AECL EACL

Assessment Document

Identification and Initial Assessment of US NRC Generic Safety Issues Applicable to ACR

ACR USA

108US-01321-ASD-001
Revision 0

Prepared by
Rédigé par
C. Xu
PSA and Safety Design

Reviewed by
Vérifié par
R. Ion
Design Certification - Technical

Reviewed by
Vérifié par
E. Oesterle
Bechtel Power Corporation

Approved by
Approuvé par
V. Langman
Program Manager - Design Certification
AECL Technologies

2003 April
Avril 2003

CONTROLLED - Licensing
CONTRÔLÉ - Permis

2251 Speakman Drive
Mississauga, Ontario
Canada L5K 1B2

2251, rue Speakman
Mississauga (Ontario)
Canada L5K 1B2

AECL EACL

Assessment Document

Identification and Initial Assessment of US NRC Generic Safety Issues Applicable to ACR

ACR USA

108US-01321-ASD-001
Revision 0

2003 April

Avril 2003

CONTROLLED - Licensing

CONTRÔLÉ - Permis

2251 Speakman Drive
Mississauga, Ontario
Canada L5K 1B2

2251, rue Speakman
Mississauga (Ontario)
Canada L5K 1B2

AECL EACL

Release and Revision History / Liste des documents et des révisions

0939B Rev. 13

Document Details / Détails sur le document

Title / Titre	Total no. of pages / Nbre total de pages
Identification and Initial Assessment of US NRC Generic Safety Issues Applicable to ACR	

CONTROLLED – Licensing / CONTRÔLÉ - Permis

Release and Revision History / Liste des documents et des révisions

Release / Document No./N°	Release / Document Date	Revision / Révision No./N°	Revision / Révision Date	Purpose of Release; Details of Rev./Amendement / Objet du document; détails des rév. ou des modif.	Prepared by / Rédigé par	Reviewed by / Examiné par	Approved by / Approuvé par
1		D1	2003-03-06	Issued for review and comments.	C. Xu	R. Ion C. Reid	V. Langman M. Soulard
2		0	2003-04-25	Issued as "approved for use".	C. Xu	R. Ion E. Oesterle	V. Langman

DCS/RMS Input / Données SCD ou SGD

Rel. Proj. / Proj. conn.	Project / Projet	SI	Section	Serial / Série	Sheet / Feuille No. / N°	Of / De	Unit No.(s) / Tranche n°
	108US	01321	ASD	001	1	1	

dsfpdfmb

TABLE OF CONTENTS

SECTION **PAGE**

TABLES

LIST OF ACRONYMS

AC	Alternating Current
ACR™	Advanced CANDU® Reactor™*
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AECB	Atomic Energy Control Board (Now CNSC)
AECL	Atomic Energy of Canada Limited
AECLT	AECL Technologies Inc.
AEOD	Office of Analysis and Evaluation of Operational Data
AFW	Auxiliary Feedwater
AFWS	Auxiliary Feedwater System
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOV	Air-Operated Valve
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
B&OTF	Bulletin and Orders Task Force
B&W	Babcock and Wilcox Company
BOP	Balance-Of-Plant
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium, registered trademark of AECL
CCAFF	Core Cooling in the Absence of Forced Flow
CCF	Common Cause Failure
CDF	Core Damage Frequency
CE	Combustion Engineering Company
CFR	Code of Federal Regulations
CNSC	Canadian Nuclear Safety Commission
COL	Combined License
CP	Construction Permit
CR	Control Room

® CANDU® is a registered trademark of Atomic Energy of Canada Limited (AECL).

* ACR™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Limited (AECL).

CRD	Control Rod
CSA	Canadian Standards Association
CSB	Containment System Branch
CSS	Containment Spray System
DC	Direct Current
DCD	Design Control Document
DG	Diesel Generator
DHR	Decay Heat Removal
DOE	U.S. Department Of Energy
DOR	Division of Operating Reactors, NRR (defunct)
DSI	Division of Systems Integration, NRR (defunct)
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
ECI	Emergency Coolant Injection
EDG	Emergency Diesel Generator
EMP	Electromagnetic Pulse
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EQ	Environmental Qualification / Equipment Qualification
ERDS	Emergency Response Data System
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
ESFS	Engineered Safety Feature System
ESRP	Environmental Standard Review Plan
ESW	Essential Service Water
ETSB	Effluent Treatment System Branch, NRR (defunct)
FEMA	U.S. Federal Emergency Management Agency
FIST	Fuel Integral Simulation Test
FW	Feedwater
GDC	General Design Criteria
GE	General Electric Company
GL	Generic Letter
GSI	Generic Safety Issue
HOV	Hydraulic-Operated Valve
HP	High Pressure
HPCI	High Pressure Coolant Injection

HPI	High Pressure Injection
HTGR	High Temperature Gas Cooled Reactor
HTS	Heat Transport System
HVAC	Heating, Ventilating and Air-Conditioning
IAEA	International Atomic Energy Agency
ICC	Inadequate Core Cooling
ICS	Integrated Control System
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	Intergranular Stress Corrosion Cracking
INPO	Institute for Nuclear Power Operations
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISA	Instrument Society of America
ISI	In-Service Inspection
ISLOCA	Interfacing System Loss-Of-Coolant Accident
KI	Potassium Iodide
LBHS	Large Bore Hydraulic Snubber
LBLOCA	Large Break Loss-Of-Coolant Accident
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LMFBR	Liquid Metal Fast Breeder Reactor
LO	Low
LOCA	Loss-Of-Coolant Accident
LOFT	Loss-Of-Fluid Test
LOFW	Loss Of Feedwater
LOOP	Loss Of Offsite Power
LOSG	Loss Of Steam Generator
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LTC	Long Term Cooling
LTOP	Low Temperature Overpressure Protection
LWR	Light Water Reactor
MMI	Man-Machine Interface
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
MSIVLCS	Main Steam Isolation Valve Leakage Control System
MSP	Maintenance and Surveillance Program

MU/HPI	Make-Up High Pressure Injection
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSP	Nuclear Steam Plant
NSSS	Nuclear Steam Supply System
NTOL	Near-Term Operating License
NUREG	NRC publication
OBE	Operating Basis Earthquake
OIE	Office of Inspection and Enforcement
OL	Operating License
OSC	Operational Support Center
OTSG	Once-Through Steam Generator
PAM	Post Accident Management
PARV	Power Actuated Relief Valve
PLEx	Plant Life Extension
PLiM	Plant Life Management
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSB	Power Systems Branch, DSI, NRR (defunct)
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
PZR	Pressurizer
QA	Quality Assurance
QC	Quality Control
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RCW	Recirculated Cooling Water
RDA	Rod Drop Accident
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RHR	Residual Heat Removal
RO	Reactor Operator

RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RSB	Reactor Systems Branch, DSI, NRR (defunct)
RSW	Raw Service Water
RTRG	Rosemount Transmitter Review Group
RTS	Reactor Trip System
RV	Reactor Vessel
RVSS	Reactor Vessel Support Structure
RWCU	Reactor Water Cleanup
RWS	Reserve Water System
RWST	Reactor Water Storage Tank
SBLOCA	Small Break Loss-Of-Coolant Accident
SCS	Shutdown Cooling System
SDS1	Shutdown System # 1
SDS2	Shutdown System # 2
SDV	Scram Discharge Volume
SEP	Systematic Evaluation Program
SER	Safety Evaluation Report
SEU	Slightly Enriched Uranium
SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SONGS	San Onofre Nuclear Generating Station
SOV	Solenoid-Operated Valve
SPDS	Safety Parameter Display System
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSC	Structure, System, and Component
SSE	Safe Shutdown Earthquake
SSWS	Station Service Water System
STS	Standard Technical Specification(s)
SV	Safety Valve
TDI	Transamerica DeLaval Inc.
TIP	Traveling In-core Probe
TMI	Three Mile Island Nuclear Plant

TS	Technical Specification(s)
TSC	Technical Support Center
TSIP	Technical Specification Improvement Program
TT	Turbine Trip
USI	Unresolved Safety Issue
UT	Ultrasonic Test
UV	Undervoltage

1. INTRODUCTION

AECL will be applying to the US NRC for the Design Certification of the Advanced CANDU Reactor (ACR) design. In support of this application, AECL Technologies Inc (AECLT) has assessed the ACR design with respect to the Generic Safety Issues (GSIs) identified and issued by the US NRC in NUREG-0933 ([1]). This report discusses this assessment, in which all GSIs included in the latest version of NUREG-0933 are screened for identification of those GSIs that may be applicable to the ACR design. Further assessment of the compliance of the ACR design with the GSIs identified as potentially applicable to the ACR design will be performed and provided in the Design Control Document (DCD).

Since the inception of the generic issues program in 1976, US NRC has identified and issued 842 reactor safety issues grouped into TMI Action Plan Items, Task Action Plan Items, New Generic Items, Human Factor Issues, and Chernobyl Issues that are collectively called Generic Safety Issues in this report. Section (a)(1)(iv) of 10 CFR 52.47 ([2]) requires technical resolutions of those Unresolved Safety Issues (USIs) and medium and high-priority GSIs identified in the version of NUREG-0933 current on the date six months prior to application and which are technically relevant to the design. To meet this requirement, 6 months prior to submitting the ACR DCD, AECLT will ensure that if a newer version of NUREG-0933 ([1]) is in place, any changes from the version which is currently reviewed will be addressed in the DCD.

This report is organized as follows:

- Chapter 2 provides the scope and objectives of this report.
- Chapter 3 provides the Screening Criteria used for categorization of the US NRC Generic Safety Issues with respect to the applicability to the ACR design.
- Chapter 4 provides the Screening Results of the US NRC GSIs with screening justification for each GSI provided in Table 1.
- Chapter 5 summarizes the overall findings of this screening.

2. OBJECTIVES

This report summarizes the results of a screening process performed to identify the potential applicability of the US NRC Generic Safety Issues (GSIs) to the ACR design.

Justification of the screening is provided for each GSI. When a GSI is identified as potentially applicable to the ACR design, the relevance is briefly assessed, the ACR compliance is preliminarily assessed as appropriate, and direction on further assessment is provided.

This report covers the 841 GSIs identified in NUREG-0933 ([1]) through supplement 25. In addition, one new GSI (192: Secondary Containment Drawdown Time) identified in US NRC Policy Issue Information, SECY-02-0148 ([3]), of August 2, 2002, is also covered in this report.

3. SCREENING CRITERIA

All 842 GSIs as listed in Table 1 have been screened against the following Screening Criteria.

The GSI will be excluded from further assessment if it meets one of the following five Exclusion Criteria:

a) Issue has been prioritised by US NRC as Low, Drop or has not been prioritized.

b) Issue is not an ACR design issue. Issue is only applicable to other designs (e.g., BWR, PWR, or B&W Reactor).

d) Issue is not a design issue (Environmental, Licensing, Regulatory Impact Issue, or US NRC internal issue; or covered in an existing NRC program). Licensing Issues are those not directly related to protecting public health and safety or the environment, but relate to improving the NRC staff's capability and the regulatory efficiency and effectiveness (e.g. increasing the NRC staff's knowledge, certainty, and understanding of safety in order to increase its confidence in assessing levels of safety) ([1]).

e) Issue is superseded by one or more issues.

f) Issue is not an ACR design certification issue. Issue is applicable to current operating plants or responsibility of COL applicant.

The GSIs that do not meet any of the above five Exclusion Criteria are deemed, to some extent, relevant to the ACR design and are screened against the following three Inclusion Criteria:

c) Issue is resolved for US PWRs and BWRs with no new requirements established. This means that PWRs and BWRs do not need to address the GSIs meeting this criterion. But it does not necessarily mean the same for the ACR design. The ACR design is or will be assessed with respect to these GSIs to ensure their closure.

g) Issue is resolved by establishment of new regulatory requirements and/or guidance. The ACR design is or will be assessed with respect to these GSIs to ensure the ACR design complies, where appropriate, with the new regulatory requirements and/or guidance.

h) Issue is unresolved pending generic resolution (for example, prioritized as High, Medium, or possible resolution identified). The ACR design will be assessed with respect to these GSIs to ensure the ACR design has addressed or will address, as appropriate, the safety concerns identified in these GSIs.

The summary of the ACR compliance with the GSIs meeting any Inclusion Criterion will be provided in the DCD.

4. GSI SCREENING RESULTS

Table 1 lists all GSIs identified and issued by the US NRC as presented in NUREG-0933 ([1]) through supplement 25, including one new GSI identified in US NRC Policy Issue Information, SECY-02-0148 ([3]), of August 2, 2002. One of the applicable Screening Criteria as provided in Section 3 has been assigned to each GSI. The justification for the screening has also been provided for each GSI. The GSI numbers of all GSIs identified as potentially applicable to the ACR design are collectively listed in Table 2.

5. CONCLUSIONS

All 842 GSIs have been screened against the Screening Criteria as provided in Section 3. The results of the screening show that most GSIs do not need to be addressed by the ACR design because they either have been dropped by the NRC or are specific to other designs not applicable to the ACR, are operational issues, or are NRC licensing/internal issues. In addition, the GSIs superseded by other issues will not be addressed separately. 147 of the 842 GSIs have been identified as potentially applicable to the ACR design. These 147 GSIs are categorized into three categories:

"c": 55 GSIs are identified as category “c”, which raised safety concerns and resulted in no new regulatory requirements. These safety concerns are considered to be applicable to some extent to the ACR design. The ACR design is or will be assessed with respect to these GSIs to ensure the safety concerns are addressed/resolved.

"g": 86 GSIs are identified as category “g”, which raised safety concerns and resulted in some new regulatory requirements. The safety concerns and the resultant new requirements are considered to be applicable to some extent to the ACR design. The ACR design is or will be assessed with respect to these GSIs to ensure the design addresses these safety concerns, and the requirements are met.

"h": 6 GSIs, all new GSIs, are identified as category “h”, which raised safety concerns and either resulted in new regulatory requirements or are still under assessment by the NRC staff. The safety concerns raised by, and the new regulatory requirements resulting from, these 6 GSIs are considered to be applicable to some extent to the ACR design. The ACR design will be assessed with respect to these 6 GSIs to ensure the design addresses these safety concerns, and the requirements are met.

6. REFERENCES

[1] NUREG-0933, “A Prioritisation of Generic Safety Issues”, main report and Supplements 1-25, U.S. Nuclear Regulatory Commission, June 2001.

[2] 10 CFR 52.47, “Contents of Applications”.

[3] US NRC Policy Issue Information, SECY-02-0148, August 2, 2002.

[4] 10 CFR 50.54, “Conditions of licenses”.

[5] Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants”, March 1971, (Rev. 1) September 1975, (Rev. 1R) May 1977, (Rev. 2) April 1987, (Rev. 3) May 2000.

[6] NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980. (Supplement 1) January 1983.

[7] Generic Letter 82-33, "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability," U.S. Nuclear Regulatory Commission, December 1982.

[8] NUREG-0799, “Draft Criteria for Preparation of Emergency Operating Procedures,” U.S. Nuclear Regulatory Commission, June 1981.

[9] NUREG-0899, “Guidelines for Preparation of Emergency Operating Procedures – Resolution of Comments on NUREG-0799,” U.S. Nuclear Regulatory Commission, September 1982.

[10] NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants -- LWR Edition," U.S. Nuclear Regulatory Commission.

[11] Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," U.S. Nuclear Regulatory Commission, May 1973.

[12] NUREG-0700, "Guidelines for Control Room Design Reviews," U.S. Nuclear Regulatory Commission, September 1981.

[13] NUREG/CR-1440’ “Light Water Reactor Status Monitoring during Accident Conditions,” U.S. Nuclear Regulatory Commission, May 1980.

[14] NUREG/CR-2100, “Boiling Water Reactor Status Monitoring during Accident Conditions,” U.S. Nuclear Regulatory Commission, May 1981.

[15] NUREG/CR-2278, “Light Water Reactor Engineered Safety Features Status Monitoring,” U.S. Nuclear Regulatory Commission, October 1981.

[16] Regulatory Guide 1.97, Rev.3, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following an Accident," U.S. Nuclear Regulatory Commission, May 1983.

[17] 10 CFR 50.34, "Additional TMI-related Requirements."

[18] 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants".

[19] AECB, "Requirements for Containment Systems for CANDU Nuclear Power Plants", Regulatory Document R-7, February 21, 1991.

[20] NRC Letter to Alabama Power Company, “Containment Purging and Venting During Normal Plant Operation,” 28 November 1978.

[21] Regulatory Guide 1.106, “Thermal Overload Protection for Electric Motors on Motor-Operated Valves,” November 1975, (Rev. 1) March 1977.

[22] NRC Letter to All Licensees of Operating Nuclear Power Plants and Holders of Construction Permits for Nuclear Power Plants, “safety-Related motor-Operated valve Testing and Surveillance (Generic Letter 89-10) – 10 CFR 50.54 (f), “ June 28, 1989, (Supplement 1) June 13, 1990, (Supplement 2) August 3, 1990, (Supplement 3) October 25, 1990, (Supplement 4) February 12, 1992, (Supplement 5) June 28, 1993, (Supplement 6) March 8, 1994.

[23] 10 CFR 21, “Reporting of Defects and Noncompliance”.

[24] 10 CFR 50.55a, "Codes and Standards."

[25] 10 CFR 50, Appendix K, “ECCS Evaluation Models”.

[26] 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors”.

[27] NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” February 1980, (Rev. 1) November 1980.

[28] NUREG-0555, “Standard Review Plan for the Environmental Review of Construction Permit Applications for Nuclear Power Plants", U.S. Nuclear Regulatory Commission.

[29] Regulatory Guide 8.25, "Calibration and Error Limits of Air Sampling Instruments for Total Volume of Air Sampled," U.S. Nuclear Regulatory Commission, August 1980.

[30] 10 CFR 20, “Standards For Protection Against Radiation”.

[31] NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1984.

[32] NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Piping Systems," U.S. Nuclear Regulatory Commission, November 1980.

[33] NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," U.S. Nuclear Regulatory Commission, September 1988.

[34] Generic Letter 85-02, Staff Recommended Actions Stemming from NRC Integrated Program for Resolution of USIs Regarding Steam Generator Tube Integrity, U.S. Nuclear Regulatory Commission, April 1985.

[35] NUREG-0661, Supplement 1, “ Safety Evaluation Report for the MARK I Containment Long-term Program,” U.S. Nuclear Regulatory Commission, August 1982.

[36] NUREG-0808, “MARK II Containment Program Evaluation and Acceptance Criteria,” U.S. Nuclear Regulatory Commission, August 1981.

[37] 10 CFR 50.62, "Requirements for Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

[38] NUREG-0577, Revision 1, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," U. S. Nuclear Regulatory Commission, October 1983.

[39] NUREG/CR-2963, “Planning Guidance for Nuclear Power Plant Decontamination,” U. S. Nuclear Regulatory Commission, June 1983.

[40] NUREG-1174, "Evaluation of Systems Interaction in Nuclear Power Plants - Technical Findings Related to Unresolved Safety Issue A-17," U.S. Nuclear Regulatory Commission, May 1989.

[41] IEEE-323-1983, “Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations”. Institute of Electrical and Electronics Engineers.

[42] NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," U.S. Nuclear Regulatory Commission, July 1981.

[43] 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plant".

[44] Regulatory Guide 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1984.

[45] Regulatory Guide 1.100, Revision 2, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1988.

[46] CAN3–N289.3–M81 (R1998), “Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants”.

[47] CAN3-N289.4–M86 (R1998), “Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants”.

[48] Regulatory Guide 1.75, Rev. 2, "Physical Independence of Electric Systems," U.S. Nuclear Regulatory Commission, September 1978.

[49] IEEE Standard 384-1981, "Criteria for Separation of Class 1E Equipment and Circuits," Institute of Electrical and Electronics Engineers.

[50] NUREG-0224, “Final Report on Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors,” U.S. Nuclear Regulatory Commission, September 1978.

[51] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1980.

[52] CSA-B167-64, “General Purpose Electric Overhead Travelling Cranes”.

[53] Regulatory Guide 1.82, Rev. 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, November 1985.

[54] 10 CFR 50.2, "Definitions".

[55] 10 CFR 50.63, “Loss of All Alternating Current Power”.

[56] Regulatory Guide 1.155, "Station Blackout," U.S. Nuclear Regulatory Commission, August 1988.

[57] Generic Letter 89-19, "Request for Action Related to Resolution of USI A-47, Pursuant to 10 CFR 50.54(f)," U.S. Nuclear Regulatory Commission, September 1989.

[58] IEEE Standard 317-1976, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Plants", Institute of Electrical and Electronics Engineers.

[59] Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Rev. 2) July 1978, (Rev. 3) February 1987.

[60] NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U. S. Nuclear Regulatory Commission, June 1978.

[61] Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post-Accident Engineered Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1978.

[62] Regulatory Guide 1.160, Rev. 0, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," June 1993.

[63] 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants".

[64] Regulatory Guide 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Rev. 3) July 1993.

[65] Guide 1.133, Rev. 1, "Loose-Part Detection Program for the Primary System of Light Water-Cooled Reactors," U.S. Nuclear Regulatory Commission, May 1981.

[66] Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," U.S. Nuclear Regulatory Commission, August 1998.

[67] 10 CFR 50.82, "Termination of License".

[68] 10 CFR 50.33, "Contents of Applications: General Information".

[69] 10 CFR 61.56, "Waste Characteristics".

[70] Regulatory Guide 1.105, Revision 2, "Instrument Setpoints For Safety-Related Systems," U.S. Nuclear Regulatory Commission, February 1986.

[71] Standard ISA-S67.04-1987, "Setpoints for Nuclear Safety-Related Instrumentation Used In Nuclear Power Plants," Instrument Society of America.

[72] Regulatory Guide 1.33, "Quality Assurance Program Requirements," U.S. Nuclear Regulatory Commission, November 1972, (Rev. 1) February 1977, (Rev. 2) February 1978.

[73] Regulatory Guide 1.151, "Instrument Sensing Lines," U.S. Nuclear Regulatory Commission, July 1983.

[74] WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1975.

[75] NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants." U.S. Nuclear Regulatory Commission, June 1990.

[76] NUREG-1445, “Regulatory Analysis for the Resolution of Generic Safety Issue-29: Bolting Degradation or Failure in Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, September 1991.

[77] NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, “Generic Safety Issue 29, ‘Bolting Degradation or Failure in Nuclear Power Plants,’ (Generic Letter, No. 91-17),” October 17, 1991.

[78] NUREG-1275, “Operating Experience Feedback Report – Air Systems Problems,” U.S. Nuclear Regulatory Commission, (Vol. 1) July 1987, (Vol. 2) December 1987.

[79] NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, “Instrument Air Supply System Problems Affecting Safety-Related Equipment (Generic Letter, 88-14),” August 8, 1988.

[80] NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, “Service Water System Problems Affecting Safety-Related Equipment (Generic Letter, 89-13),” July 18, 1989.

[81] NRC Letter to All Pressurized Water Reactor Licensees and Construction Permit Holders, “Resolution of Generic Issue 70, ‘Power-Operated Relief Valve and Block Valve Reliability,’ and Generic Issue 94, ‘Additional Low Temperature Overpressure Protection for Light-Water Reactors,’ Pursuant to 10 CFR 50.54(f) (Generic Letter 90-06),” June 25, 1990.

[82] NUREG-0977, “NRC Fact Finding Task Force Report on the ATWS events at Salem Nuclear Generating Station Unit 1 on February 22 and 25, 1983,” U.S. Nuclear Regulatory Commission, March 1983.

