SOUTH CAROLINA ELECTRIC & GAS COMPANY

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T. C. NICHOLS, JR.
VICE PRESIDENT AND GROUP EXECUTIVE
NUCLEAR OPERATIONS

May 12, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Responses to Comments on SCE&G NUREG 0588 Report, Rev. 4

Dear Mr. Denton:

On February 19, 1982, South Carolina Electric and Gas Company (SCE&G) submitted to you Revision 4 of the SCE&G report on NUREG 0588, Equipment Qualification. Attached we provide responses to NRC staff comments on our February 19, 1982, submittal and a sample radiation calculation as requested by the staff. These responses were discussed previously with the NRC in recent telephone communications. This attachment should provide all the required information to address the NRC's concerns.

If you have additional questions, please let us know.

Very truly yours,

T. C. Nichols, Jr.

T. C. Nichols, Jr.

NEC:TCN:lkb
Attachment
cc: V. C. Summer (w/o attach.)
G. H. Fischer (w/o attach.)
H. N. Cyrus
T. C. Nichols, Jr. (w/o attach.)
M. B. Whitaker, Jr.
J. P. O'Reilly
H. T. Babb
D. A. Nauman
C. L. Ligon (NSRC)
W. A. Williams, Jr.
R. B. Clary
O. S. Bradham
A. R. Koon
M. N. Browne
G. J. Braddick
J. C. Ruoff
J. L. Skolds
J. B. Knotts, Jr.
B. A. Bursey
NPCF
File

A046
51/40 on Shelf

1. COMMENT:

Rev. 4 of the submittal did not contain the additional data that VCSNS was to provide supporting the conclusion that the mechanical integrity of penetrations with non-1E circuits will not be degraded when these circuits are submerged.

RESPONSE:

This concern is discussed in the V. C. Summer FSAR Response to Regulatory Guide 1.63, Electrical Penetration Assemblies In Containment Structure For Water-Cooled Nuclear Power Plants, (pg. 3A-94). This discussion describes the basis for the qualification of the penetrations to maintain containment integrity even with the single failure of any overcurrent protective device.

2. COMMENT:

VCSNS commited, during the audit, to provide proof of qualification of the Gulf Oil Company lubricants by fuel load. The Rev. 4 submittal states documentation of qualification is under development and will not be completed by fuel load, but by full power operation. Justification for interim use should be provided. Also, the lubricants are not listed in Table 1.2-1 "Class 1E Equipment Subject to Harsh Environmental Conditions, NSSS Tables, nor BOP Tables.

RESPONSE:

The Summer Station has an approved program for the specification of lubricants for safety-related use. Oils and greases are procured to industry standards which envelope the design conditions expected to be seen by the lubricant. These industry standards, however, do not address the radiation resistance properties of the lubricants. The specific concern is that insufficient documentation exists to show that the

lubricants used in certain Class 1E motors are capable of performing their function in the harsh environmental conditions postulated to occur following certain accidents. Included in this concern are the lubricants used with certain mechanical equipment associated with the motors.

Westinghouse (W) has provided qualification data on certain Mobil and Texaco oils and Chevron SRI-2 grease used in W equipment that were tested at W testing facilities. It is reasonable to expect that W qualification data on I.S.O. grade 46 oils can be used to extrapolate qualification justification for I.S.O. grade 32 and 68 oils of the same family. Discussions with manufacturers' representatives indicated that oils of the same family are formulated of the same identical ingredients and additives but are blended to differing viscosities. Our latest information is that the results of the radiation testing for the oils by W has not been documented in an official report. However, there is no specific documentation that critical breakdown of the oils would be expected by 2 x 10^8 rads. There is also sufficient information from W in the form of letters and test results to have a degree of assurance in the radiation resistance of these oils. The assurance of radiation resistance for the Chevron SRI-2 grease is better documented with information from W as well as from Chevron.

By letter of August 4, 1981, Gulf Research and Development Company (Gulf R & D) provided the available qualification data for Gulf lubricating oil and grease. The letter notes that the data for grease is preliminary and the data for oil is dated and should not be applied to current Gulf products.

A telecon was held on December 18, 1981, with Mr. G. J. Schreuders of Gulf R & D. The testing of the Gulfcrown EP-2 grease is now completed up to 3 x 10^8 rads using a slow gamma radiation source. Concerning the radiation testing of the Gulf Harmony series of oils, Gulf has contracted with National Laboratories at Oak Ridge to do radiation testing of Gulf Oils. The Harmony series will be among the first group of oils tested.

Mr. Schreuders forwarded this information by letter dated December 21, 1981, to SCE&G. In further discussions with Mr. Schreuders on April 23, 1982, the test plan for radiation testing has been finalized and testing is expected to commence within two or three weeks. It may be three or four months before the results of the testing are available. Mr. Schreuders expects the results to be similar to those of a 1969 Gulf radiation testing program which showed degradation starting at 4×10^7 rads but the oil (Gulf Harmony 32) was still considered suitable for further service after 1×10^8 rads. Since the time of this study some changes have been made in the type and character of the additives used in these oils. The information therefore cannot be specifically applied to today's product, hence the new testing program soon to be underway

It is the judgement of SCE&G Nuclear Engineering that we procure Gulf lubricants as stipulated in our lubrication manual. The Gulfcrown EP-2 grease has been judged acceptable for use by Gulf R & D after exposure to 1.5×10^8 rads. Chevron SRI-2 grease has been tested by Westinghouse and is not expected to breakdown until levels higher than 2×10^8 rads. Either of these greases are acceptable. Westinghouse has tested the Mobil DTE Medium and the Texaco Regal "B" oils (industry alternates to Gulf Harmony #46) to 1×10^8 rads Critical breakdown of these oils is expected by 2×10^8 rads. Gulf R & D is contracting their own independent radiation testing of their oils to include the Harmony series with results expected in 1982. The Gulf Harmony series of oils, which meet the same industry lubrication specifications as its Texaco and Mobil counterparts, cannot be considered qualified based on the Westinghouse radiation testing programs. SCE&G Nuclear Engineering is not aware, however, of any evidence that Gulf oils should be considered suspect to significant radiation degradation and does not see sufficient reason to justify a change to the Mobil and Texaco oils until the results of the Gulf contracted testing program would show a need to do so. When the testing program is completed, the results will be reviewed for adequate radiation resistance for the intended usage of the Gulf oils, and any action required in switching to Mobil or Texaco oils would be taken at that time.

The establishment of qualification information on a series of lubricants does not relieve SCE&G from the responsibility of maintaining an "on going" assurance of quality throughout future procurements. This will be accomplished by establishing appropriate means to verify that the lubricants are equal or better than that which was qualified. To this end, Quality Assurance has investigated a bearing manufacturer's controls over lubricants in sealed bearings. Assurance that lubricants themselves are of a consistently "equivalent" quality shall be established through procurement requirements developed by Engineering in cooperation with QA. Manufacturer controls over product batch testing, identification and sealing of containers, product distribution, and our own history or satisfactory product use are some of the means available for use in the development of assurance of the quality of lubricants procured for safety related application.

As can be seen by the above information, SCE&G is not inactive in its quest for finalizing the lubrication qualification program. We do not, however, see sufficient reason at this time to revise our current lubrication program with radiation testing of Gulf oils about to commence. The lubricants used at the Summer Station are considered consumable "components" of the Class 1E equipment and hence have not been individually listed in Table 1.2-1 nor the specific NSSS or BOP tables. Qualification of lubricants will, however, continue to be an outstanding item to be fully addressed by full power operation. We feel the current status of our program gives sufficient interim justification until that time.

3. COMMENT:

VCSNS states that Valcor solenoid valves corrosion problem has been corrected. The submittal does not give the cause or corrective action for the corrosion nor does it provide the surveillance program which will be performed to ensure the problem will not reoccur.

RESPONSE:

The corrosion was in the attached D. G. O'Brien hermetic connector pins, causing an electrical short to ground. Corrective action consisted of an analysis of the corrosion found, and the following positive hardware modifications.

1. Stainless steel nipples are to be used with the stainless steel D. G. O'Brien connectors on all Valcor valves instead of galvanized nipples.

2. The area inside the D. G. O'Brien connector where the wires are attached to the connector pins is to be potted with Dow Corning DC170 potting compound. This potting material is the same as used in D. G. O'Brien connectors.

3. Water tight connections between the nipple and the O'Brien connector and the Valcor valve body will be accomplished by using Graphoil tape in the threaded connections.

4. ECN 1747 dated 10/26/81 was issued to install a voltage reducing circuit for the Valcor valve circuits.

5. D. G. O'Brien's "Instruction Manual 1060 Connector Assembly" was revised to incorporate steps 1, 2 and 3.

The VCSNS surveillance and maintenance program will be utilized to monitor this resolution to insure this problem will not reoccur.

4. COMMENT

The submittal should state in the BOP Tables and/or BOP detailed reports (DR's) the 20 year qualified life for the Rosemount level transmitter depends on a 5 year replacement schedule of the O-ring. What action is planned at 20 years?

RESPONSE:

At the end of the 20 year qualified life, one of several actions could take place. These actions could include:

1. Replacement with identical but new transmitter.

2. Replacement with a qualified transmitter of a different or newer type which would have to meet all technical and qualification criteria.

3. Replacement with a different brand (manufacturer) which would have to meet all technical and qualification criteria.

5. COMMENT:

The DR for the Samuel Moore instrument cable (DR 82) should state the cable was qualified for low voltage use (24-48 vdc); and, if it were to be used for an application with higher voltage, the qualification would be reevaluated for this higher voltage.

RESPONSE:

This DR could be revised in a future revision to the submittal. However, whenever any piece of 1E equipment is used in a new application, the qualification of the equipment is reviewed for acceptability in accordance with approved procedures. We agree that if the cable were desired to be used in higher voltage application, such a review would be necessary. This type of condition would exist however for any new or differing application of 1E electrical equipment.

6. COMMENT:

VCSNS should provide information in the DR's for the PYCO RTD's (DR-01, 46) which states whether the RTD's were tested with the new sealant material, the RTD's operated satisfactorily, and that there was no ingress of moisture.

RESPONSE:

Here again SCE&G took a conservative approach to assure functioning of these RTD's in the event of an MSLB or LOCA event. These RTD's have only two possible moisture entry paths, one is directly through the conduit opening and the other would be at the threaded cap (provided for wire/cable termination). This threaded cap also utilizes a stainless steel O-ring for sealing purposes. SCE&G has now installed qualified D. G. O'Brien hermetic connectors on the conduit entry points, and also used qualified DC-170 potting sealant on the threaded caps. As a result, SCE&G does not consider it necessary to require retesting of this combination by PYCO. In discussions with PYCO, it was considered that the threaded cap with O-ring did not need to have sealant to properly perform it's function, however, SCE&G considered this to be a minimal cost addition to provide redundant sealing of the cap. PYCO considered the conduit entry point to be the area of concern for line break/LOCA qualification. As a result, SCE&G has used the hermetic connector approach to positively provide a water tight boundary at the conduit entrance. This could be documented in any next revision to the submittal if required.

7. COMMENT:

VCSNS did not address the CCW flow instrument for cooling water to the RHR motor. This was to be done by fuel load.

RESPONSE:

Concerning the component cooling water (CCW) flow indicators for cooling supply to the residual heat removal pumps, an evaluation has been performed to confirm the instruments are properly classified as non-1E and do not have to be environmentally qualified. The function of the CCW supply to the RHR pumps is to provide cooling for the pump seals, because the RHR system handles high temperature fluids during normal plant cooldowns or during post-accident operation. The CCW supply for each RHR pump flows directly from one of two independent, redundant CCW supply loops through (in respective order) a manual isolation valve, an orifice, another manual isolation valve, the RHR pump seal heat exchanger, a second orifice (for the flow transmitter), and two manual isolation valves to the CCW system return loops. The CCW flow rate to the RHR pumps is established by orifice size during system flow balancing and is not adjusted during normal operation or post-accident operation. The position of the manual isolation valves in the supply and return lines from the RHR pump seals are administratively controlled to assure that CCW flow is maintained at all times except during any maintenance that would require securing CCW flow to the RHR pumps. The purpose of the flow transmitters in the CCW supply to the RHR pumps is to provide additional assurance, by means of a low-flow alarm, that the manual valves are not closed inadvertently during maintenance and testing activities in the vicinity of the valves in the auxiliary building. This alarm function is not required during post-accident operation of the CCW system because maintenance, testing and manual operation of equipment in the auxiliary building is conducted on a limited basis and under strict procedural controls. Also, the plant operators do not rely on this flow indication to perform any actions following an accident; and failure of the instrument (as a result of high radiation fields in the vicinity of the transmitters) would not cause the operator to perform actions adverse to safety because of the procedural controls on post-accident activities and the availability to the operator of other indications that the CCW system is functioning properly. Therefore, the flow transmitters are properly categorized as non-1E and are not required to be environmentally qualified.

8. COMMENT:

The Reliance fan motors had not originally been considered qualified because the proprietary data at the Reliance Cleveland office had not been audited. On 6/6/81, a GAI audit was performed at Cleveland on the Reliance fan motors and found acceptable. This method of qualification used is not generally allowed and would be acceptable only on a special case-by-case basis. The audit performed by GAI would have to include a detailed summary of the test report and the qualifying environmental data as it pertains to VCSNS. The VCSNS should make every effort to obtain qualifying data from Reliance to be placed in the equipment file.

RESPONSE:

SCE&G and GAI were not successful in being able to obtain hardcopy proprietary data from Reliance. This is not an uncommon occurrence although fortunately only a few of the V. C. Summer equipment suppliers have taken such a position. The audit performed by GAI was completed by individuals involved with the NUREG 0588 review and also knowledgeable with the necessary documentation required to fulfill qualification criteria for the V. C. Summer Nuclear Station. These efforts have resulted in a review which SCE&G considers acceptable to determine that the installed motors are indeed qualified to the parameters so stated in Rev. 4 to the NUREG 0588 Submittal.

9. COMMENT:

DR-48 (3) for the Rosemount Reactor Building Sump Level Transmitters states that margin is not applicable for an arbitrary chosen value of time. I think more explanation is required as to what margin VCSNS is talking about and why it is not applicable.

RESPONSE:

The margin VCSNS is discussing in DR-48, Item 3, concerns the first few seconds of the test LOCA profile versus the first few seconds of the VCSNS postulated LOCA temperature profile. The main steam line break postulated profile has a quicker temperature rise than the test profile during the first few seconds of the LOCA event. The vendor's profile uses an arbitrary value for temperature rise based on a general knowledge of requirements from nuclear power plants. The lack of margin during the first few seconds of the test is considered acceptable since the remainder of the test includes a considerable margin over the VCSNS postulated profile.

10. COMMENT:

The operability requirement of the Westinghouse RHR pump motor is listed as 4 months in the Rev. 4 submittal, but during the July audit the operability was listed as 1 year. Which is correct? If the Rev. 4 submittal is not correct, then VCSNS should check the operability requirement of the listed equipment to ensure correctness and qualification.

RESPONSE:

The Rev. 4 required operability time of 4 months is the correct value, and as such, the demonstrated operability was evaluated against this 4 month criteria.

11. COMMENT:

The aging temperature and time for the Westinghouse RHR pump motor is listed as "NA". This motor has been aged and the aging data should be entered into the submittal. The aging temperature and time for all other equipment in the submittal which has been listed as "NA", "See Kerite Report" or "NOTE 51", should be checked and the missing information, if

any, entered into the submittal. Also, the method of analysis and justification (i.e., arrhenius, 10°C Rule, etc.) used to extend qualified life should be stated in the submittal.

RESPONSE:

For the RHR pump motor, this listing was marked NA since the entire and complete motor was not qualified as a singular unit. Instead, motorette testing was utilized. As stated in DR-6W(2) the materials in the insulation system were subjected to thermal and radiation aging. Thermal aging was performed at 200°C for 21 days for the equivalent of 105°C for 40 years. As stated in DR-6W(1), this thermal aging followed the 10°C "rule".

The Kerite information was considered proprietary and as a result was not included in the submittal. We did however obtain permission to reproduce the accident profiles which we included in the back of the submittal. These Kerite proprietary reports are however on file at the VCSNS central file. In regard to other instances, SCE&G described within the DR's the method of analysis and justification used, such as, for Okonite where the Arrhenius concept was discussed in DR-05(8).

12. COMMENT:

Some equipment was to have problems resolved by fuel load now have been changed to full power operation.

1) Veritrak Transmitter

2) Motor Lubricants

3) Hydrogen Recombiners

RESPONSE:

These resolutions had not been committed by SCE&G to be resolved by fuel load. However, SCE&G expects successful testing of the Veritrak Transmitter to be completed in mid 1982, thus far Westinghouse has not found any test results that would indicate that it will not satisfactorily pass testing. Also contained herein (Question #2) is a status and justification for the SCE&G proposal to complete lubricant qualification by full power operation. In addition, SCE&G has done a material review of the Hydrogen Recombiner in accordance with NUREG 0588 Category II 4(2) aging requirements. The Hydrogen Recombiners have been qualified to IEEE 323-1974 by Westinghouse as we are sure the NRC is aware, and at this point in time SCE&G is actively pursuing a purchase order with Westinghouse to upgrade our qualification documentation with a hard copy of this IEEE 323-1974 testing. This is expected to be completed by full power operations.

13. A) COMMENT:

VCSNS stated that ASCO solenoid valves would be installed to replace the following Westinghouse supplied air operated valves.

XVT-8149A/B/C-CS
XVD-8047-RC
XVT-8871-SI
LCV-1003-WL
XVD-7126-WL
XVT-8152-CS
XVT-8880-SI
XVT-8860-SI
XVT-8961-SI
PCV-44B-RC
PCV-445A/B-RC

A) RESPONSE:

This is a correct statement, solenoid valves have been installed.

B) COMMENT:

The submittal did not contain qualification information on the following valves.

XVT-8149A/B/C-CS
XVD-8047-RC
XVT-8871-SI
LCV-1003-WL
XVD-7126-WL
XVT-8152-CS
XVT-8880-SI
XVT-8860-SI
XVT-8961-SI

B) RESPONSE:

This is not correct, the submittal table 1.2-1 and the BOP foldout pages contain the qualification information on these valves.

C) COMMENT:

The tag numbers of the ASCO valves which have "Note 64" under "Operability Demonstrated" do not match the ones listed in Table 1.6-2.

C) RESPONSE:

All 1E harsh environment ASCO solenoids have been qualified and installed. Any "Note 64" in the submittal is a clerical error and should not be considered pertinent.

14. COMMENT:

The submittal did not state whether the ASCO solenoid valves XVG 9627A/B, IFV-3531, 3536, 3541, 3546, 3551, 3556, XVT-2662A/B, XVT-6384A/B, XVT-6385A/B, and XVT-2660 would be tested to ensure qualification of the conduit connection seals (see DR-60).

RESPONSE:

The conduit connections were sealed by SCE&G to provide an added measure of assurance that the valves would function in the event of LOCA environmental conditions. This sealing was performed by the use of qualified D. G. O'Brien hermetic connectors as contained and listed in Rev. 4 of the NUREG Submittal. SCE&G does not consider it necessary to require retesting of the ASCO solenoid valves to ensure the D. G. O'Brien connectors can prevent moisture and/or spray intrusion.

15. COMMENT:

The submittal does not indicate if a test report of the -N adhesive is part of the qualification package for the Raychem Nuclear Motor Connection Kits (see DR-80(10)).

RESPONSE:

The -N adhesive was tested an an integral part of the entire kit and is therefore covered by the same test report, not a different one.

16. COMMENT:

There is no detailed qualification report on the Gould-Shawmut fuses (page B-34). Is there a surveillance program to enusre no premature failure due to corrosion or other means?

RESPONSE:

As discussed in DR-19, a premature failure of the fuse is totally acceptable. These fuses simply isolate the non-1E heat tracing circuits from their 1E power supply sources. As a result, any premature failure by any mechanism would be acceptable and provide no safety concern whatsoever.

17. COMMENT:

DR-76 for "Phenolic Type" blocks states ".... temperature and radiation are the only parameters that would appear to cause the fuse block to fail." Has this been documented in the qualification file? Why was humidity not addressed.

RESPONSE:

This definitely was documented in the VCSNS qualification Central File. At the time of review, these blocks had not completed the full IEEE 323-1974 LOCA and MSLB testing. As a result, SCE&G chose to only use these blocks in areas where there would not be LOCA or MSLB high moisture and humidity conditions. As a result, we qualified these blocks for their normal environmental conditions plus LOCA radiation that they might see. This we have been able to amply document in our qualification file. It should be noted that subsequently these blocks have passed the full IEEE 323-1974 qualification.

18. COMMENT:

Note 50 on page B-38 is very general. The equipment whose qualified life depends on the periodic replacement of components (i.e., seals, solenoids, gaskets, etc.) should state when and what equipment/components are to be replaced due to aging.

Example: The Allied solenoid valves have a 40 year qualified life with a 5 year replacement program for seals and solenoids.

The ASCO solenoid valves indicate qualification is based on the periodic replacement of equipment components, but "what" and "when" is not given.

RESPONSE:

SCE&G intended to provide this information in the DR's wherever it was applicable. This was an oversight to not include it for the ASCO solenoid valves. However, the elastomers and coils will be replaced every 4 years to maintain qualification. This was documented in the SCE&G review and has been incorporated into the SCE&G documentation central file.

19. COMMENT:

Equipment whose qualified life is less than 40 years should have in the submittal what action will be taken at the end of its qualified life (i.e., replacement with new equipment, requalification of original equipment, etc.).

RESPONSE:

As discussed in Comment #4, SCE&G will make this determination on a case-by-case basis prior to the end of qualified life. However, SCE&G will install qualified equipment which will be adequately documented for its intended normal and accident functions.

