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RS-22-004 10 CFR 50.90

January 4, 2022

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

James A. FitzPatrick Nuclear Power Plant
Renewed Facility Operating License No. DPR-59
NRC Docket No. 50-333

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Nine Mile Point Nuclear Station, Unit 2
Renewed Facility Operating License No. NPF-69
NRC Docket No. 50-410

Peach Bottom Atomic Power Station, Units 2 and 3
Subsequent Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

R.E. Ginna Nuclear Power Plant
Renewed Facility Operating License No. DPR-18
NRC Docket No. 50-244

Subject: Supplement to Application to Adopt TSTF-554, Revision 1, "Revise Reactor Coolant Leakage Requirements"

Reference: Letter from P.R. Simpson (Exelon) to U.S. NRC, "Application to Adopt TSTF-554, Revision 1, 'Revise Reactor Coolant Leakage Requirements'," dated June 30, 2021 (ML21181A180)

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC), proposed a change to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License (FOL) Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2; Renewed FOL Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2; Renewed FOL Nos. DPR-53 sand DPR-69 for Calvert Cliffs Nuclear Power Plant, Units 1 and 2; FOL No. NPF-62 for Clinton Power Station, Unit 1; Renewed FOL Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station, Units 2 and 3; Renewed FOL No. DPR-59 for James A. FitzPatrick Nuclear Power Plant; Renewed FOL Nos. NPF-11 and NPF-18 for LaSalle County Station, Units 1 and 2; Renewed FOL Nos. NPF-39 and NPF-85 for Limerick Generation Station, Units 1 and 2; Renewed FOL No. NPF-69 for Nine Mile Point Nuclear Station, Unit 2; Renewed FOL Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station, Units 2 and 3; Renewed FOL Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station, Units 1 and 2; and Renewed FOL No. DPR-18 for R.E. Ginna Nuclear Power Plant. The proposed changes are consistent with previously NRC-approved Industry/Technical Specifications Task Force Traveler 554 (TSTF-554), Revision 1, "Revise Reactor Coolant Leakage Requirements."

The markup of Limerick TS 1.28 provided in the initial amendment request proposes to add the following sentence: “LEAKAGE past seals, packing, and gaskets is not PRESSURE BOUNDARY LEAKAGE.” During preparation and review of clean TS pages it was identified that the word “LEAKAGE” by itself should not be in all capital letters since this is not a defined term in the Limerick TS. An additional grammatical consistency issue was also identified regarding the markup to delete an existing semicolon at the end of the Unidentified LEAKAGE definition for certain plants, which if incorporated as originally submitted would punctuate one item in a list differently than the rest of the list. This grammatical change to not delete the existing semicolon affects one page of the previously submitted TS markups for Clinton,

Dresden, Fitzpatrick, LaSalle, Peach Bottom, and Quad Cities. The updated marked-up TS pages for each affected plant are included in Attachments 1 through 9. These pages supersede the corresponding markup page included in the referenced document. The remaining pages are unchanged from those submitted in the referenced document.

EGC has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Reference 1. EGC has concluded that the information provided in this response does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92. In addition, EGC has concluded that the information in this supplemental letter does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), a copy of this supplement, with attachments, is being provided to the designated State Officials.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Ms. Rebecca L. Steinman at (630) 657-2831.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 4th day of January 2022.

Respectfully,

Patrick R. Simpson
Sr. Manager Licensing
Exelon Generation Corporation, LLC

Attachments (only changed page(s) included):

1. Revised Markup of Clinton Power Station, Unit 1 Technical Specifications Pages
2. Revised Markup of Dresden Nuclear Power Station, Units 2 and 3 Technical Specifications Pages
3. Revised Markup of James A. FitzPatrick Nuclear Power Plant Technical Specifications Pages
4. Revised Markup of LaSalle County Station, Units 1 and 2 Technical Specifications Pages
5. Revised Markup of Limerick Generating Station, Unit 1 Technical Specifications Pages
5 Revised Markup of Limerick Generating Station, Unit 2 Technical Specifications Pages
7. Revised Markup of Peach Bottom Atomic Power Station, Unit 2 Technical Specifications Pages