[83] NUREG-1000, “Generic Implications of ATWS events at Salem Nuclear Power Plant,” U.S. Nuclear Regulatory Commission, (Vol. 1) April 1983, (Vol. 2) August 1983.

[84] NRC Letter to All Licensees of Operating Boiling Water Reactors (BWRs), and Holders of Construction Permits, “NRC Position on IGSCC in BWR Authenitic Stainless Steel Piping (Generic Letter 88-01),”January 25, 1988.

[85] Generic Letter 88-17, "Loss of Decay Heat Removal," U.S. Nuclear Regulatory Commission, October 1988.

[86] Regulatory Guide 1.35, “Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments,” U.S. Nuclear Regulatory Commission, February 1973, (Rev.1) June 1974, (Rev. 2) January 1976, (Rev. 3) July 1990.

[87] Regulatory Guide 1.35.1, “Determining Prestressing Forces for Inspection of Prestressed Concrete Containments,” U.S. Nuclear Regulatory Commission, July 1990.

[88] Regulatory Guide 1.22, “Periodic Testing of Protection System Actuation Functions,” U.S. Nuclear Regulatory Commission, February 1972.

[89] Regulatory Guide 1.118, “Periodic Testing of Electric Power and Protection Systems,” U.S. Nuclear Regulatory Commission, June 1976, (Rev.1) November 1977, (Rev. 2) June 1978, (Rev. 3) April 1995.

[90] 10 CFR 50.59, “Changes, Tests and Experiments”.

[91] NRC Letter to All Holders of Operating Licensees, “Resolution of Generic Issue A-30, ‘Adequacy of Safety-Related DC Power Supplies,’ Pursuant to 10 CFR 50.54(f) (Generic Letter 91-06),” April 29, 1991.

[92] NRC Letter to All Holders of Operating Licensees, “Resolution of Generic Issue 48, ‘LCOs for Class 1E Vital Instrument Buses,’ and 49, ‘Interlocks and LCOs for Class 1E Tie Breakers,’ Pursuant to 10 CFR 50.54(f) (Generic Letter 91-11),” July 18, 1991.

[93] Generic Letter 91-13, “Request for Information Related to the Resolution of Generic Issue 130, ‘Essential Service Water Pump Failures at Multiplant Sites,’ Pursuant to 10 CFR 50.54(f) (Generic Letter 91-13),” September 19, 1991.

[94] Regulatory Guide 1.3, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors,” U.S. Nuclear Regulatory Commission, November 1970, (Rev.1) June 1973, (Rev. 2) June 1974.

[95] Regulatory Guide 1.4, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," U.S. Regulatory Commission, June 1974.

[96] TID-14844, “ Calculation of Distance Factors for Power and Test Reactor Sites,” U.S. Atomic Energy Commission, March 1962.

[97] NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, February 1995.

[98] NUREG-1738, “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, February 2001.

[99] IAEA, The Convention on the Physical Protection of Nuclear Material, INFCIRC/274/Rev. 1.

[100] IAEA, The Physical Protection of Nuclear Material, INFCIRC/225/Rev. 3.

[101] 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors."

[102] 10 CFR 54.21, “Contents of application -- technical information”.

[103] NUREG/CR-3696, “Potential Human Factors deficiencies in the Design of Local Control Stations and Operator Interfaces in Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, April 1984.

[104] NUREG/CR-6146, “Local Control Stations: Human Engineering Issues and Insights,” U.S. Nuclear Regulatory Commission, September 1994.

Table 1
Listing of US NRC Generic Safety Issues

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
<u>TMI Action Plan Items</u>			
I.A.1.1	Shift Technical Advisor	f	This resolved item and the Items I.A.1.2 and I.A.1.3 were to increase the capability of the shift crews in the control room to operate the nuclear facility in a safe and competent manner, by assuring that a proper number of individuals with proper qualifications and fitness are on shift at all time. These items did not address the requirements to improve the design of the control room. Thus these items are only applicable to operating plants or the responsibility of COL applicant.
I.A.1.2	Shift Supervisor Administrative Duties	f	Same as I.A.1.1.
I.A.1.3	Shift Manning	f	Same as I.A.1.1.
I.A.1.4	Long-Term Upgrading	d	The purpose of this item was to develop changes to 10 CFR 50.54 ([4]) concerning shift staffing with licensed operators and working hours of licensed operators. Therefore it is ranked as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.A.2.1(1)	Qualifications -Experience	f	This resolved item together with items I.A.2.1(2) and I.A.2.1(3) required all operating plant licensees and all license applicants to provide specific improvements in training and qualifications of senior operators and control room operators. This item is only applicable to operating plant or the responsibility of COL applicant.
I.A.2.1(2)	Training	f	Same as I.A.2.1(1).
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	f	Same as I.A.2.1(1).
I.A.2.2	Training and Qualifications of Operations Personnel	d	This resolved item called for reactor licensees to review their training and qualification program for all operations personnel including licensed and auxiliary operators, technicians, maintenance personnel, and supervisors. The review was to be conducted to examine existing practices in light of the safety significance of the duties of the operations staff. If the review determined that the existing practices adequately assured proper safety-related staff conduct, then documentation of the justification for this determination was required; this documentation did not require submittal to the NRC but was required to be maintained on site. If the review uncovered inadequacies, the licensee was required to update the training and qualification practices to ensure adequate performance of operations personnel. Thus, this is a licensing issue. It is the operating plant's or COL applicant's responsibility to address this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.A.2.3	Administration of Training Programs	f	This resolved item called for the NRC staff to develop criteria and procedures to be used in auditing training programs and to increase the amount of auditing. Only the operating plant or COL applicant needs to address the requirements resulted from the closure of this item.
I.A.2.4	NRR Participation in Inspector Training	d	NUREG-0933 ranked this item as a Licensing Issue.
I.A.2.5	Plant Drills	f	This resolved item was to upgrade operator training by requiring operating personnel to conduct plant drills during shifts. It is only applicable to operating plants or the responsibility of COL applicant.
I.A.2.6(1)	Revise Regulatory Guide 1.8	f	This resolved item called for the NRC staff to revise Regulatory Guide 1.8 ([5]) in order to incorporate some upgrading requirements on training and qualifications of operations personnel. The revisions were published in 1987 that should only be addressed by operating plants or COL applicant.
I.A.2.6(2)	Staff Review of NRR 80-117	f	This resolved item called for the NRC staff to review its internal report, NRR-80-117 (on operator licensing), to make recommendations with respect to the revision of the Regulatory Guide 1.8 ([5]). Only operating plant or COL applicant should address this item.
I.A.2.6(3)	Revise 10 CFR 55	e	Superseded by I.A.2.2.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.A.2.6(4)	Operator Workshops	f	This resolved item was to conduct seminar-type workshops to exchange information on operations experience between the NRC and licensees and among licensees. The NRC staff conducted three workshops and a mail survey. It was not recommended to conduct such workshop or survey on an annual basis. But rather, it was preferable to conduct the workshop or survey on an as needed basis. Thus this item should only be applicable to operating plants or the responsibility of COL applicant.
I.A.2.6(5)	Develop Inspection Procedures for Training Programs	d	This item called for the NRC staff to develop inspection procedures for training programs. No new requirements were issued as the resolution of this item. Since training is the responsibility of licensees or COL applicants, this item is not applicable to ACR design certification.
I.A.2.6(6)	Nuclear Power Fundamentals	a	NUREG-0933 dropped this item.
I.A.2.7	Accreditation of Training Institutions	f	This resolved item required NRR to complete a study to establish the procedures and requirements for NRC accreditation of reactor operator training programs. As a result of the closure of this item, the NRC adopted a Policy Statement on Training and Qualifications which made the training accreditation program managed by INPO the focus of training improvement in the industry. Thus, only the operating plant or COL applicant should address this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	f	This resolved item, together with I.A.3.2, I.A.3.3, and I.A.3.4, was to upgrade the requirements for operation personnel licensing. Thus, it is applicable to operating plant or the responsibility of COL applicant.
I.A.3.2	Operator Licensing Program Changes	f	Same as I.A.3.1.
I.A.3.3	Requirements for Operator Fitness	f	Same as I.A.3.1.
I.A.3.4	Licensing of Additional Operations Personnel	f	Same as I.A.3.1.
I.A.3.5	Establish Statement of Understanding with INPO and DOE	d	NUREG-0933 ranked this item as a Licensing Issue.
I.A.4.1(1)	Short-Term Study of Training Simulators	f	This resolved item called for the NRC staff to conduct a short-term study of training simulators to collect and develop corrections for identified weakness. Such a study was completed and documented and is of interest for the operating plant or COL applicant only.
I.A.4.1(2)	Interim Changes in Training Simulators	f	This resolved item called for the development of requirements to correct specific training simulator weakness. Only operating plants or COL applicant should address these requirements.
I.A.4.2(1)	Research on Training Simulators	f	This resolved item and items I.A.4.2(2), I.A.4.2(3) and I.A.4.2(4) called for long-term simulator upgrading. The closure of these issues resulted a series of new requirements on training simulators for operating plants and COL applicant.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.A.4.2(2)	Upgrade Training Simulator Standards	f	Same as I.A.4.2(1).
I.A.4.2(3)	Regulatory Guide on Training Simulators	f	Same as I.A.4.2(1).
I.A.4.2(4)	Review Simulators for Conformance to Criteria	f	Same as I.A.4.2(1).
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	d	NUREG-0933 ranked this item as a Licensing Issue.
I.A.4.4	Feasibility Study of NRC Engineering Computer	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.1.1(1)	Prepare Draft Criteria	f	This resolved item, together with items I.B.1.1(2 - 7), dealt with implementation of long-term organization and management improvements of licensees. These items were to improve licensee safety performance and ability to respond to accidents by upgrading licensee groups responsible for radiation protection and plant operation. These items are only applicable to operating plant or the responsibility of COL applicant.
I.B.1.1(2)	Prepare Commission Paper	f	Same as I.B.1.1(1).
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	f	Same as I.B.1.1(1).
I.B.1.1(4)	Review Responses to Determine Acceptability	f	Same as I.B.1.1(1).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.B.1.1(5)	Review Implementation of the Upgrading Activities	f	Same as I.B.1.1(1).
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	e	Superseded by I.A.2.6(1), 75.
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	e	Superseded by I.A.2.6(1), 75.
I.B.1.2(1)	Prepare Draft Criteria	f	This resolved item, together with items I.B.1.2(2, 3), required the NRC staff to evaluate organization and management capabilities of Near-Term Operating License applicants before license issuance. These items should only be applicable to operating plants or the responsibility of COL applicant.
I.B.1.2(2)	Review Near-Term Operating License Facilities	f	Same as I.B.1.2(1).
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	f	Same as I.B.1.2(1).
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.2.1(3)	Follow-up on Completed Maintenance Work Orders to Ensure Proper Testing and Return to Service	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.2.1(6)	Observe Routine Maintenance	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.2.1(7)	Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.2.2	Resident Inspector at Operating Reactors	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.2.3	Regional Evaluations	d	NUREG-0933 ranked this item as a Licensing Issue.
I.B.2.4	Overview of Licensee Performance	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.C.1(1)	Small Break LOCAs	f	This resolved item, together with items I.C.1(2-4) addressed the need for improvement in the quality of operational information provided to plant operations and staff personnel in order to enhance normal plant operation and the prevention and mitigation of plant transients or accidents. Following the Three Mile Island Unit 2 (TMI-2) accident, new guidance was established to improve the quality of operational information for dealing with emergency events. The objective of the guidance identified in NUREG-0737 ([6]) and supplemented by Generic Letter 82-33 ([7]) was to improve the quality of procedures to provide greater assurance that operator and staff actions are technically correct, by making them explicit and easily understood for normal, transient and accident conditions. The overall content, wording, and format of procedures that affect plant operation, administration, maintenance, testing and surveillance were to be evaluated by the NRC in accordance with NUREG-0737 ([6]) and Generic Letter 82-33 ([7]). Items I.C.1(1-3) involved analyses and preparation of guidelines for the preparation of emergency operating procedures for small break LOCAs, recognition and prevention of impending core uncovery, and operation of the plant in natural circulation. Item I.C.1(4) addressed NRC review of procedures, guidelines, and the supporting analyses of various transients, and as such is a Licensing Issue. The ultimate responsibility for meeting NUREG-0737 ([6]), Supplement 1 and Generic Letter GL 82-33 ([7]), rests with the COL applicant. Therefore Items I.C.1(1-3) are only applicable to

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			the COL applicants. AECL, however, will assist the COL applicant in establishing these procedures and training the plant operators and staff by providing Emergency Operations Guidelines in the COL application process.
I.C.1(2)	Inadequate Core Cooling	f	Same as I.C.1(1).
I.C.1(3)	Transients and Accidents	f	Same as I.C.1(1).
I.C.1(4)	Confirmatory Analyses of Selected Transients	d	Same as I.C.1(1).
I.C.2	Shift and Relief Turnover Procedures	f	This item addressed shift and relief turnover procedures of plants, and as such is not a design certification issue. This item is the responsibility of operating plants or COL applicants.
I.C.3	Shift Supervisor Responsibilities	f	This item addressed shift supervisor responsibilities of plants, and as such is not a design certification issue. This item is the responsibility of operating plants or COL applicants.
I.C.4	Control Room Access	f	This item addressed the specific operating procedure, control room access, and as such is related to the administration of plant and is not a design certification issue. This item is the responsibility of operating plants or COL applicants.
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	f	This item required that each applicant for an operating license prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and re-training programs.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			It is the responsibility of the prospective COL applicant of ACR to address this item. This item is not a design certification issue.
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	f	This item called for the implementation of verification of operating activities. It is applicable to operating plants or the responsibility of COL applicant. It is not a design certification issue.
I.C.7	NSSS Vendor Review of Procedures	f	This item addressed the adequacy of operating procedures, and as such is not a design certification issue.
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	f	This item addressed the adequacy of emergency procedures, and as such is not a design certification issue.
I.C.9	Long-Term Program Plan for Upgrading of Procedures	f	This item was to develop a long-term program plan for the upgrading of plant procedures. This plan would incorporate and expand on current efforts associated with the development, review, and monitoring of procedures. Consideration of studies to ensure clear procedures with particular emphasis on diagnostic aids for off-normal conditions was called for. The interrelationships of administrative, operating, maintenance, test, and surveillance procedures were to be considered. The topics of emergency procedures, reliability analysis, human factors engineering, crisis management, and operator training were also to be addressed. The part of this item related to emergency operating procedures (EOP), was implemented in accordance with Item I.C.1 of NUREG-0737 ([6]). In regard to the EOPs, SECY-82-111

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			requested Commission approval of a set of basic requirements for emergency response capability and approval for the staff to work with licensees to develop plant specific implementation schedules. A significant amount of work on emergency operating procedures had been completed. All four US NSSS vendors had submitted technical guidelines based on reanalysis of accidents and transients. In the area of human factors, a survey of current practices, research on EOPs, and pilot monitoring of some NTOL plants had been completed and criteria for development of EOPs were published for public comment in NUREG-0799 ([8]). NUREG-0899 ([9]) was published in final form in September 1982 and incorporated resolution of comments received on NUREG-0799 ([8]). Resolution of this item was expected to have a significant impact on plant procedures. The changes in procedures, in turn, were expected to improve the safety-related performance of all plant operations staff. This would apply to both routine and abnormal operating conditions. This item was clarified in Supplement 1 to NUREG-0737 ([6]) and was resolved with the issuance of SRP ([10]) Section 13.5.2, Rev. 1, and Section 13.5.2, Appendix A, Rev. 0. No new requirements for US PWRs and BWRs were issued. The prospective COL applicant for the ACR design should be responsible for the development of operating procedures. The COL applicant will be provided with necessary input from AECL for the development of these procedures, but this will be done in the COL application process. Therefore this item is ranked only applicable to the operating plants or the

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			responsibility of the COL applicant.
I.D.1	Control Room Design Reviews	g	This item called for the operating plants and operating license applicant to conduct a detailed control room design review to identify and correct human factor problems. This item was clarified in NUREG-0737 ([6]), requirements were issued, and multi-plant action item F-08 was established for implementation purposes. Human factor engineering principles have been well applied throughout the ACR design, including the control room design. This issue is resolved for the ACR design. The compliance of the ACR design with this GSI will be confirmed in the DCD.
I.D.2	Plant Safety Parameter Display Console	g	This item required to provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. A safety parameter display panel is provided in the control room of ACR to display important plant variables that will help operator to timely determine the plant status. This issue is resolved for the ACR design. The compliance of the ACR design with this GSI will be confirmed in the DCD.
I.D.3	Safety System Status Monitoring	c	This item recommended that a study be undertaken to determine the need for all licensees and applicants not committed to Regulatory Guide 1.47 ([11]) to install a bypass and inoperable

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			status indication system or a similar system. Implementation of a well-engineered bypass and inoperable status indication system could provide the operator with timely information on the status of the plant safety systems. This operator aid could help eliminate operator errors such as those resulting from valve misalignment due to maintenance or testing errors. This item was resolved for U.S. PWRs and BWRs with no new requirements established. Safety systems are provided at ACR that are equivalent to US reactor protection system and engineered safety feature actuation system. An assessment will be performed and provided in the DCD to show that the ACR design conforms to Regulatory Guide 1.47 ([11]).
I.D.4	Control Room Design Standard	c	This item emphasized a need for guidance on the design of control room to incorporate human factors considerations. NRC issued NUREG-0700 ([12]) for the guidance of the detailed control room design reviews for conformance to accepted human factors principles. SRP ([10]) Section 18.1 was issued to guide NRC review process for both existing and advanced control room designs. Additional human factors engineering guidance on various aspects of control room design was also developed in NUREG/CR reports and various industry publications. The NRC staff concluded that the documents noted above provided the *de facto* control room design standards for

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			evaluating commercial nuclear power plants in the U.S. No new requirements for US PWRs and BWRs were issued. Human factor engineering principles have been well applied to the ACR control room design. This issue is resolved for the ACR design. It will be confirmed in the DCD that the ACR design addresses the safety concerns of this GSI.
I.D.5(1)	Operator-Process Communication	c	This item addressed the man-machine interface in the control room with reference to the use of lights, alarms, and annunciators. The method of presentation of information can significantly enhance the performance of the control room operators and thereby potentially affect operator error. It was proposed that existing practice and use of lights, alarms, and annunciators be reviewed to assess how well they facilitate operator-machine interaction and minimize errors. NRC issued no new requirements for US PWRs and BWRs. ACR communication system is based on accepted engineering practices and the state-of-the-art human factors considerations. This issue is resolved for the ACR design. It will be confirmed in the DCD that the ACR design addresses the safety concerns of this GSI.
I.D.5(2)	Plant Status and Post-Accident Monitoring	g	This item focused on the need to improve the ability of reactor operators to prevent, diagnose, and properly respond to accidents. The emphasis was on the information needs (i.e., indication of plant status) of the operator. In order for operators to perform their functions, it is necessary that they receive all the necessary information on the plant status. This can enhance

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			operator performance (and therefore reduce operator error). Accident sequences should be analyzed to determine the information required to provide unambiguous indication of plant status. Specific instrumentation and Engineered Safety Features (ESF) status monitoring needs would then be determined. PWR instrumentation requirements were analyzed in NUREG/CR-1440 ([13]), and BWR instrumentation requirements were analyzed in NUREG/CR-2100 ([14]). ESF status monitoring requirements were also studied in NUREG/CR-2278[15]). Revision 2 to Regulatory Guide 1.97 ([16]) was issued in December 1980. This Regulatory Guide identifies criteria for design and qualification of the instrumentation divided into three categories, designated 1, 2 and 3, which provide a graded approach to requirements based on the importance to safety of the variable being monitored. ACR design includes provisions for plant safety status monitoring and a Post-Accident Management (PAM) process that provides the plant operator with qualified, reliable information to monitor and control post-accident conditions in the unit. Further assessment will be performed and provided in the DCD to show the ACR design compliance with this GSI.
I.D.5(3)	On-Line Reactor Surveillance System	c	This item addressed the benefit to plant safety and operations of continuous on-line automated surveillance systems. Continuous on-line surveillance systems which automatically monitor reactors can benefit plant operations and safety by providing diagnostic information which can predict anomalous behavior and thus be used to maintain safe conditions.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			No new requirements for US PWRs and BWRs were issued. ACR design includes automated surveillance and monitoring provisions that are to enhance plant safety and operational performance. This issue is resolved for the ACR design. It will be confirmed in the DCD that the ACR design addresses the safety concerns of this GSI. .
I.D.5(4)	Process Monitoring Instrumentation	c	This item called for the NRC staff to explore the feasibility of using new concepts for measuring certain reactor parameters. A directly related issue, II.F.2 in NUREG –0737 ([6]), mandated that industry develop and implement PWR liquid level detection systems. This item addressed the major concern on the inadequate core cooling (ICC) of a reactor. No new requirements for US PWRs and BWRs were issued as resolution of this item. ACR deploys advanced and reliable monitoring instrumentation for the primary heat transport system (HTS)*. This issue is resolved for the ACR design. It will be confirmed in the DCD that the ACR design addresses the safety concerns of this GSI.
I.D.5(5)	Disturbance Analysis Systems	d	NUREG-0933 ranked this item as a Licensing Issue.
I.D.6	Technology Transfer Conference	d	NUREG-0933 ranked this item as a Licensing Issue.