20. COMMENT:

The submittal should list for each equipment item the method of qualification and the report I.D. for each environmental parameter. In some cases it was difficult to tell which environmental parameters were qualified by test and which were qualified by analysis.

RESPONSE:

This comment apparently only applies to that equipment which was stated in the foldout pages as being qualified by a combination of test and analysis. Wherever the qualification did not specifically meet the NUREG 0588 requirements, a (Deviation Report) DR was written to provide the technical reasoning and thought process for acceptance. The complete and detailed review is contained in the VCSNS documentation central file and it was not considered necessary to address this specifically in each foldout page. SCE&G considers the method of analysis has been adequately discussed in the DR's to demonstrate the reason and method for acceptance.

21. COMMENT:

The submittal should list all the references used for the abnormal or accident environment for each parameter.

RESPONSE:

Section 1.3 of the submittal discusses the method and FSAR references for the development of the environmental service conditions. SCE&G did attempt to provide a reference figure for each temperature and pressure parameter in each line item on the foldout pages for abnormal and accident conditions. Radiation and spray parameters are discussed in Section 1.3 and it was not considered necessary to reference these other than to indicate post-LOCA operability time radiation dose when calculated for specific equipment.

22. COMMENT:

Equipment that is not qualified at this time should have the qualification status stated in the W or B pages under "Qualified Life".

Example: 1) Hydrogen Recombiner

2) Veritrak Transmitter

3) W Accoustic Leak Monitor

RESPONSE:

These were left blank intentionally and the status was specifically listed in detail in Table 1.6-1. In any possible future revision, SCE&G could reference this table in the foldout page under the heading of "Qualified Life".

23. COMMENT:

VCSNS should commit to periodically placing the results from reviews of component/equipment failures during "Programmed Preventative Maintenance" and "Corrective Maintenance" into the submittal to keep the documentation up to date.

RESPONSE:

SCE&G does not consider this necessary, since the VCSNS documentation central file will be performing this function. In addition, SCE&G is awaiting the new 10CFR50.49 rulemaking which will require a submittal of information. It is our understanding that the NUREG 0588 Submittal requirements will be deleted upon issuance of the new rulemaking.

24. COMMENT:

On some equipment, the demonstrated operability does not seem to include adequate margin over the required operability.

RESPONSE:

SCE&G specifically reviewed all equipment to meet the margin requirements of NUREG 0588. Wherever there was any difference between

the testing qualification margin and NUREG requirements, the difference was discussed and justified in the DR section of the submittal.

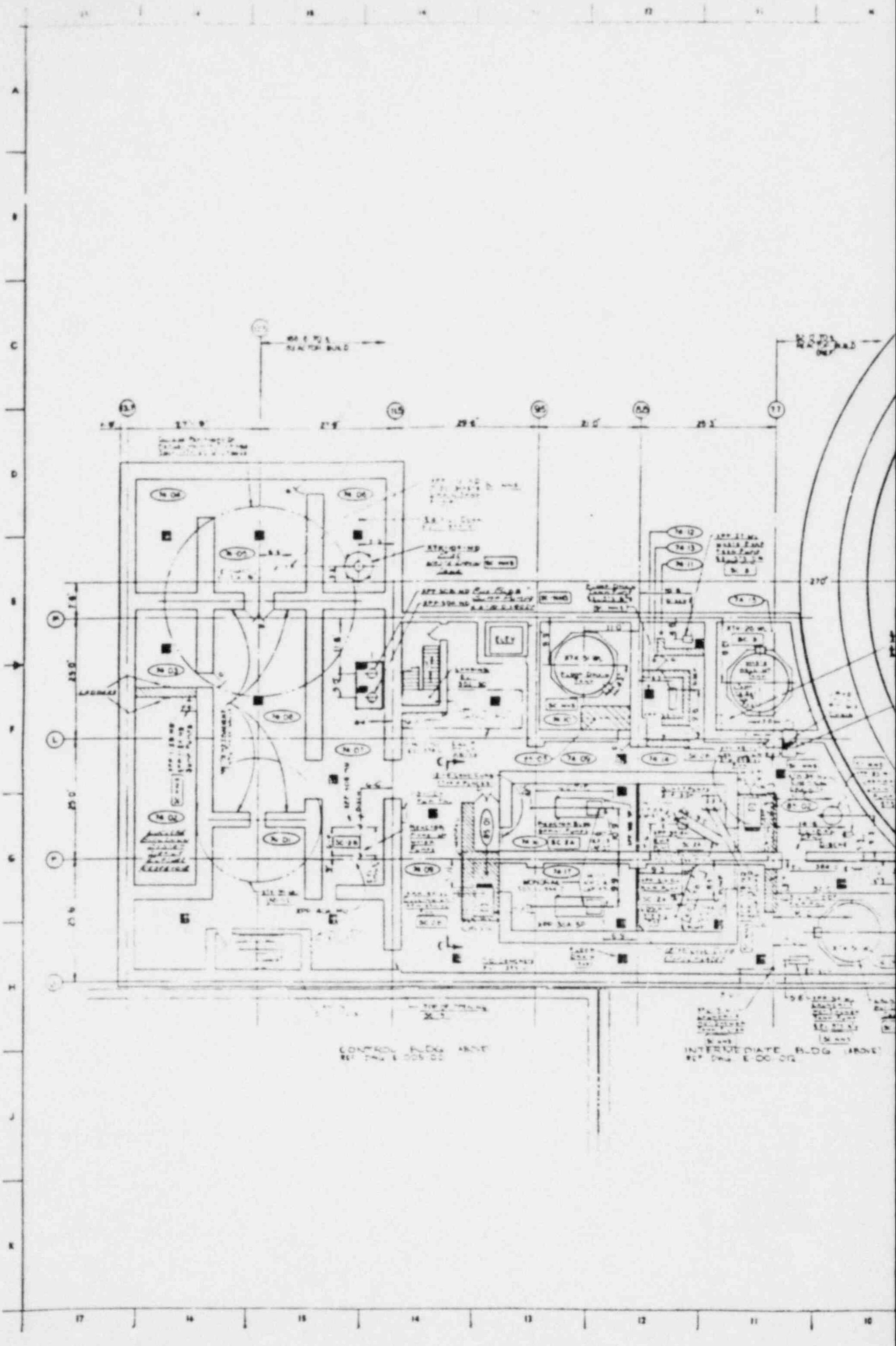
25. COMMENT:

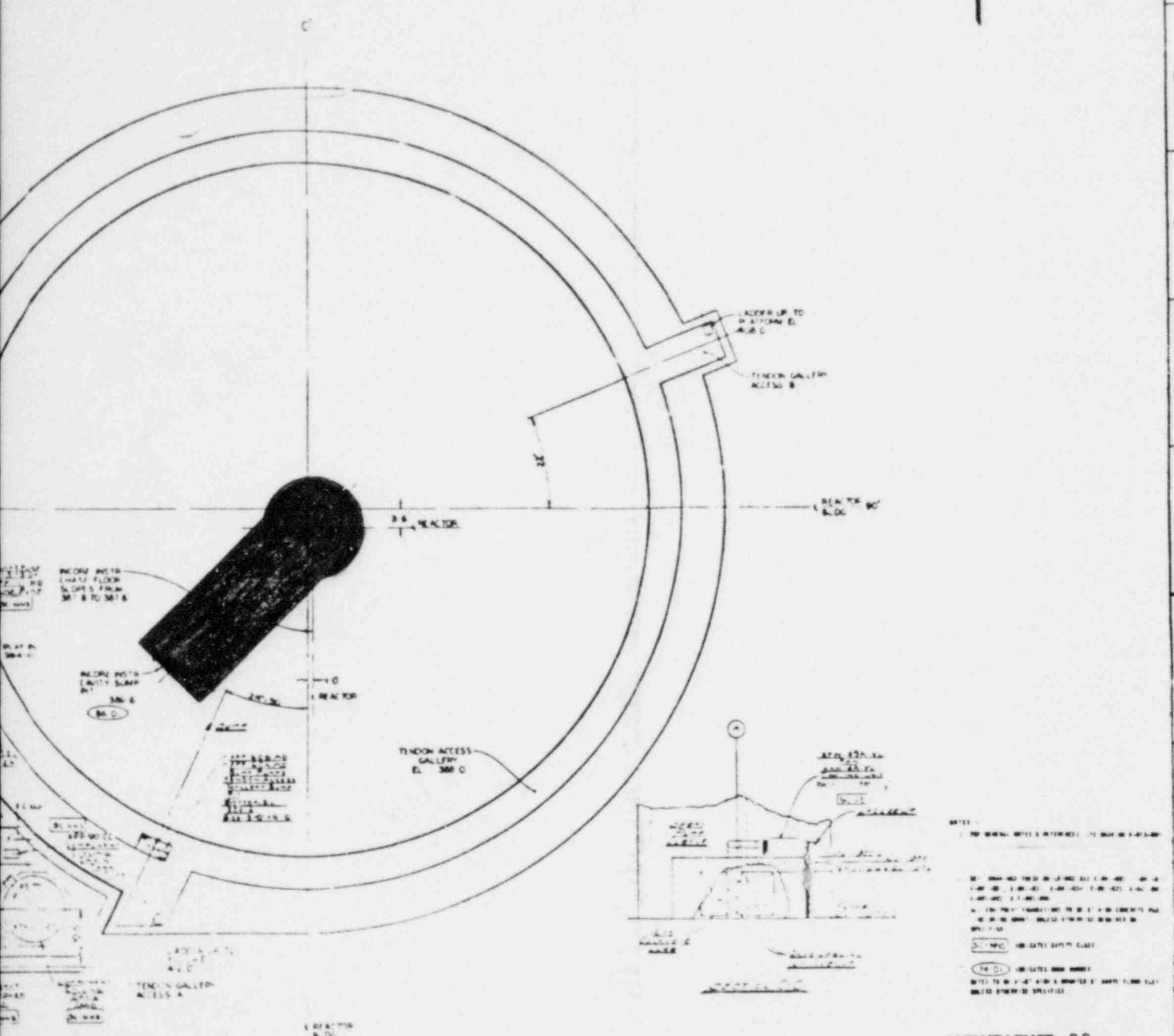
BOP Tables do not provide information of the concentration of chemical spray used.

RESPONSE:

For each piece of 1E equipment the spray testing and results were compared to the required concentrations and durations as stated in Section 1.3-4 of Rev. 4 to the NUREG 0588 Submittal. This review is documented for each piece of equipment on the Check/Summary Sheets as contained in Appendix I to the NUREG Submittal. All of these reviews are filed in the V. C. Summer Central File for environmental qualification. This was a part of the total documentation as audited by the NRC Environmental Qualification Team during the July, 1981 site review trip.

The submittal BOP foldout pages however only have the spray duration listed. However, as stated above, the tested spray concentrations were compared to required criteria and documented. Where any deviation existed in regard to either concentration or duration, this was documented and discussed in the Deviation Report for that piece of equipment in the NUREG Submittal. As a result, SCE&G did perform a total review to encompass both duration and concentration criteria for chemical spray qualification.


ELEV
CONTROL BLDG ABOVE
INTERMEDIATE BLDG (ABOVE)



AMENDMENT 22
NOVEMBER, 1980

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Auxiliary and Reactor Buildings,
Sub-basements Plan Above
Elevation 374'-0"

Figure 1.2-2
(GAI Dwg. E-001-001)

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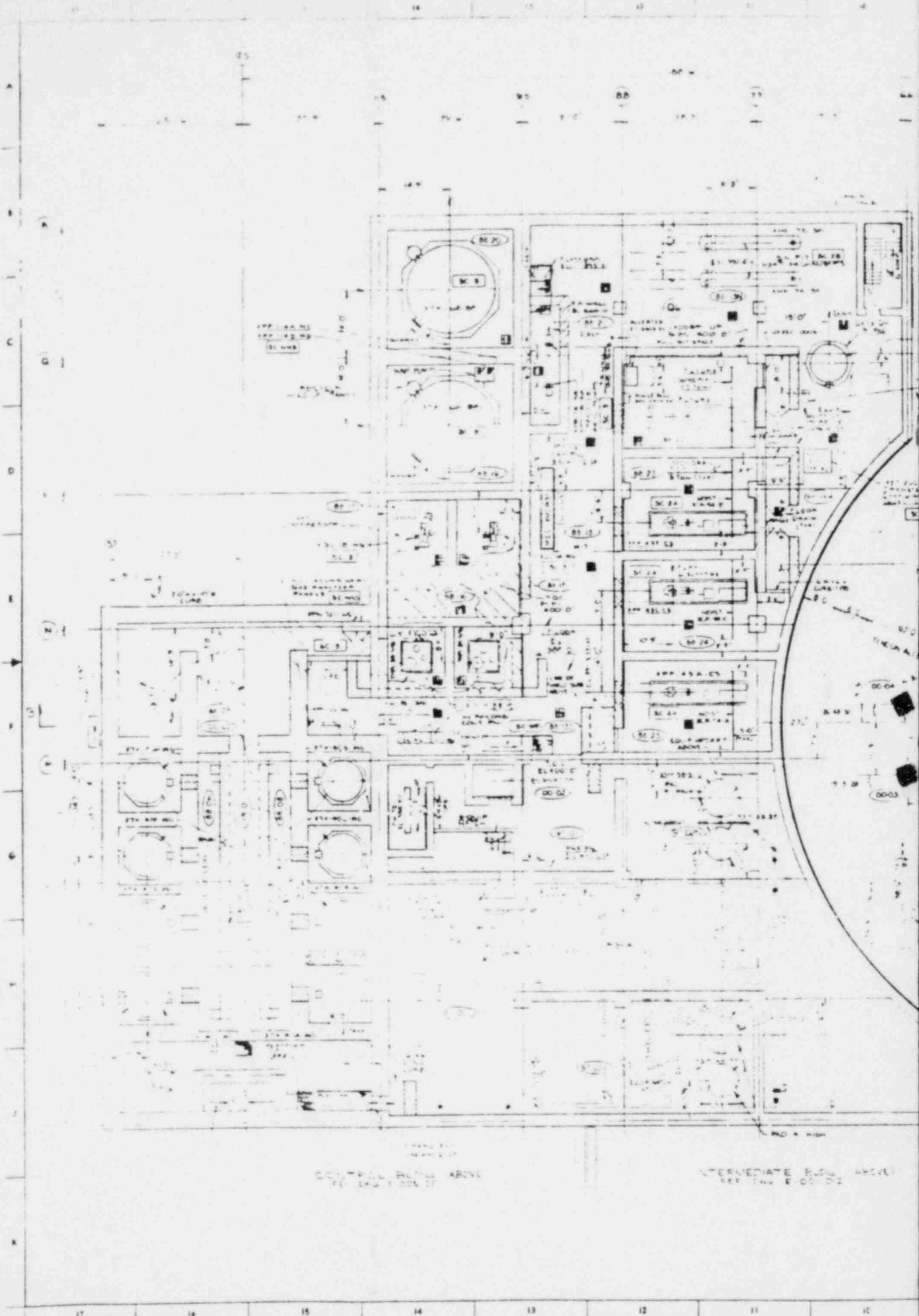
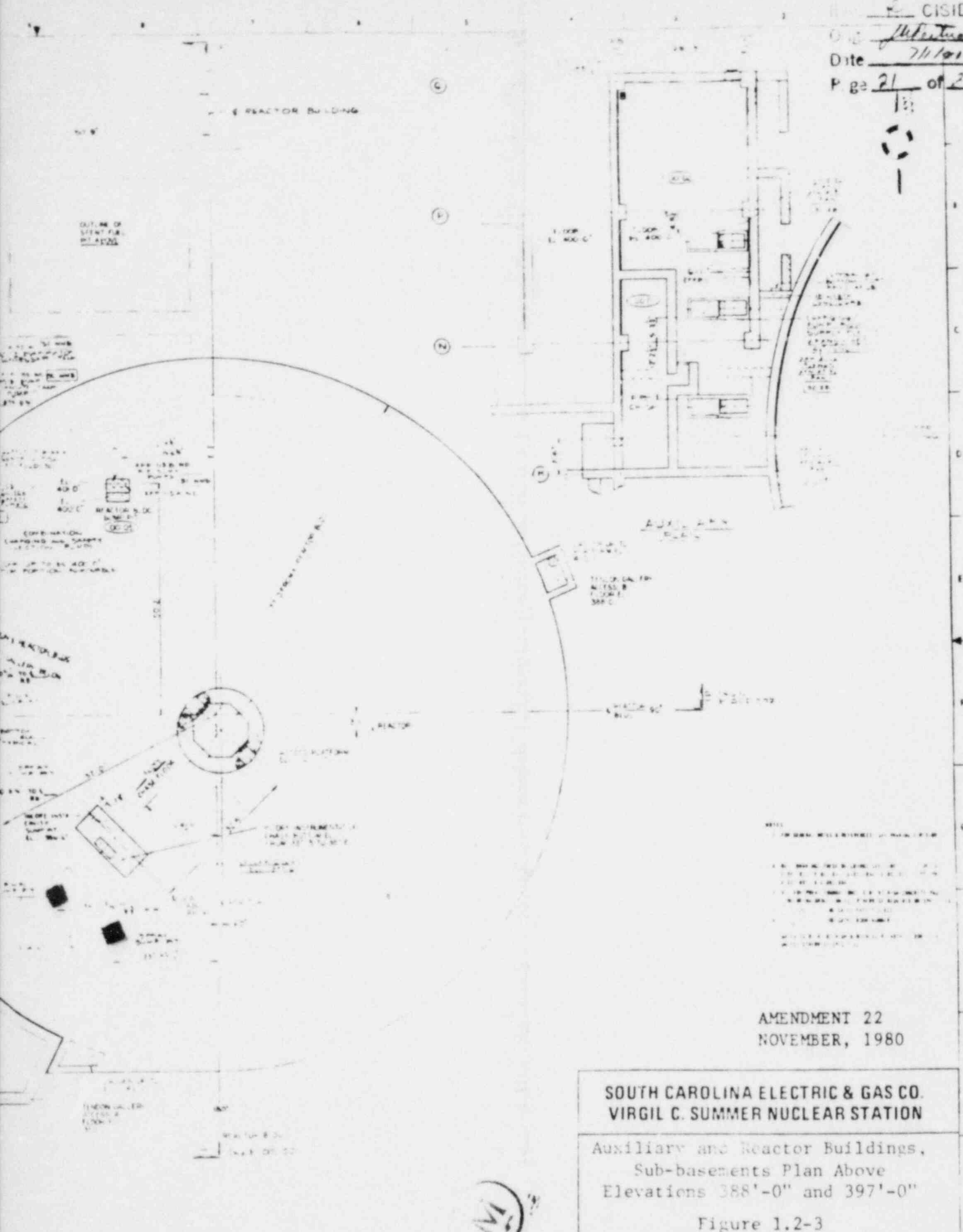