8. Revised Markup of Peach Bottom Atomic Power Station, Unit 3 Technical Specifications Pages
9. Revised Markup of Quad Cities Nuclear Power Station, Units 1 and 2 Technical Specifications Pages

cc: NRC Regional Administrator, Region I
NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Clinton Power Station
NRC Senior Resident Inspector – Dresden Nuclear Power Station
NRC Senior Resident Inspector – James A. FitzPatrick Nuclear Power Plant
NRC Senior Resident Inspector – LaSalle County Station
NRC Senior Resident Inspector – Limerick Generating Station
NRC Senior Resident Inspector – Peach Bottom Atomic Power Station
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety
Director, Bureau of Radiation Protection – Pennsylvania Department of Environmental Protection
A.L. Peterson, NYSERDA
Bridget Frymire, NYSPSC

ATTACHMENT 1

Revised Markup of Clinton Power Station, Unit 1 Technical Specifications Pages

Facility Operating License No. NPF-62

NRC Docket No. 50-461

<u>Markup of Technical Specifications Page</u>

1.0-5

1.1 Definitions (continued)

END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial movement of the associated turbine stop valve or turbine control valve to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known ~~either~~ to not ~~to~~ interfere with the operation of leakage detection systems ~~or not to be pressure boundary LEAKAGE~~;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a ~~nonisolable~~ fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.

(continued)

ATTACHMENT 2

Revised Markup of Dresden Nuclear Power Station, Units 2 and 3 Technical Specifications Pages

Renewed Facility Operating License Nos. DPR-19 and DPR-25

NRC Docket Nos. 50-237 and 50-249

Markup of Technical Specifications Page

1.1-5

1.1 Definitions

DRAIN TIME (continued)

c. The penetration flow paths required to be evaluated per paragraph b) are assumed to open instantaneously and are not subsequently isolated, and no water is assumed to be subsequently added to the RPV water inventory;

d. No additional draining events occur; and

e. Realistic cross-sectional areas and drain rates are used.

A bounding DRAIN TIME may be used in lieu of a calculated value.

INSERVICE TESTING PROGRAM

The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known ~~either~~ to not ~~to~~ interfere with the operation of leakage detection systems ~~or not to be pressure boundary LEAKAGE~~;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

(continued)

ATTACHMENT 3

Revised Markup of James A. FitzPatrick Nuclear Power Plant
Technical Specifications Pages

Renewed Facility Operating License No. DPR-59

NRC Docket No. 50-333

Markup of Technical Specifications Page

1.1-3

1.1 Definitions (continued)

ISOLATION INSTRUMENTATION RESPONSE TIME

The ISOLATION INSTRUMENTATION RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valve receives the isolation signal (e.g., de-energization of the main steam isolation valve solenoids). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known ~~either~~ to not ~~to~~ interfere with the operation of leakage detection systems ~~or not to be pressure boundary LEAKAGE~~;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a ~~nonisolable~~ fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, or gaskets is not pressure boundary LEAKAGE.

(continued)

ATTACHMENT 4

Revised Markup of LaSalle County Station, Units 1 and 2 Technical Specifications Pages

Renewed Facility Operating License Nos. NPF-11 and NPF-18

NRC Docket Nos. 50-373 and 50-374

<u>Markup of Technical Specifications Page</u>

1.1-7

1.1 Definitions (continued)

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known to ~~either~~ not ~~to~~ interfere with the operation of leakage detection systems ~~or not to be pressure boundary LEAKAGE~~;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a ~~nonisolable~~ fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.