* The Heat Transport System (HTS) of the ACR is equivalent to the Reactor Coolant System (RCS) of PWRs or BWRs.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.E.1	Office for Analysis and Evaluation of Operational Data	d	NUREG-0933 ranked this item as a Licensing Issue.
I.E.2	Program Office Operational Data Evaluation	d	NUREG-0933 ranked this item as a Licensing Issue.
I.E.3	Operational Safety Data Analysis	d	NUREG-0933 ranked this item as a Licensing Issue.
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	d	NUREG-0933 ranked this item as a Licensing Issue.
I.E.5	Nuclear Plant Reliability Data Systems	d	NUREG-0933 ranked this item as a Licensing Issue.
I.E.6	Reporting Requirements	d	NUREG-0933 ranked this item as a Licensing Issue.
I.E.7	Foreign Sources	d	NUREG-0933 ranked this item as a Licensing Issue.
I.E.8	Human Error Rate Analysis	d	NUREG-0933 ranked this item as a Licensing Issue.
I.F.1	Expand QA List	c	This item addressed the issue of systems that are important to safety that are not on the Quality Assurance List. The suggestion was made that equipment important to safety be ranked and that ranking used to determine systems that should be added to the Quality Assurance List. This approach has not been implemented by the NRC. NUREG-0933 stated that this item was resolved for US PWRs and BWRs with no new requirements established. However, a general requirement was mandated in 10 CFR 50.34 f(3)(ii) ([17]): “ensure that the Quality Assurance (QA) List required by Criteria II App. B, 10 CFR Part 50 includes all

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			structures, systems and components important to safety (I.F.1).” The Quality Assurance program of the ACR design has provisions to address systems important to safety. Further assessment will be performed and provided in the DCD to ensure appropriate requirements in 10 CFR have been properly addressed and this item resolved by the ACR design.
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	a	NUREG-0933 prioritized this item as Low.
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	g	This item required including the QA personnel in the review and approval of plant operational maintenance and surveillance procedures and quality-related procedures associated with design, construction, and installation. ACR’s QA program is compliant with Canadian national standards and based on accepted engineering practices. Further assessment of the compliance of the ACR design with this GSI will be performed and provided in the DCD.
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	g	This item required including QA personnel in all design, construction, installation, preoperational and startup testing, and operation. ACR’s QA program is compliant with Canadian national standards and based on accepted engineering practices. Further assessment of the compliance of the ACR design with this GSI will be performed and provided in the DCD.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	a	NUREG-0933 prioritized this item as Low.
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	a	NUREG-0933 prioritized this item as Low.
I.F.2(6)	Increase the Size of Licensees' QA Staff	f	This item specifically required increasing the size of licensee's QA staff. This is only applicable to operating plants or the responsibility of COL applicant.
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	a	NUREG-0933 prioritized this item as Low.
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	a	NUREG-0933 prioritized this item as Low.
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	c	This item specifically required clarifying organizational reporting levels for the QA organization during the plant design, construction and operation stages. The ACR design follows the ACR QA Manual and related sub-tier documents which meet the Canadian Standards requirements. Further assessment will be performed and provided in the DCD to ensure the ACR design addresses the safety concerns of this GSI.
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	a	NUREG-0933 prioritized this item as Low.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
I.F.2(11)	Define Role of QA in Design and Analysis Activities	a	NUREG-0933 prioritized this item as Low.
I.G.1	Training Requirements	f	This item called for new operating licensees to define training plan prior to fuel loading. This item is the responsibility of COL applicant.
I.G.2	Scope of Test Program	g	The major thrust of TMI Action Plan Task I.G. was to use the pre-operational and startup test programs as a training exercise for the operating crews. In contrast to this, Item I.G.2 called for a more comprehensive test program to search for anomalies in a plant's response to a transient. The safety significance of this issue lies in the early discovery of anomalies or unanticipated plant behavior. This item was resolved with the revisions to SRP ([10]) Section 14 and the OIE manual. The ACR startup test program is based on accepted engineering practices and is in accordance with current Canadian regulatory requirements and Standards. Further assessment of the compliance of the ACR design with this GSI will be performed and provided in the DCD.
II.A.1	Siting Policy Reformulation	f	The objective of this item was to provide an added contribution to safety through the development of siting criteria for new power plants. However, NRC decided that, before new siting efforts were undertaken, a new radioactive source term must be approved and the evaluation of the safety goal must be completed. Upon completion of these two tasks, the need for a

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			revised siting rule was to be reassessed and, if necessary, a new generic safety issue was to be established to address siting rulemaking. Thus, this item was resolved with no new requirements established. A COL applicant should address NRC's concern in this GSI when selecting a candidate site for the plant.
II.A.2	Site Evaluation of Existing Facilities	e	Superseded by V.A.1.
II.B.1	Reactor Coolant System Vents	g	This item addressed the requirements in 10 CFR 50 and NUREG-0737 ([6]) to install reactor coolant system (RCS) and reactor vessel high point vents. NRC determined that there was a need for vents in the high points of the reactor coolant system and reactor vessel. The purpose of these vents is to release non-condensable gases from the RCS which may inhibit core cooling during natural circulation. Since the vents are part of the reactor coolant pressure boundary, the design of the vents must conform to the requirements of 10 CFR 50, Appendix A ([18]). In addition, the NRC determined that the vents should not cause an unacceptable increase in the probability of a loss-of-coolant accident (LOCA), should not challenge containment integrity, and should be designed with sufficient redundancy to assure a low probability of inadvertent or irreversible actuation. The ACR design complies with the intent of the requirements noted above. Further assessment of the compliance of the ACR design with this GSI will be performed and provided in the DCD.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	g	This item required that each licensee perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post accident operations of these systems. Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility. The ACR radiation and shielding-design, and the post accident dose limits, were based on compliance with Canadian standards. Further assessment of the compliance of the ACR design with this GSI and the US standards will be performed and provided in the DCD.
II.B.3	Post-Accident Sampling	g	This item required licensees to provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain radioactive materials without undue radiation exposures to any individual. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage, hydrogen in the containment atmosphere, dissolved gases, chloride, and boron

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			concentrations. The ACR design includes Heat Transport Sampling System, Moderator Sampling System, and Post Accident Management process. These systems can provide the sampling capability as required by this item. This issue is resolved for the ACR design. The compliance of the ACR design with this GSI will be confirmed in the DCD.
II.B.4	Training for Mitigating Core Damage	f	This item required developing and implementing a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. This requirement is only the responsibility of COL applicant. AECL would provide necessary support, if needed.
II.B.5(l)	Behavior of Severely Damaged Fuel	d	NUREG-0933 ranked this item as a Licensing Issue.
II.B.5(2)	Behavior of Core Melt	d	NUREG-0933 ranked this item as a Licensing Issue.
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structures	d	NUREG-0933 ranked this item as a Licensing Issue.
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	f	This item required the review of operating reactors in areas of high population density to determine what additional measures and/or design changes could be implemented that would further reduce the probability of a severe reactor accident, and would reduce the consequences of such an accident by reducing the amount of radioactive releases and/or by delaying any radioactive releases, and thereby provide additional time for evacuation near the sites.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			This requirement is only applicable to operating plants.
II.B.7	Analysis of Hydrogen Control	e	Superseded by II.B.8.
II.B.8	Rulemaking Proceedings on Degraded Core Accidents	g	This item envisioned both a short-term and a long-term rulemaking to establish policy, goals, and requirements to address accidents resulting in core damage greater than the present design basis. In the past, safety reviews concentrated on how to prevent a core from being damaged. Consequently, little attention was given to how a severely damaged core could be dealt with after damage occurred. Other subtasks within Task II.B were concerned with the study of the characteristics of degraded and melted cores (research programs) plus some immediate actions to be taken at plants in operation. Work performed by the NRC on the hydrogen control aspect of this item resulted in a Hydrogen Control Rule that was approved and published in the Federal Register on January 25, 1985. The severe accident portion of this item was addressed by a Policy Statement that set forth the NRC's intentions for resolving safety issues related to reactor accidents more severe than design basis accidents. The ACR design has addressed the hydrogen control issue by providing hydrogen control provisions, such as passive autocatalytic hydrogen recombiners and local air coolers. The ACR design has also adequately considered the impact of severe accidents and provided mitigating capability as appropriate. Further assessment of the compliance of the ACR design with this GSI will be performed and provided in the DCD.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.C.1	Interim Reliability Evaluation Program	d	This item called for the NRC staff to complete the interim multiplant reliability evaluation activities that were to develop and standardize the reliability methodology involved in performing reliability and safety studies. The completion of the activities resulted in the publication of a few staff reports. This item is ranked as a resolved licensing issue with no new requirements established.
II.C.2	Continuation of Interim Reliability Evaluation Program	d	This item called for the NRC staff to implement an extended interim reliability evaluation program. The completion of the planned activities resulted in an NRC's Severe Accident Policy Statement that states that OL holders are expected to perform plant-specific PRAs to find instances of particular vulnerability to a core-melt or poor containment performance, given a core-melt. No new requirements were issued. Therefore, this item is ranked as a resolved licensing issue.
II.C.3	Systems Interaction	e	Superseded by A-17.
II.C.4	Reliability Engineering	c	This item called for the NRC to develop a requirement for licensees to develop and implement a reliability assurance program. However, in resolving this item, the NRC staff concluded that the safety concern of this item was addressed in other NRC programs and the issue was considered resolved for US PWRs and BWRs with no new requirements. It has been a tradition to maintain an acceptable level of reliability of safety systems of CANDU reactors. This is confirmed through probabilistic studies. A PRA is performed

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			for the ACR design to ensure that the current core damage frequency goal is met. The results of the PRA will serve as a key input to the COL applicant's operational reliability assurance program. This issue is resolved for the ACR design. It will be confirmed in the DCD that the ACR design addresses the safety concerns of this GSI. .
II.D.1	Testing Requirements	g	This item called for applicants and licensees to conduct testing to qualify reactor coolant relief valves, safety valves, block valves, and associated discharge piping for all operating conditions and design basis accidents. The ACR design includes safety valves and relief valves in its Heat Transport System (HTS) and Pressure and Inventory Control system. All these valves and associated discharge piping are qualified by adequate means including testing. This issue is resolved for the ACR design. The compliance of the ACR design with this GSI will be confirmed in the DCD.
II.D.2	Research on Relief and Safety Valve Test Requirements	a	NUREG-0933 prioritized this item as Low.
II.D.3	Relief and Safety Valve Position Indication	g	This item simply called for all OL holders and applicants to provide the RCS relief and safety valves with position indication in the control room. Relief and safety valves are used in the ACR design and their position indication (open or closed) is provided in the control room. This will be confirmed in the DCD.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.E.1.1	Auxiliary Feedwater System Evaluation	g	This item called for all PWR plants to reevaluate the auxiliary feedwater systems (AFWs). This action includes: (1) Perform a simplified AFW reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages; (2) Perform a deterministic review of the AFW system using the acceptance criteria of SRP ([10]) Section 10.4.9 and the associated Branch Technical Position (BTP) ASB 10-1; and (3) Reevaluate the AFW system flowrate design bases and criteria. The ACR design includes two auxiliary feedwater pumps and associated piping and valves that are intended to be used during normal plant operation and accident events, and are backed up by the RWS that will supply emergency feedwater by gravity in case of a total loss of feedwater to the SGs. Further assessment will be performed and provided in the DCD to show that the ACR design meets the intent of this GSI.
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	g	The requirements established in the resolution of this item included two parts: Part 1: Auxiliary Feedwater System Automatic Initiation Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 ([18]) with respect to the timely initiation of the auxiliary feedwater system (AFWS), the following requirements shall be implemented in the short term: (1) the design shall provide for the automatic

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			initiation of the AFWS; (2) the automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of AFWS functions; (3) testability of the initiating signals and circuits shall be a feature of the design; (4) the initiating signals and circuits shall be powered from the emergency buses; (5) manual capability to initiate the AFWS from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of AFWS functions; (6) the AC motor-driven pumps and valves in the AFWS shall be included in the automatic actuation of the loads onto the emergency buses; and (7) the automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room. In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements. Part 2: Auxiliary Feedwater System Flowrate Indication Consistent with satisfying the requirements of General Design Criterion 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented: (1) safety-grade indication of auxiliary flow to each steam generator shall be provided in the control room; and (2) the auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the AFWS set forth in Auxiliary Systems Branch Technical Position 10-1 of the SRP ([10]) Section 10.4.9.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			The ACR design includes two auxiliary feedwater pumps and associated piping and valves that are intended to be used during normal plant operation and accident events, and are backed up by the RWS that will supply emergency feedwater by gravity in case of a total loss of feedwater to the SGs. Further assessment of its instrumentation will be performed and provided in the DCD to ensure it meets the requirements listed above.
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	d	NUREG-0933 ranked this item as a Licensing Issue.
II.E.2.1	Reliance on ECCS	e	Superseded by II.K.3(17).
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	d	This item called for the NRC staff to conduct the research on small breaks and transients. It included experimental research in the loss-of-fluid test (LOFT) Semiscale, BWR full integral simulation test (FIST), and B&W integral system test facilities, system engineering, and material effects programs, as well as analytical methods development and assessments in the code development program. On the completion of the tests, the NRC staff concluded that the ECCS will provide adequate core cooling for SBLOCA. No new requirements were issued.
II.E.2.3	Uncertainties in Performance Predictions	a	NUREG-0933 prioritized this item as Low.
II.E.3.1	Reliability of Power Supplies for Natural Circulation	g	This item addressed the need for reliable power supplies to pressurizer heaters to ensure that natural circulation can be maintained in the reactor coolant system.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			The resolution of this item resulted in the issuance of requirements for: (1) upgrading the pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions; and (2) establishing new procedures and training for maintaining the RCS at hot standby conditions with only onsite power available. The ACR design includes a Pressure and Inventory Control System which would facilitate the natural circulation cooling. An assessment of its power supply will be performed and provided in the DCD to ensure it meets the requirements listed above.
II.E.3.2	Systems Reliability	e	Superseded by A-45.
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	e	Superseded by A-45.
II.E.3.4	Alternate Concepts Research	d	This item called for the NRC staff to conduct a specific study related to the usefulness of installing an add-on decay heat removal system in existing nuclear power plants to improve the overall operational reliability of decay heat removal. This NRC action item resulted in no new requirements.
II.E.3.5	Regulatory Guide	e	Superseded by A-45.
II.E.4.1	Dedicated Penetrations	b	This item required that the plant designs with external hydrogen recombiners provided redundant dedicated containment penetrations so that, assuming a single failure, the recombiner

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			systems can be connected to the containment atmosphere. The ACR design does not include external hydrogen recombiners. The ACR hydrogen recombiner system is located inside the containment. Thus, this item is not applicable to the ACR design.
II.E.4.2	Isolation Dependability	g	This item required: (1) containment isolation system designs shall comply with the recommendations of SRP ([10]) Section 6.2.4; (2) all plant personnel shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the result of the reevaluation to the NRC; (3) all nonessential systems shall be automatically isolated by the containment isolation signal; (4) the design of control systems for automatic containment valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action; (5) the containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions; (6) containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position II.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days; and (7) containment purge and vent isolation valves must close on a high radiation

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			signal. The ACR containment isolation design is based on compliance with the requirements of Canadian regulatory document, R-7 ([19]) and an assessment of compliance with US NRC requirements will be performed and provided in the DCD.
II.E.4.3	Integrity Check	d	This item called for the NRC staff to perform a feasibility study to evaluate the need and possible methods for performing a periodic or continuous test to detect unknown gross openings in the containment structure. The study indicated that containment leakage provides only a small contribution to the total exposure from postulated design basis accidents. Increasing the containment leakage up to a factor of 10 results in only a very small increase in total risk. Therefore, this NRC staff action item resulted in no new requirements.
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	g	This item addressed the concerns relative to potential failures affecting the purge penetration valves which could lead to a degradation in containment integrity and, for PWRs, a degradation in ECCS performance because of insufficient containment back pressure. In order to reduce the probability of these potential accident scenarios, the NRC was to issue letters to licensees of operating plants requesting limited purging of containment and justification for additional purging. This item was resolved with the issuance of a letter ([20]) to all licensees of operating plants on 28 November 1978 requiring compliance with specific requests.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			The ACR design includes the Reactor Building Ventilation System that meets current Canadian regulatory requirements. The ACR design also includes a Reserve Water System that can assist with the NPSH requirements for the containment sump pumps by providing an emergency source of water to the containment sumps for recovery by the LTC system (a sub-system of the ECCS) in the event of a LOCA. An assessment will be performed to ensure the concerns mentioned above are addressed in the ACR design.
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Valve	g	By letter ([20]) dated 28 November 1978 (mentioned in Item II.E.4.4(1)), the NRC requested all licensees of operating reactors to respond to generic concerns about containment purging or venting during normal plant operation. The generic concerns were two-fold: (1) events occurred where licensees overrode or bypassed the safety actuation isolation signals to the containment isolation valves. These events were determined to be abnormal occurrences; and (2) licensing reviews required tests or analyses to show that containment purge or vent valves would shut without degrading containment integrity during the dynamic loads of a design basis LOCA. This item was resolved with the issuance of a letter ([20]) to all licensees of operating reactors requesting an interim commitment from all licensees of operating reactors. The ACR design includes containment isolation monitors installed at the Reactor Building Ventilation System and the D_2O vapour recovery system exhaust ducts, downstream of the isolation dampers. The isolation dampers close on receipt of isolation signal. Further assessment will be performed and

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			provided in the DCD to ensure the generic concerns mentioned above are addressed by the ACR design.
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	g	By letter ([20]) dated 28 November 1978 (mentioned in Item II.E.4.4(1)), the NRC requested all licensees of operating reactors to respond to generic concerns about containment purging or venting during normal plant operation. As a result of the review of licensee responses to this letter ([20]), NRC learned that at least three valve vendors reported that their valves may not close against ascending differential pressure and the resulting dynamic loading during the design basis LOCA. For plants utilizing valves from these manufacturers, it was determined that the containment integrity could be sufficiently assured by maintaining the valves in the closed position or by restricting the angular opening of valves whenever primary containment integrity is required. In resolving this item, the NRC issued a letter containing guidelines to all affected licensees in order to ensure operability of purge or vent valves. The ACR design includes containment isolation system consisting of valves or dampers in the ventilation ducts and certain process lines penetrating the containment envelope. Further assessment will be performed and provided in the DCD to ensure the generic concerns mentioned above are addressed by the ACR design.
II.E.4.4(4)	Evaluate Purging and Venting during Normal Operation	a	The value/impact score estimated by the NRC staff indicated a low priority ranking for this GSI.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.E.4.4(5)	Issue Modified Purging and venting Requirement	a	The value/impact score estimated by the NRC staff indicated a low priority ranking for this GSI.
II.E.5.1	Design Evaluation	b	The objective of this resolved item was to reduce the sensitivity of B&W plants to feedwater transients, with emphasis on the overcooling transients that had been observed at B&W operating plants. Thus, it is a B&W plant specific item.
II.E.5.2	B& W Reactor Transient Response Task Force	b	Same as II.E.5.1.
II.E.6.1	Test Adequacy Study	g	This item was to establish the adequacy of existing requirements for safety-related valve testing. It recommended a study that would result in recommendations for alternate means of verifying performance requirements. This item was divided into four parts during resolution: (1) pressure isolation valves; (2) check valves; (3) reevaluation of thermal-overload protection provisions of Regulatory Guide 1.106 ([21]) for MOVs; and (4) in-situ testing of MOVs. Part (1) and (2) were integrated into the resolution of Item 105. Regulatory Guide 1.106 ([21]) was determined to be adequate and letters were sent to the pertinent IEEE and ASME subcommittees encouraging the development of standards for thermal overload protection. In-situ testing of MOVs was resolved with the issuance of Generic Letter 89-10 ([22]). The startup, preoperational and in-service inspections program for safety-related valves in ACR is set up in accordance with current Canadian regulatory requirements and standards. An assessment of the ACR design compliance with the NRC's

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			requirements will be performed and provided in the DCD.
II.F.1	Additional Accident Monitoring Instrumentation	g	The general objective of this item was to provide instrumentation to monitor plant variables and systems during and following an accident. This item specifically required monitoring of: (1) radiological noble gas effluent; (2) containment radiation level; (3) containment pressure; (4) containment water level; and (5) containment hydrogen concentration. In addition, the provisions for continuous sampling of plant effluents for postaccident release of radioactive iodines and particulates and onsite laboratory capabilities were also required. The Post Accident Management process of the ACR complies with the requirements listed above. This will be confirmed in the DCD.
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	g	This item required licensees to provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). The instrumentation of the ACR design includes features to monitor reactor status including cooling status. Further assessment will be performed and provided in the DCD to show that the ACR design meets the intent of the requirement listed above.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.F.3	Instruments for Monitoring Accident Conditions	g	The resolution of this item resulted in the publication of Revision 2 to Regulatory Guide 1.97 ([16]). The ACR's Post Accident Management process design is based on compliance with the current Canadian regulatory requirements. Further assessment of compliance with US NRC requirements, Regulatory Guide 1.97 ([16]), will be performed and provided in the DCD.
II.F.4	Study of Control and Protective Action Design Requirements	a	NUREG-0933 dropped this GSI.
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	d	This resolved item called for the NRC to revise some regulatory documents, and as such is ranked as a Licensing Issue. The resolution resulted in no new requirements.
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	g	This item required: "Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 ([18]) for the event of loss-of-offsite power, the following requirements shall be implemented: power supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators (1) motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available; (2) motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			available; (3) motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements; and (4) the pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when the offsite power is not available. ” The ACR design includes a Pressure and Inventory Control system. An assessment will be performed and provided in the DCD to ensure this system meets the intent of the requirements listed above.
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	b	This item covered the efforts by NRC to monitor, review, and assess the safety and environmental impact of the post–accident operation, cleanup, and possible recovery operations at TMI-2 in order to maintain safety of TMI-2 and minimize its environmental impact. This GSI is only applicable to TMI-2.
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	b	Same as II.H.1.
II.H.3	Evaluate and Feed Back Information Obtained from TMI	e	Superseded by II.H.2.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	d	NUREG-0933 ranked this item as a Licensing Issue.
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	d	NUREG-0933 ranked this item as a Licensing Issue.
II.J.1.2	Modify Existing Vendor Inspection Program	d	NUREG-0933 ranked this item as a Licensing Issue.
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	d	NUREG-0933 ranked this item as a Licensing Issue.
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	d	NUREG-0933 ranked this item as a Licensing Issue.
II.J.2.1	Reorient Construction Inspection Program	d	NUREG-0933 ranked this item as a Licensing Issue.
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	d	NUREG-0933 ranked this item as a Licensing Issue.
II.J.2.3	Assign Resident Inspectors to All Construction Sites	d	NUREG-0933 ranked this item as a Licensing Issue.
II.J.3.1	Organization and Staffing to Oversee Design and Construction	e	Superseded by I.B.1.1.
II.J.3.2	Issue Regulatory Guide	e	Superseded by I.B.1.1.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.J.4.1	Revise Deficiency Reporting Requirements	d	This item called for the NRC to revise, as necessary, the event-reporting requirements of 10 CFR 21 ([23]). This item was resolved with the amendments to 10 CFR 21 ([23]) and 10 CFR 50.55(e) ([24]). Since the event-reporting is the responsibility of a licensee, this item is not a design certification issue.
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	e	This item called for all operating plants to effect short-term changes in emergency operating procedures and operator training in order to improve the capability of plants to mitigate the likelihood of SBLOCAs and loss-of-feedwater events. For OL applicants, this item was determined to be covered by Items I.A.2.2 and I.A.3.1.
II.K.1(2)	Review Transients Similar to TMI-2 that have Occurred at other Facilities and NRC Evaluation of Davis-Besse Event	b	NUREG-0933 stated that this item affected all B&W operating plants. For OL applicants with B&W reactor, this item was determined to be covered by Items I.A.2.2 and I.A.3.1. This is a B&W specific issue.
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	e	This item is applicable to all operating PWRs. For OL applicants with PWRs, this item was determined to be covered by Item I.C.1