CONTROL BLDG ABOVE
INTERMEDIATE BLDG ABOVE

CISID
Date 7/1/01
Page 21 of 27

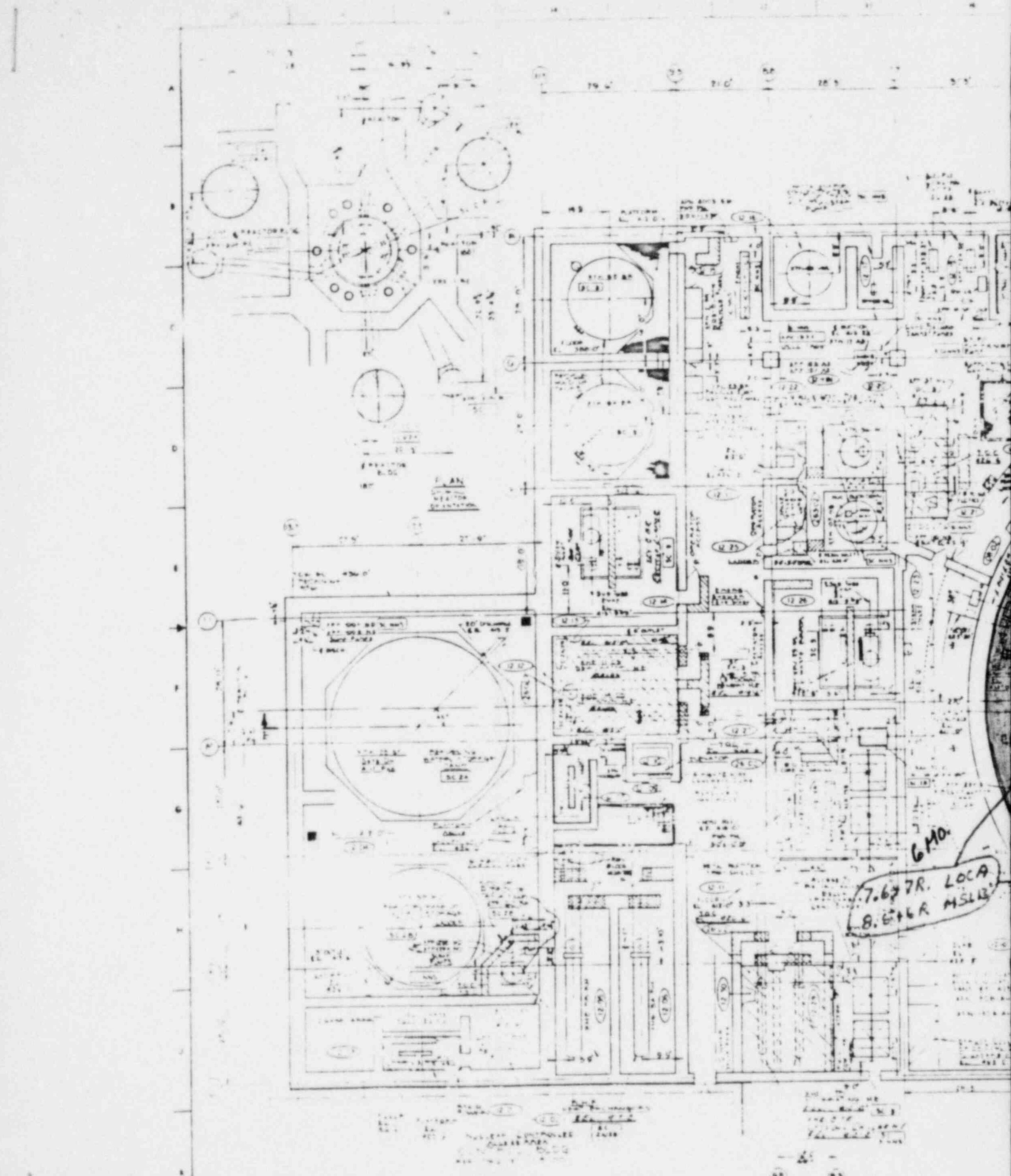


AMENDMENT 22
NOVEMBER, 1980

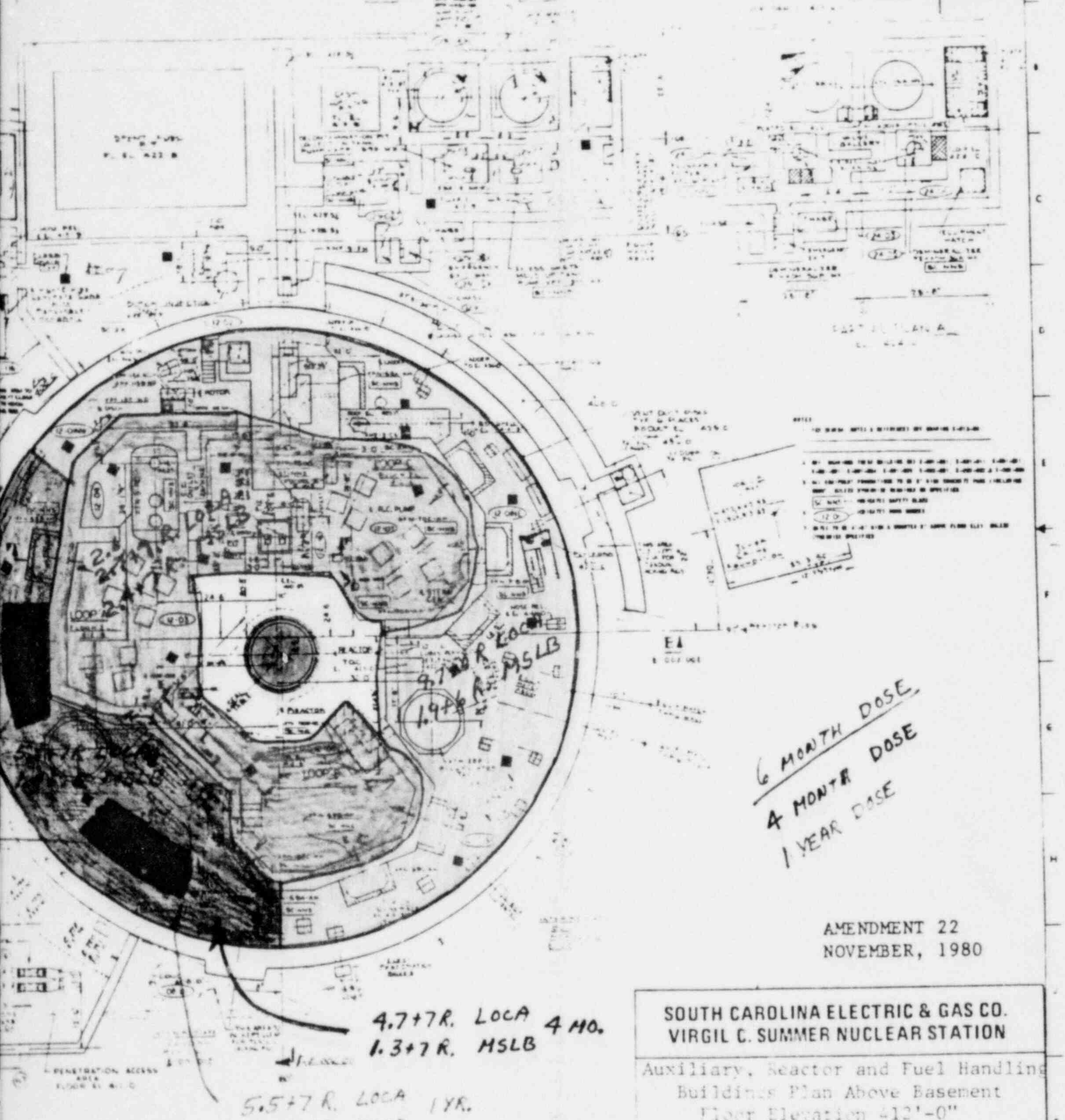
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Auxiliary and Reactor Buildings,
Sub-basements Plan Above
Elevations 388'-0" and 397'-0"

Figure 1.2-3
(CAI Dwg. E-001-002)


PLAN
6 MO.
7.67R. LOCA
8.64R MSLB

Subj Equip Doses in Cont
Rev. 2 CISID 1.8.5.6
Orig. [illegible]
Date [illegible]
Page 22 of 27

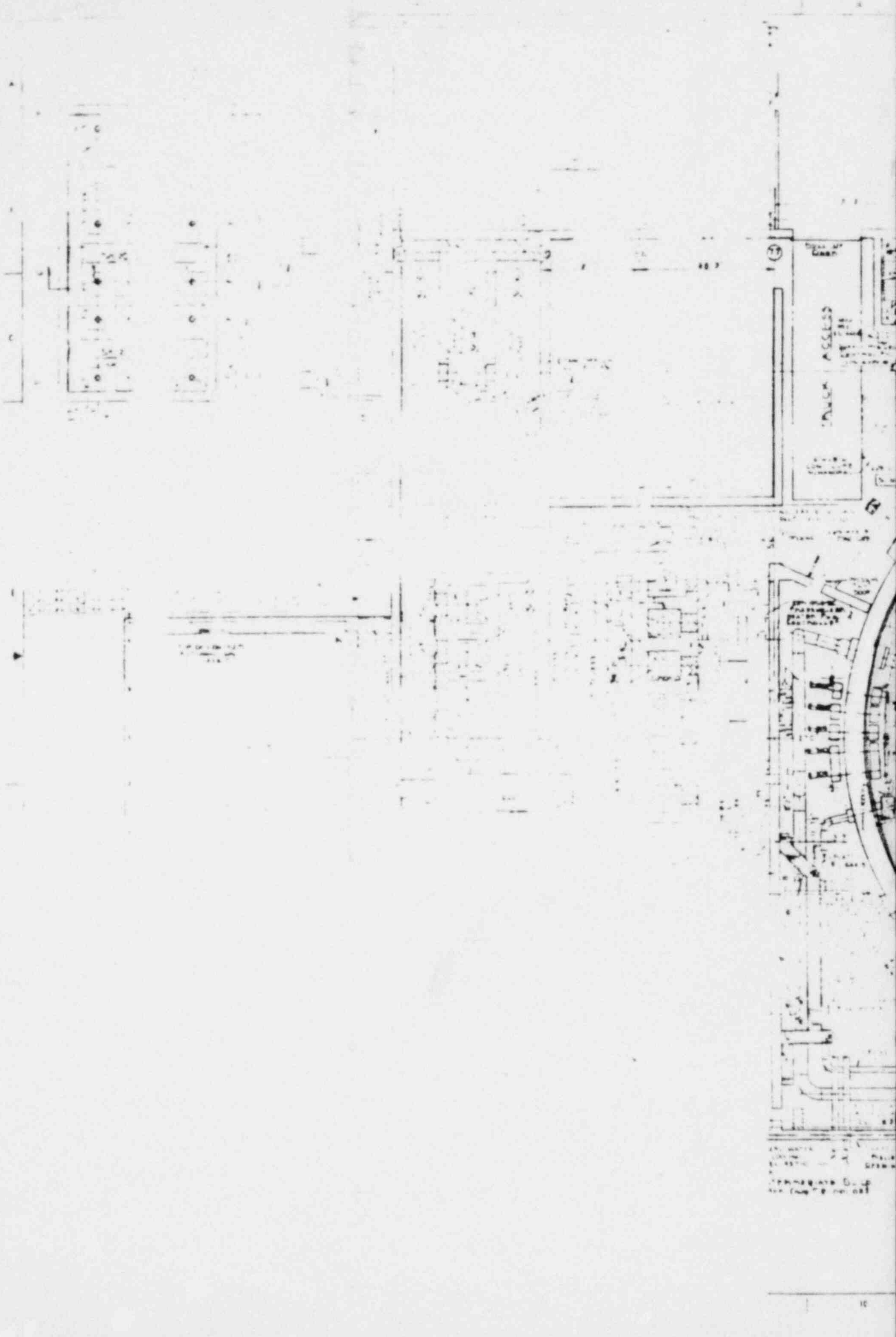


AMENDMENT 22
NOVEMBER, 1980

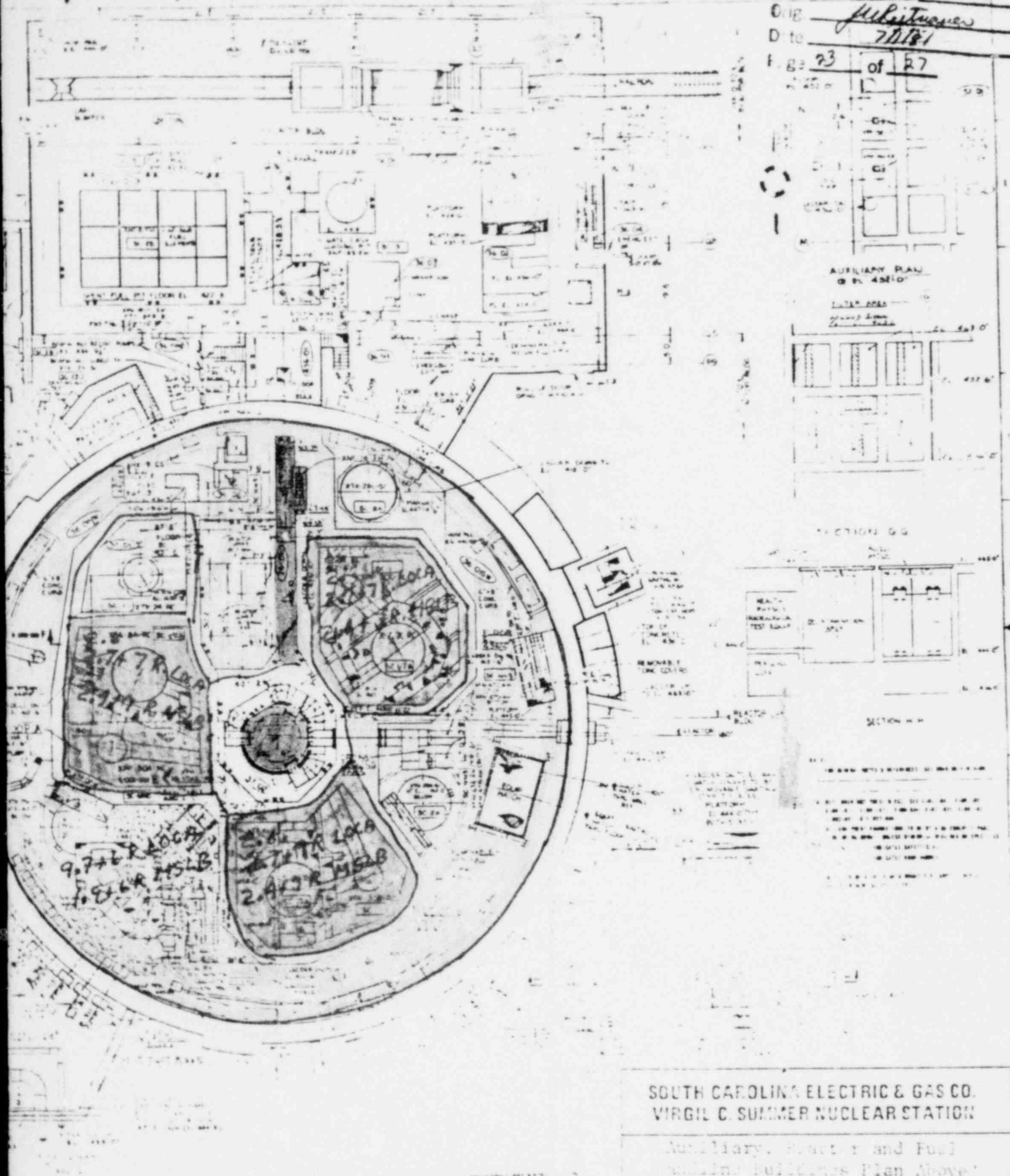
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Auxiliary, Reactor and Fuel Handling Buildings Plan Above Basement Floor Elevation 412'-0"

Figure 1.2-4
(GAI Dwg. E-001-011)


TRUCK ACCESS

Subj. Equip Doses in Cont.
Rev. 2 CISID 1.8.5.6
Orig. [illegible]
Date 7/1/81
Page 23 of 27

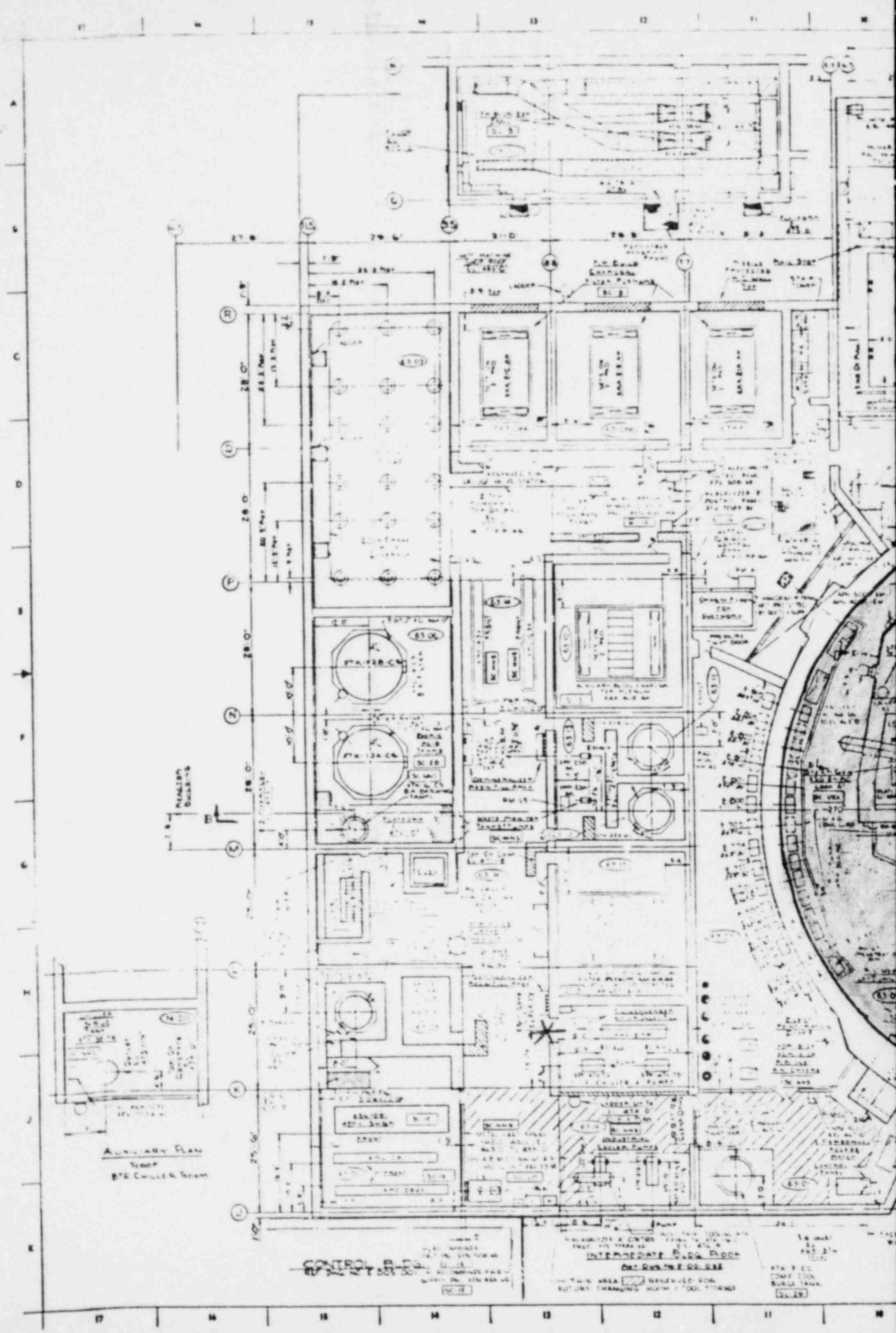


AMENDMENT 12
NOVEMBER, 1980

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

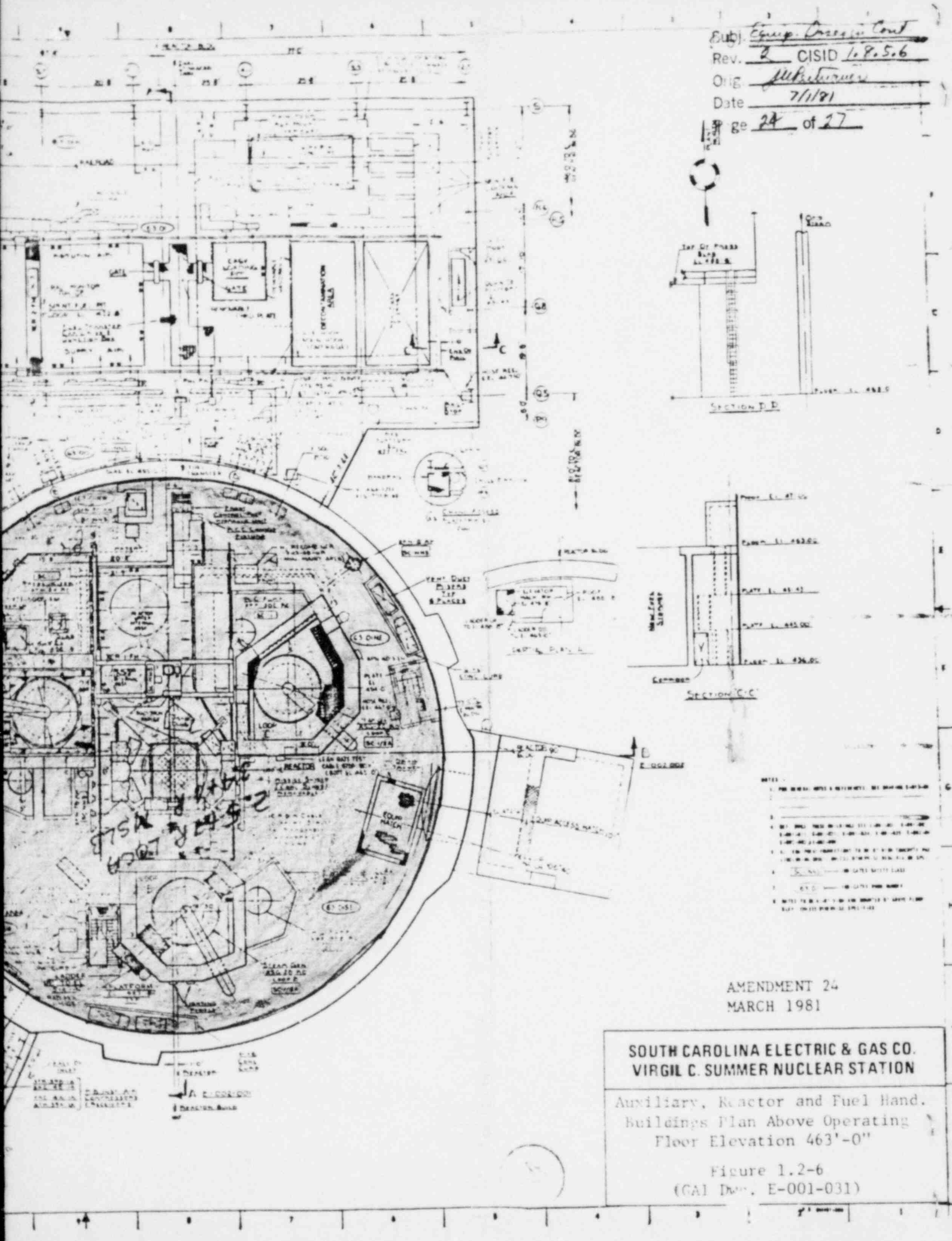
Auxiliary, Reactor and Fuel Handling Buildings Plan Above Operating Floor Elevation 436'-0"

Figure 1.2-5
(GAI Dwg. E-001-021)



AUXILIARY PLAN
CONTROL BLDG
INTERMEDIATE BLDG ROOF
THIS AREA RESERVED FOR
FUTURE CHANGING ROOM / TOOL STORAGE
SURGE TANK

Subj. Equip. Design Cont
Rev. 2 CISID 1.8.5.6
Orig. [illegible]
Date 7/1/81
Page 24 of 27

