations selected had been performed at various times from 1976 through 1984, and all were previously verified. The reviewers were directed to do a complete formal check and design re-verification of each calculation to determine if the calculation's conclusions were correct, and to note any deficiencies. A deficiency was defined as any deviation from Ebasco procedure E-30 or any technical error. The reviewers were instructed to pay particular attention to the type of deficiencies noted by the IDI team. However, the effect of the new Westinghouse source term manual noted in item #2 above was to be addressed separately.

This re-verification effort found that all 10 calculations had valid results. No significant deficiencies were found in either the Thermal-Hydraulic or the Applied Mechanics calculations. Some deficiencies were noted in the shielding calculations, mainly in the documentation and use of data input. However, all the results were determined to be correct. These 10 calculations along with the reviewers comments and supporting documentation, such as reviewers independent calculations, were shown to the IDI team during the reinspection.

Recognizing the small sample size of the 10 calculations and the fact they they were restricted to the Shearon Harris project, a second larger review was performed. This review was conducted by the Chief Engineer and the three Applied Physics supervisors. It covered approximately 100 calculations across all projects. This review was not as rigorous as the re-verification effort described above; but through spot checking, examination of supporting documentation and a review of the methodology as presented, it was deemed sufficient to confirm the quality level of the calculation and the design verification. While this review was not without some findings, it did indicate a high quality level of the technical content and that design verification was conscientiously applied.

3) REVIEW OF PAST CALCULATIONS (Cont'd)

As a result of these reviews, the Applied Physics Chief Engineer has concluded the following:

a. Applied Physics calculations are being performed in an accurate manner and a workable design verification effort has existed.

b. The deficiencies noted in the Shearon Harris shielding calculations were not typical of Applied Physics calculations in general. Corrective action in this specific area was called for which includes a continuing review of the Shearon Harris shielding effort, more direct supervisory participation, and the assignment of a senior radiation protection engineer to the Shearon Harris project.

c. While the Applied Physics design verification effort was workable, the review has indicated certain weaknesses that must be upgraded and actions must be taken to prevent the deficiencies noted during the IDI from recurring. Departmental corrective actions are detailed in items 4 and 5 below.

4) DESIGN VERIFICATION UPGRADE

To prevent recurrence of the deficiencies noted during the IDI, a program has been instituted to upgrade the design verification effort. This program has three aspects:

a. Formal Training

Over the past several months, formal training sessions have been conducted covering those Ebasco procedures that affect AP work. Most Applied Physics' members have completed these sessions, the exceptions being those members currently on field assignments or otherwise unavailable. These sessions will continue until all members have completed them, and they will be repeated periodically if deemed necessary.

b. Verification Awareness

A series of department directives have been issued and informal meetings and discussions have been held to increase the awareness, by the engineers, of the requirements inherent in producing verified calculations. The IDI report has been used as an example for the engineers. A heightened awareness among the individuals in the department has been evident as a result of this program; and these measures will continue.

c. Direct Supervisory Participation

Discussions have been held with all Applied Physics supervisors regarding design verification, and they have been directed to become directly involved in the design verification effort by training their group, identifying individuals that need improvement, and directly reviewing completed calculations to assess the quality of the calculation and verification. The supervisor signs the Shearon Harris calculation as objective evidence of this review and its acceptability for project application. This effort is ongoing.

5) ADVANCED TECHNOLOGY TECHNICAL REVIEW

Due to the highly technical nature of the work done in Applied Physics and other departments under Advanced Technology, Dr R C Iotti, Vice President of Advanced Technology, has announced the formation of a Technical Review Committee. The charter and procedure for this committee are in the formulation stage, however the committee will consist of senior level personnel within Advanced Technology. The committee will have the mission and organizational freedom to assess the technical accuracy or quality of any design document calculation, study or computer code produced in Advanced Technology. The committee will have the freedom to obtain necessary expertise outside the originating department, or outside Ebasco if necessary, to perform its function.

ENCLOSURE 7

Ebasco Quality Assurance has recently completed an audit of eleven Applied Physics (AP) calculations. This audit was initiated in April of this year to review those AP calculations that were performed to address the concerns identified during the Integrated Design Inspection (IDI). The purpose of the audit was to ensure that the corrective measures initiated in AP, as reflected in the new calculations, satisfactorily address the IDI concerns.

The eleven calculations which were the subject of this audit were of a radiation dose/shielding nature reflecting the types of calculations reviewed during the IDI. The audit was conducted in two phases, the first of which was initiated in April and completed in early July of this year. During this phase, which covered two of the eleven AP calculations, nine concerns were identified. These concerns were classified into the following five areas:

1. Unsatisfactory compliance with requirements to identify sources of design inputs and assumptions, justification of selection of inputs, and identification of assumptions and inputs that must be confirmed as the design progresses (five items).

2. Incomplete documentation of design review/checking (one item).

3. Written presentation with characteristics that may not be reproducible (one item).

4. Clerical inaccuracy in presentation of calculation (one item).

5. Documentation not available from official files showing the basis for acceptability of commercial computer program used by AP (SPAN 4).

These concerns were formally transmitted to the department chief engineer and group supervisor for corrective action.

The second phase of the audit was conducted in the latter half of July, during which the remaining nine AP calculations were reviewed. No additional concerns were identified during this period.

In total, the audit covered hundreds of pages of calculations. The nine concerns identified during the first phase, which have all been corrected by AP, did not affect the technical bases of the calculations. Taken alone, the results of this first phase indicate that the programs instituted in AP subsequent to the IDI have yielded substantial improvement. The completion of the second phase with no additional concerns demonstrate that these programs in conjunction with internal audits, modified to include written feedback to the chief engineer and his written response, will result in calculations fully in compliance with industry and corporate quality standards.

ENCLOSURE 8

CIRCUIT AND RELAY CHANGES

The following items were closed by the IDI team; however, implementation is subject to inspection by NRC Region II.

Unresolved Item	U5.3-1
Deficiency	D5.4-1
Deficiency	D5.4-2
Deficiency	D5.4-3
Deficiency	D5.4-4
Deficiency	D5.5-2
Deficiency	D5.5-3
Deficiency	D.5.-4
Deficiency	D5.6-1
Deficiency	D5.7-1
Deficiency	D5.8-1
Deficiency	D5.9-1
Deficiency	D5.10-1