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.1(4)	Review Operating Procedures and Training Instructions	e	This item was to ensure: a) That operator do not override ESF actions unless continued operation is unsafe; b) HPI system operation; c) RCP operation; and d) That operators are instructed not to rely on level indication alone in evaluating plant conditions. These requirements affected various operating plants. For OL applicants, it was determined this item was covered by Items I.C.1, I.C.7, I.C.8, I.G.1, I.A.1.3, I.A.3.1, or II.F.2.
II.K.1(5)	Safety-Related Valve Position Description	e	This item was to ensure: a) review all valves positions and positioning requirements and positive controls along with all related test and maintenance procedures to assure proper ESF functioning, if required; and b) verify that AFW valves are in the open position. Part a) affected all operating plants. For OL applicants, part a) was determined to be covered by Items I.C.2, and I.C.6. Part b) affected all B&W operating plants. For OL applicants with B&W reactors, part b) was determined to be covered by Items I.C.2, and I.C.6.
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	e	This item was to assure isolation of all lines that do not degrade safety features or cooling capability upon automatic initiation of SI. For all OL applicants, it was determined this item was covered by Item II.E.4.2.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	b	NUREG-0933 stated that this item affected all B&W operating plants. For OL applicants with B&W plants, this item was determined to be covered by Item II.E.1.1. This is a B&W specific issue.
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	b	This item required all operating B&W plants to immediately implement procedures that assure two independent 100% AFW flow or specify explicitly LCO with reduced AFW capacity. For OL applicants with B&W plants, this item was determined to be covered by Item II.E.1.1. This is a B&W specific issue.
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	e	This item required all operating plants to review their procedures to assure that radioactive liquids and gases are not transferred out of containment inadvertently, especially upon ESF reset. All applicable systems and interlocks were required to be listed. For all OL applicants, it was determined this item was covered by Items II.E.4.2 and I.C.6.
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	e	This item required all operating plants to review and modify (as required) their procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. For all OL applicants, it was determined this item was covered by Items I.C.2 and I.C.6.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	e	This item affected all operating plants. For all OL applicants, it was determined this item was covered by Items I.A.2.2 and I.A.3.1.
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	e	This item affected all operating plants. For all OL applicants, it was determined this item was covered by Items I.E.6 and III.A.3.3.
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	f	This item required all operating plants to propose TS changes reflecting implementation of all Bulletin items, as required. This item is only applicable to operating plants. The ACR Technical Specifications are based on proven CANDU technology, in accordance with current Canadian regulatory requirements, and were extensively reviewed against experience feedback. Should any possible changes to the Technical Specifications be perceived necessary for the ACR, AECL will provide assistance to the COL applicant for assessment and implementation of the change.
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	e	This item affected all operating plants with Westinghouse, CE, and GE reactors. For all OL applicants with these reactors, it was determined this item was covered by Items II.B.4, II.B.7, II.E.4.1 and II.F.1.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	e	This item affected all operating plants with Westinghouse and CE reactors. For all OL applicants with these reactors, it was determined this item was covered by Item II.E.1.2.
II.K.1(16)	Implement Procedures That Identify PZR PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	e	This item affected all operating plants with Westinghouse and CE reactors. For all OL applicants with these reactors, it was determined this item was covered by Items I.C.1 and II.D.3.
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	c	This item required all operating plants and OL applicants with Westinghouse reactors to trip the pressurizer level bistable so that the pressurizer low pressure (rather than the pressurizer low pressure and pressurizer low level coincidence) would initiate safety injection. The ACR design includes a Pressure and Inventory Control system. The emergency coolant injection will not be initiated by the pressurizer low pressure and pressurizer low level coincidence. This will be confirmed in the DCD. A detailed discussion of the trip logic and reactor physics will also be provided in the DCD.
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	e	This item affected all operating B&W plants. For all OL applicants with B&W reactors, it was determined this item was covered by Items I.C.1 and I.G.1.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	e	This item required all operating B&W plant to describe their design and procedure modifications (based on analysis) to reduce the likelihood of automatic pressurizer PORV actuation in transients. For all OL applicants with B&W reactors, it was determined this item was covered by Item II.E.5.
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	b	This item affected B&W reactors only.
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	b	This item affected B&W reactors only.
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	g	This item required the design of auxiliary heat removal systems such that necessary and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. Although this item affected BWRs only, the ACR design has well met the requirement by providing: (1) Reserve Water System (RWS) that can supply emergency feedwater (auxiliary feedwater) by gravity to the SGs in case of a total loss of feedwater; (2) Long Term Cooling (LTC) system that can cooldown the reactor after reacter is shutdown or after an accident; and (3) moderator system as a heat sink in case of loss of primary coolant. These systems can be either automatically

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			or manually activated. Further assessment will be performed and provided in the DCD to demonstrate that the ACR design addresses this GSI.
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	b	This item required all BWRs to describe their uses and types of reactor vessel indication for automatic and manual initiation safety systems, and their alternative instrumentation. This item affected BWRs only.
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip	e	This item affected all operating PWRs. For all OL applicants with PWRs, it was determined this item was covered by Item I.C.1.
II.K.1(25)	Develop Operator Action Guidelines	e	This item required all operating PWRs to develop operator action guidelines, based on the analyses performed in response to Item II.K.1(24). For OL applicant with PWRs, this item was determined to be covered by Item I.C.1.
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	e	This item required all operating PWRs to revise their emergency procedures and train ROs and SROs, based on guidelines developed in response to Item II.K.1(25). For OL applicant with PWRs, this item was determined to be covered by Items I.A.3.1, I.C.1 and I.G.1.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	e	This item required all operating PWRs to provide analyses and develop guidelines and procedures for inadequate core cooling conditions, and to define their RCP restart criteria. For OL applicant with PWRs, this item was determined to be covered by Items I.C.1 and II.F.2.
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	e	This item affected all operating PWRs. For all OL applicants with PWRs, it was determined this item was covered by Item II.K.3(5).
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	b	Item II.K.2 contains 21 requirements (II.K.2(1-21)) for 7 operating plants with B&W reactors. Only the 7 B&W operating plants were requested to complete II.K.2(1). For OL with B&W reactors, this item was determined to be covered by Items II.E.1.1 and II.E.1.2.
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	b	Only the 7 B&W operating plants were requested to complete this item. This item was also applicable to OL applicants with B&W reactors.
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	b	Only the 7 B&W operating plants were requested to complete this item.
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	b	Only the 7 B&W operating plants were requested to complete this item. For OL with B&W reactors, this item was determined to be covered by Items I.A.3.1 and I.C.1.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	b	Only the 7 B&W operating plants were requested to complete this item. For OL with B&W reactors, this item was determined to be covered by Item I.A.2.6.
II.K.2(6)	Reevaluate Analysis of Dual-Level Setpoint Control	b	This item only affected Davis-Besse 1 plant.
II.K.2(7)	Reevaluate Transient of September 24, 1977	b	This item only affected Davis-Besse 1 plant.
II.K.2(8)	Continued Upgrading of AFW System	e	Superseded by II.E.1.1, II.E.1.2.
II.K.2(9)	Analysis and Upgrading of Integrated Control System	b	This item called for B&W plants to provide failure mode effects analysis on the integrated control system. Only the operating B&W plants and OL applicants with B&W reactors were required to complete this item.
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	b	This item called for B&W plants to provide a design and schedule for implementation of a safety-grade reactor trip upon loss of feedwater, turbine trip, and significant reduction in steam generator level. Only the operating B&W plants and OL applicants with B&W reactors were required to complete this item.
II.K.2(11)	Operator Training and Drilling	b	This item called for continued operator training and drilling to assure a high state of preparedness. Only the operating B&W plants were required to complete this item. For OL applicants with B&W reactors, this item was determined to be covered by Items I.A.2.2, I.A.2.5, I.A.3.1 and I.G.1.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	e	Superseded by I.C.1(3).
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	c	This item called for the affected plants to demonstrate that sufficient mixing of the high pressure injection water would occur with the reactor coolant so that significant thermal shock effects to the reactor vessel would be precluded. Only the operating B&W plants and OL applicants with B&W reactors were required to complete this item. All PWRs were later required to address this item. Since this item focused on the integrity of reactor vessel and there is no such a vessel at ACR, there is likely no similar concern for the ACR. This will be confirmed in the DCD.
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	b	Only the operating B&W plants and OL applicants with B&W reactors were required to complete this item.
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	b	This item called for the affected plants to assess the loading on steam generator tube sheets induced from slug flow during natural circulation cooldown. Only the operating B&W plants and OL applicants with B&W reactors were required to complete this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	g	This item called for the investigation of the consequences of losing coolant to the seals of the reactor coolant pumps during loss of offsite power. Only the operating B&W plants and OL applicants with B&W reactors were required to complete this item. The ACR RCP seals will be assessed in the DCD to address the concerns noted above.
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	e	This item called for the affected plants to determine the consequences of voiding in the reactor vessel and the hot legs during normal anticipated transients. Only the operating B&W plants were required to complete this item. For OL applicants with B&W reactors, this item was determined to be covered by Item I.C.1. This item was later determined to be applicable to all PWRs. The ACR design will address this GSI by addressing Item I.C.1.
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	e	Superseded by I.C.1(3).
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	b	This item called for the evaluation of the steam generator model in the small-break licensing code (CRAFT-2) by predicting the Crystal River Asymmetric cooldown start-up test. Only the operating B&W plants were required to complete this item. For OL applicants with B&W reactors, this item was determined to be covered by Item I.C.1. This item was later determined to be applicable to all PWRs. This item focused on a specific computer code that will not be used for the ACR design and analysis. Thus, this item is not applicable to the ACR design.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA	b	This item called for the assessment of small-break LOCAs which result in pressurization of the primary system to the PORV setpoint. Only the operating B&W plants were required to complete this item. For OL applicants with B&W reactors, this item was determined to be covered by Item I.C.1.
II.K.2(21)	LOFT L3-1 Predictions	b	All 7 operating B&W plants were required to benchmark their small-break LOCA model. No OL applicant was affected by this item.
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	g	This item required all operating PWRs to provide a system that uses the PORV block valve to protect against a small-break LOCA. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. OL applicants with PWRs were also required to complete this item. The ACR design includes provisions for overpressure protection. An assessment will be performed and provided in the DCD to address this issue for the ACR.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	g	This item called for all operating PWRs to document the action to be taken to decrease the probability of a small-break LOCA caused by a stuck-open PORV. OL applicants with PWRs were also required to complete this item. Clarifications were issued in NUREG-0737 ([6]) to require licensees to perform an analysis to determine the probability of a small break LOCA caused by a stuck open PORV. The ACR design incorporates provisions for overpressure protection. An assessment will be performed and provided in the DCD to address this issue for the ACR.
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	f	This item called for all operating plants and OL applicants to report safety and relief valve failures promptly and challenges annually. Thus, this item is only applicable to operating plants and the responsibility of COL applicant.
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non- Safety Equipment for Small-Break LOCA Mitigation	e	Superseded by I.C.1, I.C.2, I.C.3.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	g	This item required all operating PWRs to study the need for automatic trip of Reactor Coolant Pumps (RCPs) and to modify procedures or design, as appropriate. OL applicants were also required to complete this item. It was required to consider the effects of automatically tripping the RCPs upon the occurrence of a small-break LOCA to assist in accident mitigation. Clarifications were issued in NUREG-0737 ([6]). The ACR design provides provisions to prevent, detect and mitigate small-break LOCA. An assessment of the need for automatic trip of RCPs in case of a small-break LOCA will be performed and provided in the DCD.
II.K.3(6)	Instrumentation to Verify Natural Circulation	e	Superseded by I.C.1(3), II.F.2, II.F.3.
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	g	This item required all PWR plants to document that their PORVs would open in less than 5% of all anticipated overpressure transients. Clarifications were issued in NUREG-0737 ([6]). ACR uses power operated valves for overpressure protection. An assessment will be performed and provided in the DCD to demonstrate that the ACR design complies with the intent of this GSI.
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	e	Superseded by II.C.1, II.E.3.3.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.3(9)	Proportional Integral Derivative Controller Modification	b	This item required all Westinghouse PWRs to implement the Westinghouse-recommended modification that is to raise the interlock bistable trip setting to preclude derivative action from opening PORV. This is only applicable to Westinghouse PWRs.
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	f	This item required that the anticipatory trip modification proposed by some licensees to confine the range of use of high-power levels not be made until it could be shown that the probability of a small-break LOCA resulting from a stuck-open PORV was substantially unaffected by the modification. The applicability of this item was to be determined on a plant-by-plant basis.
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	f	This item required plants to justify the use of PORVs that had failed during testing. The applicability of this item was to be determined on a case-by-case basis.
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	c	This item required all Westinghouse plants to confirm that their plants have an anticipatory reactor trip upon turbine trip. The behaviour of ACR is slightly different from that of PWRs. ACR would setback upon a turbine trip. An assessment of the trip logic and reactor physics of the ACR will be performed and provided in the DCD.
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	b	This GE-specific item required all GE plants to analyze the benefits to be gained from separating HPCI (High Pressure Coolant Injection) and RCIC (Reactor Core Isolation Cooling) initiation levels and providing auto-start of RCIC on low-low level.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			The interfaces between the Emergency Coolant Injection (ECI) system and Long Term Cooling (LTC) system of the ACR is different from that between the HPCI and the RCIC of a BWR, and thus, this item is not relevant to the ACR design.
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	b	This GE-specific item required all GE plants with isolation condensers to increase the availability of the isolation condensers as heat sinks by providing high radiation isolation signals at the vent rather than at the steam lines. This item is not relevant to the ACR design.
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	b	This GE-specific item required all GE plants to modify their pipe break detection circuitry to prevent isolation of systems due to startup pressure transient. This item is not relevant to the ACR design.
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	b	This item required all GE plants to study the reduction in challenge and failure rates of relief valves to minimize the most possible cause of a small-break LOCA. Since this is a GE-specific item and the ACR design will be assessed with respect to similar GSIs (II.K.3(2, 7)), no assessment will be performed on this GSI.
II.K.3(17)	Report on Outage of ECC Systems -Licensee Report and Technical Specification Changes	b	This GE-specific item required all GE plants to review data on ECC system outages to determine if cumulative outage time limitations should be incorporated in technical specifications. The availability of the ECC system of ACR has been clearly defined and analysed. Thus, this GE-specific item is not relevant to the ACR design.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.3(18)	Modification of ADS Logic -Feasibility Study and Modification for Increased Diversity for Some Event Sequences	b	This GE-specific item required all GE plants to modify their ADS (Automatic Depressurization System) actuation logic to eliminate the need for manual actuation to assure adequate core cooling. This GE-specific item is not relevant to the ACR design.
II.K.3(19)	Interlock on Recirculation Pump Loops	b	This GE-specific item required all GE operating plants with non-jet pumps to install interlocks to assure that level measurements are representative of the level in the core. This GE-specific item is not relevant to the ACR design.
II.K.3(20)	Loss of Service Water for Big Rock Point	b	This item required Big Rock Point to evaluate the acceptability or the consequences of a loss of service water. This plant-specific item is not relevant to the ACR design.
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level- Design and Modification	b	This GE-specific item required all GE plants to modify their core spray and LPCI (Low Pressure Coolant Injection) system logic so that these systems would restart, if required, to assure adequate core cooling. This GE-specific item is not relevant to the ACR design.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.3(22)	Automatic Switchover of RCIC System Suction -Verify Procedures and Modify Design	b	This GE-specific item required all GE plants to implement an automatic switchover of the RCIC system suction. The RCIC system takes suction from the condensate storage tank with manual switchover to the suppression pool when the condensate storage tank level is low. The NRC staff believed the switchover should be made automatically. This GE-specific item is not relevant to the ACR design.
II.K.3(23)	Central Water Level Recording	e	Superseded by I.D.2, III.A.1.2(1), III.A.3.4.
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	b	This GE-specific item required all GE plants to verify that HPCI and RCIC are designed to withstand loss of offsite power for at least 2 hours. This GE-specific item is not relevant to the ACR design.
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	g	This item required all GE plants to verify the adequacy of pump seals to withstand loss of cooling water due to loss of AC power for at least 2 hours. Clarifications were issued in NUREG-0737 ([6]) to include all BWRs, Westinghouse and CE operating plants and all OL applicants. The ACR HTS pump seals have been designed to withstand the impact caused by loss of AC power supply. Further assessment will be performed and provided in the DCD to demonstrate that the ACR design complies with this GSI.
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	e	Superseded by II.E.2.1.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	b	This GE-specific item required all GE plants to reference all reactor vessel level instruments to the same point. It was believed that different reference points of the various reactor vessel instruments could cause operator confusion. This GE-specific item is not relevant to the ACR design.
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	b	This GE-specific item required all GE plants to assure that air or nitrogen accumulators for ADS valves had sufficient capacity to cycle the valves open five times at design pressure. Since ACR does not have such an ADS system, thus, this GE-specific item is not relevant to the ACR design.
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensable	b	This GE-specific item required all GE operating plants with isolation condensers to demonstrate the adequacy of isolation condensers with non-condensable. This GE-specific item is not relevant to the ACR design.
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	g	This item required all operating plants and OL applicants to revise and submit for NRC approval the analyses used by NSSS vendors and/or fuel suppliers for small-break LOCA analysis in compliance with 10 CFR 50, Appendix K ([25]). The revised analyses were to account for comparisons with experimental data, including data from the LOFT and semiscale test facilities. Small-break LOCA analyses have been conducted for the ACR design. A summary of the analyses will be provided in the DCD to demonstrate that the ACR design meets the intent of this GSI.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	g	This item required all operating plants and OL applicants to submit for NRC approval plant-specific calculations using NRC-approved models for small-break LOCA, to show compliance with 10 CFR 50.46 ([26]). Small-break LOCA analyses have been conducted for the ACR design by using models developed by AECL. A summary of the analyses will be provided in the DCD to demonstrate that the ACR design meets the intent of this GSI.
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	e	Superseded by II.E.2.2.
II.K.3(33)	Evaluate Elimination of PORV Function	e	Superseded by II.C.1.
II.K.3(34)	Relap-4 Model Development	e	Superseded by II.E.2.2.
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	e	Superseded by I.C.1(3).
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	e	Superseded by I.C.1(3).
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	e	Superseded by I.C.1(3).
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the e Pressurizer Spray Line	e	Superseded by I.C.1(3).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by BPI and CFT Flows	e	Superseded by I.C.1(3).
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA	e	Superseded by I.C.1(16).
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	e	Superseded by I.C.1(3).
II.K.3(42)	Submit Requested Information on the Effects of Non Condensable Gases	e	Superseded by I.C.1(3).
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	e	Superseded by I.C.1(15).
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	b	This GE-specific item required all GE plants to show that transients combined with the worst single failure would not result in significant fuel damage. The ACR design is significantly different from BWR design in operation modes, safety systems, and etc. The ACR design will be assessed on the assumption of a single failure to ensure fuel integrity during anticipated transients. However, this will not be demonstrated by addressing this BWR-specific item. Thus, this item is ranked not applicable to the ACR design.
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	b	This GE-specific item required all GE plants to analyze depressurization modes other than full ADS for possible inclusion in emergency procedures. The NRC staff believed that slower depressurization would reduce the possibility of