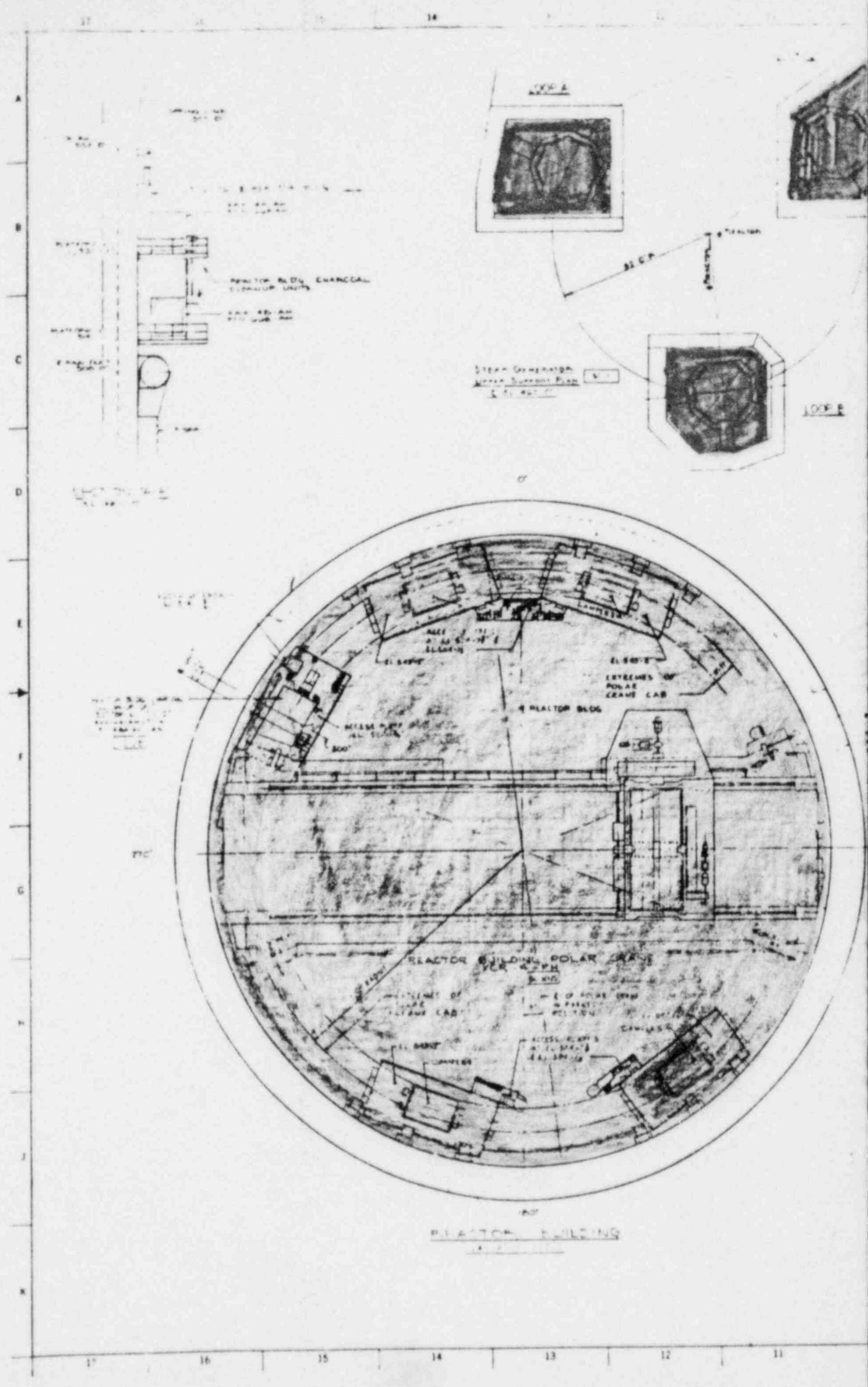


AMENDMENT 24
MARCH 1981

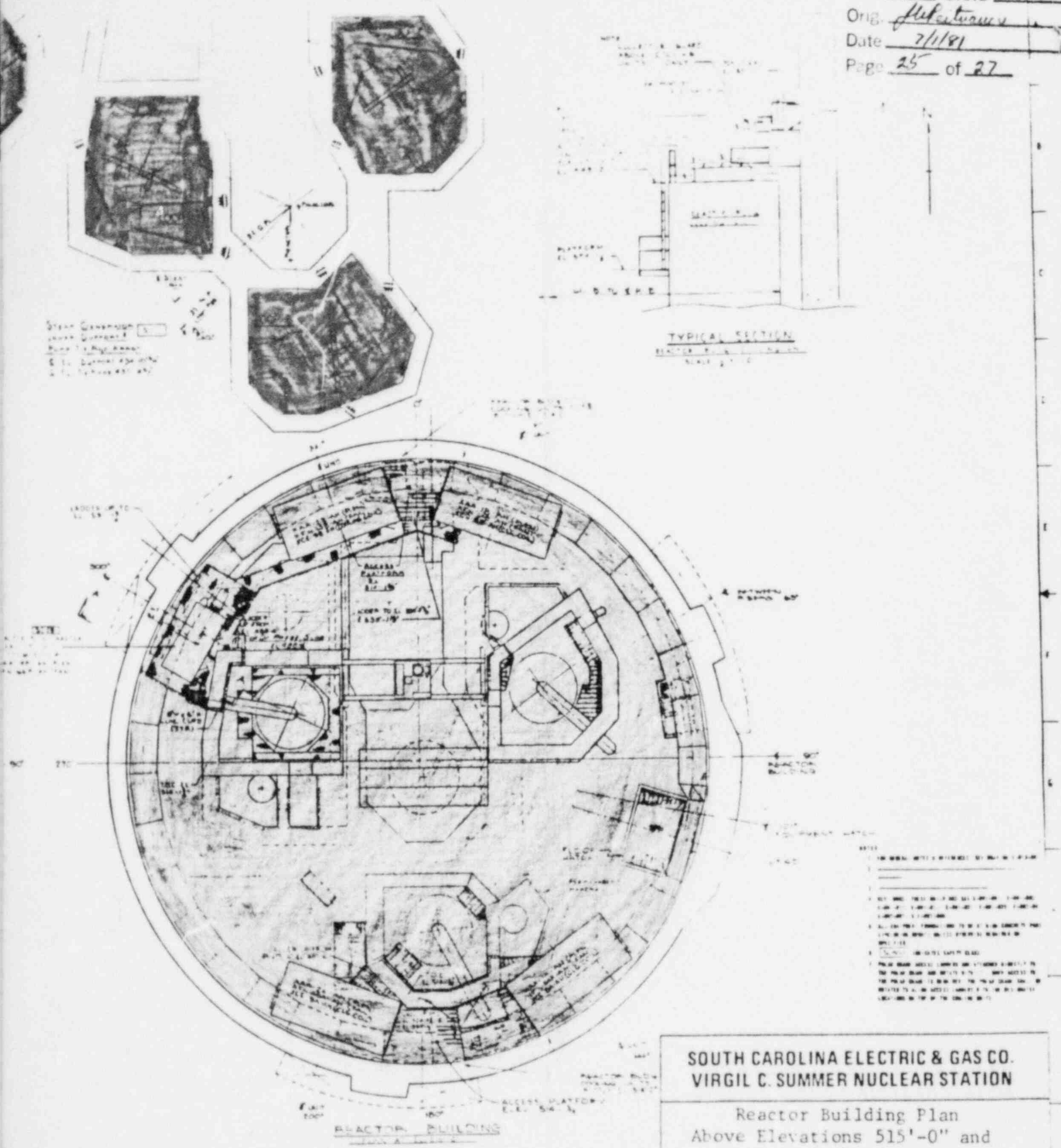
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Auxiliary, Reactor and Fuel Hand. Buildings Plan Above Operating Floor Elevation 463'-0"

Figure 1.2-6
(GAI Dwg. E-001-031)


LOOP A
REACTOR BLDG CHARCOAL CLEANUP UNITS
STEAM GENERATOR UPPER SUPPORT PLAN
LOOP B
EXTREMES OF POLAR CRANE CAB
REACTOR BLDG
REACTOR BUILDING POLAR CRANE
REACTOR BUILDING

Subj. Equip. [illegible]
Rev. 2 CISID 1.8.5.6
Orig. [illegible]
Date 7/1/81
Page 25 of 27

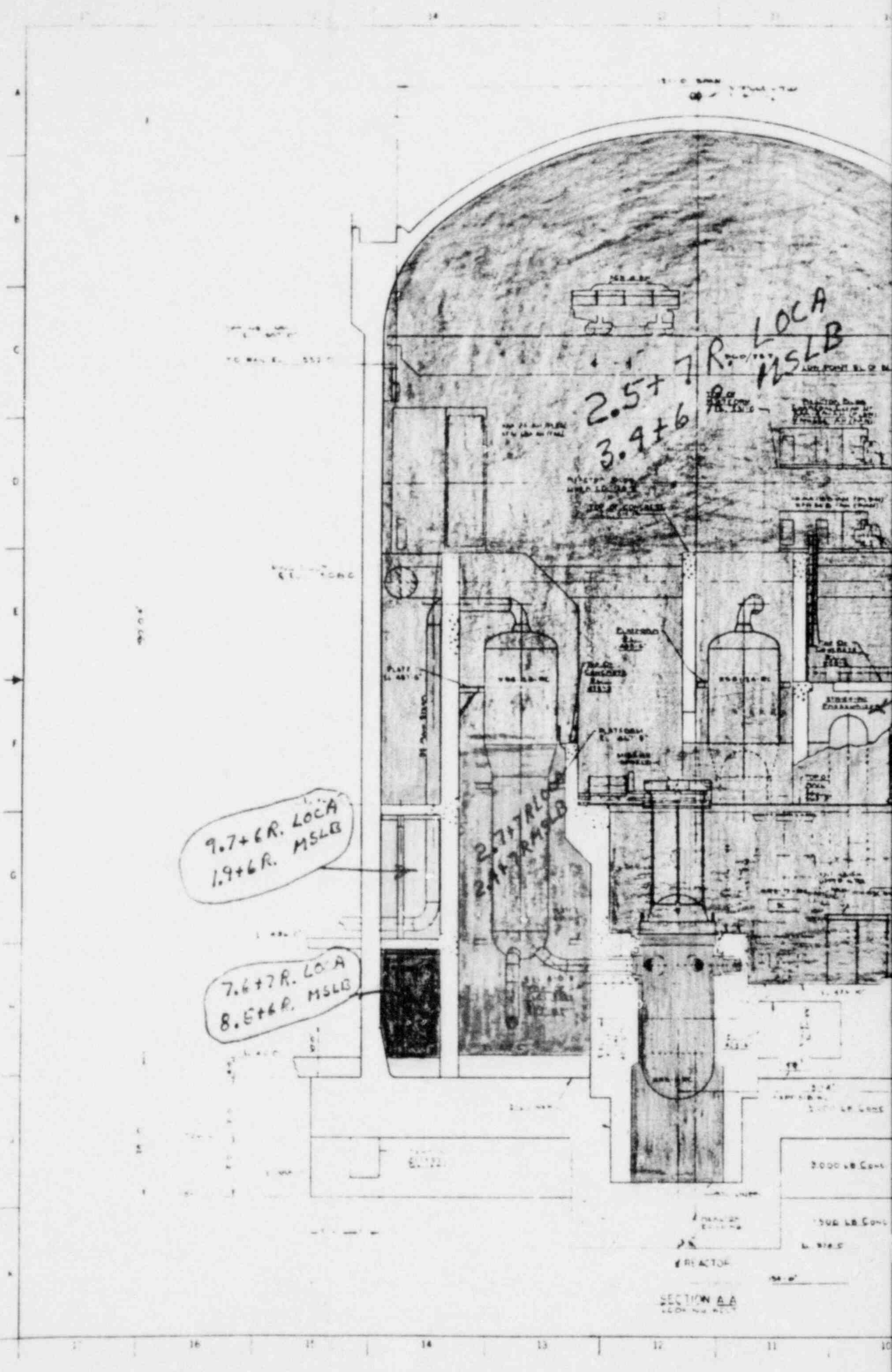


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

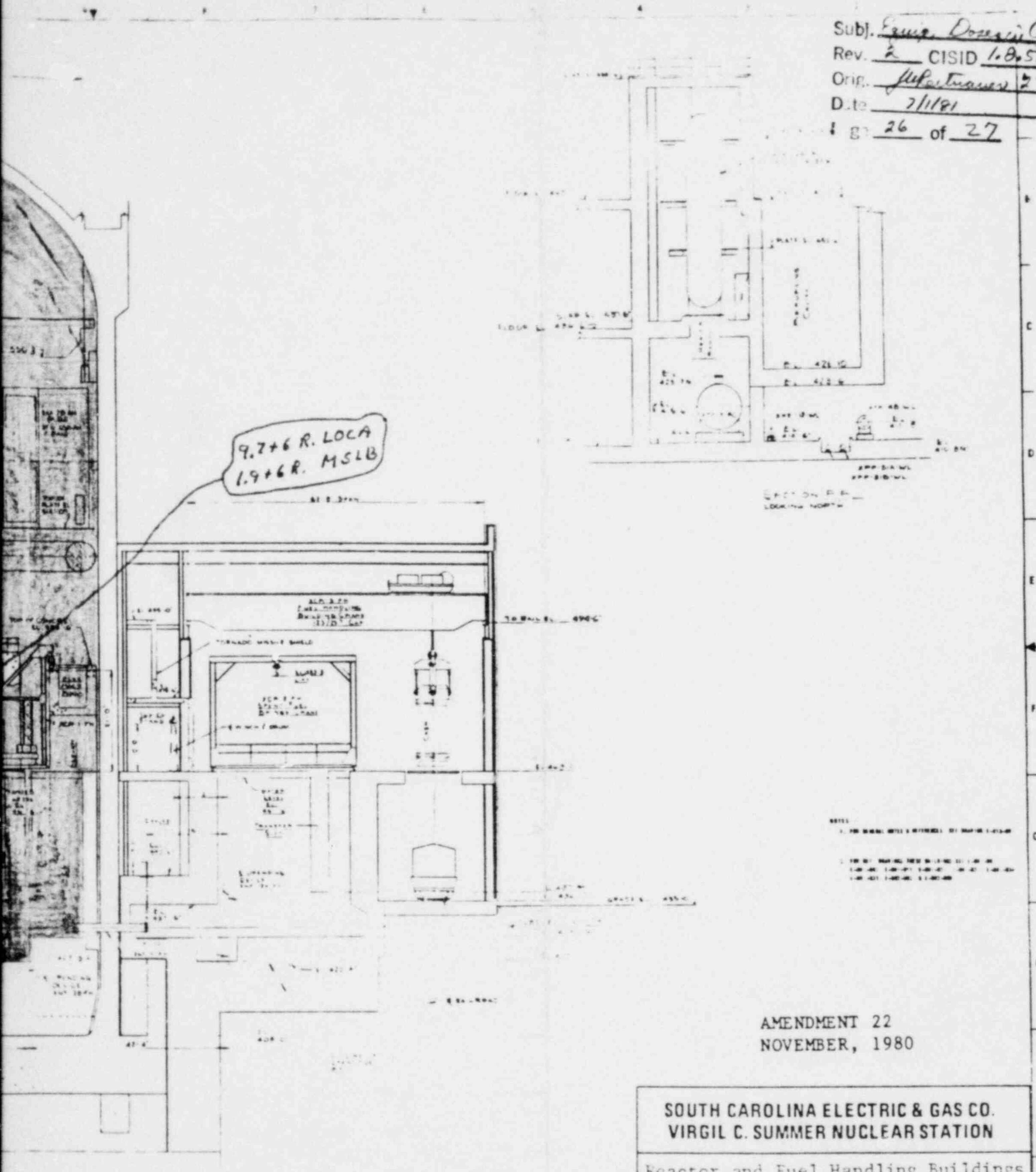
Reactor Building Plan
Above Elevations 515'-0" and 552'-0"

Figure 1.2-7
(GAI Dwg. E-001-034)

AMENDMENT 22
NOVEMBER, 1980


2.5+7 R. LOCA
3.4+6 R. MSLB
9.7+6R. LOCA
1.9+6R. MSLB
7.6+7R. LOCA
8.E+6R. MSLB
REACTOR
SECTION A-A

Subj. Equip. Dose in Cont.
Rev. 2 CISID 1.8.5.6
Orig. [illegible] 27
Date 7/1/81
Pg 26 of 27



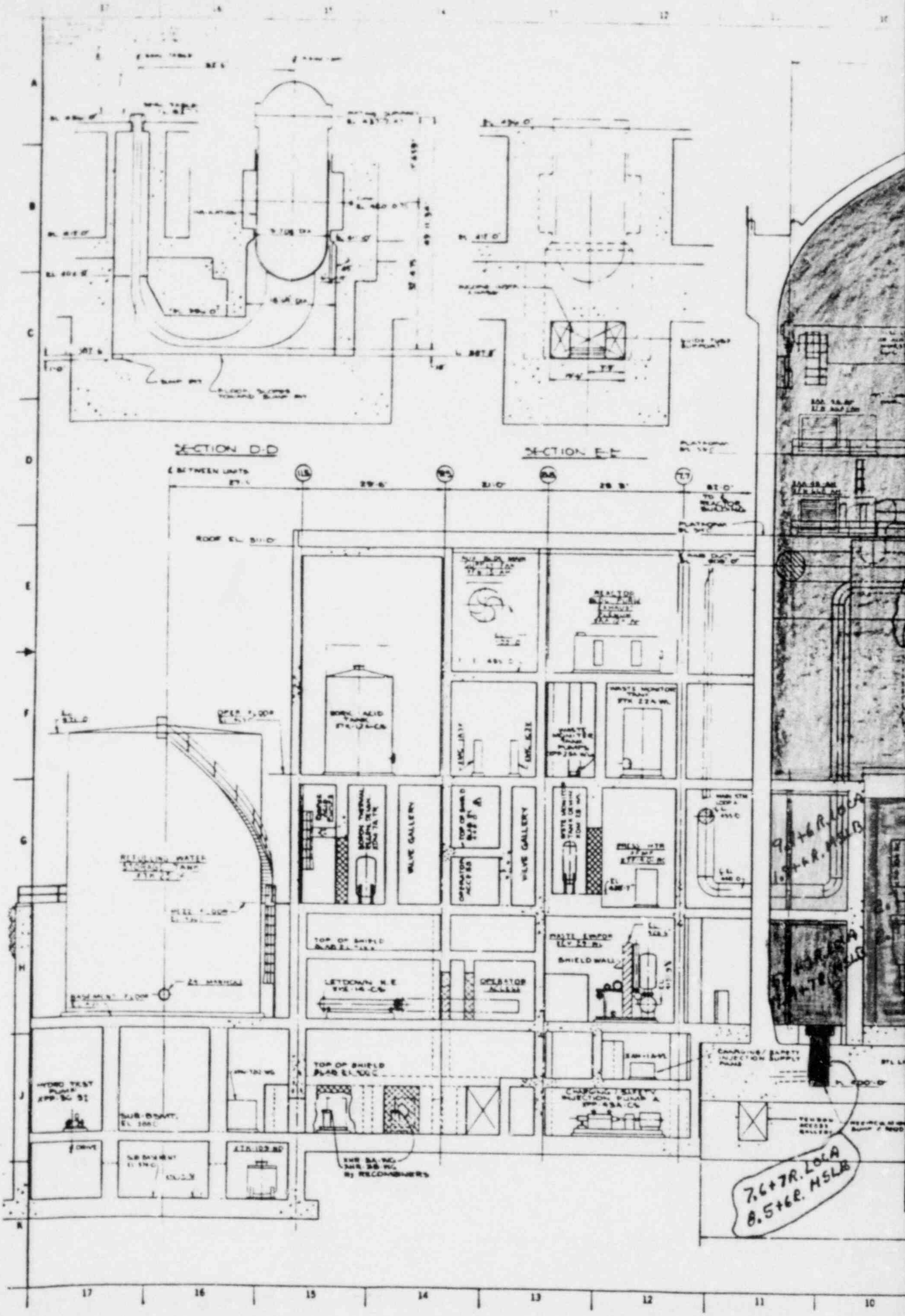
AMENDMENT 22
NOVEMBER, 1980

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

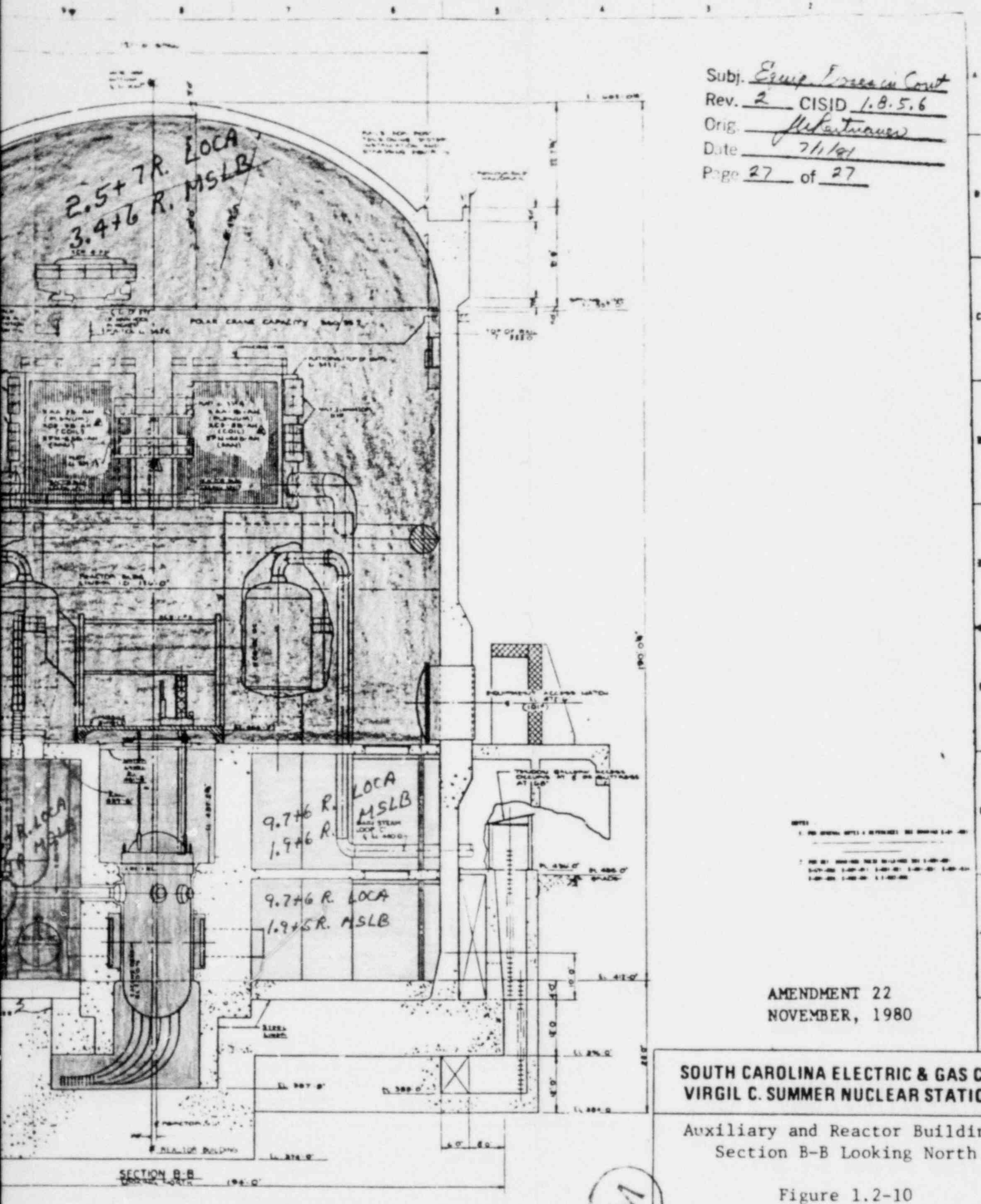
Reactor and Fuel Handling Buildings
Section A-A Looking West

Figure 1.2-9

(GAI Dwg. E-002-001)


SECTION D-D
SECTION E-E
VALVE GALLERY
VALVE GALLERY
OPERATOR ACCESS
SHIELD WALL
TOP OF SHIELD
OPERATOR
7.6+7R. LOCA
8.5+6R. HSLB
9.4+6R. LOCA
17
16
15
14
13
12
11
10

Subj. Equip. Environ Cont
Rev. 2 CISID 1.8.5.6
Orig. [illegible]
Date 7/1/81
Page 27 of 27



AMENDMENT 22
NOVEMBER, 1980

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Auxiliary and Reactor Buildings
Section B-B Looking North

Figure 1.2-10

(GAI Dwg. E-002-002)

Gilbert Commonwealth Companies

DESIGN VERIFICATION RECORD

FILING CODE 1.6.1

PROJECT: V. C. Summer Nuclear Station, Unit 1

SUBJECT: CONTAINMENT POST-LOCA DOSE

REVISION: 2
DATE: ~~12-18-80~~ 4-10-81

SECTION NAME AND NUMBER Applied Engineering Analysis Department Nuclear (2109) and Mechanical (2107) Sections

W.O. 04-4461-120

Safety Class	M.M. WASELUS	J.M. Reitmaner	12/8/80
CLASSIFICATION	ORIGINATOR	PROJECT ENGINEER	DATE

THIS ANALYSIS CONTAINS ASSUMPTIONS WHICH REQUIRE LATER CONFIRMATION.