LINEAR HEAT GENERATION RATE (LHGR)

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

(continued)

ATTACHMENT 5

Revised Markup of Limerick Generating Station, Unit 1 Technical Specifications Pages

Renewed Facility Operating License No. NPF-39

NRC Docket No. 50-352

Markup of Technical Specifications Page

1-5

OPERATIONAL CONDITION - CONDITION

1.26 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.28 PRESSURE BOUNDARY LEAKAGE shall be leakage through a ~~nonisolable~~ fault in a reactor coolant system component body, pipe wall or vessel wall.

Leakage past seals, packing, and gaskets is not PRESSURE BOUNDARY LEAKAGE.

PRIMARY CONTAINMENT INTEGRITY

1.29 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. The primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the solidification or dewatering and packaging of radioactive wastes results in a waste package with properties that meet the minimum and stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site. With solidification or dewatering, the PCP shall identify the process parameters influencing solidification or dewatering, based on laboratory scale and full scale testing or experience.

ATTACHMENT 6

Revised Markup of Limerick Generating Station, Unit 2 Technical Specifications Pages

Renewed Facility Operating License No. NPF-85

NRC Docket No. 50-353

Markup of Technical Specifications Page

1-5

OPERATIONAL CONDITION - CONDITION

1.26 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.28 PRESSURE BOUNDARY LEAKAGE shall be leakage through a ~~nonisolable~~ fault in a reactor coolant system component body, pipe wall or vessel wall.

Leakage past seals, packing, and gaskets is not PRESSURE BOUNDARY LEAKAGE.

PRIMARY CONTAINMENT INTEGRITY

1.29 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. The primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the SOLIDIFICATION or dewatering and packaging of radioactive wastes results in a waste package with properties that meet the minimum and stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site. With SOLIDIFICATION or dewatering, the PCP shall identify the process parameters influencing SOLIDIFICATION or dewatering based on laboratory scale and full scale testing or experience.

ATTACHMENT 7

Revised Markup of Peach Bottom Atomic Power Station, Unit 2
Technical Specifications Pages

Renewed Facility Operating License No. DPR-44

NRC Docket No. 50-277

<u>Markup of Technical Specifications Page</u>

1.1-3a

1.1 Definitions (continued)

END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME

The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

INSERVICE TESTING PROGRAM

The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known ~~either~~ to not ~~to~~ interfere with the operation of leakage detection systems ~~or not to be pressure boundary LEAKAGE~~;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a ~~nonisolable~~ fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.

LINEAR HEAT GENERATION RATE (LHGR)

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

(continued)

ATTACHMENT 8

Revised Markup of Peach Bottom Atomic Power Station, Unit 3
Technical Specifications Pages

Renewed Facility Operating License No. DPR-56

NRC Docket No. 50-278

<u>Markup of Technical Specifications Page</u>

1.1-3a

1.1 Definitions (continued)

END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known ~~either~~ to not ~~to~~ interfere with the operation of leakage detection systems ~~or not to be pressure boundary LEAKAGE~~;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a ~~nonisolable~~ fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

(continued)

ATTACHMENT 9

Revised Markup of Quad Cities Nuclear Power Station, Units 1 and 2 Technical Specifications Pages

Renewed Facility Operating License Nos. DPR-29 and DPR-30

NRC Docket Nos. 50-254 and 50-265

<u>Markup of Technical Specifications Page</u>

1.1-5

1.1 Definitions

DRAIN TIME (continued)

c. The penetration flow paths required to be evaluated per paragraph b) are assumed to open instantaneously and are not subsequently isolated, and no water is assumed to be subsequently added to the RPV water inventory;

d. No additional draining events occur; and

e. Realistic cross-sectional areas and drain rates are used.

A bounding DRAIN TIME may be used in lieu of a calculated value.

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The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

LEAKAGE

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a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known to ~~either~~ not ~~to~~ interfere with the operation of leakage detection systems ~~or not to be pressure boundary LEAKAGE~~;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

(continued)