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			exceeding vessel integrity limits by rapid cooldown. Since the slower depressurization modes of a BWR will not occur at the ACR, this BWR-specific item is deemed not applicable to the ACR design.
II.K.3(46)	Response to List of Concerns from ACRS Consultant	b	This GE-specific item required all GE plants to respond to concerns raised by ACRS consultants. Since the concerns were specific to BWRs, this item is applicable to BWRs only.
II.K.3(47)	Test Program for Small-Break LOCA Model Verification, Pretest Prediction, Test Program, and Model Verification	e	Superseded by I.C.1(3), II.E.2.2.
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	e	Superseded by II.C.1, II.C.2.
II.K.3(49)	Review of Procedures (NRC)	e	Superseded by I.C.8, I.C.9.
II.K.3(50)	Review of Procedures (NSSS Vendors)	e	Superseded by I.C.7, I.C.9.
II.K.3(51)	Symptom-Based Emergency Procedures	e	Superseded by I.C.9.
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	e	Superseded by I.B.1.1, I.C.2, I.C.5.
II.K.3(53)	Two Operators in Control Room	e	Superseded by I.A.1.3.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	e	Superseded by I.A.4.1(2).
II.K.3(55)	Operator Monitoring of Control Board	e	Superseded by I.C.1(3), I.D.2, I.D.3.
II.K.3(56)	Simulator Training Requirements	e	Superseded by I.A.2.6(3), I.A.3.1.
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	b	This GE-specific item required all GE operating plants to revise their emergency procedures to include verification that low pressure cooling systems are available prior to manual ADS. For OL applicants, this GSI was determined to be covered by Item I.C.1. ACR does not have an Automatic Depressrization System or equivalent such as used in BWRs. This is a BWR-specific item and not applicable to ACR.
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	f	This item called for licensees to promptly upgrade their overall state of emergency preparedness for accidents, including the integration of onsite and offsite emergency preparedness. This item is only applicable to operating plants or the responsibility of COL applicant.
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	f	This item called for the NRC staff to perform a long-term integrated assessment of the implementation of the actions required by Item III.A.1.1(1). For the same reason as for Item III.A.1.1(1), this item is only applicable to operating plants or the responsibility of COL applicant.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
III.A.1.2(1)	Technical Support Center	g	This item called for a dedicated Technical Support Center (TSC) to provide a place for management and technical personnel to support reactor control functions, to evaluate and diagnose plant conditions, and for a more orderly conduct of emergency operations. The TSC was required to be separate from but near the control room and was expected to have the capability to display and transmit plant status to those individuals knowledgeable of and responsible for engineering and management support of reactor operations, in the event of an accident. The ACR design includes a technical support center that provides space for the required staff complement and safety parameter displays and related emergency support documentation. This will be confirmed in the DCD.
III.A.1.2(2)	On-Site Operational Support Center	g	This item called for the establishment of an Operational Support Center (OSC) separate from the control room as a place in which operations support personnel could assemble in an emergency situation to receive instructions from the operating staff. The OSC was to be provided with communication capability with the plant control room, TSC, and the near-site Emergency Operations Facility (EOF). The ACR design includes PAM process and a secondary control building that will serve as an OSC. Further assessment will be performed and provided in the DCD to demonstrate that the ACR design complies with this GSI.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
III.A.1.2(3)	Near-Site Emergency Operations Facility	f	This item called for a near-site EOF to provide a planned, organized, central focal point for coordination of on-site and off-site activities for reactor emergency situations. The EOF was required to be operated by licensees and sized and equipped to function as a center for: (1) licensee command and control functions of on-site operations and evaluation and coordination of all on-site and off-site licensee activities related to an emergency having actual or potential environmental consequences; and (2) analysis of plant effluent monitors, meteorological conditions, and off-site radiation measurements, and for off-site dose projections. The EOF is the responsibility of the COL applicant and is out of the ACR design scope. However, the interface requirements will be provided by the designer of the ACR, AECL. AECL will also provide the COL applicant with assistance in the establishment of the EOF on an as required basis.
III.A.1.3(1)	Maintain Supplies of Thyroid-Blocking Agent -Workers	f	This item and Item III.A.1.3(2) required licensees to have adequate supplies of potassium iodine (KI) available for onsite personnel and for offsite emergency response support personnel, including offsite agencies, which is a requirement for the operating plants and OL applicants.
III.A.1.3(2)	Maintain Supplies of Thyroid-Blocking Agent -Public	f	Same as III.A.1.3(1).
III.A.2.1(1)	Publish Proposed Amendments to the Rules	d	This resolved item called for the NRC to revise 10 CFR 50 and as such is ranked as an NRC Internal Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
III.A.2.1(2)	Conduct Public Regional Meetings	d	This resolved issue called for the NRC to conduct public meeting with state and local authorities and, therefore, is ranked as an NRC Internal Issue.
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	d	This resolved item called for the NRC staff to prepare a final Commission Paper recommending the adoption of effective rules and is therefore ranked as an NRC Internal Issue.
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	d	This resolved issue called for the NRC staff to revise its inspection program and is therefore ranked as an NRC Internal Issue.
III.A.2.2	Development of Guidance and Criteria	d	This resolved item called for the NRC and FEMA to use NUREG –0654 ([27]) as interim guidance and upgraded criteria in judging the adequacy of various emergency plan and preparedness. The licensees, and state and local governments may need to address this item. It is not a design certification issue.
III.A.3.1(1)	Define NRC Role in Emergency Situations	d	This resolved item was to define the NRC role in emergency situations and as such is ranked as an NRC Internal Issue.
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	d	Same as III.A.3.1(1).
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	d	Same as III.A.3.1(1).
III.A.3.1(4)	Prepare Commission Paper	d	Same as III.A.3.1(1).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	d	Same as III.A.3.1(1).
III.A.3.2	Improve Operations Centers	d	This resolved item called for the NRC to upgrade its operation center, and as such is ranked as an NRC Internal Issue.
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	d	This resolved item was to improve the communication system for NRC emergency preparedness, and as such is ranked as an NRC Internal Issue.
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	d	Same as III.A.3.3(1).
III.A.3.4	Nuclear Data Link	d	This resolved item was to enable the NRC staff to obtain more current and reliable plant data, so that the staff can help develop and evaluate accident mitigating actions. Therefore this item is ranked as an NRC internal Issue.
III.A.3.5	Training, Drills, and Tests	d	This resolved item was to slowly expand the scope of the NRC headquarters and regional drills and is therefore ranked as an NRC Internal Issue.
III.A.3.6(1)	Interaction of NRC and Other Agencies -International	d	This resolved item was to establish interaction agreements between NRC and other agencies for cooperation during emergency situations, and as such is ranked as a Regulatory Impact Issue.
III.A.3.6(2)	Federal	d	Same as III.A.3.6(1).
III.A.3.6(3)	State and Local	d	Same as III.A.3.6(1).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
III.B.1	Transfer of Responsibilities to FEMA	d	This resolved item called for the NRC to transfer its lead responsibility with regard to state and local government radiological emergency preparedness to FEMA in accordance with a president order. Therefore this item is ranked as a Regulatory Impact Item.
III.B.2(1)	The Licensing Process	d	Same as III.B.1.
III.B.2(2)	Federal Guidance	d	Same as III.B.1.
III.C.1(1)	Review Publicly Available Documents	d	NUREG-0933 ranked this item as a Licensing Issue.
III.C.1(2)	Recommend Publication of Additional Information	d	NUREG-0933 ranked this item as a Licensing Issue.
III.C.1(3)	Program of Seminars for News Media Personnel	d	NUREG-0933 ranked this item as a Licensing Issue.
III.C.2(1)	Develop Policy and Procedures for Dealing With Briefing Requests	d	NUREG-0933 ranked this item as a Licensing Issue.
III.C.2(2)	Provide Training for Member of the Technical Staff	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	g	It was required to implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. The ACR design includes design features and access control to ensure that the dose limit for the areas outside containment will not be exceeded and the ALARA principle is applied. This will be confirmed in the DCD.
III.D.1.1(2)	Review Information on Provisions for Leak Detection	a	NUREG-0933 dropped this item.
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	a	NUREG-0933 dropped this item.
III.D.1.2	Radioactive Gas Management	a	NUREG-0933 dropped this item.
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications	a	NUREG-0933 dropped this item.
III.D.1.3(2)	Review and Revise SRP	a	NUREG-0933 dropped this item.
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	a	NUREG-0933 dropped this item.
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Absorber	d	This resolved item required the NRC to evaluate the charcoal absorber radioiodine collection performance under accident conditions, and as such is ranked as an NRC Internal Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	a	NUREG-0933 dropped this item.
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	a	NUREG-0933 prioritized this item as Low.
III.D.2.1(2)	Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released to the Atmosphere	a	NUREG-0933 prioritized this item as Low.
III.D.2.1(3)	Revise Regulatory Guides	a	NUREG-0933 prioritized this item as Low.
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	d	This resolved item called for the NRC to undergo further research for improving understanding of Radioiodine, Carbon-14, and Tritium Behavior. Therefore it is ranked as an NRC Internal Issue.
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	e	Superseded by III.D.2.5.
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air- Water-Steam Mixtures	e	Superseded by III.D.2.5.
III.D.2.2(4)	Revise SRP and Regulatory Guides	e	Superseded by III.D.2.5.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	d	Items III.D.2.3(1-4) were concerned with improving public radiation protection in the event of a nuclear power plant accident by improving the control of radioactivity released into the liquid pathway. This control could be accomplished by the application of various interdictive measures at the source of the release and/or along the liquid pathway. Techniques were developed and were being used to evaluate the liquid pathway effects of an accident for each reactor site. Sites that might require interdictive measures related to liquid pathway releases were to be determined. Interdictive measures were to be assessed as to their effectiveness in improving public radiation protection. It was the NRC staff's responsibility to complete these items. The NRC staff: a) Completed a liquid pathway analysis for Zion, b) Performed a liquid pathway analysis for Indian Point, c) Published a simplified analysis for liquid pathway, and d) Drafted and published Section 7.1.1 of Environmental Standard Review Plan (ESRP) ([28]) with no new requirements for licensees or applicants. COL applicant should refer to NRC publications noted above in evaluation of the site-specific liquid pathway.
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	d	Same as III.D.2.3(1).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	d	Same as III.D.2.3(1).
III.D.2.3(4)	Prepare a Summary Assessment	d	Same as III.D.2.3(1).
III.D.2.4(1)	Study Feasibility of Environmental Monitors	d	This resolved item called for the NRC staff to study the feasibility of environmental monitors capable of measuring real-time rates of exposures to noble gases and radioiodines, and it is therefore ranked as an NRC Internal Issue.
III.D.2.4(2)	Place 50 TLDs Around Each Site	d	NUREG-0933 ranked this item as a Licensing Issue.
III.D.2.5	Offsite Dose Calculation Manual	f	This item called for the NRC staff to prepare a manual to be used by the NRC and plant personnel to estimate the maximum individual doses and population doses during an accident. This item was resolved with the publication of the manual requested. No new requirements were issued. Since the offsite dose calculation is related to the site characteristics and the specific accident scenario, the manual prepared by NRC should be used by the plant personnel or the COL applicants. Thus, this is not a design certification issue.
III.D.2.6	Independent Radiological Measurements	d	NUREG-0933 ranked this item as a Licensing Issue.
III.D.3.1	Radiation Protection Plans	f	This item was to improve nuclear power plant worker radiation protection programs by better defining the performance criteria and responsibility for such programs. This item resulted in no new requirements.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Since this item mainly dealt with the responsibility of the plant personnel involved, and as such should be addressed by licensees or COL applicants.
III.D.3.2(1)	Amend 10 CFR 20	d	NUREG-0933 ranked this item as a Licensing Issue.
III.D.3.2(2)	Issue a Regulatory Guide	d	NUREG-0933 ranked this item as a Licensing Issue.
III.D.3.2(3)	Develop Standard Performance Criteria	d	NUREG-0933 ranked this item as a Licensing Issue.
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	d	NUREG-0933 ranked this item as a Licensing Issue.
III.D.3.3(1)	Issue Letter requiring Improved Radiation Sampling Instrumentation	g	Items III.D.3.3(1-4) addressed the need for additional survey equipment and radiation monitors in vital areas. NUREG-0737 ([6]) clarified the requirements for improving plant iodine monitoring instrumentation and for in-containment high-range radiation monitors, for use during and following an accident. These requirements were also incorporated in Regulatory Guide 1.97 ([16]), and SRP ([10]) Sections 12.3 and 12.5, were revised to include additional monitor requirement criteria. Regulatory Guide 8.25 ([29]) was also issued to prescribe acceptable methods for calibration of air sampling instruments, and a revision of 10 CFR 20 ([30]), Paragraph 20.501(c) provided acceptable methods for calibration of personnel radiation monitors. The ACR design provides a radiation monitoring system to:

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			a) detect the presence of and locate failed fuel in the core; b) measure the external radiation fields around the plant, and survey personnel, work locations and equipment for any sources of contamination; c) monitor personal dosimetry; d) monitor the release of airborne and liquid effluents from the station and alarm upon the release of activity concentration above some preset level; e) provide offsite monitoring of radioactive material in environmental media; f) provide a low level radiation laboratory for the analysis of environmental and biological samples and the monitoring of individuals; and g) provide a calibration facility to calibrate all radiation monitors used in the plant. Further assessment will be performed and provided in the DCD to demonstrate that the ACR design addresses the requirements of the relevant Sections of SRP ([10]), 10 CFR and Regulatory Guides as noted above.
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	g	Same as III.D.3.3(1).
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation Monitoring Instruments	g	Same as III.D.3.3(1).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
III.D.3.3(4)	Issue a Regulatory Guide	g	Same as III.D.3.3(1).
III.D.3.4	Control Room Habitability	g	NUREG-0737 ([6]) clarified this item, which stated that "licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions". NUREG-0737 ([6]) also specified the Sections of SRP ([10]) that should be met and Regulatory 1.78 and 1.95 that should be used for guidance. An HVAC system is provided to ensure the habitability of ACR main control room during normal operation. This system, together with other systems (e.g., radiation monitoring system), is also capable to ensure the post-accident habitability of ACR control room. Further assessment will be performed and provided in the DCD to demonstrate that the ACR design complies with this GSI.
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	d	NUREG-0933 ranked this item as a Licensing Issue.
III.D.3.5(2)	Investigate Methods of Obtaining Employee Health Data by Nonlegislative Means	d	NUREG-0933 ranked this item as a Licensing Issue.
III.D.3.5(3)	Revise 10 CFR 20	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.A.1	Seek Legislative Authority	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
IV.A.2	Revise Enforcement Policy	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	d	This resolved item called for the NRC staff to extend the lessons learned from TMI to other key NRC programs, and as such is ranked as an NRC Internal Issue.
IV.D.1	NRC Staff Training	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.E.1	Expand Research on Quantification of Safety Decision-Making	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.E.2	Plan for Early Resolution of Safety Issues	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.E.3	Plan for Resolving Issues at the CP Stage	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.E.4	Resolve Generic Issues by Rulemaking	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.E.5	Assess Currently Operating Reactors	d	This item called for the NRC staff to develop an integrated program of safety decisionmaking and was finally resolved with no new requirement established. Thus it is ranked as an NRC Internal Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	d	This resolved item was to enhance public safety through the reduction of disincentives to safety resulting from financial pressure on the utility at various stages. Thus it is ranked as a Licensing Issue.
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	d	Same as IV.F.1.
IV.G.1	Develop a Public Agenda for Rulemaking	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.G.3	Improve Rulemaking Procedures	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.G.4	Study Alternatives for Improved Rulemaking Process	d	NUREG-0933 ranked this item as a Licensing Issue.
IV.H.1	NRC Participation in the Radiation Policy Council	d	NUREG-0933 ranked this item as a Licensing Issue.
V.A.1	Develop NRC Policy Statement on Safety	d	NUREG-0933 ranked this item as a Licensing Issue.
V.B.1	Study and Recommend, as Appropriate, Elimination of Nonsafety Responsibilities	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
V.C.1	Strengthen the Role of Advisory Committee on Reactor Safeguards	d	NUREG-0933 ranked this item as a Licensing Issue.
V.C.2	Study Need for Additional Advisory Committees	d	NUREG-0933 ranked this item as a Licensing Issue.
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	d	NUREG-0933 ranked this item as a Licensing Issue.
V.D.1	Improve Public and Intervenor Participation in the Hearing Process	d	NUREG-0933 ranked this item as a Licensing Issue.
V.D.2	Study Construction-During-Adjudication Rules	d	NUREG-0933 ranked this item as a Licensing Issue.
V.D.3	Reexamine Commission Role in Adjudication	d	NUREG-0933 ranked this item as a Licensing Issue.
V.D.4	Study the Reform of the Licensing Process	d	NUREG-0933 ranked this item as a Licensing Issue.
V.E.1	Study the Need for TMI-Related Legislation	d	NUREG-0933 ranked this item as a Licensing Issue.
V.F.1	Study NRC Top Management Structure and Process	d	NUREG-0933 ranked this item as a Licensing Issue.
V.F.2	Reexamine Organization and Functions of the NRC Offices	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
V.F.3	Revise Delegations of Authority to Staff	d	NUREG-0933 ranked this item as a Licensing Issue.
V.F.4	Clarify and Strengthen the Respective Roles of Chairman, Commission, and Executive Director for Operations	d	NUREG-0933 ranked this item as a Licensing Issue.
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	d	NUREG-0933 ranked this item as a Licensing Issue.
V.G.1	Achieve Single Location, Long-Term	d	NUREG-0933 ranked this item as a Licensing Issue.
V.G.2	Achieve Single Location, Interim	d	NUREG-0933 ranked this item as a Licensing Issue.
<u>Task Action Plan Items</u>			
A-1	Water Hammer (former USI)	g	This item addressed identifying the probable causes of water hammer and minimizing the susceptibility of fluid systems and components to water hammer by correcting design and operational deficiencies. Water hammer issue was raised after the occurrence of various incidents of water hammer that involved steam generator feedrings and piping. Water hammer was observed in many fluid systems including residual heat removal, containment spray, service water, feedwater systems, and main steam lines. The incident was attributed to such causes as rapid condensation

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			of steam pockets, steam driven slugs of water, pump startup with partially empty lines, and rapid valve cycling. Regardless of the initiating event, water hammer and the resulting fluid accelerations can cause damage to the affected fluid system. The level of severity of damage depends upon the event, and can range from minor damage such as overstressed pipe hangers to major damage to restraints, piping and components. As a result of the closure of this item, NRC issued NUREG-0927 ([31]) that states that water hammer can be induced by operator/maintenance actions and by design inadequacies. Experience has shown that water hammer events reported on LERs are about equally divided between operator or maintenance actions and design deficiencies. The NRC implemented SRP ([10]) changes relative to the design, operation, and maintenance of new plants to minimize the probability and effects of water hammer. According to NUREG-0933, this item is applicable to all future plants. CANDU plants have adequately addressed the effects of water hammer. The occurrence probability of water hammer at CANDU plants has been minimized by design features based on piping stress analysis and by following well-established maintenance and operating procedures. The ACR design follows the accepted engineering practices identified at CANDU plants in resolving water hammer related issue. Further assessment will be performed and provided in the DCD to demonstrate the compliance of the ACR design with

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			NUREG-0927 ([31]) and relevant Sections of SRP ([10]).
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	g	This item addressed asymmetric blowdown loads imposed on the reactor vessel (RV) as a result of a loss of coolant accident (LOCA). The resultant forces from these loads could affect reactor vessel support integrity, thus jeopardizing plant safety. The NRC reviewed the predicted asymmetric loadings and developed acceptance criteria and guidelines, which were documented in NUREG-0609 ([32]) that requires that the design of the reactor primary coolant system shall demonstrate that the asymmetric loads on the reactor vessel, internals, primary coolant loop, and components shall not exceed the limits imposed by the applicable codes and standards. The ACR design provides protection against postulated pipe ruptures. The ACR design also addresses the loading on the reactor from the impact of fuel at the inlet end of channels downstream of an inlet header break that would cause flow reversal. Further assessment will be performed and provided in the DCD to show the compliance of the ACR design with this GSI.
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	g	Items A-3, A-4 and A-5 addressed the measures necessary to verify and maintain PWR steam generator tube integrity, and to mitigate the consequences of a Steam Generator Tube Rupture (SGTR) event. The NRC implemented a program to assess the lessons that were learned from domestic SGTR events, as well as identifying and evaluating potential new requirements. The results of the

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			program were documented in NUREG-0844 ([33]). As a part of the program, the NRC staff issued a Generic Letter 85-02 ([34]) to all PWR licensees and applicants to inform them of the staff-recommended actions and to request a description of their overall program for ensuring steam generator tube integrity and SGTR mitigation. Future plant was required to facilitate adoption of the measures recommended in the Generic Letter 85-02 ([34]), to the extent they are applicable to that design. These measures are as follows: a) Prevention and detection of loose parts by visual inspection of the secondary side of the steam generators, and by quality assurance/quality control procedures to preclude introduction of foreign objects into the primary or secondary sides during inspection, maintenance and repair operations. Precautions should be taken to minimize the potential for corrosion while the tube bundle is exposed to air. b) In-service inspection of the steam generator tubes should include full tube length examinations. The maximum allowable time between eddy current inspections of an individual steam generator should be limited consistent with the NRC Standard Technical Specifications (STS) and in addition should not extend beyond 72 months. c) Secondary water chemistry program should incorporate the guidelines in EPRI-NP-2704, "PWR Secondary Water Chemistry Guidelines", and corrective action procedures for out-of specification conditions. There should also be an inservice inspection program for the condenser. The program should include identification and location of

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			leakage sources, methods of repair, and a preventive maintenance program. d) Technical specification limits for primary to secondary leakage rates should not be less restrictive than the NRC STS limits. e) Technical specification limits and surveillance for coolant iodine activity should not be less restrictive than the NRC STS limits. Additional restrictions are recommended for plants having low head high pressure safety injection pumps. f) The safety injection signal reset logic should minimize the loss of safety function associated with safety injection reset during an SGTR event. Automatic switchover of safety injection pumps from the boric acid storage tank to the containment sump should take into account the concurrent status of the safety injection signal. Item f) above is not applicable to the ACR design, since no liquid poison is to be added to the primary coolant system. The ACR design adequately addresses the items a) – e) above by including design features, technical specifications and operating and maintenance guidelines that facilitate the adoption of the NRC recommendations. Further assessment will be performed and provided in the DCD to show ACR's compliance with requirements mentioned above except item f).
A-4	CE Steam Generator Tube Integrity (former USI)	g	Same as A-3.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
A-5	B&W Steam Generator Tube Integrity (former USI)	b	The B&W steam generators are once-through straight tubes but ACR steam generators are similar to CE designs. This B&W plant specific item is not applicable to the ACR design.
A-6	Mark I Short-Term Program (former USI)	b	This item called for a reassessment of the MARK I containment system design (BWR). During the evaluation of an advanced design BWR pressure suppression containment system (MARK III), new suppression hydrodynamic loads associated with a postulated LOCA were identified which had not been explicitly included in the original design of the MARK I containment systems. This item was determined applicable to operating plant with MARK I containment systems only and was resolved. The ACR design does not include such type of containment system. Thus, this item is not applicable to the ACR design.
A-7	Mark I Long-Term Program (former USI)	b	This item called for conducting a long-term program to provide a generic basis to define suppression pool hydrodynamic loads and the structural acceptance criteria, such that a comprehensive reassessment of each MARK I containment system would be performed. This program was implemented by MARK I Owners Group. This item was resolved with the issuance of Supplement 1 to NUREG-0661 ([35]) and SRP ([10]) Section 6.2.1.1C. For the same reason as for Item A-6, this item is not applicable to the ACR design.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
A-8	Mark II Containment Pool Dynamic Loads Long-Term Program (former USI)	b	This item called for GE BWRs with Mark II containment to address issues identified in the evaluation of an advanced design BWR pressure suppression containment system (MARK III). This item was resolved with the issuance of NUREG-0808 ([36]) and SRP ([10]) Section 6.2.1.1C. For the same reason as for Item A-6, this item is not applicable to the ACR design.
A-9	ATWS (former USI)	g	This item called for assuring that the reactor can attain safe shutdown after incurring an anticipated transient with a failure of the Reactor Trip System (RTS). 10 CFR 50.62 ([37]) requires that the reactor must be capable of reaching a safe shutdown condition after incurring an anticipated transient and a RTS failure. The ACR design provides two completely redundant and diverse shutdown systems (SDS1 and SDS2). In addition to the control rods which are capable of shutting down reactor for all design basis accidents, each SDS on its own is capable of safely shutting down the reactor for any accident. The diversity extends even to the poison portion of each system, unlike the common shutdown rods used in most PWRs. A number of anticipated operational occurrences included in NRC ATWS rulemaking process are already within the design bases for the shutdown systems of the ACR. Further assessment will be performed and provided in the DCD to ensure the ATWS concerns are addressed for the ACR design and the requirements in 10 CFR 50.62 ([37]) have been met.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
A-10	BWR Feedwater Nozzle Cracking (former USI)	b	This item addressed the cracking in the feedwater nozzles of reactor vessels identified during the inspections of operating BWRs. This BWR specific item is not applicable to the ACR design.
A-11	Reactor Vessel Materials Toughness (former USI)	g	This item addressed the degradation in reactor vessel material fracture toughness due to neutron irradiation as the plant accumulates more and more service time. Unlike PWRs and BWRs, reactor coolant in ACR is contained in 284 individual pressure tubes. These tubes are designed for a service lifetime of 30 years and can be replaced if necessary before then. The mechanism of tube cracking has been well understood, attributed to delayed hydride cracking, and adequately addressed in the ACR design. The fitness for service guidelines are used to monitor the pressure tube performance. Further assessment will be performed and provided in the DCD to ensure the ACR design complies with this GSI.
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	g	This item addressed minimizing the susceptibility for lamellar tearing and low fracture toughness of steam generator and reactor coolant pump supports. During the course of licensing the North Anna Units 1 and 2, a number of questions were raised as to the potential for lamellar tearing (a cracking phenomenon that occurs beneath welds involving rolled steel plate) and low fracture toughness of the steam generator and reactor coolant pump (RCP) support materials. Concerns regarding the supports at North Anna were determined to apply to all PWRs. This item was resolved with the issuance of NUREG-0577

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			([38]). It was required that the major RCS component supports must meet the requirements specified in Subsection NF of the ASME Code. The SG and RCP supports in the ACR design are classified as ASME Code NF components and are specified to meet the requirements of subsection NF of the ASME Code, for design, material selection, fabrication and inspection. The compliance of the ACR design with this GSI will be confirmed in the DCD.
A-13	Snubber Operability Assurance	g	This item addressed snubber selection and operability for safety related systems and components. Snubbers are utilized primarily as seismic and pipe whip restraints at operating plants. Their safety function is to operate as rigid supports for restraining the motion of systems or components under dynamic load conditions such as earthquakes and severe hydraulic transients, e.g., pipe breaks. NUREG-0933 stated that a substantial number of Licensee Event Reports (LERs), concerning snubber operability, were issued by utilities. A review of these LERs showed that a variety of methods were employed to determine the operability of the snubbers and that different types of snubbers were used for systems with similar configurations. To resolve this issue, it is required that the design, specification, installation, and inservice operability of snubbers must meet the intent of the guidance given in SRP ([10]) Section 3.9.3. For ACR, snubbers are used to restrain piping systems under dynamic loading conditions while also accommodating thermal

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			expansion. An assessment will be performed and provided in the DCD to show that the ACR design meets the intent of SRP ([10]) Section 3.9.3.
A-14	Flaw Detection	a	NUREG-0933 dropped this item.
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	c	This item called for the NRC staff to develop criteria for decontamination of the primary coolant system and steam generator of nuclear power plant. It was believed that periodic removal of activated corrosion products from the primary coolant system and steam generator would reduce occupational exposure due to maintenance and inspection activities. While some activated corrosion products are removed by the reactor's water chemistry system, a small amount is continually deposited or plated out on the primary coolant system's internal surfaces. The criteria for decontamination were reported in NUREG/CR-2963 ([39]). The HTS and moderator system are the major sources of radioactive material at ACR. The ACR design ensures a pretty low occupational exposure due to maintenance and inspection activities. This is achieved by: a) Shielding, and the shield tank design that allows access to many areas of the plant and parts of systems, both during power operation and at shutdown. b) Well designed cleanup systems (e.g., the Heat Transport Purification System keeps the heat transport system free of impurities, such as activated corrosion products, fission products and non-active ionic impurities; the ACR design

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			also includes considerations on chemical cleaning of both the primary and secondary sides of the steam generator.), c) Deliberate selection of the materials used in HTS and moderator systems, and d) Radiation monitoring system. The compliance of the ACR design with this GSI will be confirmed in the DCD.
A-16	Steam Effects on BWR Core Spray Distribution	b	This GE BWR-specific item addressed the concerns over BWR core spray distribution and called for a series of relevant tests. This BWR-specific item is not applicable to the ACR design.
A-17	Systems Interactions in Nuclear Power Plants (former USI)	c	This item was to investigate the potential that unrecognized, subtle dependencies among structures, systems, and components (SSCs) have remained hidden and that they could lead to safety-significant events. A number of significant, plant-specific events have involved unintended or unrecognized dependencies among the SSCs. Some of these events have involved subtle dependencies between safety-related SSCs and other SSCs. Some events have also involved subtle dependencies between redundant safety-related SSCs that were believed to be independent. This item was resolved for US PWRs and BWRs with no new requirements established. However, the NRC staff identified a few actions that should be taken. For future plants, it was required that attention shall be paid in the detailed plant design to detecting and minimizing the potential for adverse system interactions due to the effects of flooding and water intrusion

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			from internal plant sources, such as the incidents at operating plants referenced in NUREG-1174 ([40]). In the ACR design, independence between each safety system ensures that the remaining safety systems are not impaired by the one that becomes unavailable and that they can provide the required safety functions. The safety systems are physically and functionally independent from the process systems** to the greatest practical extent, to ensure that the required safety function is not lost as a consequence of a process system failure. The fire, seismic, and flooding PRAs will address physical dependencies between systems for those events. Further assessment will be performed and provided in the DCD to ensure the concerns noted above by the NRC are addressed in the ACR design.
A-18	Pipe Rupture Design Criteria	a	NUREG-0933 dropped this item.
A-19	Digital Computer Protection System	d	NUREG-0933 ranked this item as a Licensing Issue.
A-20	Impacts of the Coal Fuel Cycle	d	NUREG-0933 ranked this item as a Licensing Issue.
A-21	Main Steamline Break Inside Containment -Evaluation of Environmental Conditions for Equipment Qualification	a	NUREG-0933 dropped this item.