____ YES (IDENTIFY BY PAGE NUMBER BELOW) X NO

COMPUTER PROGRAMS

X NOT USED ____ USED CERTIFIED PROGRAM PER DCP1.50

____ USED, PROGRAM NOT CERTIFIED BUT VERIFIED PER CAP1.00

M.M. Waselus — ORIGINATOR — 12/18/80 — DATE

NO VERIFICATION REQUIRED, DESIGN DUPLICATES PREVIOUSLY VERIFIED DESIGN NOTED BELOW:

NAME ____

FILING CODE ____ DATE ____

VERIFICATION IS REQUIRED (SEE DCP 2.05, SECTION 2.7):

METHOD DESIGN REVIEW

(IF QUALIFICATION TESTS ARE USED ATTACH COPY OF TEST CONDITIONS AND ACCEPTANCE CRITERIA)

VERIFIER K A SLIVKA — J.M. Reitmaner — PROJECT ENGINEER — 4/10/81 — DATE

X SELECTION OF VERIFIER

____ USE OF QUALIFICATION TEST VERIFICATION METHOD, IF APPLICABLE

M.M. Waselus — SECTION MANAGER — 4/10/81 — DATE

VERIFIER'S ATTESTATION:

THE (DESIGN DOCUMENT HAS) (REVISIONS TO THE DOCUMENT OCCURRING SINCE THE PREVIOUS VERIFICATION HAVE) BEEN REVIEWED BY ME IN ACCORDANCE WITH DCP 2.05. ANY FINDINGS UNCOVERED DURING THE COURSE OF MY REVIEW HAVE BEEN DIRECTED TO THE ORIGINATOR AND RESOLVED.

K.A. Slivka — VERIFIER — 4-10-81 — DATE

P.E. REVIEW AND APPROVAL (FOR DESIGN ANALYSIS AND CALCULATION ONLY)

J.M. Reitmaner — PROJECT ENGINEER — 4/10/81 — DATE

Gilbert Associates, Inc. Reading, Pennsylvania ANALYSIS/CALCULATION	SUBJECT Containment Post-LOCA Dose		CISID 1.6.1		PAGE 2 OF 5 PAGES
	REV.	0	1	2	3
	MICROFILMED				
	ORIGINATOR		M.Waselus		
	DATE		12/18/80		

Object:

To calculate acceptable values for the containment post-LOCA integrated dose for V. C. Summer for equipment qualification using the information contained in References 1 and 2.

References:

1. IE Bulletin No. 79-01B, "Environmental Qualification of Class IE Equipment," January 14, 1980.
2. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December, 1979.
3. V. C. Summer FSAR Section 6.2.
4. V. C. Summer FSAR Figures 1.2-4, 1.2-5, and 1.2-10.

A. Containment Centerline Dose for 30 Days

Assumptions:

1. Containment free volume - 1.84×10^6 ft^3 (Ref. 3 - Table 6.2-1)
2. Reactor Power - 2910 Mwt (Ref. 3 - Table 6.2-2)

Results:

From Figure 1 of Reference 1 the resultant integrated dose is given as 2×10^7 R.

For qualification a value of $\underline{2.4 \times 10^7 \text{ R}}$ is specified to include a 10% factor to cover potential β exposure and a 10% margin for conservatism.

For equipment qualification for a MSLB accident, a value of 2.4×10^6 R is acceptable (Figure 1, Reference 1).

B. Conservative estimate of Post-LOCA integrated dose to "short term" equipment on the 412' and 436' elevations of V. C. Summer reactor building outside the steam generator compartments.

Assumptions:

1. The equipment is conservatively assumed to be located in a compartment equivalent to one half the free volume of the elevation. This is conservative in that no credit is taken for reduction of the volume by any equipment or concrete walls.

M

2. The required operational time is 1 hour.

3. The reduction factors are taken from Reference 1, Figures 2, 3, and 4.

4. A minimum of 24" concrete shielding is available (Reference 4) to reduce the dose from the centerline cloud dose.

Calculation:

1. The total integrated centerline dose as given in Item A (above) is 2.4×10^7 R.

2. From Figure 2 (Reference 1) at a dose of 2.4×10^7 R and 24" of concrete the resultant dose is 7.0×10^4 R. To this, the dose from inside the compartment must be added.

3. The free volume of the elevations are given as follows:

412' $V = \pi r^2 h$

$= \pi(63\ ft)^2\ (434-412)\ ft$

$= 274{,}320\ ft^3$

436' $v = \pi(63\ ft)^2\ (461-436)\ ft$

$= 311{,}724\ ft^3$

Where the configuration and dimensions are given in Reference 4.

Using 1/2 the larger volume, the dose correction factor from Reference 1, Figure 3 is ~~0.18~~ 0.36 (conservative assumption since the larger volume results in a higher dose).

The sum of the doses is then given as =

$7 \times 10^4\ R + (2.4 \times 10^7\ R)\ (\cancel{0.18}\ 0.36) = \cancel{4.39}\ 8.71 \times 10^6\ R$

4. From Figure 4 (Reference 1) the correction factor for 1 hour operational time is 0.15. Therefore, the resultant integrated dose for equipment at 412' and 436' elevations is ~~6.6×10^5~~ 1.3×10^6 R. (One hour required function time.)

C. Conservative estimate of the containment post-LOCA centerline dose for 4 months and 1 year.

The containment integrated LOCA dose provided in Item A was for a time period of 30 days. Subsequently, the need was identified for integrated doses for longer time periods. In order to expedite an answer to meet client requirements, conservative doses were established based on extrapolating the data given in Reference 1 and Item A.

Assumptions:

1. The 30 day integrated dose is 2.4×10^7 R as given in Item A.

2. Figure 1 (attached) p.5 is a plot of the integrated dose values from Tables D-5 and D-8 of Reference 1. The doses for 4 months and 1 year are conservatively estimated by extrapolating the 30 day design value of a constant slope. The maximum value is based on the containment sump curve and is to be used as the design basis for equipment qualification.

Calculation:

$$\frac{\text{Relative Dose at time of interest}}{\text{Relative Dose at 30 days}} \times \text{Design 30 day dose} = \text{Design dose at time of interest}$$

At 4 Months: $\frac{6.6 \times 10^6}{3.96 \times 10^6} \times 2.4 \times 10^7 \text{ R} = \underline{4.0 \times 10^7 \text{ R}}$

At 1 Year: $\frac{1.0 \times 10^7}{3.96 \times 10^6} \times 2.4 \times 10^7 \text{ R} = \underline{6.1 \times 10^7 \text{ R}}$

Note: The above are acceptable values for equipment qualification post-LOCA inside containment except for equipment located directly above the sump, in the vicinity of filters or equipment recirculating post-LOCA fluids, or submerged in contaminated liquids.

Results:

The results were documented in the following memos:

1. Memo from M. M. Waselus to J. W. Reitnauer, "V. C. Summer Nuclear Station Post-LOCA Containment Integrated Dose," December 12, 1980.

2. Memo from M. M. Waselus to J. W. Reitnauer, "V. C. Summer Nuclear Station Post-LOCA Containment Integrated Dose," December 18, 1980.

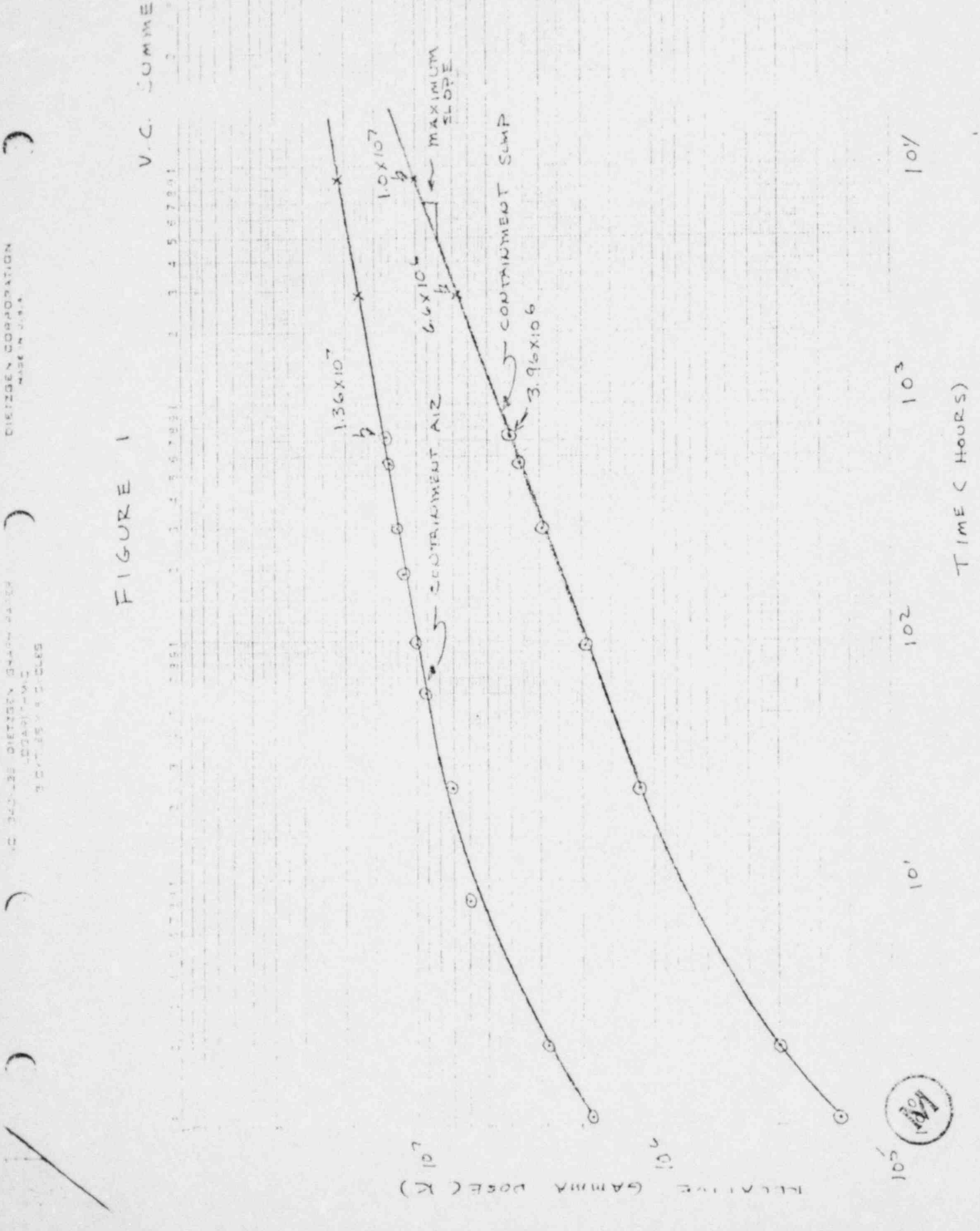
Copies of the memos are given as an Attachment to this calculation.

3. Memo from M.M. Waselus to J.W. Reitnauer, "V.C. Summer Nuclear Station Post-LOCA Containment Integrated Dose", April 10, 1981.

M

Containment Post-LOCA DC + F.C. 1.6.1 Rev. 2 P 5/5

M Wasselus
12/18/80



Attachment 1
F.C. 1.6.1 Rev.2
M Waselus
12/12/80

ATTACHMENT 1

Conservative estimate of Post-LOCA integrated dose to "short term" equipment on the 412' and 436' elevations of V. C. Summer reactor building.

Assumptions:

1. The equipment is conservatively assumed to be located in a compartment equivalent to one half the free volume of the elevation. This is conservative in that no credit is taken for reduction of the volume by any equipment or concrete walls.

2. The required operational time is 1 hour.

3. The reduction factors are taken from Reference 1, Figures 2, 3, and 4.

4. A minimum of 24" concrete shielding is available.

Calculation:

1. The total integrated dose as given in the memo is 2.4×10^7 R.

2. From Figure 2 at a dose of 2.4×10^7 R and 24" of concrete the resultant dose is 7.0×10^4 R. To this, the dose from inside the compartment must be added.

3. The free volume of the elevations are given as follows:

412' $V = \pi r^2 h$

$= \pi (63 \text{ ft})^2 \ (434-412) \text{ ft}$

$= 274{,}320 \text{ ft}^3$

Attachment 1
F.C. 1.6.1 Rev. 2
M Waselus
12/12/80

436' $v = \pi\ (63\ ft)^2\ (461\text{-}436)\ ft$

$= 311,724\ ft^3$

Using 1/2 the larger volume, the dose correction factor from Reference 1, Figure 3 is 0.18.

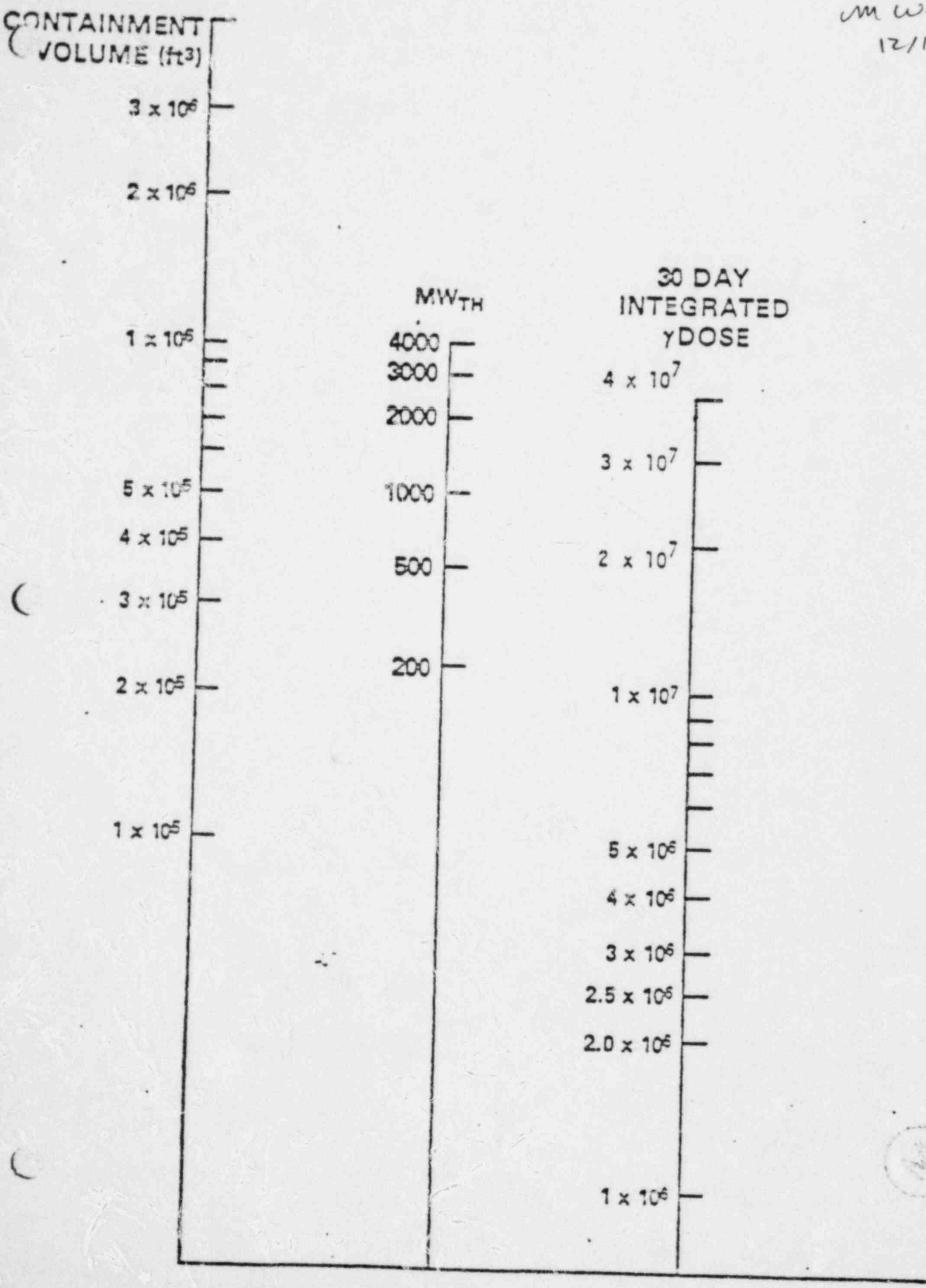
The sum of the doses is then given as =

$7 \times 10^4\ R + (2.4 \times 10^7\ R)\ (0.18) = 4.39 \times 10^6\ R$

4. From Figure 4 (Reference 1) the correction factor for 1 hour operational time is 0.15. Therefore, the resultant integrated dose for equipment at 412' and 436' elevations is 6.6 x 10^5 R. (One hour required functional time).

FIGURE 1
NOMOG   FOR CONTAINMENT VOLUME AND R   TOR POWER
LOCA DOSE CORRECTIONS*

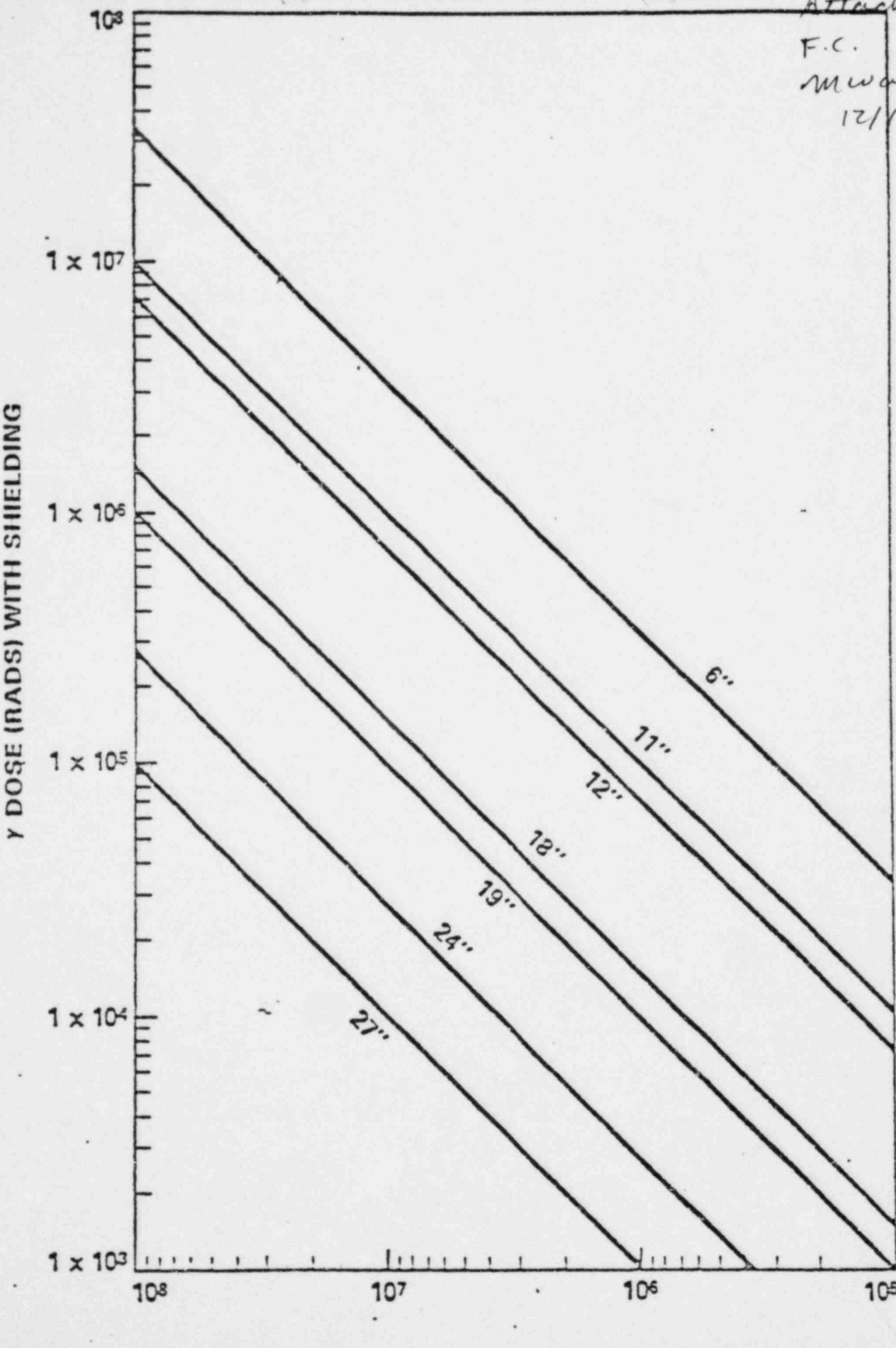
Attachment 1
F.C. 1.6.1. Rev
M Waselus
12/12/80



*MSLB ACCIDENT DOSES SHOULD BE READ AS A FACTOR OF ...

DOSE CORRECTION FACTOR FOR CONCRETE SHIELDING
(γ ONLY)

Attachment 1
F.C. 1.6.1 Rev 2
M Waselus
12/12/80



γ DOSE (RADS) WITHOUT SHIELDING (FROM FIGURE 1)

Attachment 1
F.C. 1.6.1 Rev. 2
M. Wasley
12/12/80

FIGURE 3
DOSE CORRECTION FACTOR FOR COMPARTMENT VOLUME

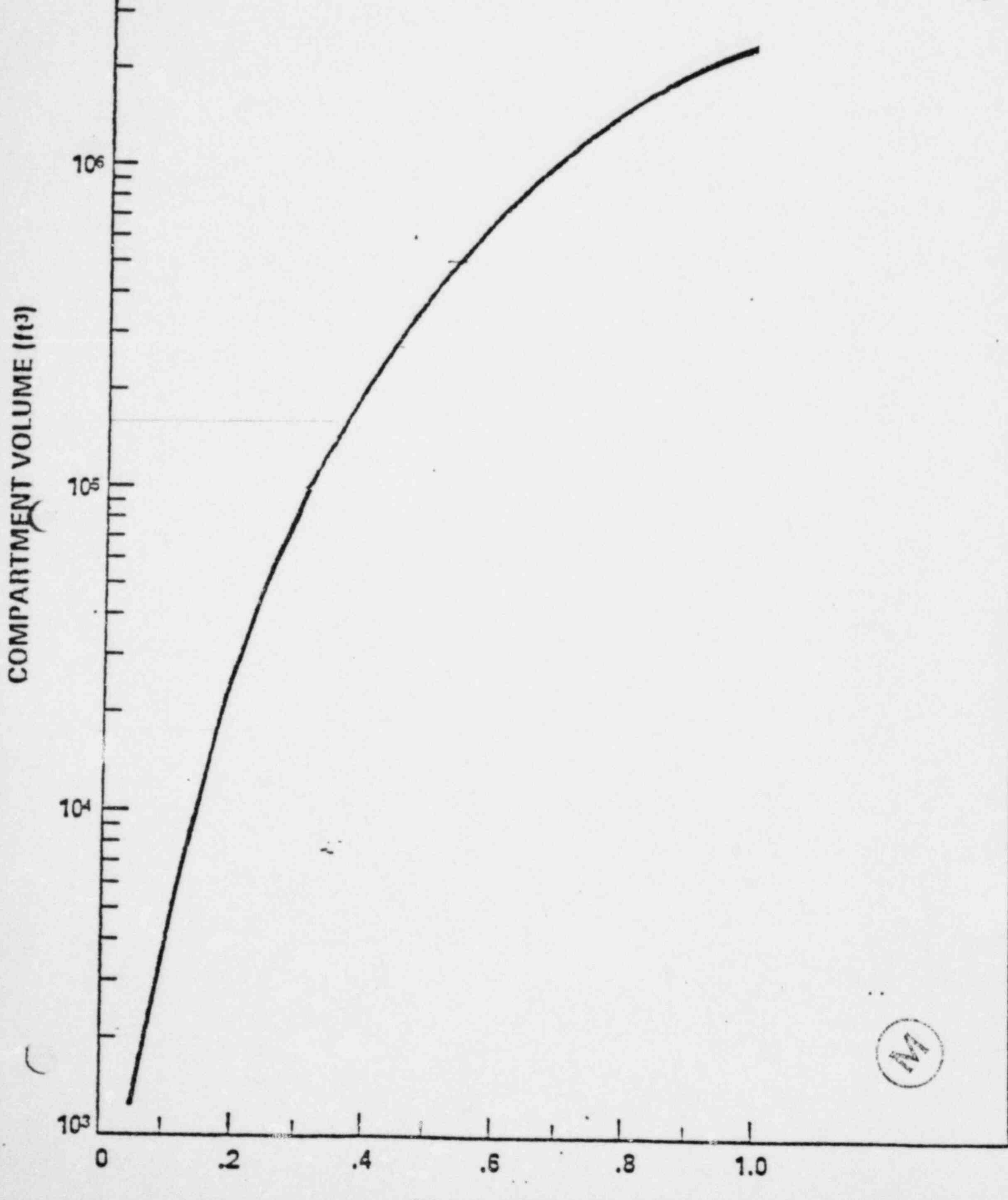
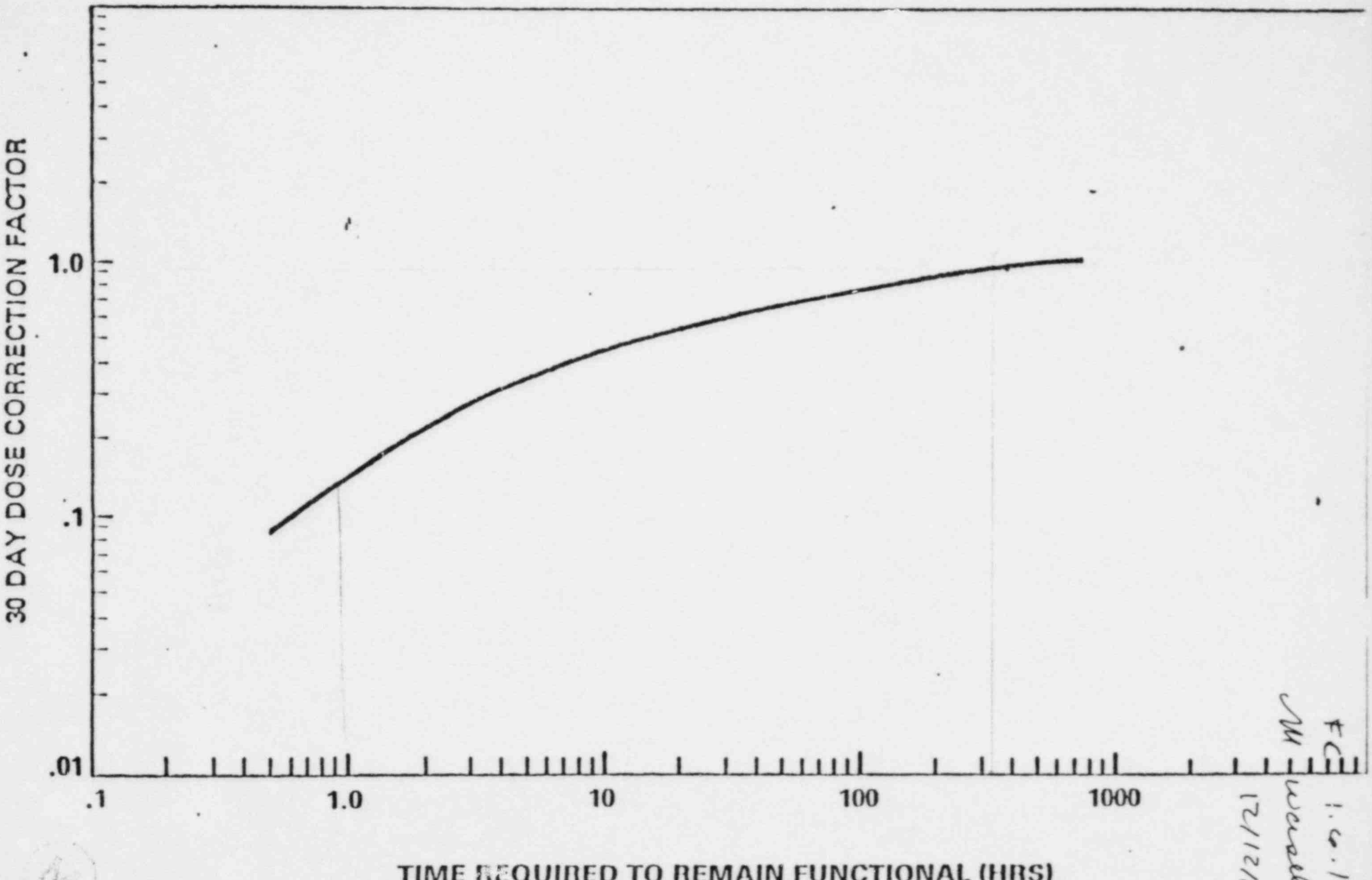


FIGURE 4

DOSE CORRECTION FOR TIME REQUIRED TO REMAIN FUNCTIONAL



FC 1.6.1 Rev. 2
M Weisselberg
12/12/80

memorandum Gilbert/Commonwealth

Attachment 7
F.C. 1.6.1 Rev 2
MMWaselus
12/18/80

to: J. W. REITNAUER

from: M. M. Waselus

subject: V. C. Summer Nuclear Station
Post-LOCA Containment Integrated Dose — December 18, 1980

Ref: 1. Memo from M. M. Waselus to J. W. Reitnauer, "V. C. Summer Nuclear Station Post-LOCA Containment Integrated Dose," December 12, 1980.

2. IE Bulletin No. 79-01B, "Environmental Qualification of Class IE Equipment," January 14, 1980.

The containment integrated LOCA dose provided in Reference 1 was for a time period of 30 days. Subsequently, the need was identified for integrated doses for longer time periods. In order to expedite an answer to meet client requirements, conservative doses were established based on extrapolating the data given in References 1 and 2. The following are acceptable values for equipment qualification post-LOCA inside containment except for equipment located directly above the sump, in the vicinity of filters or submerged in contaminated liquids:

4 month integrated dose - 4.0×10^7 R
one year integrated dose - 6.1×10^7 R

The method by which these values were determined is provided in Attachment 1 to this memo.

M. M. Waselus
M. M. WASELUS
NUCLEAR ANALYSIS

MMW:mm

Attachments

cc: H. E. Yocom
G. J. Braddick
F. X. Rehill
G. M. Kowal

GAI 35 12 71

Attachment 2
F.C. 1.6.1 Rev. 2
M Wasely
12/18/80

ATTACHMENT 1

Conservative estimate of Post-LOCA containment integrated dose for four months and one year. (V. C. Summer Nuclear Station)

Assumptions:

1. The 30 day integrated dose is 2.4×10^7 R as given in Reference 1.

2. Figure 1 of the attachment is a plot of the integrated dose values from Tables D-5 and D-8 of Reference 2. The doses for 4 months and 1 year are conservatively estimated by extrapolating the 30 day design value at a constant scope. The maximum value is based on the containment sump curve and is to be used as the design basis for equipment qualification.

Calculation:

$$\frac{\text{Relative Dose at time of interest}}{\text{Relative Dose at 30 days}} \times \text{Design 30 day dose} = \text{Design dose at time of interest}$$

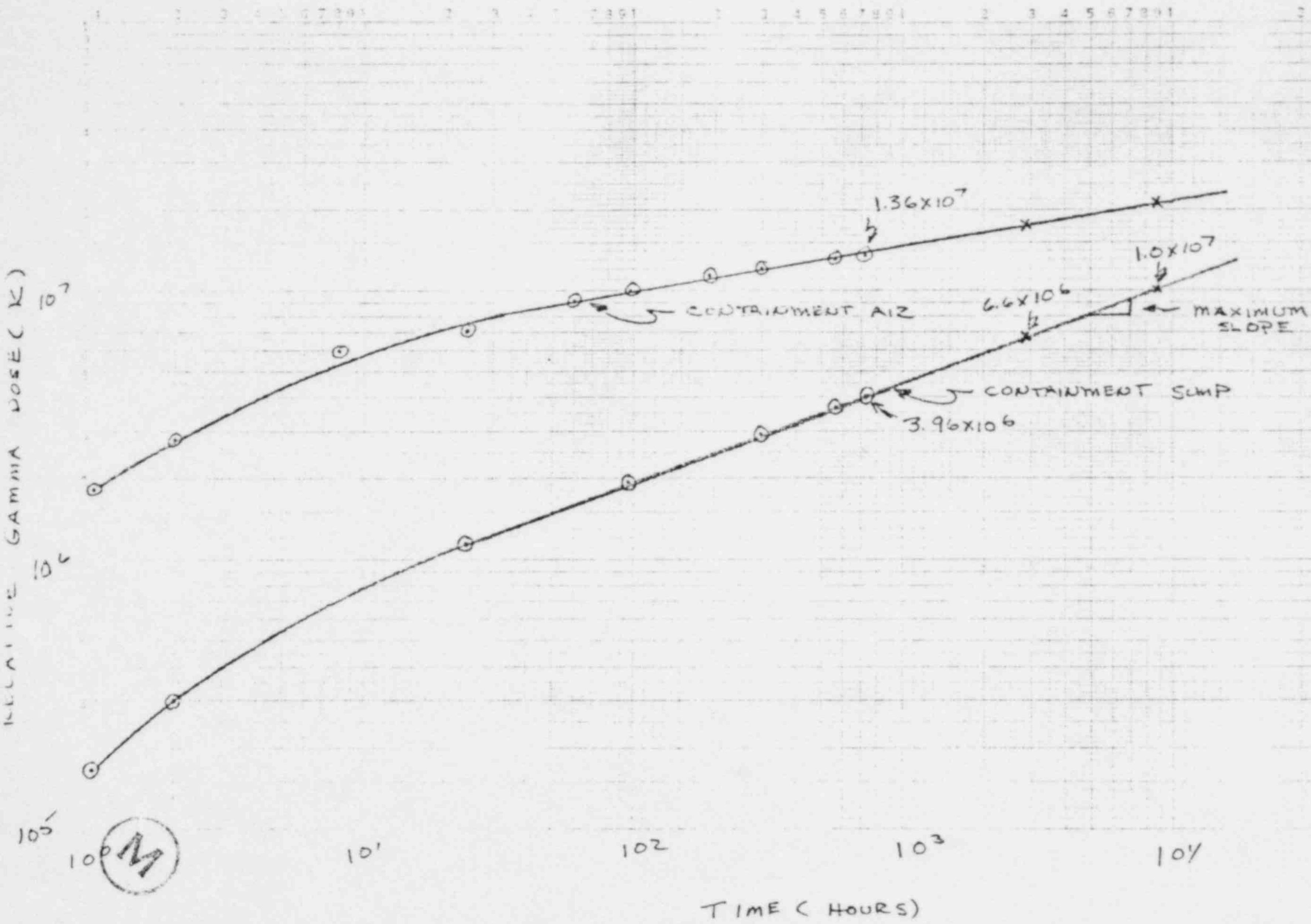
At 4 Months : $\dfrac{6.6 \times 10^6}{3.96 \times 10^6} \times 2.4 \times 10^7 \text{ R} = \underline{4.0 \times 10^7 \text{ R}}$

At 1 Year : $\dfrac{1.0 \times 10^7}{3.96 \times 10^6} \times 2.4 \times 10^7 \text{ R} = \underline{6.1 \times 10^7 \text{R}}$

M

FIGURE 1

V.C. SUMMER



Attachment 2
F.C. 1.6.1 Rev. 2
MWasdens
12/18/80

J. W. Reitnauer

ATTACHMENT 3
F.C. 1.6.1 REV 2
M. Waselus
12/18/80

memorandum Gilbert/Commonwealth

to: J. W. REITNAUER

from: M. M. Waselus

subject: V. C. Summer Nuclear Station
Post-LOCA Containment Integrated Dose April 10, 1981

Ref.: 1. Memo from M. M. Waselus to J. W. Reitnauer, "V. C. Summer Nuclear Station Post-LOCA Containment Integrated Dose," December 12, 1980.

During design review of the calculation package, an error was discovered in one of the values presented in Reference 1.

The incorrect data was for the one hour integrated dose to equipment at the 412' and 436' elevations outside the steam generator compartments. The correct value is 1.3×10^6 R which supersedes the previously reported value of 6.6×10^5 R (Ref. 1). All other reported values were determined to be correct.

MMWaselus
M. M. WASELUS
NUCLEAR ANALYSIS

MMW:mm

cc: H. E. Yocom
G. J. Braddick
G. M. Kowal

GAI 35 12/75

Gilbert Commonwealth Companies	DESIGN VERIFICATION RECORD	PAGE 1 OF 3
	PROJECT: V. C. Summer Nuclear Station, Unit 1	FILING CODE 1.6.1A
	SUBJECT: CONTAINMENT POST-LOCA DOSE, 4 MONTH OPERABILITY REQUIREMENT	REVISION: 0 DATE: 6/4/81
SECTION NAME AND NUMBER Applied Engineering Analysis Department Nuclear (2109) and Mechanical (2107) Sections		W.O. 04-4461-

Safety Class — CLASSIFICATION
K.A. SLIVKA — ORIGINATOR
J.L. Reitnauer — PROJECT ENGINEER
6/4/81 — DATE

THIS ANALYSIS CONTAINS ASSUMPTIONS WHICH REQUIRE LATER CONFIRMATION.

____ YES (IDENTIFY BY PAGE NUMBER BELOW) ✓ NO

COMPUTER PROGRAMS

✓ NOT USED ____ USED CERTIFIED PROGRAM PER DCP1.50

____ USED, PROGRAM NOT CERTIFIED BUT VERIFIED PER CAP1.00

K.A. Slivka — ORIGINATOR
6-4-81 — DATE

NO VERIFICATION REQUIRED, DESIGN DUPLICATES PREVIOUSLY VERIFIED DESIGN NOTED BELOW:

NAME ____

FILING CODE ____ DATE ____

VERIFICATION IS REQUIRED (SEE DCP 2.05, SECTION 2.7):

METHOD by design review

(IF QUALIFICATION TESTS ARE USED ATTACH COPY OF TEST CONDITIONS AND ACCEPTANCE CRITERIA)

VERIFIER S.S. Musser
J.L. Reitnauer — PROJECT ENGINEER
6/4/81 — DATE

✓ SELECTION OF VERIFIER

____ USE OF QUALIFICATION TEST VERIFICATION METHOD, IF APPLICABLE

W.M. Waselus — SECTION MANAGER
6/4/81 — DATE

VERIFIER'S ATTESTATION:

THE (DESIGN DOCUMENT HAS) (REVISIONS TO THE DOCUMENT OCCURRING SINCE THE PREVIOUS VERIFICATION HAVE) BEEN REVIEWED BY ME IN ACCORDANCE WITH DCP 2.05. ANY FINDINGS UNCOVERED DURING THE COURSE OF MY REVIEW HAVE BEEN DIRECTED TO THE ORIGINATOR AND RESOLVED.

S.S. Musser — VERIFIER
6/4/81 — DATE

P.E. REVIEW AND APPROVAL (FOR DESIGN ANALYSIS AND CALCULATION ONLY)

J.L. Reitnauer — PROJECT ENGINEER
6/4/81 — DATE

PURPOSE: Calculation of Containment post-LOCA integrated dose for 4 month operability duration at 436'-0" elevation.

COMPUTER CODES: None used.

INPUT & ASSUMPTIONS:

1. 30 day post-LOCA integrated dose to equipment on 436' elev. = 8.71×10^6 R (reference 1)
2. A graph of relative dose vs. post-LOCA time (Figure 1 of ref. 1 — attachment 1 of this Calc.) from Tables D-5 & D-8 of ref. 2 was used to calculate the 4-month operability design dose. (see next pg.)
3. others as noted.

REFERENCES:

1. VCS File 1.6.1, "Containment post-LOCA Dose", REV 2, M.M. Wasilus, 4-10-81.
2. IE Bulletin No. 79-01B, "Environmental Qualification of Class IE Equipment", Jan. 14, 1980.

RESULTS: 4 month integrated dose = 1.45×10^7 RADS.