** The ACR process systems include Heat Transport System and auxiliaries, Moderator system and auxiliaries and Steam and Feedwater systems.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
A-22	PWR Main Steamline Break -Core, Reactor Vessel and Containment Building Response	a	NUREG-0933 dropped this item.
A-23	Containment Leak Testing	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	g	This item addressed the adequacy of environmental qualification methods and acceptance criteria for Class 1E electrical equipment. The NRC initially required license applicants to qualify all safety-related equipment to IEEE-323 ([41]). Some of the industry qualification methods and concepts proposed in accordance with this standard, such as testing margins, aging effects, and the simulation of worst case environments, were not resolved to the satisfaction of the NRC. It was therefore decided that a generic approach should be developed to expedite the review and assessment of equipment qualification methods used by NSSS and BOP vendors. This item was resolved with the publication of NUREG-0588 ([42]). In a subsequent rulemaking, 10 CFR 50.49 ([43]) established the requirement for an environmental qualification program for Class 1E electrical equipment together with rules for its content. NUREG-0588 ([42]) and 10 CFR 50.49 comprise the bases for the rules. Regulatory Guide 1.89 ([44]) was then revised to describe an acceptable method for complying with 10 CFR 50.49 ([43]). Dynamic and seismic qualification of Class 1E electrical equipment was not included in the scope of

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			10 CFR 50.49 ([43]). Existing dynamic and seismic qualification requirements are identified in Regulatory Guide 1.100 ([45]). The Class 1E electrical equipment in the ACR design is environmentally qualified in accordance with IEEE-323 ([41]) that is endorsed by Regulatory Guide 1.100 ([45]). Dynamic and seismic qualification is conducted according to Canadian Standards, CAN3-N289.3 ([46]) and CAN3-N289.4 ([47]). Further assessment will be performed and provided in the DCD to demonstrate the compliance of the ACR design with the requirements noted in this item.
A-25	Non-Safety Loads on Class 1E Power Sources	g	This item addressed whether nonsafety related loads should be allowed to share class 1E power sources with safety-related plant systems. Past regulatory practices has allowed the connection of nonsafety-related loads in addition to the required safety loads to class 1E power sources by imposing some restrictions. This item called for the NRC staff to determine whether the reliability of the Class 1E power sources is significantly affected by the sharing of safety and nonsafety-related loads to a Class 1E power source. This item was resolved with the issuance of Revision 2 to Regulatory Guide 1.75 ([48]) that includes special requirements for connection of nonsafety-related loads to Class 1E power sources. Standard IEEE-384 ([49]) that is consistent with Regulatory Guide 1.75 ([48]) has been applied to the ACR design. Further assessment will be performed and provided in the DCD to

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			confirm that the ACR design meets the intent of the Regulatory Guide 1.75 ([48]).
A-26	Reactor Vessel Pressure Transient Protection (former USI)	g	This item addressed the need to provide PWR reactor vessel overpressure protection whenever plants are in a cold shutdown condition. NUREG-0933 states that there have been, since 1972, many events of pressure transients which have exceeded the pressure-temperature limits of pressurized water reactor vessels. The majority of these events occurred at relatively low reactor vessel temperatures at which the material has less toughness and is more susceptible to failure through brittle fracture. Therefore, the margin of safety to vessel failure under low temperature conditions is reduced. This item was resolved with the publication of NUREG-0224 ([50]) and SRP ([10]) Section 5.2. There is no pressure vessel in the ACR design. However, an assessment will be performed and provided in the DCD for the potential functionally equivalent aspects of the ACR design to which this issue may apply.
A-27	Reload Applications	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
A-28	Increase in Spent Fuel Pool Storage Capacity	g	This item addressed the safety significance of damage to spent fuel, primarily from a lack of adequate cooling in the spent fuel pool, that could result in the release of radioactivity. This item was resolved with the issuance of an NRC letter containing the criteria used by the NRC staff in evaluating applications for spent fuel pool modifications. The ACR design ensures adequate cooling of the spent fuel in the pool. This will be confirmed in the DCD.
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	c	This item addressed potential methods to reduce vulnerability to sabotage. This item was resolved for US PWRs and BWRs with no new requirements established. The NRC staff concluded that existing requirements dealing with plant physical security, controlled access to vital areas, screening for reliable personnel, etc., appear to be effective. The staff believed that licensees should continue to monitor and assess security practices in terms of (1) hiring reliable personnel, and (2) surveillance procedures to prevent, detect, and mitigate adverse insider acts. The ACR design provides high-level security by (1) locating safety-related equipment in secure areas, (2) providing for separation, independence, and redundancy of safety systems, (3) providing for strict access control to vital areas, and etc. Further assessment of the physical security considerations for the ACR will be performed in the DCD to ensure that the ACR design complies with this GSI.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
A-30	Adequacy of Safety-Related DC Power Supplies	e	Superseded by 128.
A-31	RHR Shutdown Requirements (former USI)	g	This item addressed the safe shutdown of the reactor, following an accident or abnormal condition other than a LOCA, from a hot standby condition to a cold shutdown condition. Considerable emphasis has been placed on long-term cooling which is typically achieved by the residual heat removal (RHR) system which starts to operate when the reactor coolant pressure and temperature are substantially lower than the hot standby values. Even though it may generally be considered safe to maintain a reactor in a hot-standby condition for a long time, experience has shown that there have been abnormal occurrences that required long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. For this reason, the ability to transfer heat from the reactor to the environment, after a shutdown resulting from an accident or abnormal occurrence, is an important safety function. It is essential that a power plant be able to go from hot standby to cold shutdown conditions under any accident condition. This item was resolved with the issuance of SRP ([10]) Section 5.4.7. The ACR design includes a Long Term Cooling (LTC) system as a subsystem of ECCS for long-term recirculation/recovery, after LOCA. The LTC system is also used for long term cooling of the reactor after shutdown. Further assessment of the LTC system will be performed and provided in the DCD to confirm

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			the compliance of the ACR design with the intent of SRP ([10]) Section 5.4.7.
A-32	Missile Effects	e	Superseded by B-68.
A-33	NEPA Review of Accident Risks	d	NUREG-0933 ranked this item as an Environmental Issue.
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	e	Superseded by II.F.3.
A-35	Adequacy of Offsite Power Systems	g	This item addressed the susceptibility of safety-related electric equipment to offsite power source degradation. This item was resolved with the issuance of SRP ([10]) Section 8.3.1, Appendix A, Branch Technical Position BTP PSB 1, “Adequacy of Station Electric Distribution System Voltages”. The ACR design includes an on-site electric power system and provision for redundant connections to the off-site power grid to assure the functioning of structures, systems, and components important to safety. Further assessment will be performed and provided in the DCD to confirm the compliance of the ACR design with BTP PSB 1 of SRP ([10]) Section 8.3.1.
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	g	This item addressed the need to review requirements, facility designs, and technical specifications regarding the movements of heavy loads near spent fuel. This item was resolved with the publication of NUREG-0612 ([51]) and SRP ([10]) Section 9.1.5. Unlike PWRs and BWRs, the transfer of spent fuel from ACR

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			does not involve the use of a heavy load crane. On rare occasions, a heavy load (such as a shipping cask) will be carried over the spent fuel reception bay by an overhead crane. Safety of this overhead crane is ensured by designing it to meet CSA Standard B167 (Class A service) ([52]) with many additional safety features. Further assessment will be performed and provided in the DCD to ensure that the ACR design conforms to NUREG-0612 ([51]) and SRP ([10]) Section 9.1.5.
A-37	Turbine Missiles	a	NUREG-0933 dropped this item.
A-38	Tornado Missiles	a	NUREG-0933 dropped this item.
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	b	This GE-specific item addressed the operation of BWR primary system pressure relief valves that can result in hydrodynamic loads on the suppression pool retaining structures or those structures located within the pool. These loads result from initial vent clearing of relief valve piping and steam quenching due to high local pool temperature. This item was resolved with the issuance of SRP ([10]) Section 6.2.1.1.C and a series of NUREG reports for BWRs. Due to the significant difference between ACR and BWR in the operation of pressure relief valves, this BWR-specific item is not applicable to the ACR design.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
A-40	Seismic Design Criteria -Short Term Program (former USI)	g	This item addressed the need to identify and quantify conservatism in the seismic design process. The SRP ([10]) Sections 2.5.4, 3.7.1, 3.7.2, and 3.7.3 provide clarification of development of site-specific spectra, justification for use of single synthetic time-history by power spectral density function, location and reductions of input ground motion for soil-structure interaction, and design of above-ground vertical tanks. The ACR design applies well established seismic evaluation methodology and enveloping seismic design criteria that complies with Canadian standards. Further assessment will be performed and provided in the DCD to ensure the compliance of the ACR design with the relevant sections of the SRP ([10]).
A-41	Long Term Seismic Program	d	This item called for the NRC staff to initiate a long-term research program to quantify the inherent seismic safety margins in NRR’s seismic design requirements. However, a reevaluation of this item revealed that the programs covered by this item were intended to gather and develop information; there were no plans to revise regulatory requirements upon completion of the programs. It was determined that the programs were long range on-going activities and were being adequately tracked. Therefore, this item is ranked as an NRC internal item resolved with no new requirements established.
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	b	This item addressed the pipe cracking occurred in the heat-affected zones of welds in primary system piping in BWRs. These cracks have occurred mainly in type 304 stainless steel which is the type used in most operating BWRs.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			The ACR design does not use type 304 stainless steel in its HTS. Thus, this item is not relevant to the ACR design.
A-43	Containment Emergency Sump Performance (former USI)	g	This item dealt with a concern for the availability of adequate recirculation cooling water following a LOCA when long-term recirculation of cooling water from the PWR containment sump, or the BWR RHR system suction intake, must be initiated and maintained to prevent core-melt. The specific concerns are: a) Containment sump hydraulic performance under post-LOCA adverse conditions resulting from potential vortex formation, air ingestion and subsequent failure of the recirculation pumps. b) The possible transport of large quantities of LOCA-generated insulation debris to the sump screen(s) and the potential for blockage, reducing net positive suction head (NPSH) margin below that required for the recirculation pumps to maintain long-term cooling. c) The capability of RHR and containment spray system pumps to continue pumping when subjected to possible air, debris, or other effects such as particulate ingestion on pump seals and bearing systems. To resolve this item, the SRP ([10]) Section 6.2.2 and Regulatory Guide 1.82 were revised to reflect the NRC staff's technical findings. The safety functions of the Long Term Cooling (LTC) system of the ACR are to provide fuel cooling in the long-term (recovery stage) of a LOCA and to remove decay heat indefinitely in the

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			long-term of transients and accidents with the HTS pressure boundary intact. For a LOCA, the LTC system is initiated following operation of the Emergency Coolant Injection (ECI) system. On a LOCA signal, water is automatically introduced into the containment sumps and the LTC pumps start automatically. When the water tanks are nearly empty, the ECI tank isolation valves close and the recovery stage begins by pumping water from the sumps into the HTS via the LTC heat exchangers and thus the LTC system is initiated. The compliance of the ACR design with the SRP ([10]) Section 6.2.2 and Regulatory Guide 1.82 ([53]) will be confirmed in the DCD.
A-44	Station Blackout (former US I)	g	This item addressed the concern that the complete loss of all alternating current (AC) electrical power to the essential and non-essential switchgear buses in a nuclear power plant could lead to a severe core damage accident. 10 CFR 50.2 ([54]), "Definitions", defines station blackout as the loss of the offsite electric power system concurrent with a turbine trip and unavailability of the onsite emergency AC power system. Since many systems required for core decay heat removal and containment heat removal depend on AC power, a station blackout can result in unacceptable consequences unless AC power is restored in a timely manner or AC power is supplied from an alternate source. The issue involves the likelihood and duration of station blackout and the potential for core damage as a result. The issue of station blackout arose because of the historical experience regarding the reliability of AC power supplies. There had been numerous reports of emergency diesel

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			generators failing to start and run in operating plants. In addition, a number of operating plants experienced a total loss of offsite electrical power. After performing an evaluation of station blackout accidents at nuclear power plants and a Regulatory Impact Analysis, a new rule, 10 CFR 50.63 ([55]), and Regulatory Guide 1.155 ([56]), "Station Blackout" were published. Specifically, 10 CFR 50.63 ([55]) provides requirements that each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in 10 CFR 50.2 ([54]). An assessment will be performed and provided in the DCD to demonstrate that the ACR design complies with 10 CFR 50.63 ([55]) and RG 1.155 ([56]).
A-45	Shutdown Decay Heat Removal Requirements (former USI)	c	This item addressed the Decay Heat Removal (DHR) function after a plant shutdown after normal operation or due to abnormal events or Loss-of-Coolant Accidents (LOCAs), and to prepare the plant for cold shutdown conditions. NRC performed DHR PRA study on six operating plants in 1987. This DHR PRA study concentrated on the DHR systems and their contribution to core melt frequencies. This study assessed the consequences of both internal and external initiators. The study found that DHR-related core damage risk is in a range between $7x10^{-5}$ and $4x10^{-4}$ per reactor year with an average value of $2x10^{-4}$. After the DHR PRA study was conducted, the NRC staff selected a goal that core damage due to failure of the DHR function should be less than $1x10^{-5}$ per

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			reactor year. This goal shall be demonstrated by a Level 1 Shutdown Cooling System (SCS) PRA. This item was resolved with the requirement for plant-specific analyses to be conducted under the individual plant examination program. A level 1 PRA will be conducted of the ACR design including an assessment of the core damage frequency due to failure of the DHR function. The compliance of the ACR design with this GSI will be confirmed in the DCD.
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	g	During the course of the licensing of commercial nuclear power plants, the NRC found that the margins of safety provided in equipment to resist seismically-induced loads and to perform their intended safety functions could vary considerably. Therefore, this item was established to reassess the seismic qualification of equipment in operating plants to ensure the ability of plants to achieve a safe shutdown condition when subject to a seismic event. An assessment will be performed and provided in the DCD to show the compliance of the ACR design with this GSI.
A-47	Safety Implications of Control Systems (former USI)	g	This item was to perform an in-depth review of the non-safety-related control systems and to assess the effect of control system failures on plant safety. This item called for the identification of potential control system failures that, either singly or in selected combinations, could cause overpressure, overcooling, overheating, overfilling, or reactivity events. The NRC staff recommended that plants: (1) provide systems to

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			protect against reactor vessel/steam generator overfill events and to prevent steam generator dryout; (2) include in their plant procedures and technical specifications provisions to periodically verify the operability of these systems; and (3) modify selected emergency procedures to ensure safe plant shutdown following a small-break LOCA. This item was resolved with the issuance of a generic letter 89-19 ([57]) that provided guidance for licensees. In the safety analysis for the ACR design, the control system failures are bounded by design basis accidents. SBLOCA has also been adequately addressed. This will be confirmed in the DCD.
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	e	NUREG-0933 stated that the major concern of this item was covered by issue 121.
A-49	Pressurized Thermal Shock (former USI)	g	This item addressed reactor vessel integrity under conditions of pressurized thermal shock (PTS). Reactor vessel integrity could be threatened by a combination of neutron embrittlement, overcooling, excessive reactor vessel pressure, and a small flaw or crack. As plants accumulate service time, neutron irradiation reduces the material toughness of the reactor vessel. Decreased fracture toughness makes it more likely that a crack already present in the vessel inner wall could grow to a size that might threaten vessel integrity, should a combination of vessel overcooling and overpressure occur.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Due to the facts that: (1) the ACR design does not include a thick-walled pressure vessel; and (2) the pressure tubes of the ACR design are intended to be replaced after about 30 years, which can avoid severe neutron embrittlement, the pressurized thermal shock is unlikely a safety concern for the ACR design. An assessment of the generic applicability of this GSI will be performed and provided in the DCD.
B-1	Environmental Technical Specifications	d	NUREG-0933 ranked this item as an Environmental Issue.
B-2	Forecasting Electricity Demand	d	NUREG-0933 ranked this item as an Environmental Issue.
B-3	Event Categorization	d	NUREG-0933 ranked this item as a Licensing Issue.
B-4	ECCS Reliability	e	Superseded by II.E.3.2.
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	c	This item has two parts. Part I - Ductility of Two-Way Slabs and Shells This part identified the concern over the lack of information on the behaviour of two-way reinforced concrete slabs loaded dynamically in biaxial membrane tension, flexure, and shear. The NRC staff concluded that there is sufficient information pertaining to the design of two-way slabs subjected to dynamic loads and biaxial tension to enable a reasonable accurate analysis. Part II - Buckling Behavior of Steel Containments Part II identified a concern over the lack of a uniform, well defined approach for design evaluation of steel containments. Of particular interest was potential instability of the shell during

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			dynamic loadings. The structural design of a steel containment vessel subjected to unsymmetrical dynamic pressure loadings may be governed by the instability of the shell. For this type of loading, the current design verification methods, analytical techniques, and the acceptance criteria may not be as comprehensive as they should be. The NRC staff concluded that existing steel containments had adequate margins against buckling, and that the issue of steel containment buckling had very little safety impact and was not worth pursuing further. Thus, part II was resolved for US PWRs and BWRs with no new requirements established. With respect to part I of this item, the ACR concrete containment design is based on proven technology as used in previous CANDU reactors and is in accordance with current Canadian standards. Further assessment will be performed and provided in the DCD to ensure the ACR design addresses the possible concerns as noted above. Part II of this item is not applicable to the ACR design, since the ACR design does not include a steel containment vessel.
B-6	Loads, Load Combinations, Stress Limits	e	Superseded by 119.1.
B-7	Secondary Accident Consequence Modeling	d	NUREG-0933 ranked this item as a Licensing Issue.
B-8	Locking Out of ECCS Power Operated Valves	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
B-9	Electrical Cable Penetrations of Containment	c	This item addressed a concern over the failures of low voltage penetration modules due to electrical short-circuits. Based on the thought that existing requirements contained in IEEE 317 ([58]) and Regulatory Guide 1.63 ([59]) provide adequate guidance for the design of containment electrical cable penetrations, the NRC concluded that no further action on this issue was required. The containment electrical cable penetrations of ACR have been designed to current Canadian regulatory requirements and standards. An assessment will be performed and provided in the DCD to ensure that this item remains closed for the ACR design.
B-10	Behavior of BWR Mark III Containments	b	The description of this item given in NUREG-0471 ([60]) is as follows: “This is a ACRS generic concern. Evaluation and approval is required of various aspects of the Mark III containment design which differs from the previously reviewed Mark I and Mark II designs. The task involves the completion of the staff evaluation of the Mark III containment and documentation of the method used to validate the analytical models and assumptions needed to predict the containment pressures in the event of a LOCA.” This BWR-specific item is not applicable to the ACR design.
B-11	Subcompartment Standard Problems	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
B-12	Containment Cooling Requirements (Non-LOCA)	d	This item called for the NRC staff to develop a basic understanding of the consequences of a loss of normal containment cooling including the impact, if any, on the operability of safety systems and control systems. The NRC staff concluded that further study of this issue was not warranted since the potential impact of a loss of normal containment cooling on the operability of safety systems and control systems is minimal, and this item was resolved with no new requirements established.
B-13	Marviken Test Data Evaluation	d	NUREG-0933 ranked this item as a Licensing Issue.
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	e	Superseded by A-48.
B-15	CONTEMPT Computer Code Maintenance	d	NUREG-0933 ranked this item as a Licensing Issue.
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	e	Superseded by A-18.
B-17	Criteria for Safety-Related Operator Actions	c	This item concerns current plant designs (LWRs) that rely on the operator to take action in response to certain transients. In addition, some current PWR designs require manual operations to accomplish the switchover from the injection mode to the recirculation mode following a LOCA. The required time for the ECCS realignment operations is dependent on pipe break

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			size and the operation must be accomplished before the inventory in the borated water storage tank is depleted. This item involved the development of a time criterion for safety-related operator actions including a determination of whether or not automatic actuation will be required. The evaluation of this issue included consideration of Item 27. Development and implementation of criteria for safety-related operator actions would result in the automation of some actions that were being performed by operators. The use of automated redundant safety-grade controls in lieu of operator actions is expected to reduce the frequency of improper action during the response to or recovery from transients and accidents by removing the potential for operator error. This in turn could reduce the expected frequency of core damaging events and, therefore, reduce the public risk accordingly. Plants would be required to perform task analysis, simulator studies, and analysis and evaluation of operational data to assess current ESF and safety-related control system designs for conformance to new criteria. Where non-conformance is identified, modification to existing designs and hardware would be required. For plants at the CP stage of review, changes and additions to the ESF control systems are anticipated but replacement equipment costs are not anticipated. Each of the safety systems, SDS1, SDS2, ECCS and Containment of ACR incorporates the instrumentation, logic and hardware to measure specific safety parameters and automatically initiate safety system action in response to an event. The adequacy of the parameters selected and their