CALCULATION:

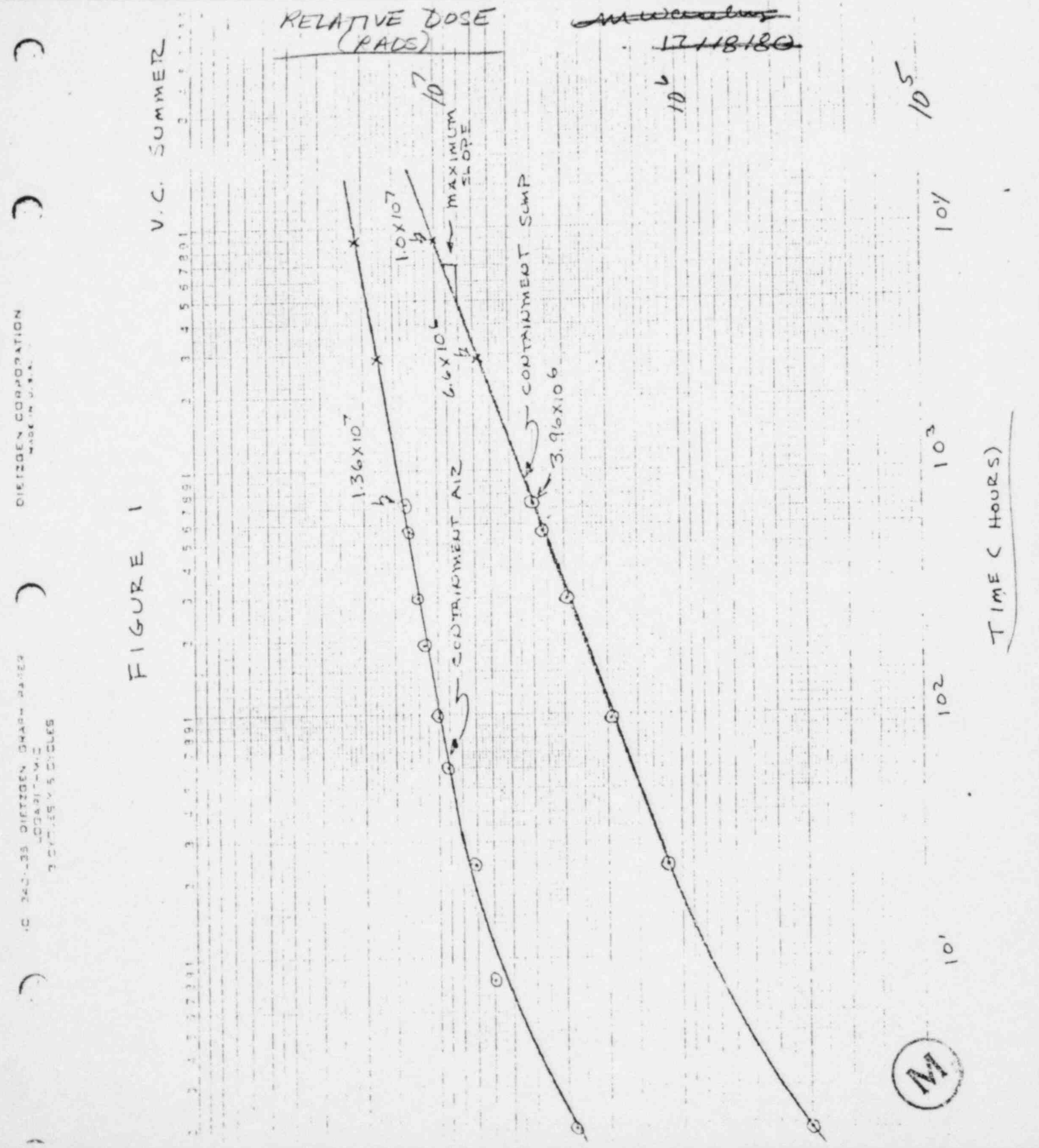
1. 30-day post-LOCA dose (ref. 1) = 8.71×10^6 RADS at 412' & 436' elevations outside stm. gen. Compartments

2. $\dfrac{\text{Relative dose @ 4 months}}{\text{Relative dose @ 30 days}} \times$ Design 30 day dose

= Design dose @ 4 months
(see attach. 1 - contain. sump curve).

$$\frac{6.6 \times 10^6 R}{3.96 \times 10^6 R} \times 8.71 \times 10^6 R = \underline{1.45 \times 10^7 \text{ RADS}}$$

VCS File 1.6.1.A
K.A. Aliwka, 6-4-81
ATTACHMENT #1



	DESIGN VERIFICATION RECORD	PAGE 1 OF 3
Gilbert Commonwealth Companies	PROJECT: V. C. Summer Nuclear Station, Unit 1	FILING CODE 1.6.1B
	SUBJECT: CONTAINMENT POST-LOCA DOSE, 1 YEAR OPERABILITY REQUIREMENT	REVISION: 0 DATE: 9/21/81
SECTION NAME AND NUMBER Applied Engineering Analysis Department Nuclear (2109) and Mechanical (2107) Sections		W.O. 04-4461-120

Safety Class	J.W. REITNAUER	JWReitnauer	9/21/81
CLASSIFICATION	ORIGINATOR	PROJECT ENGINEER	DATE

THIS ANALYSIS CONTAINS ASSUMPTIONS WHICH REQUIRE LATER CONFIRMATION.

____ YES (IDENTIFY BY PAGE NUMBER BELOW) ✓ NO

COMPUTER PROGRAMS

✓ NOT USED ____ USED CERTIFIED PROGRAM PER DCP1.50

____ USED, PROGRAM NOT CERTIFIED BUT VERIFIED PER CAP1.00

JWReitnauer — ORIGINATOR 9/21/81 — DATE

NO VERIFICATION REQUIRED, DESIGN DUPLICATES PREVIOUSLY VERIFIED DESIGN NOTED BELOW:

NAME CONTAINMENT POST-LOCA DOSE, 4 MONTH OPERABILITY REQUIREMENT

FILING CODE 1.6.1A, REVISION 0 DATE 6/4/81

VERIFICATION IS REQUIRED (SEE DCP 2.05, SECTION 2.7):

METHOD ~

(IF QUALIFICATION TESTS ARE USED ATTACH COPY OF TEST CONDITIONS AND ACCEPTANCE CRITERIA)

VERIFIER ~ ~ PROJECT ENGINEER ~ DATE

____ SELECTION OF VERIFIER

____ USE OF QUALIFICATION TEST VERIFICATION METHOD, IF APPLICABLE

~ SECTION MANAGER ~ DATE

VERIFIER'S ATTESTATION:

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~ VERIFIER ~ DATE

P.E. REVIEW AND APPROVAL (FOR DESIGN ANALYSIS AND CALCULATION ONLY)

JWReitnauer — PROJECT ENGINEER 9/21/81 — DATE

Gilbert Associates, Inc. Reading, Pennsylvania ANALYSIS/CALCULATION	JECT CONTAINMENT POST-LOCA DOSE, 1 YEAR OPERABILITY REQUIREMENT		CISID 1.6.1B		PAGE 2 OF 3 PAGES
	REV. 0	1	2	3	
	MICROFILMED				
	ORIGINATOR M.E. Turner				
	DATE 9/21/81				

Purpose

The calculation of post-LOCA integrated dose to equipment located on the 436'-0" and 412'-0" elevations of the reactor building over a 1 year post-LOCA period.

References

(1) File 1.6.1, revision 2: Containment Post-LOCA Dose. M.M. Warabus, originator, 12/18/80. K.A. Slivka, verifier 4/10/81.

(2) File 1.6.1A, revision 0: Containment Post-LOCA Dose, 4-month Operability Requirement. K.A. Slivka, originator, 6/4/81. S.S. Murser, verifier 6/4/81.

Computer Codes

No computer codes used.

Assumptions

Assumptions, if any, included within analysis.

Analysis

The 30-day post-LOCA integrated dose to equipment on elevations 412'-0" and 436'-0" is calculated in reference (1) as 8.71×10^6 R.

Using page 4 of reference (1), the relationship between the doses at 30 days and 1 year can be determined:

$$\frac{\text{Relative dose at 1 year}}{\text{Relative dose at 30 days}} = \frac{\text{Design dose at 1 year}}{\text{Design dose at 30 days}}$$

$$\frac{1.0 \times 10^7 \text{ Rads}}{3.96 \times 10^6 \text{ Rads}} = \frac{\text{Design dose at 1 year}}{8.71 \times 10^6 \text{ Rads}}$$

$\therefore$ Design dose at 1-year $= 2.20 \times 10^7$ Rads.

Summary

The equipment located on elevations 412'-0" and 436'-0" of the Reactor Building will receive a airborne dose of 2.20×10^7 Rads, except in the areas of the sump and the RCS piping, during the 1-year post-LOCA period.

Gilbert Commonwealth Companies

DESIGN VERIFICATION RECORD

PROJECT: V. C. Summer Nuclear Station, Unit 1	FILING CODE 1.6.1C
SUBJECT: CONTAINMENT POST-LOCA DOSE, 7 DAY OPERABILITY REQUIREMENT	REVISION: 0 DATE: 9/24/81
SECTION NAME AND NUMBER Applied Engineering Analysis Department Nuclear (2109) and Mechanical (2107) Sections	W.O. 04-4461-120

Safety Class	J.W. REITNAUER	JWReitnauer	9/24/81
CLASSIFICATION	ORIGINATOR	PROJECT ENGINEER	DATE

THIS ANALYSIS CONTAINS ASSUMPTIONS WHICH REQUIRE LATER CONFIRMATION.

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JWReitnauer — ORIGINATOR 9/24/81 — DATE

NO VERIFICATION REQUIRED, DESIGN DUPLICATES PREVIOUSLY VERIFIED DESIGN NOTED BELOW:

NAME CONTAINMENT POST-LOCA DOSE, 4 MONTH OPERABILITY REQUIREMENT

FILING CODE 1.6.1A, REVISION 0 DATE 6/4/81

VERIFICATION IS REQUIRED (SEE DCP 2.05, SECTION 2.7):

METHOD ~

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VERIFIER ~ ~ PROJECT ENGINEER ~ DATE

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~ SECTION MANAGER ~ DATE

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P.E. REVIEW AND APPROVAL (FOR DESIGN ANALYSIS AND CALCULATION ONLY)

M

JWReitnauer — PROJECT ENGINEER 9/24/81 — DATE

Gilbert/Commonwealth

POWER ENGINEERING - READING

GAI 468 6-80

Purpose

The calculation of post-LOCA integrated dose to equipment located on the 436'-0" and 412'-0" elevations of the reactor building over a 7 day post-LOCA period.

References

(1) File 1.6.1, revision 2: Containment Post-LOCA Dose. M.M. Waselus, originator, 12/18/80. K.A. Slivka, verifier, 4/10/81.

(2) File 1.6.1A, revision 0: Containment Post-LOCA Dose, 4 Month Operability Requirement. K.A. Slivka, originator, 6/4/81. S.S. Musser, verifier, 6/4/81.

Computer Codes

No computer code used.

Assumptions

Assumptions, if any, included within analysis.

Analysis

The 30 day post-LOCA integrated dose to equipment on elevations 412'-0" and 436'-0" is calculated in reference (1) as 8.71×10^6 Rads.

Using the methodology on page 4 of reference (1), the relationship between the doses at 30 days and 7 days can be determined:

$$\frac{\text{Relative dose at 7 days}}{\text{Relative dose at 30 days}} = \frac{\text{Design dose at 7 days}}{\text{Design dose at 30 days}}$$

$$\frac{2.6 \times 10^6 \text{ Rads}}{3.96 \times 10^6 \text{ Rads}} = \frac{\text{Design dose at 7 days}}{8.71 \times 10^6 \text{ Rads}}$$

$$\text{Design dose at 7 days} = 8.71 \times 10^6 \text{ Rads} \times \frac{2.6 \times 10^6 \text{ Rads}}{3.96 \times 10^6 \text{ Rads}}$$

$$= 5.72 \times 10^6 \text{ Rads.}$$

Gilbert Associates, Inc.
Reading, Pennsylvania

ANALYSIS/CALCULATION

SUBJECT Containment Post-LOCA Dose 7 Day Operability Requirement			JOB ID [illegible]		PAGE 3 OF 3 PAGES
REV.	0	1	2	3	
MICROFILMED					
ORIGINATOR	[illegible]				
DATE	9/24/81				

Summary

The equipment located on elevations 412'-0" and 436'-0" of the Reactor Building will receive a airborne dose of 5.72×10^6 Rads over a 7-day post-LOCA period, except in the areas of the sump and RCS piping.

Gilbert Commonwealth Companies	**DESIGN VERIFICATION RECORD**	
	PROJECT: V. C. Summer Nuclear Station, Unit 1	FILING CODE 1.8.5.6
	SUBJECT: EQUIPMENT DOSES IN CONTAINMENT	REVISION: 3 DATE: 11/17/81
SECTION NAME AND NUMBER Applied Engineering Analysis Department Nuclear (2109) and Mechanical (2107) Sections		W.O. 04-4461-120

Safety Class — CLASSIFICATION
J.W. REITNAUER — ORIGINATOR
J.W. Reitnauer — PROJECT ENGINEER
11/12/81 — DATE

THIS ANALYSIS CONTAINS ASSUMPTIONS WHICH REQUIRE LATER CONFIRMATION.

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COMPUTER PROGRAMS

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J.W. Reitnauer — ORIGINATOR
11/16/81 — DATE

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VERIFICATION IS REQUIRED (SEE DCP 2.05, SECTION 2.7):

METHOD by design review

(IF QUALIFICATION TESTS ARE USED ATTACH COPY OF TEST CONDITIONS AND ACCEPTANCE CRITERIA)

VERIFIER K.A. Slivka
J.W. Reitnauer — PROJECT ENGINEER
11/16/81 — DATE

✓ SELECTION OF VERIFIER

____ USE OF QUALIFICATION TEST VERIFICATION METHOD, IF APPLICABLE

W.M. Wascher — SECTION MANAGER
11/17/81 — DATE

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K.A. Slivka — VERIFIER
11/17/81 — DATE

P.E. REVIEW AND APPROVAL (FOR DESIGN ANALYSIS AND CALCULATION ONLY)

J.W. Reitnauer — PROJECT ENGINEER
11/17/81 — DATE

Gilbert Associates, Inc. Reading, Pennsylvania ANALYSIS/CALCULATION	SUBJECT Equipment Doses in Containment			[illegible]	PAGE 2 OF 27 PAGES
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	ORIGINATOR	[illegible]	[illegible]	[illegible]	
	DATE	[illegible]	[illegible]	[illegible]	

Objective of Revision 1

The objective of Revision 1 to this analysis is the calculations of the total integrated dose received by the equipment listed below for the operability requirements given:

Tag Number	Location	Operability requirement	Category
ILT-1969-LD	RB 12-01SW	7 day	A1, A2
ILT-1970-LD	RB 12-01SW	7 day	A1, A2

Objective of Revision 2

(1) The calculation of the total integrated dose received by equipment located in the vicinity of the reactor building recirculation sump for the following times post-accident: 4 months, 1 year

(2) The revision of the 7 day total integrated dose calculated in revision 1 to correct an error in the methodology used.

Objective of Revision 3

(1) The calculation of the total integrated dose received by equipment located in the containment below floor elevation [illegible] for the following times post-accident: 1 week, 4 months, 1 year.

(2) The calculation of the total integrated dose received by the equipment located in the vicinity of the reactor building recirculation sump for 1 month operability post-accident.

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Gilbert Associates, Inc. Reading, Pennsylvania ANALYSIS/CALCULATION	SUBJECT Equipment Doses in Containment			ISID 04-4461-120 1.8.5.6	PAGE 3 OF 27 PAGES
	REV.	0	1	2	3
	MICROFILMED				
	ORIGINATOR	J.W. Reitnauer	J.W. Reitnauer	J.W. Reitnauer	
	DATE	4/15/81	6/4/81	7/1/81	

Objective

The objective of this analysis is the calculation of the total integrated dose received by the equipment located in the containment following either a Loss-of-Coolant Accident (LOCA) or a Main Steam Line Break (MSLB) upon the last day of the 40th year of operation.

Authorization

Part of the NUREG-0588 review effort.

References

(1) NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December, 1979.

(2) IE Bulletin No. 79-01B, "Environmental Qualification of Class 1E Equipment," January 14, 1980.

(3) File 1.6.1, Revision 2: "Containment Post-LOCA Dose" M.M. Waselus, originator 12/18/80. K.A. Slivka, verifier 4/10/81.

(4) File 1.8.9.5, Revision 1: "Containment Sump Dose to Sump Equipment." J.W. Reitnauer, originator, 6/30/80. K.A. Slivka, verifier, 7/1/81. ②

(5) File 1.6.2.1, Revision 0: "Steam Generator Compartment Piping Doses - Normal and Accident." J.W. Reitnauer originator 4/22/81. S.S. Musser verifier, 4/24/81. ②

(6) File 1.6.2, Revision 0: "Steam Generator Compartment Doses." J.O. Burtt originator 8/19/79. J.T. Kamphouse, verifier, 8/15/79.

(7) "Qualification of Gilbert Sump Level Nuclear Transmitter By Similarity to the 1153 Series A." RMT Report No. 17931, revision A. GAI identifier 1MS-92-3542-1. ②

(8) File 1.6.1A, revision 0: "Containment Post-LOCA Dose, 4 month Operability Requirement." K.A. Slivka, originator, 6/4/81. S.S. Musser, verifier, 6/4/81.

(9) File 1.6.1B, revision 0: "Containment Post-LOCA Dose, 1 Year Operability Requirement." J.W. Reitnauer, originator, 9/21/81.

(10) File 1.6.1C, revision 0: "Containment Post-LOCA Dose, 7 Day Operability Requirement." J.W. Reitnauer, originator, 9/24/81.

3

Assumptions

The assumptions used are contained within the analysis/calculation section.

Analysis/Calculations

Integrated Dose Received During Normal Plant Operating Conditions

The normal operating doses are calculated using the plant radiation zone designations contained in Chapter 12 of the V.C. Summer FSAR.

The area dose is assumed to be the dose received in the area due to a constant maximum area radiation dose rate given in Chapter 12 over the expected 40-year life of the plant.

Dose Rate (DR_{NORMAL})

≤ 1 mR/hr	$ID_{NORMAL} = (1\ mR/hr)(40\ yr)(8765.8\ hr/yr)(10^{-3}\ R/mR)$	$= 3.51\times10^2$ R.
≤ 2.5 mR/hr	(2.5 mR/hr)	$= 8.77\times10^2$ R
≤ 15 mR/hr	(15 mR/hr)	$= 5.26\times10^3$ R
≤ 100 mR/hr	(100 mR/hr)	$= 3.51\times10^4$ R

The specified dose limits shall be:

Dose Rate (DR_{NORMAL})	Calculated Dose	Specified Integrated Dose (ID_{NORMAL})
≤ 1 mR/hr	3.51×10^2 R.	$\leq 5\times10^2$ R
≤ 2.5 mR/hr	8.77×10^2 R	$\leq 10^3$ R
≤ 15 mR/hr	5.26×10^3 R	$\leq 10^4$ R
≤ 100 mR/hr	3.51×10^4 R	$\leq 10^6$ R
> 100 mR/hr	~	$\leq 10^7$ R

(3)

Integrated Dose Due to Accident Environment

Using the methodology contained in NUREG-0588 and IE Bulletin 79-01B (references (1) and (2)), the following accident-related doses were calculated in references (3), (8), (9), and (10):

(a) At and above operating floor elevation 463'-0", the containment centerline accident-caused dose are as follows:

(1) Due to LOCA

for 1 month duration = 2.4×10^7 Rads

for 4 month duration = 4.0×10^7 Rads

for 1 year duration = 6.1×10^7 Rads

(2) Due to MSLB

for 1 month duration = 2.4×10^6 Rads

for 4 month duration = 4.0×10^6 Rads

for 1 year duration = 6.1×10^6 Rads

(b) Below the operating floor elevation 463'-0" and outside the steam generator cubicles, the airborne accident-related doses are as follows:

(1) Due to LOCA

for 1 hour duration = 1.3×10^6 Rads

for 1 week duration = 5.72×10^6 Rads

for 1 month duration = 8.71×10^6 Rads

for 4 month duration = 1.45×10^7 Rads

for 1 year duration = 2.20×10^7 Rads

(2) Due to MSLB

16% of the above LOCA dose

(c) The above data is not acceptable for equipment located above the sump, in the vicinity of filters, or equipment recirculating fluids or submerged in contaminated liquids.

The total integrated dose (TID) at the containment centerline for points at and above the operating floor elevation 463'-0" (airborne activity):

LOCA, 30 day duration: $TID = \text{Normal 40-yr. dose} + \text{accident ID}$
$= 10^6 \text{ Rad} + 2.4 \times 10^7 \text{ Rad}$
$= 2.5 \times 10^7 \text{ Rads.}$

LOCA, 4 month duration: $TID = 10^6 \text{ Rad} + 4.0 \times 10^7 \text{ Rad} = 4.1 \times 10^7 \text{ Rads.}$

LOCA, 1 year duration: $TID = 10^6 \text{ Rad} + 6.1 \times 10^7 \text{ Rad} = 6.2 \times 10^7 \text{ Rads.}$

MSLB, 30 day duration: $TID = 10^6 \text{ Rad} + 2.4 \times 10^6 \text{ Rad} = 3.4 \times 10^6 \text{ Rad}$

MSLB, 4 month duration: $TID = 10^6 \text{ Rad} + 4.0 \times 10^6 \text{ Rads} = 5.0 \times 10^6 \text{ Rad.}$

MSLB, 1 year duration: $TID = 10^6 \text{ Rad} + 6.1 \times 10^6 \text{ Rad} = 7.1 \times 10^6 \text{ Rads.}$

The total integrated dose (TID) in the areas outside the steam generator compartments and below the operating floor elevation 463'-0" due to airborne activity:

LOCA, 1 hour duration: $TID = 10^6 \text{ Rads} + 1.3 \times 10^6 \text{ Rad} = 2.3 \times 10^6 \text{ Rad}$

LOCA, 30 day duration: $TID = 10^6 \text{ Rad} + 8.71 \times 10^6 \text{ Rad} = 9.71 \times 10^6 \text{ Rad}$

MSLB, 1 hour duration: $TID = 10^6 \text{ Rad} + 1.3 \times 10^5 \text{ Rad} = 1.13 \times 10^6 \text{ Rad}$

MSLB, 30 day duration: $TID = 10^6 \text{ Rad} + 8.71 \times 10^5 \text{ Rad} = 1.87 \times 10^6 \text{ Rad}$

NOTE: The above data contains a 10% factor to cover potential β-ray exposure and a 10% margin for conservatism.

LOCA, 1 week duration: TID = 10^6 Rads + 5.72×10^6 Rads = 6.7×10^6 Rads. ③

LOCA, 4 month duration: TID = 10^6 Rads + 1.45×10^7 Rads = 1.6×10^7 Rads

LOCA, 1 year duration: TID = 10^6 Rads + 2.20×10^7 Rads = 2.3×10^7 Rads

MSLB, 1 week duration: TID = 10^6 Rads + 5.72×10^5 Rads = 1.6×10^6 Rads

MSLB, 4 month duration: TID = 10^6 Rads + 1.45×10^6 Rads = 2.5×10^6 Rads

MSLB, 1 year duration: TID = 10^6 Rads + 2.20×10^6 Rads = 3.2×10^6 Rads.

Gilbert Associates, Inc. Reading, Pennsylvania	PROJECT Equipment Doses in Containment		CISID 04-4461-120 1.8.5.6		PAGE 9 OF 27 PAGES
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	DATE		7/6/81		

The total integrated dose (TID) inside the steam generator compartments due to reactor coolant system piping:

The normal and post-accident doses received by equipment in the vicinity of the hot and cold leg piping within the steam generator cubicles following a LOCA was calculated in reference (5). The results of the analysis are as follows:

Dose Point Location	DBA	Total Integrated Dose (Rads)					
		1 week	2 weeks	30 days	4 month	6 month	1 year
Cold Leg Pipe Exterior Surface	LOCA	3.53+7	1.85+7	1.89+7	2.01+7	2.07+7	2.22+7
	MSLB	1.65+7	1.65+7	1.66+7	1.67+7	1.67+7	1.69+7
Hot Leg Pipe Exterior Surface	LOCA	2.50+7	2.53+7	2.57+7	2.65+7	2.68+7	2.72+7
	MSLB	2.35+7	2.35+7	2.35+7	2.36+7	2.36+7	2.37+7
Hot Leg Pipe Thermwell	LOCA	1.06+8	1.10+8	1.16+8	1.26+8	1.30+8	1.35+8
	MSLB	8.77+7	8.81+7	8.87+7	8.98+7	9.01+7	9.06+7

The cold leg pipe thermwell TIDs were not calculated, since reference (5) showed that a lower dose rate exists at the dose receptor point that that shown for the comparable point at the hot leg piping.

(M)

The total integrated dose (TID) in the vicinity of or submerged in the reactor building sump in reactor building cubicle 12-01SW:

Due to the presence of two 2-ft. concrete floor slabs at elevations 463'-0" and 436'-0", the contribution of the airborne centerline dose will be minimal:

From reference (3) → total integrated centerline dose = 2.9×10^7 R.

From reference (2), figure 2 → the 4 feet of concrete shielding will attenuate the centerline dose to $< 10^3$ R.

In reference (3), the source terms used in the dose considered the sump water to contain the following:

100% of the noble gas core inventory

50% of the halogen core inventory

1% of all other nuclides in the core inventory.

Thus, the consideration of the dose contributions from both the sump water and the compartment airborne activity would result in a overly conservative dose value.

Using the methodology outlined in reference (3), the dose received from the compartment airborne activity has the following maximum value for the cubicle 12-01SW:

Due to location of the sumps within one quadrant of the 412'-0" elevation of the reactor building, assume the free volume of the elevation applicable to this problem to be:

from reference (3) = Volume $\doteq 274{,}320 ft^3$

effective compartment volume = $(0.25)(274{,}320 ft^3)$

$= 68{,}580 ft^3$

Dose correction factor for compartment volume = 0.28

Total airborne dose = 1.0×10^3 R. + $(0.28)(2.9 \times 10^7 R)$

$\doteq 6.72 \times 10^6$ R.

Therefore instead of considering the sum of both the sump dose and the compartment airborne doses we shall consider only the greater of the two quantities as being the integrated post-accident dose.

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The post-accident doses received in the vicinity of or submerged in the reactor building sump were calculated in reference (4) as follows:

(a) TID for 6 mo. post-LOCA to submerged equipment = 7.49×10^7 Rads.
(b) TID for 6 mo. post-LOCA at sump surface edge = 3.72×10^7 Rads.
(c) TID for 6 mo. post-LOCA at sump surface center = 3.65×10^7 Rads.
(d) TID for 6 mo. post-LOCA at el. 420'-0" = 3.18×10^7 Rads
el. 425'-0" = 1.33×10^7 Rads
el. 430'-0" = 6.41×10^6 Rads

The examination of the sump and airborne doses indicates that a conservative decision can be made to consider the post-LOCA integrated dose within the cubicle to be 3.72×10^7 Rads. The results of other analytical work indicates that this will be a reasonably conservative maximum value for cubicle 12-01SW.

TID 6 mo. post-LOCA to submerged equipment

$= ID_{NORMAL} + ID_{LOCA}^{6 MO.}$
$= 1.0 \times 10^6$ Rad $+ 7.49 \times 10^7$ Rad
$= 7.59 \times 10^7$ Rads.

TID 6 mo. post-LOCA [illegible] surface and above:

$= ID_{NORMAL} + ID_{LOCA}^{6 MO}$ + 10% beta margin
$= 1.0 \times 10^7$ Rad $+ 3.72 \times 10^7$ Rad $+ (0.10 \times 3.72 \times 10^7$ Rad$)$
$= 5.09 \times 10^7$ Rads.

TID 6 mo. post-MSLB to submerged equipment

$= ID_{NORMAL} + ID_{MSLB}^{6 MO}$
$= ID_{NORMAL} + 0.10 \times ID_{LOCA}^{6 MO}$
$= 1.0 \times 10^6$ Rad $+ 0.10 \times 7.49 \times 10^7$ Rad
$= 8.49 \times 10^6$ Rad.

TID 6 mo. post-MSLB at sump surface and above

$= ID_{NORMAL} + ID_{MSLB}^{6 MO} + ID_{BETA}$
$= 1.0 \times 10^7$ Rad $+ 0.10 \times ID_{LOCA}^{6 MO} + 0.01 \times ID_{LOCA}^{6 MO}$
$= 1.0 \times 10^7$ Rad $+ 0.10 \times 3.72 \times 10^7$ Rad $+ 0.01 \times 3.72 \times 10^7$ Rad.
$= 1.41 \times 10^7$ Rads.

The total integrated doses (TID) to equipment located within the R.B. sump during the operability period requirements of 1week, 1month, 4month, +1 years:

The following data is taken from reference (4):

Time Interval Post-LOCA	Dose Received From Accident Source Within Time Interval				
	Shielded + submerged	Unshielded + submerged	Unshielded + @ surface edge	Unshielded + @ surface center	Unshielded + @ el. 420'-0"
0hr to 1hr	2.105+5 R.	1.997+6 R.	9.935+5 R.	1.023+6 R.	8.789+5 R.
1hr to 8hr.	5.986+5 R.	6.407+6 R.	3.191+6 R.	3.243+6 R.	2.788+6 R.
8hr to 20day	2.457+6 R.	3.622+7 R.	1.805+7 R.	1.782+7 R.	1.545+7 R.
20day to 4mo.	1.521+6 R.	2.497+7 R.	1.241+7 R.	1.194+7 R.	1.051+7 R.
20day to 6mo.	1.868+6 R.	3.024+7 R.	1.502+7 R.	1.442+7 R.	1.272+7 R.
6mo to 1year	5.552+5 R.	8.710+6 R.	4.320+6 R.	4.087+6 R.	3.628+7 R.
Time Post-LOCA	Total Accumulated Dose Received From Accident				
1hr	2.105+5 R.	1.997+6 R.	9.935+5 R	1.023+6 R.	8.789+5 R.
8hr	8.091+5 R.	8.404+6 R.	4.185+6 R.	4.266+6 R.	3.667+6 R.
20day	3.266+6 R.	4.462+7 R.	2.223+7 R.	2.209+7 R.	1.912+7 R.
4 mo.	4.787+6 R.	6.959+7 R.	3.464+7 R.	3.403+7 R.	2.963+7 R.
6 mo.	5.134+6 R.	7.486+7 R.	3.725+7 R.	3.651+7 R.	3.184+7 R.
1 yr.	5.689+6 R.	8.357+7 R.	4.157+7 R.	4.059+7 R.	6.812+7 R.

The above total accumulated dose received due to the accident are graphed on the following pages on a per needed basis. Our current needs are for 1week dose received by ILT-1969-LD and ILT-1970-LD. Reference (7) states that these instruments are shielded. The above "shielded and submerged" case considers this situation.

From the graph, $ID_{LOCA}^{1WK} = 2.45 \times 10^6$ Rads and $ID_{LOCA}^{1MO.} = 3.56 \times 10^6$ Rads for shielded + submerged

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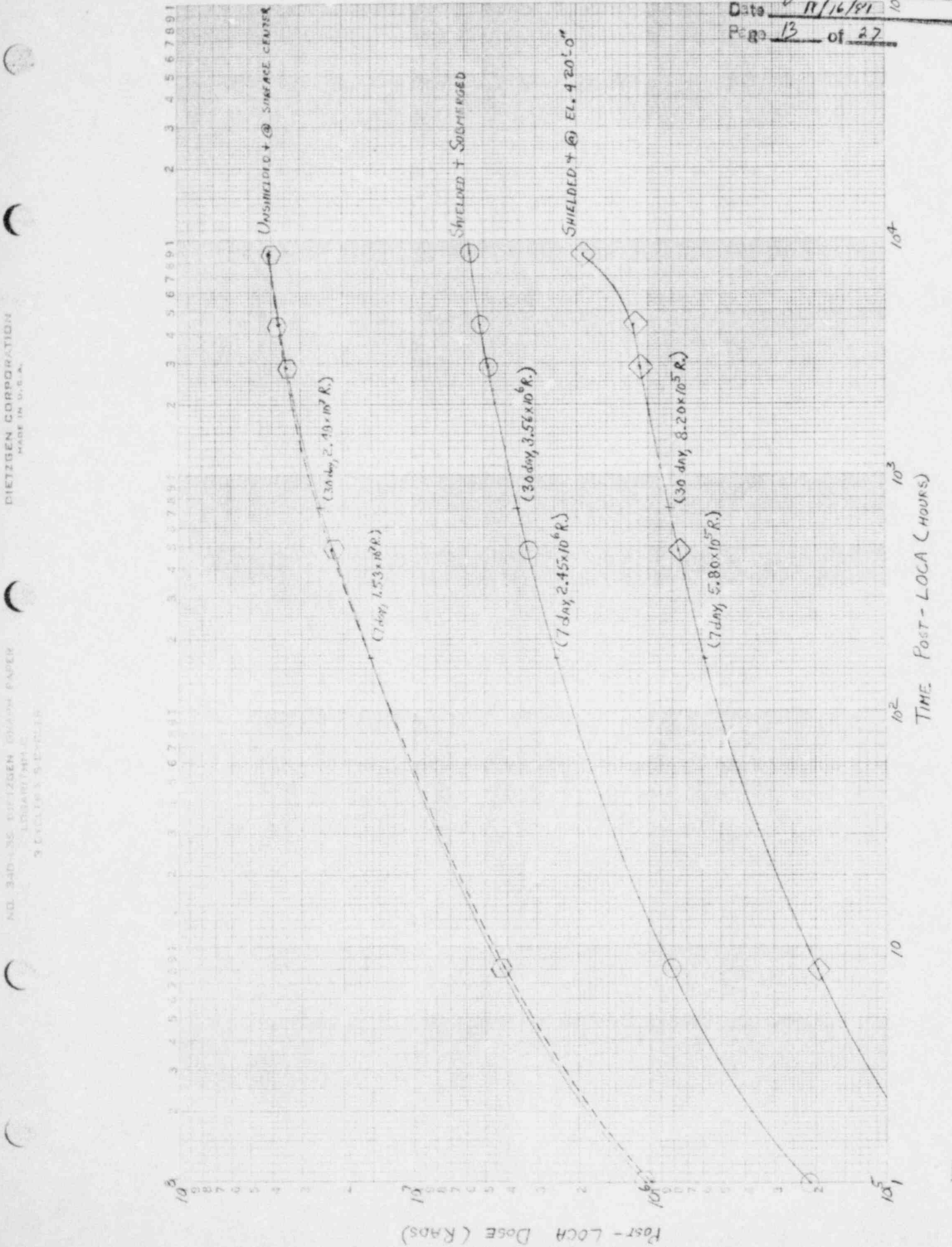

The dose to equipment located at elevation 420'-0" on the centerline shall be calculated using the following data: (2)

Time Interval Post-LOCA	Interval Dose Shielded + @ el. 420'-0"
0hr to 1hr	5.233+4 R.
1hr to 8hr	1.448+5 R.
8hr to 20day	5.572+5 R.
20day to 4mo.	3.267+5R.
20day to 6mo.	3.775+5 R.
6mo. to 1year	7.328+5R.
Time Post-LOCA	**T.A.B.**
1hr	5.233+4R.
8hr	1.971+5 R.
20day	7.543+5R.
4mo.	1.081+6 R.
6mo.	1.154+6R.
1yr.	1.887+6

From the graph, $ID^{1WK}_{LOCA} = 5.80\times10^5$ Rads
$ID^{1MO}_{LOCA} = 8.20\times10^5$ Rads.

(M)

(2)

TID 1 week post-LOCA to shielded, submerged equipment:

$= ID_{NORMAL} + ID_{LOCA}^{1WK}$

$= 1.0\times10^{6}R. + 2.45\times10^{6} R.$ (10^{7} written above 10^{6})

$= 3.45\times10^{6}R.$

TID 1 week post-MSLB to shielded, submerged equipment

$= ID_{NORMAL} + 0.10\times ID_{LOCA}^{1WK}$

$= 1.0\times10^{6}R. + 0.10\times 2.45\times10^{6}R.$

$= 1.25\times10^{6}R.$

TID 30 days post-LOCA to shielded, submerged equipment

$= ID_{NORMAL} + ID_{LOCA}^{30d.}$

$= 1.0\times10^{6} R. + 3.56\times10^{6}$ Rads

$= 4.56\times10^{6} R.$

TID 30 days post MSLB to shielded, submerged equipment

$= ID_{NORMAL} + 0.10\times ID_{LOCA}^{30d}$

$= 1.0\times10^{6} R. + 0.10\times 3.56\times10^{6} R.$

$= 1.36\times10^{6} R.$

TID 1 week post-LOCA to shielded equipment at el. 420'-0"

$= ID_{NORMAL} + ID_{LOCA}^{1WK}$ + 10% beta margin

$= 1.0\times10^{7} R. + 5.80\times10^{5}R. + 0.10\times5.80\times10^{5}R.$

$= 1.06\times10^{7}R.$

TID 1 week post-MSLB to shielded equipment at el. 420'-0"

$= 1.0\times10^{7}R + 0.1\times 5.80\times10^{5} R.$

$= 1.01\times10^{7} R.$

TID 30 days post-LOCA to shielded equipment at el. 420'-0"

$= 1.0\times10^{7}R + 8.20\times10^{5}R. + 0.10\times 8.20\times10^{5}R.$

$= 1.09\times10^{7} R.$

Y

②

TID 30 days post-MSLB to shielded equipment at el. 420'-0"

$= 1.0 \times 10^7 R + 0.10 \times 8.20 \times 10^5 R.$

$= 1.01 \times 10^7 R.$

TID 4 month post-LOCA to unshielded equipment at sump surface and above

$= ID_{NORMAL} + ID_{LOCA}^{4mo} + 10\%$ beta margin

$= 1.0 \times 10^7 R. + 3.403 \times 10^7 R. + 0.10 \times 3.403 \times 10^7 R.$

$= 4.74 \times 10^7 R.$

TID 4 month post-MSLB to unshielded equipment at sump surface and above

$= ID_{NORMAL} + 0.10 \times ID_{LOCA}^{4MO.}$

$= 1.0 \times 10^7 R. + 0.10 \times 3.403 \times 10^7 R.$

$= 1.34 \times 10^7 R.$

TID 1 year post-LOCA to unshielded equipment at sump surface and above

$= ID_{NORMAL} + ID_{LOCA}^{1YR} + 10\%$ beta margin

$= 1.0 \times 10^7 R. + 4.059 \times 10^7 R. + 0.10 \times 4.059 \times 10^7 R.$

$= 5.46 \times 10^7 R.$

TID 1 year post-MSLB to unshielded equipment at sump surface and above

$= ID_{NORMAL} + 0.10 \times ID_{LOCA}^{1YR}$

$= 1.0 \times 10^7 R + 0.10 \times 4.059 \times 10^7 R$

$= 1.41 \times 10^7 R.$

The dose to unshielded equipment located at the sump surface center shall be calculated using the graph found on page 13. The graph contains the data listed on page 10 as "unshielded + @ surface edge."

From the graph, $ID_{LOCA}^{1mo.} = 2.48 \times 10^7$ Rads

TID 30 day post-LOCA to unshielded equipment at sump surface center

$= ID_{NORMAL} + ID_{LOCA}^{1mo}$

$= 1.0 \times 10^7$ Rads $+ 2.48 \times 10^7$ Rads

$= 3.5 \times 10^7$ Rads

TID 30 day post-MSLB to unshielded equipment at sump surface center

$= ID_{NORMAL} + 0.10 \times ID_{LOCA}^{1mo}$

$= 1.0 \times 10^7$ Rads $+ 0.10 \times 2.48 \times 10^7$ Rads

$= 1.25 \times 10^7$ Rads.

The dose to unshielded equipment located at the sump surface edge:

From the graph, $ID_{LOCA}^{1mo} = 2.55 \times 10^7$ Rads

TID 30 day post-LOCA to unshielded equipment at sump surface edge

$= ID_{NORMAL} + ID_{LOCA}^{1mo}$

$= 1.0 \times 10^7$ Rads $+ 2.55 \times 10^7$ Rads

$= 3.55 \times 10^7$ Rads

TID 30 day post-MSLB to unshielded equipment at sump surface edge

$= ID_{NORMAL} + 0.10 \times ID_{LOCA}^{1mo}$

$= 1.0 \times 10^7$ Rads $+ 0.10 \times 2.55 \times 10^7$ Rads

$= 1.26 \times 10^7$ Rads

Due to the minor differences in doses shown above, we shall use the following for all sump surface doses at 30 days post-accident:

3.6×10^7 Rads 30 days post-LOCA

1.3×10^7 Rads 30 days post-MSLB

(3)

M

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	MICROFILMED				
	ORIGINATOR		J.M. Petrowsky	J.M. Petrowsky	
	DATE		7/8/81	11/16/81	

Summary

The total integrated doses received by the equipment located in the containment following either a loss-of-coolant (LOCA) or a main steam line break (MSLB) accident upon the last day of the 40th year of operation are as follows:

Airborne dose above el. 463'-0":

Operability requirement	LOCA	MSLB
30 day	2.5+7 R.	3.4+6 R.
4 month	4.1+7 R.	5.0+6 R.
1 year	6.2+7 R.	7.1+6 R.

Doses below el. 463'-0":

Operability Requirement	Outside S.G. LOCA	Outside S.G. MSLB	Inside S.G. (Airborne) LOCA	Inside S.G. (Airborne) MSLB	Inside S.G. (RCS Pipe) LOCA	Inside S.G. (RCS Pipe) MSLB	
1 hour	2.3+6 R.	1.1+6 R.			N.R.	N.R.	
1 week	6.7+6 R.	1.6+6 R.	Not significant		3.53+7 R.	1.65+7 R.	3
2 week	N.R.	N.R.	in comparison		2.53+7 R.	2.35+7 R.	
30 days	9.7+6 R.	1.9+6 R	to RCS Pipe		2.57+7 R.	2.35+7 R.	
4 month	1.6+7 R.	2.5+6 R.	doses		2.65+7 R.	2.36+7 R.	3
6 month	N.R.	N.R.			2.68+7 R.	2.36+7 R.	
1 year	2.3+7 R.	3.2+6 R.			2.72+7 R.	2.37+7 R.	3

RB cubicle 12-01SW:

Position of Dose Point	Operability Requirement	LOCA	MSLB	
① Submerged	6 month	7.6+7 R.	8.5+6 R.	
② Submerged, shielded	1 week	3.5+6 R.	1.3+6 R.	
	30 day	4.6+6 R.	1.4+6 R.	
③ Sump surface + above	1 month	3.6+7 R.	1.3+7 R.	3
	4 month	4.7+7 R.	1.3+7 R.	
	6 month	5.1+7 R.	1.4+7 R.	
	1 year	5.5+7 R.	1.4+7 R.	

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	ORIGINATOR		MKritvanov		
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		LOCA	MSLB
(4) Shielded eq. at el. 420'-0"	1 week	1.1+7R.	1.0+7R.
	1 month	1.1+7R.	1.0+7R.

The above data is summarized for worst case situation on pages ~~18~~ 20 through ~~23~~ 27 as it applies to the various areas of the reactor building.
The color key to the data given is as follows.

black numbers → 4 month TID
blue numbers → 6 month TID
green numbers → 1 year TID

For that equipment ... as cable, wire, and termination blocks which are routed or located throughout the reactor building we shall use the worst case data that can be found, which is as follows:

Due to 40 years of normal operating conditions → $\leq 1.0\times10^7$ R.

Operability requirement	LOCA	MSLB
30 day	2.5+7R.	2.4+7R.
4 month	4.7+7R.	2.4+7R.
6 month	5.1+7R.	2.4+7R.
1 year	6.2+7R.	2.4+7R.

Equipment assumed not to be flooded.