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			setpoints is confirmed by safety analysis. In Canadian practice, no operator action is credited in the safety analysis until a defined period following receipt of a clear and unambiguous signal in the control room, which would allow the operator to diagnose the event. This will be confirmed in the DCD.
B-18	Vortex Suppression Requirements for Containment Sumps	e	Superseded by A-43.
B-19	Thermal-Hydraulic Stability	b	This item addressed the concerns on the thermal-hydraulic instability in a BWR that had been investigated by GE. Hydrodynamic flow instabilities may occur in a BWR when two-phase flow exists in a channel with critical dimensions and particular flow parameters. BWR plants would be required to revise their technical specifications that will restrict operation of the reactor in regions of potential thermal-hydraulic instability and/or provide for surveillance and corrective measures under conditions of marginal stability. This BWR-specific item is not applicable to the ACR design.
B-20	Standard Problem Analysis	d	NUREG-0933 ranked this item as a Licensing Issue.
B-21	Core Physics	d	NUREG-0933 ranked this item as a Licensing Issue.
B-22	LWR Fuel	a	NUREG-0933 dropped this item.
B-23	LMFBR Fuel	d	NUREG-0933 ranked this item as a Licensing Issue.
B-24	Seismic Qualification of Electrical and Mechanical Components	e	Superseded by A-46.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
B-25	Piping Benchmark Problems	d	NUREG-0933 ranked this item as a Licensing Issue.
B-26	Structural Integrity of Containment Penetrations	d	This item called for the NRC staff to review the containment penetrations and to determine whether or not the configuration and accessibility of the welds in the proposed design and the procedures proposed for performing volumetric examination permit inservice examination requirements of Section XI of the ASME Code at an augmented frequency in break exclusion regions, as required by SRP ([10]) Section 3.6.2. The specific containment penetrations involved in this item include only the high-energy fluid systems. However, after reevaluation of this item, the NRC staff believed that the increase in occupational radiation exposure from additional inspections would negate the small potential risk reduction associated with this item. Thus, this item was resolved with no new requirements established.
B-27	Implementation and Use of Subsection NF	d	NUREG-0933 ranked this item as a Licensing Issue.
B-28	Radionuclide/Sediment Transport Program	d	NUREG-0933 ranked this item as an Environmental Issue.
B-29	Effectiveness of Ultimate Heat Sinks	d	NUREG-0933 ranked this item as a Licensing Issue.
B-30	Design Basis Floods and Probability	d	NUREG-0933 ranked this item as a Licensing Issue.
B-31	Dam Failure Model	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
B-32	Ice Effects on Safety-Related Water Supplies	e	Superseded by 153.
B-33	Dose Assessment Methodology	d	NUREG-0933 ranked this item as a Licensing Issue.
B-34	Occupational Radiation Exposure Reduction	e	Superseded by III.D.3.1.
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors	d	NUREG-0933 ranked this item as a Licensing Issue.
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	g	This item involved developing revisions to current guidance and staff technical positions regarding ESF and normal ventilation system air filtration and adsorption units. This involved developing revisions to Regulatory Guide 1.52 ([61]) and BTP ETSB 11.2 to address technical advances that have shown that some current positions are unjustifiably conservative, some are unnecessary, and in some cases additional positions are necessary. The ACR ventilation systems are designed in accordance with current Canadian regulatory requirements and standards. An assessment will be performed and provided in the DCD for the ACR ventilation systems with respect to NRC regulatory documents noted above.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
B-37	Chemical Discharges to Receiving Waters	d	NUREG-0933 ranked this item as an Environmental Issue.
B-38	Reconnaissance Level Investigations	d	NUREG-0933 ranked this item as an Environmental Issue.
B-39	Transmission Lines	d	NUREG-0933 ranked this item as an Environmental Issue.
B-40	Effects of Power Plant Entrainment on Plankton	d	NUREG-0933 ranked this item as an Environmental Issue.
B-41	Impacts on Fisheries	d	NUREG-0933 ranked this item as an Environmental Issue.
B-42	Socioeconomic Environmental Impacts	d	NUREG-0933 ranked this item as an Environmental Issue.
B-43	Value of Aerial Photographs for Site Evaluation	d	NUREG-0933 ranked this item as an Environmental Issue.
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	d	NUREG-0933 ranked this item as an Environmental Issue.
B-45	Need for Power -Energy Conservation	d	NUREG-0933 ranked this item as an Environmental Issue.
B-46	Cost of Alternatives in Environmental Design	d	NUREG-0933 ranked this item as an Environmental Issue.
B-47	Inservice Inspection of Supports -Classes 1,2,3, and MC Components	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
B-48	BWR CRD Mechanical Failure	b	This item concerns the cracks discovered in control rod drive internal parts at some operating plants. This BWR-specific item does not need to be addressed by the ACR design.
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	d	NUREG-0933 ranked this item as a Licensing Issue.
B-50	Post-Operating Basis Earthquake Inspections	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	e	Superseded by A-40.
B-52	Fuel Assembly Seismic and LOCA Responses	e	Superseded by A-2.
B-53	Load Break Switch	g	NUREG-0933 ranked this item as a Regulatory Impact Issue. However, it did result in new requirements to resolving this item. This item was resolved with the issuance of Appendix A to SRP ([10]) Section 8.2. This item addressed the use of the generator load break switch for isolating the generator from the stepup transformer following turbine trip. The ACR design includes generator load break switches to ensure a reliable source of AC power to the electrical system. The acceptable reliability of the generator load break switches of

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			the ACR design will be confirmed in the DCD.
B-54	Ice Condenser Containments	b	This item addressed two concerns regarding the ice condenser containment design. The first concern is the lack of computer code with ice condenser capability used for independent confirmatory calculation by the NRC staff. The second concern is the ice loss experienced in some plants. The ACR design does not have an ice condenser containment. Thus, this item is not applicable to the ACR design.
B-55	Improved Reliability of Target Rock Safety Relief Valves	b	This item addressed the concerns on the malfunction of pressure relief system valves in BWR. The BWR pressure relief system is designed to prevent overpressurization of the reactor coolant pressure boundary (RCPB) under the most severe abnormal operational transient: closure of the main steam line isolation valves (MSIVs) with failure of the MSIV position switches to scram the reactor. This design function is accomplished through the use of a plant-unique combination of safety valves (SVs), power actuated relief valves (PARVs), and dual function safety/relief valves (SRVs). In addition to the RCPB overpressure protection design functions of the BWR pressure relief system, a specified number of the PARVs or SRVs utilized in the pressure relief system of each BWR plant are used in the automatic depressurization system (ADS), which is one of the ECCSs. Certain safety concerns result when: (1) a valve fails to open properly on demand; (2) a valve opens spuriously and then fails to properly reseat; and (3) a valve opens properly but fails to

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			properly reseat. In resolving the issue, the NRC staff found that licensees had significantly improved the performance of Target Rock SRVs and continued to evaluate and improve their performance. Licensee compliance with existing regulations was sufficient for the NRC staff to pursue additional improvements on a plant-specific basis. No new requirements were established. This BWR-specific item is not applicable to the ACR design.
B-56	Diesel Reliability	g	This item addresses emergency diesel generator reliability. As discussed in NUREG-0933, a review of the LERs conducted by the NRC revealed that a diesel generator’s starting reliability is, on the average, about 0.94 per demand. Thus, the NRC determined that there was a need to upgrade the reliability of emergency diesel generators. A new reliability of between 0.95 and 0.975 per demand was established. The issue was resolved by the inclusion of guidance on EDG reliability in Regulatory Guide 1.160 ([62]) which was issued as part of the Maintenance Rule (10 CFR 50.65 ([63])). This guide endorsed NUMARC 93-01, “Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plant,” which addressed the optimisation of EDG reliability and availability and contained an example of an acceptable means of establishing performance criteria and/or goals for EDGs. In addition, Regulatory Guide 1.9 ([64]), Rev. 3 was issued to integrate into a single document pertinent guidance previously addressed in a series of documents. Thus, this item was resolved with new requirements established.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			There are two onsite standby diesel-generator sets for each unit of the two-unit ACR plant, with all four being available to supply the safety functions for each unit. An assessment will be performed and provided in the DCD to show that the ACR design complies with Regulatory Guide 1.9 ([64]), Rev. 3.
B-57	Station Blackout	e	Superseded by A-44.
B-58	Passive Mechanical Failures	d	This item was to investigate passive valve failures and the operation of a valve maintenance program, including the replacement of valves as necessary over the life of the plant. The distinction between active and passive failures is that active failures typically occur during valve operation while passive failures occur over a period of time, going unnoticed as the valve is rendered inoperable. In pursuing the resolution of this item, the NRC was expecting some recommendations for valve maintenance, repair or replacement that would result from its Nuclear Plant Aging Research Program. No new requirements were incorporated in NRC's regulatory documents.
B-59	(N-1) Loop Operation in BWRs and PWRs	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
B-60	Loose Parts Monitoring System	c	This item was to resolve any outstanding issues related to implementation of the Regulatory Guide on loose part detection in the primary system, including the development of NRC staff positions and guidance with respect to upgrading loose parts detection systems at operating facilities.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			This item was resolved with the reiteration that license applicants should meet the provisions of the existing Regulatory Guide 1.133 ([65]), Revision 1. ACR primary system design is significantly different from those of other LWRs. For the ACR design, the systems, e.g., control rod devices, flux detectors, and internal supports that might produce loose parts are not in the primary coolant system. Loose parts are extremely unlikely to appear in the ACR primary system (i.e. the HTS). An assessment of the generic applicability of this GSI will be performed and provided in the DCD.
B-61	Allowable ECCS Equipment Outage Periods	c	This item addressed surveillance test intervals and allowable equipment outage periods in the technical specifications for safety-related ECCS equipment. The task involved the NRC development of analytically based criteria for use in confirming or modifying the surveillance intervals and allowable equipment outage periods. In resolving this item, the NRC staff concluded that all aspects of the issue, other than the possible need for a limit on cumulative outage time, were addressed by the Technical Specification Improvement Program (TSIP) and the risk-informed Technical Specification (TS) guidance in Regulatory Guide 1.177 ([66]), cumulative outage time was addressed by the Maintenance Rule in 10 CFR 50.65 ([63]). No new requirements for US PWRs and BWRs were established. The ECCS of ACR has been designed to the CNSC's regulatory requirements and Canadian standards. An assessment will be

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			performed and provided in the DCD to ensure the ACR design addresses the safety concerns of this GSI and complies with the relevant US standards.
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	d	NUREG-0933 ranked this item as a Licensing Issue.
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	g	This item addressed the adequacy of the isolation of low-pressure systems that are connected to the reactor coolant pressure boundary. It was required that valves forming the interface between high- and low-pressure systems associated with the reactor coolant boundary had sufficient redundancy to prevent the low-pressure systems from being subjected to pressures that exceed their design limits. Because of the importance of the HP to LP interface for safety-related systems, in resolving the issue, the NRC reviewed and updated SRP ([10]) Section 3.9.6 by issuing Revision 2. This SRP ([10]) Section endorsed the ASME B&PV Code, Section XI (for the in-service testing of the boundary valves). The ACR design includes interconnections between high- and low-pressure systems, such as the long-term cooling system. Valves have been used as isolation in these system interfaces. An assessment of these interfaces with respect to SRP Section 3.9.6 will be performed and provided in the DCD.
B-64	Decommissioning of Reactors	f	This item was to address the development of some guidance on decommissioning of reactors. 10 CFR 50.82 ([67]) provides the regulations that govern the termination of licenses. The NRC

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			may require information from a licensee to demonstrate that the methods and procedures to be used for decommissioning will not adversely affect public health and safety. 10 CFR 50.33(f) ([68]) requires that OL applicants show that they posses (or have reasonable assurance of obtaining) funds necessary to cover the estimated costs of permanently shutting down their plants and maintaining them in a safe condition. This item was resolved with the issuance of the rule amendments that were to ensure that decommissioning of all licensed facilities would be accomplished in a safe and timely manner and that adequate licensee funds would be available for this purpose. This is not a design certification issue. This item is the responsibility of COL applicant who should be responsible for the funds necessary for future decommissioning.
B-65	Iodine Spiking	a	NUREG-0933 dropped this item.
B-66	Control Room Infiltration Measurements	g	This item addressed the adequacy of control room area ventilation systems and control building layout to ensure that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gases. The NRC considered this issue as being technically resolved, and criteria had been incorporated in SRP ([10]) Sections 6.4 and 9.4.1. The main control room habitability systems of ACR are designed to ensure that the operators are adequately protected against the effects of accidental releases of radioactivity in the outside air. Further assessment will be performed and provided

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			in the DCD to ensure that the ACR design complies with the SRP ([10]) Sections noted above.
B-67	Effluent and Process Monitoring Instrumentation	e	Superseded by III.D.2.1.
B-68	Pump Overspeed During LOCA	a	NUREG-0933 dropped this item.
B-69	ECCS Leakage Ex-Containment	e	Superseded by III.D.1.1(1).
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	e	Offsite power system frequency decay, depending on the rate of decay, could provide an electrical brake on the reactor coolant pump motors that could slow the pumps faster than the assumed flywheel coastdown normally used in analyzing loss of flow accidents. Item A-35 was to determine a maximum credible frequency decay rate to be used in this task. This item was to determine if any additional measures are necessary to protect against a frequency decay event. No new requirements were issued in resolving this item. Moreover, BTP ICSB-15, which addressed this item was removed from Revision 2 to SRP ([10]) Appendix 8A. It was concluded that this item was partially superseded by Item A-35 and the rest of the item required no further action.
B-71	Incident Response	e	Superseded by III.A.3.1.
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	e	Superseded by C-12.
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	g	This item addressed the concerns regarding the long-term capability of hermetically sealed instruments and equipment which must function in post-accident environments. When safety-related components within containment must function during post-LOCA accident conditions, their operability is sensitive to the ingress of steam or water. If the seals should become defective as a result of personnel errors in the maintenance of such equipment, such errors could lead to the loss of effective seals and the resultant loss of equipment operability. The establishment of a basis for confidence that sensitive equipment has a seal during the lifetime of the plant is needed. The NRC undertook a program to reevaluate the qualification of all safety-related electrical equipment at all operating reactors. As part of this program, more definitive criteria for environmental qualification of safety-related electrical equipment were developed by the staff. The Division of Operating Reactors' "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines) were completed in November 1979. The Guidelines were intended as a screening device to catch those pieces of equipment that might have qualification problems. In addition, for reactors under licensing review, the staff issued NUREG-0588 ([42]). The staff intended to evaluate the qualification of all electrical safety equipment in operating plants pursuant to the Guidelines. If problems arise, the staff

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			shall resolve them using NUREG-0588 ([42]) as a guide for their judgement. For ACR, the safety related equipment, including instrumentation and electrical equipment, are environmentally qualified in accordance with an established Environmental Qualification program. Further assessment will be performed and provided in the DCD to demonstrate the equipment operability in a post-accident environment.
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	b	This item addressed the need to design PWR containment vessels to withstand the external pressure experienced after inadvertent Containment Spray System (CSS) operation. Inadvertent operation of the CSS may result in a rapid depressurization of the containment vessel that could cause an excessive negative pressure differential across the pressure boundary. Some PWRs were provided with pressure limiting devices such as vacuum breakers as an alternative to designing the containment to withstand the differential pressure. These features were necessary because the relatively short containment depressurization transient could threaten containment integrity. The NRC staff confirmed that the analyses of containment depressurization due to inadvertent Containment Spray System operation would be reviewed in accordance with existing SRP ([10]) Section 6.2.1.1. Traditional CANDU reactors used to include a “Dousing System” that has similar function to the PWR Containment Spray System. The ACR design has removed the “Dousing System”, so a large negative pressure condition due to rapid

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			cooling will not occur. Thus, this item is not applicable to the ACR design.
C-3	Insulation Usage Within Containment	e	Superseded by A-43.
C-4	Statistical Methods for ECCS Analysis	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
C-5	Decay Heat Update	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
C-6	LOCA Heat Sources	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
C-7	PWR System Piping	c	This item addressed cracking of thin-walled piping in PWRs. A combination of fabrication, stress, and environmental conditions had resulted in isolated instances of stress corrosion cracking of low pressure Schedule 10 type 304 stainless steel piping systems. The affected piping was found to have been sensitized and, therefore, particularly vulnerable to corrosion attack. The existing licensing criteria precluded the use of sensitized piping in safety-related piping systems and placed emphasis on the use of corrosion-resistant material in such systems. This item was established to determine if augmented In-Service Inspection (ISI) requirements should be established to further enhance the reliability of such piping systems. This item was resolved with the conclusion that the existing ISI requirements for thin-walled piping in PWRs were adequate. No new requirements for US PWRs and BWRs were established. Type 304 stainless steel is not used in ACR HTS piping systems, but is used in the moderator system based on the accepted

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			engineering practices at traditional CANDU reactors. An assessment will be performed and provided in the DCD to show how the reliability of Type 304 stainless steel piping is ensured for ACR.
C-8	Main Steam Line Leakage Control Systems	b	This item addressed the concern over the main steam isolation valve leakage control system (MSIVLCS) of some BWRs. Dose calculations by the NRC staff indicated that operation of the MSIVLCS could result in higher offsite accident doses than if the system were not used and the integrity of the steam lines and condenser was maintained. This item was resolved with no new requirements established. Since there is no MSIVLCS in the ACR design, this BWR-specific item is not applicable to the ACR design.
C-9	RHR Heat Exchanger Tube Failures	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
C-10	Effective Operation of Containment Sprays in a LOCA	b	This item addressed the concern of ACRS about the effectiveness of various containment sprays to remove airborne radioactive materials which could be present within the containment following a LOCA. The concern was expanded to include the possible damage to equipment located inside containment due to an inadvertent actuation of the sprays. One of the tasks of this item was to assist the industry in writing an ANSI standard on the design of containment spray system. This item was resolved with the issuance of ANSI/ANS 56.5-1979 that is referenced in SRP ([10]) Section 6.5.2. Since there is no containment spray system in the ACR design to remove airborne radioactive materials, this item is not applicable to ACR.
C-11	Assessment of Failure and Reliability of Pumps and Valves	c	This item called for the NRC staff to evaluate the active pumps and valves with respect to their operability and reliability under accident loading, i.e., LOCA and SSE, and to implement a corrective action program specifically directed towards improved design and fabrication of active pumps and valves. In pursuing the resolution of this item, the NRC was expecting some recommendations for valve maintenance, repair or replacement that would result from its Nuclear Plant Aging Research Program. No new requirements were incorporated in NRC's regulatory documents. The pumps and valves of the ACR design have been designed to meet current Canadian regulatory requirements and international standards. Further assessment will be performed and provided

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			in the DCD to ensure that the ACR design addresses the safety concerns identified by this item.
C-12	Primary System Vibration Assessment	c	This item addressed the potential adverse effect of vibration in the primary system. There were a number of instances where components internals to the reactor coolant pressure boundary have come loose as the result of flow-induced vibration and been carried through the primary system by the coolant flow. The NRC staff was called for to develop criteria for instrumentation for monitoring excessive vibration inside the reactor vessel. However, the NRC staff concluded that the existing guidelines in SRP ([10]) 3.9.2, combined with staff positions on loose parts monitoring (Regulatory Guide 1.133 ([65])) provided sufficient basis for resolution of this item. No new requirements for US PWRs and BWRs were issued. Unlike PWRs and BWRs, the ACR primary system does not contain component internals, such as control rod devices, that are vulnerable to vibration. However, ACR design does include some provisions to monitor the vibration of rotating machines (pumps). Further assessment will be performed and provided in the DCD to ensure that the concerns noted above are addressed for the ACR design.
C-13	Non-Random Failures	e	Superseded by A-17.
C-14	Storm Surge Model for Coastal Sites	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
C-15	NUREG Report for Liquids Tank Failure Analysis	d	NUREG-0933 ranked this item as a Licensing Issue.
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	d	NUREG-0933 ranked this item as an Environmental Issue.
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	g	This item called for the NRC staff to develop criteria for acceptability of radioactive waste solidification agents. This item was resolved with the issuance of 10 CFR 61.56 ([69]) which addresses waste characteristics. Solidification of waste is not a part of ACR design. A brief description of ACR waste management system will be provided in the DCD to ensure its compliance with 10 CFR 61.56 ([69]), as appropriate.
D-1	Advisability of a Seismic Scram	a	NUREG-0933 dropped this item.
D-2	Emergency Core Cooling System Capability for Future Plants	a	NUREG-0933 dropped this item.
D-3	Control Rod Drop Accident	d	This item called for the NRC staff to assess the uncertainties in calculations of the control rod drop accident. This item was resolved with the conclusion that no further action was required.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
New Generic Items			
1	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	a	NUREG-0933 dropped this item.
2	Failure of Protective Devices on Essential Equipment	a	NUREG-0933 dropped this item.
3	Set Point Drift in Instrumentation	c	This issue addressed drift in safety-related instrumentation and controls setpoints and the potential for a delay in initiation of a safety-related system or component. Setpoint drift can be defined as a change in the input-output relationship of an instrument over a period of time. Setpoint drift can occur as a result of a number of factors including component failure, instrumentation error and environmental conditions. Setpoint drift primarily affects analog instrumentation rather than digital instrumentation (which is less sensitive to the environmental effects of temperature, humidity, etc.). Safety related instrumentation and controls systems use setpoints as a means of determining when to initiate a safety function. Should an unplanned change in the setpoint of a safety-related component occur (i.e., setpoint drift) the actual value of the measured parameter at which a particular action is specified to occur will be altered. This phenomenon can result in the delay in the initiation of a safety function. A number of Licensee Event Reports (LERs) were reviewed by

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			the NRC which dealt with setpoint drift in safety-related instrumentation and controls. Subsequently, many of these LERs were determined to have reported setpoint drift in safety-related instruments beyond their permissible technical specification limits. Therefore, the NRC determined that it was necessary to provide industry with additional guidance, Regulatory Guide 1.105 ([70]), which could be utilized in establishing and maintaining safety-related instrument setpoints. In conjunction with the NRC work, industry developed a standard, ISA S67.04-1987 ([71]) for safety-related instrument setpoints. This revised standard replaces ISA S67.04-1982 endorsed by Regulatory Guide 1.105 ([70]), Revision 2. The instrumentation of ACR is based on proven technology and conforms to up to date standards (both Canadian and applicable international), which addresses the safety concern of this issue and will be confirmed in the DCD.
4	End-of-Life and Maintenance Criteria	f	Based on an ACRS-sponsored review of LERs, it was concluded that the criteria used for setting up equipment maintenance intervals and determining equipment life expectancy required further review and evaluation. It was believed that these factors were significant contributors to the component failures in equipment such as valves, pumps, etc. The failure of safety-related components can lead to loss of reactor coolant pressure boundary integrity or loss of safety functions. Such failures could possibly be reduced by using end-of-life data and improved periodic maintenance criteria.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			NUREG-0933 clarified that a few regulatory documents and industry standards contain specific requirements for setting up maintenance criteria. These documents include: 1) NUREG-0588 ([42]), 2) DOR Guidelines, 3) IEEE 323-1974 ([41]), 4) Regulatory Guide 1.89 ([44]), 5) Regulatory Guide 1.33 ([72]), and 6) NUREG0800 (NRC Standard Review Plan) ([10]). This issue has been resolved with no new requirements established. The ultimate responsibility of the establishment of equipment maintenance program rests with the COL applicant. Therefore, this item is not a design issue.
5	Design Check and Audit of Balance-of-Plant Equipment	e	Superseded by I.F.1.
6	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	b	BWRs are operated in such a manner as to attempt to produce an axial power distribution with as little peaking as possible. When the reactor moderator is not boiling, the axial flux distribution is much more peaked. This in turn causes incremental control rod worths to peak at a certain axial location due to the non-linear neutron importance weighting. When a reactor is restarted shortly after a shutdown, the xenon distribution tends to enhance the peaking effect.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			The question then arose concerning implications for the analysis of the Rod Drop Accident (RDA). The issue was resolved by conducting a study, and no new requirements were established. Since the reactivity control mechanism of ACR is significantly different from that of BWRs, this BWR-specific issue is not applicable to the ACR design.
7	Failures Due to Flow-Induced Vibrations	a	NUREG-0933 dropped this item.
8	Inadvertent Actuation of Safety Injection in PWRs	e	Superseded by I.C.1.
9	Reevaluation of Reactor Coolant Pump Trip Criteria	e	Superseded by II.K.3(5).
10	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	a	NUREG-0933 dropped this item.
11	Turbine Disc Cracking	e	Superseded by A-37.
12	BWR Jet Pump Integrity	b	This issue addressed the failure occurred at the jet pumps used at GE BWRs. ACR will not use such type of pumps, and as such does not need to address this issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
13	Small Break LOCA from Extended Overheating of Pressurizer Heaters	a	NUREG-0933 dropped this item.
14	PWR Pipe Cracks	c	This item addressed the pipe cracking in PWRs. In particular, piping cracks associated with secondary plant high pressure systems which could result in a diminished system functional capability and safety margin. Cracking had occurred in PWR piping systems as a result of stress corrosion, vibratory and thermal fatigue and dynamic loadings. All incidents of cracking had been detected and corrective actions taken prior to any catastrophic failures. After evaluating potential enhancements of high energy secondary system piping surveillance and inspection requirements, the NRC concluded that existing regulatory guidance was satisfactory for the design of secondary system piping in future plants. Therefore no new requirements for US PWRs and BWRs were issued in resolving this item. An assessment will be performed and provided in the DCD to ensure the ACR design addresses the above pipe cracking concerns.
15	Radiation Effects on Reactor Vessel Supports	c	This item addressed the potential for failure of the reactor vessel support structure (RVSS) due to a combination of low temperature and neutron irradiation embrittlement. The item was resolved for US PWRs and BWRs with no new requirements established. ACR design does not include a reactor vessel and the associated

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			reactor vessel support structure. However, an assessment of any safety-related supports susceptible to neutron irradiation in the ACR design will be performed and provided in the DCD to ensure the ACR design addresses the safety concerns of this GSI.
16	BWR Main Steam Isolation Valve Leakage Control Systems	e	Superseded by C-8.
17	Loss of Offsite Power Subsequent to LOCA	a	NUREG-0933 dropped this item.
18	Steam Line Break with Consequential Small LOCA	e	Superseded by I.C.1.
19	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	e	Superseded by A-47.
20	Effects of Electromagnetic Pulse on Nuclear Power Plants	c	The concern was raised because of the potential for a high-altitude nuclear weapon detonation causing a large electromagnetic pulse (EMP) which subsequently could induce large currents and voltages in electrical systems. The original concern was that sensitive electronics at a nuclear power plant would be irreparably damaged by the EMP, but the NRC later believed that the failure, if it occurred, would likely be only momentary and the failed equipment could be restored to service to continue core heat removal. This item was resolved with the publication of a NUREG report that documented the NRC staff's study on the effects of the EMP

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			on nuclear power plants. No new requirements for US PWRs and BWRs were established. The ACR design includes consideration of electromagnetic effects. This will be confirmed in the DCD.
21	Vibration Qualification of Equipment	a	NUREG-0933 dropped this item.
22	Inadvertent Boron Dilution Events	c	This item addressed the possibility of core criticality during cold shutdown conditions because of an inadvertent boron dilution event. Some inadvertent boron dilution events occurred at PWRs during maintenance and refueling periods. If the boron in the RCS is sufficiently diluted and the reactor core is near the beginning of life, there is the potential for core criticality with all rods inserted. To resolve this item, the NRC concluded that existing review criteria were adequate. No new requirements for US PWRs and BWRs were established as a part of the resolution of this item. ACR design is significantly different from PWRs in this regard. However, an assessment of any possible functional equivalents unique to the ACR design will be performed and provided in the DCD.
23	Reactor Coolant Pump Seal Failures	b	This item addressed the high rate of Reactor Coolant Pump (RCP) seal failures that challenge the makeup capacity of the ECCS in PWRs. Such an event could result in a small-break loss-of-coolant accident with possibly core damage consequences. WASH-1400 ([74]) indicated that breaks in the reactor coolant

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			pressure boundary having an equivalent diameter in the range of 0.5 to 2 inches was a significant cause of core melt. Since then, a study has shown that comparable break flow rates have resulted from RCP seal failures at a frequency about an order of magnitude greater than the pipe break frequency used in WASH-1400 ([74]). It was believed that efforts could be undertaken to implement a program to improve pump design, seal design, maintenance procedures and seal auxiliary support systems. But, in resolving the issue, the NRC elected to pursue plant-specific backfits based on the staff's plant-by-plant risk analysis of the loss of component cooling water/essential service water systems. Thus, the issue was resolved with no new requirements established. The leakages from any HTS pump seal of the ACR will not lead to core damage. The ACR pump design is based on accepted engineering practices for traditional CANDU plants. Thus, this item is not applicable to the ACR design.
24	Automatic Emergency Core Cooling System Switch to Recirculation	c	This item was raised by the staff following a review of operating events that indicated a significant number of ECCS spurious actuations, particularly, the four events that occurred at Davis-Besse during 1980. ECCS operation has two different phases in PWRs: injection and recirculation. The injection phase involves initial cooling of the reactor core and replenishment of the primary coolant following a LOCA. In this phase, the ECCS pumps typically take suction from the refueling water storage tank (RWST). The recirculation phase provides long-term cooling during the

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			accident recovery period. The ECCS is realigned in the recirculation phase to take suction from the containment sump. Switchover from injection to recirculation phase involves realignment of several valves and may be accomplished by: (1) manual operations to realign the valves; (2) fully automatic realignment of the valves; or (3) automatic realignment of some valves, followed by manual completion of the process (semi-automatic). Each option is vulnerable in varying degrees to human errors, hardware failures, and common cause failures. During a LOCA, ECCS pump suction must be switched from the RWST to the containment sump before RWST inventory is lost or loss of the ECCS pumps will occur. Switching to the sump early could adversely affect the accident because the containment sump may not have enough inventory to provide pump suction. The automatic and the semi-automatic switchovers reduce the risk of human error but have a slight increase in risk for inadvertent actuations. This issue affects PWRs only. The two possible solutions to this issue are alternate cases requiring fully-automatic or semi-automatic switchover to the containment sump. The fully-automatic switchover could be implemented by a system that would monitor the water level in the RWST and, at a preset level, automatically realign the ECCS to take suction from the containment sump. The semi-automatic switchover could be implemented by a system that would involve automatic alignment of some valves and manual

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			completion of the switchover process. The solution of this item resulted in no new requirements for US PWRs and BWRs. The ECCS of ACR design consists of two subsystems, namely Emergency Coolant Injection System and Long-Term Cooling System. Pump suction switchover is not needed. This will be confirmed in the DCD.
25	Automatic Air Header Dump on BWR Scram System	b	This item concerned the slow loss of control air pressure in the scram system of BWRs. Since the ACR control rod /shutoff rod devices do not rely on control air pressure, this item is not applicable to the ACR design.
26	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	e	Superseded by 17.
27	Manual vs. Automated Actions	e	Superseded by B-17.
28	28. Pressurized Thermal Shock	e	Superseded by A-49.
29	Bolting Degradation or Failure in Nuclear Power Plants	f	This item addressed bolting degradation within safety-related components and support structures and its impact on the integrity of the reactor coolant pressure boundary. The most crucial bolting applications are those constituting an integral part of the primary pressure boundary such as closure studs and bolts on reactor vessels, reactor coolant pumps, and steam generators. Degradation of these bolts or studs could result in the loss of reactor coolant. Other bolting applications

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			such as component support and embedment anchor bolts or studs are essential for withstanding transient loads created during abnormal or accident conditions. Review of operating experience demonstrated that the owner-operator's maintenance practices significantly affect bolting degradation. In resolving this item, the NRC issued NUREG-1339 ([75]), “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants” and NUREG-1445 ([76]), “Regulatory Analysis for the Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants”. The NRC concluded that leakage of bolted pressure joint was possible, but catastrophic RCPB joint failure that could lead to significant accident sequences was highly unlikely. A Generic Letter, No. 91-17 ([77]), was issued to licensees to implement the industry bolting integrity program, as presented in an EPRI Report. No new requirements were established. This item is only related to maintenance practice concerns over bolting, and as such is not a design issue. Since the reactivity control system of the ACR design does not use large amounts of boron, degradation of bolting in the primary system has not been a problem.
30	Potential Generator Missiles -Generator Rotor Retaining Rings	a	NUREG-0933 dropped this item.
31	Natural Circulation Cooldown	e	Superseded by I.C.1.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
32	Flow Blockage in Essential Equipment Caused by Corbicula	e	Superseded by 51.
33	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	e	Superseded by A-47.
34	RCS Leak	a	NUREG-0933 dropped this item.
35	Degradation of Internal Appurtenances in LWRs	a	NUREG-0933 dropped this item.
36	Loss of Service Water	c	This item addressed the potential for the loss of both redundant trains of service water caused by the failure of a non-safety system or component. Calvert Cliffs Unit 1 experienced a loss of both redundant trains of service water when the Station Service Water System (SSWS) became air-bound as a result of the failure of a non-safety-related instrument air compressor aftercooler. This event involved two fundamental aspects in the design of safety-related systems: (1) interaction between safety and non-safety-related systems and components, and (2) common cause failure of redundant safety systems. This item was resolved for US PWRs and BWRs with no new requirements established. The ACR service water systems consist of the Raw Service Water (RSW) system, and the Recirculated Cooling Water (RCW) system. Both the RSW and RCW systems consist of two independent divisions, Division 1 and Division 2. An

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			assessment will be performed and provided in the DCD to ensure that the safety concerns noted above are addressed for the ACR design.
37	Steam Generator Overfill and Combined Primary and Secondary Blowdown	e	Superseded by A-47, I.C.1(2).
38	Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fine Debris	a	NUREG-0933 dropped this item.
39	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	e	Superseded by 25.
40	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	b	This item addressed the safety concerns associated with pipe breaks in the BWR scram system. A break or leak in the Scram Discharge Volume (SDV) piping of a BWR during a reactor scram would result in the release of water and steam into reactor building and is postulated to result in 100% relative humidity in the reactor building. Isolating the break is dependent on the ability to reset scram, which cannot be absolutely ensured. Therefore, a rupture of the SDV could result in a unisolable break outside of primary containment, which is postulated to threaten emergency core cooling equipment by flooding areas in which this equipment is located and by causing ambient temperature and relative humidity conditions for which this equipment is not qualified.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Due to the significant difference between ACR and BWR designs, the scenario noted above will not occur in ACR. Thus, this item is not applicable to the ACR design.
41	BWR Scram Discharge Volume Systems	b	This item called for the improvement of BWR Scram Discharge Volume (SDV) system. Since there is no such system in ACR design, this item is not applicable to the ACR design.
42	Combination Primary/Secondary System LOCA	e	Superseded by I.C.1.
43	Reliability of Air Systems	g	This item addressed the concern that compressed air system degradation or malfunction may cause malfunction of safety-related systems and components. U.S. LWRs rely upon air systems to actuate or control safety-related equipment during normal operation; however, air systems are not safety grade systems at most operating plants. Safety system design criteria require (and plant accident analyses assume) that safety-related equipment dependent upon air systems will either "fail safe" upon loss of air or perform its intended function with the assistance of backup accumulators. The AEOD Case Study highlights 29 failures of safety-related systems that resulted from degraded or malfunctioning air systems. These failures contradict the requirement that safety-related equipment dependent upon air systems will either "fail safe" upon loss of air or perform their intended function with the assistance of backup accumulators. Some of the systems that may be significantly degraded or failed are decay

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			heat removal, auxiliary feedwater, BWR scram, main steam isolation, salt water cooling, emergency diesel generator, containment isolation, and the fuel pool seal system. The end result of degradation or failure of safety or safety-related systems is an increase in the expected frequency of core-melt events and, therefore, an increase in public risk. The NRC published a report, NUREG-1275 ([78]), volume 2, that indicated that the performance of the air-operated safety-related components may not be in accordance with their intended safety function because of inadequacy in the design, installation, and maintenance of the instrument air system. The report also indicated that anticipated transient and system recovery procedures were frequently inadequate and that operators were not well-trained for coping with loss of instrument air conditions. The NRC issued Generic Letter 88-14 ([79]) to request that each licensee/applicant review NUREG-1275 ([78]), Volume 2, and perform a design and operations verification of instrument air systems. In addition, all licensees/applicants were requested to provide a discussion of their program for maintaining proper instrument air quality. ACR instrument air systems that supply compressed air to safety or safety-related systems are categorized as safety related systems. Safety requirements have been imposed accordingly. Further assessment will be performed and provided in the DCD for the ACR instrument air systems and their maintenance program to ensure the safety concerns noted above are addressed for the ACR design.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
44	Failure of Saltwater Cooling System	e	Superseded by 43.
45	Inoperability of Instrumentation Due to Extreme Cold Weather	g	This item addressed the inoperability of instrumentation due to extreme cold weather. This item was raised after an incident occurred at a plant where in the heat tracing system surrounding sensing lines and level transmitters for the Refueling Water Storage Tank (RWST) failed during sub-freezing weather. The failure of the heat tracing system resulted in freezing of the sensing lines and associated level transmitters causing a loss of all four RWST instrumentation channels, which would have resulted in the failure of the Emergency Core Cooling System, thus jeopardizing plant safety. In resolving this item, a Regulatory Guide 1.151 ([73]), " Instrument Sensing Lines", was issued. The NRC also revised the following SRP ([10]) Sections to incorporate the changes associated with the resolution of this item: (1) Section 7.1, Rev.3; (2) Section 7.1, Appendix A, Rev. 1; (3) Section 7.5, Rev. 3; and (4) Section 7.7, Rev.3. The ACR design includes instrument sensing lines that are well protected against postulated environmental conditions. This will be confirmed in the DCD.
46	Loss of 125 Volt DC Bus	e	Superseded by 76.
47	Loss of Off-Site Power	e	This item was raised after a US plant experienced a significant event resulting from a loss of offsite power. The analysis and evaluation identified some design and procedural deficiencies. The NRC staff made 8 recommendations based on the

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			investigations. This item was resolved with the conclusion that the 8 recommendations were being covered by other GSIs or other NRC programs, or were resolved with no new requirements established.
48	LCO for Class 1E Vital Instrument Buses in Operating Reactors	e	Superseded by 128.
49	Interlocks and LCOs for Redundant Class 1E Tie- Breakers	e	Superseded by 128.
50	Reactor Vessel Level Instrumentation in BWRs	b	This item addressed the safety concerns associated with reactor vessel level instrumentation in BWRs. BWR uses reactor level instrumentation to perform a number of functions including control functions and protective functions. The NRC concluded that, depending on specific plant instrumentation configurations, there could be the potential for adverse interactions between the control systems and the protection systems, which may lead to loss of some protection functions. This BWR-specific item is not applicable to the ACR design.
51	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	g	This item addressed the subject of service water system fouling at operating plants primarily by aquatic bivalves. The service water system is the ultimate heat sink that, during an accident or transient, cools the reactor building component cooling water heat exchangers, with in turn cool the RHR heat exchangers as well as provide cooling for safety-related pumps and area cooling coils. Fouling of safety-related service water

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			system may lead to plant shutdown or degraded mode of operation. The solution to this item recommended by the NRC staff was the implementation of a baseline fouling program which was issued to licensees in Generic Letter, 89-13 ([80]). ACR design includes two safety-related divisions of the raw service water (RSW) systems that are once-through systems that draw water from the ultimate heat sink and use it to cool the recirculated cooling water system (RCW). The RCW in turn cools various safety related and non-safety related loads in the plant. The heat absorbed by the RSW is then transferred to the ultimate heat sink. A combination of design, materials and chemistry is used to control degradation mechanisms such as fouling and erosion. The ACR design compliance with this GSI will be confirmed in the DCD. AECL can also provide assistance to the future operator of the plant in establishment of a surveillance and preventive maintenance program.
52	SSW Flow Blockage by Blue Mussels	e	Superseded by 51.
53	Consequences of a Postulated Flow Blockage Incident in a BWR	a	NUREG-0933 dropped this item.
54	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	e	Superseded by II.E.6.1.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
55	Failure of Class e Safety-Related Switchgear Circuit Breakers to Close on Demand	a	NUREG-0933 dropped this item.
56	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	e	Superseded by A-47, I.D.1.
57	Effects of Fire Protection System Actuation	c	This item addressed the potential for safety-related equipment to become inoperable because of water spray from the fire protection system. No new requirements for US PWRs and BWRs were established as the resolutions to this item. For ACR, the impacts on safety related systems caused by the operation or actuation of the other systems (including fire protection system) are limited by the physical separation of redundant divisions. This will be confirmed in the DCD.
58	Inadvertent Containment Flooding	a	NUREG-0933 dropped this item.
59	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
60	Lamellar Tearing of Reactor Systems Structural Supports	e	Superseded by A-12.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
61	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	b	This item postulated a break in the SRV discharge line in the wetwell airspace above the suppression pool of Mark I and II plants. Coupled with the line break was a failure of the relief valve to close after its actuation in response to the transient. The relief valve must be postulated to remain open for a significant amount of steam to escape, bypass the pool, and threaten overpressurization of the containment vessel with rupture in approximately ten minutes. This scenario is BWR-specific and is not applicable to the ACR design.
62	Reactor Systems Bolting Applications	e	Superseded by 29.
63	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	a	NUREG-0933 dropped this item.
64	Identification of Protection System Instrument Sensing Lines	c	This item called for the establishment of guidance for the identification of the mechanical sensing lines which feed the protection systems. Sensing lines are an integral part of the protection systems, and are essential to their reliable operation. Therefore, identification of these lines would facilitate verification that these lines are appropriately separated and protected from external hazards. Industry developed a standard for safety-related instrument sensing lines, ISA-S67.02 ([71]), which includes identification criteria. The NRC endorsed ISA-S67.02 ([71]) in Regulatory Guide 1.151 ([73]).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			This item was resolved and resulted in no change in requirements for US PWRs and BWRs. An assessment will be performed and provided in the DCD to show that the sensing lines of the ACR safety systems are appropriately separated and protected.
65	Probability of Core-Melt Due to Component Cooling Water System Failures	e	Superseded by 23.
66	Steam Generator Requirements	e	This item addressed the potential for and the safety implications of steam generator tube ruptures (SGTR) in PWRs. Unplanned radioactive effluent releases to the environment and loss of primary coolant inventory as a result of a SGTR were also addressed. PWR steam generator tube degradation can lead to leaks and/or ruptures. Various forms of steam generator tube degradation have been identified, including: stress corrosion cracking, wastage, intergranular attack, denting, erosion-corrosion, corrosion cracking, pitting, fretting, support plate degradation, and mechanical wear. This item was divided into 16 sub-items that were determined to be dropped, covered by Item 67, or Licensing Issues. NUREG-0933 also stated that this item was based on possible steam generator requirements that were to evolve from the resolution of Items A-3, A-4, and A-5, and that no new requirements were necessary.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
67.2.1	Integrity of Steam Generator Tube Sleeves	e	Superseded by 135.
67.3.1	Steam Generator Overfill	e	Superseded by A-47.
67.3.2	Pressurized Thermal Shock	e	Superseded by I.C.1.
67.3.3	Improved Accident Monitoring	g	This item called for the staff to address the accident monitoring weaknesses of the type observed at Ginna by implementation of Regulatory Guide 1.97 ([16]) and the Safety Parameter Display System (SPDS). During the event at Ginna, several weaknesses in accident monitoring were apparent. These include: (1) non-redundant monitoring of RCS pressure; (2) failure of the position indication for the steam generator relief and safety valves; and (3) the limited range of the charging pump flow indicator for monitoring charging flow during accidents. These conditions make it more difficult for correct operator action in response to such events. Had Regulatory Guide 1.97 ([16]) been implemented at Ginna before the January 1982 event, the monitoring of the event would have been substantially improved and there would have been more assurance of correct operator actions. Improved accident monitoring would also have improved the NRC's ability to assess the plant status and the appropriateness of the licensee's actions and recommendations. This item was resolved with the issuance of Generic Letter, 82-33 ([7]).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			The parameters to monitor in ACR are determined in accordance with current Canadian regulatory requirements and standards, and based on accepted engineering practices. These require safety systems to have their own dedicated monitoring indications, resulting in redundant accident monitoring instrumentation. The ACR's compliance with Regulatory Guide 1.97 ([16]) will be confirmed in the DCD.
67.3.4	Reactor Vessel Inventory Measurements	e	Superseded by II.F.2.
67.4.1	RCP Trip	e	Superseded by II.K.3(5).
67.4.2	Control Room Design Review	e	Superseded by I.D.1.
67.4.3	Emergency Operating Procedures	e	Superseded by I.C.1.
67.5.1	Reassessment of SGTR Design Basis	d	NUREG-0933 ranked this item as a Licensing Issue.
67.5.2	Reevaluation of SGTR Design Basis	d	NUREG-0933 ranked this item as a Licensing Issue.
67.5.3	Secondary System Isolation	a	NUREG-0933 dropped this item.
67.6.0	Organizational Responses	e	Superseded by III.A.3.
67.7.0	Improved Eddy Current Tests	e	Superseded by 135.
67.8.0	Denting Criteria	e	Superseded by 135.
67.9.0	Reactor Coolant System Pressure Control	e	Superseded by A-45, I.C.1(2,3).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
67.10.0	Supplement Tube Inspections	d	NUREG-0933 ranked this item as a Licensing Issue.
68	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pumps Steam Supply Line Rupture	e	Superseded by 124.
69	Make-up Nozzle Cracking in B&W Plants	b	Cracks were found in the normal make-up high pressure injection (MU/HPI) nozzles of several B&W plants. As a result, a B&W Owners' Group Task Force was established to identify the cause of the failures and recommend modifications to eliminate future failures. These cracks appeared to be directly related to loose or missing thermal sleeves. The incorporation of a thermal sleeve into a nozzle assembly is a common practice in the nuclear industry to provide a thermal barrier between the cold MU/HPI fluid and the hot high pressure injection nozzle. Based on investigation, B&W made four recommendations as the resolutions to the problem. NRC staff was in favor of these recommendations Due to the significant differences between B&W plants and ACR designs, the root cause of the failure does not exist in ACR. The piping of ACR ECCS is significantly different from that of B&W's MU/HPI system. Thus, this item is not applicable to the ACR design.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
70	PORV and Block Valve Reliability	g	PORVs and block valves were originally designed as non-safety components in the reactor pressure control system for use only when plants are in operation. The block valves were installed because of expected leakage from the PORVs. Neither the PORVs nor the block valves were required to safely shut down a plant or mitigate the consequences of accidents. In 1983, the NRC Reactor Systems Branch determined that PORVs are relied upon to mitigate a design basis SGTR. However, the acceptability of relying on non-safety grade PORVs to mitigate a design basis accident (e.g., an SGTR) was questioned by the RSB. This issue was identified to improve the reliability of PORVs and block valves. NUREG-0737 ([6]), Item II.D.1, set forth functional requirements for both PORVs and block valves. All plants were required to demonstrate the functionability of these valves for all expected flow conditions during operating and accident conditions. It was further required that the block valves be capable of closing to ensure that a stuck-open relief valve can be isolated, thereby terminating a small LOCA. In response to the Item II.D.1 requirements, PORVs were tested extensively by EPRI and the results reported to the staff. Only limited block valve testing had been performed in the EPRI program at the time this issue was evaluated. Reports describing the test program results were submitted to the staff for review. Most plants requested exemptions to the specified completion date for Item II.D.1 in order to obtain additional time for the required evaluation of piping associated with safety valves, PORVs, and block valves.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Resolution of this issue could involve specification of the PORV/block-valve combination to some or all of the requirements associated with safety grade systems, better initial qualifications for the valves, and specified maintenance and testing requirements. In resolving the item, the NRC issued Generic Letter 90-06 ([81]) which required TS revisions at PWRs with PORVs and block valves. The SRP ([10]) Sections 3.2.2, 5.2.2, and 5.4.7 were also revised. ACR design includes the use of PORVs for overpressure protection of the HTS. But the ACR does not use blocking valves. Further assessment will be performed and provided in the DCD to ensure that ACR complies with the requirements of this GSI.
71	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	a	NUREG-0933 dropped this item.
72	Control Rod Drive Guide Tube Support Pin Failures	a	NUREG-0933 dropped this item.
73	Detached Thermal Sleeves	b	This item addressed the problems with “Generation 3” thermal sleeves at PWRs designed by Westinghouse. Since ACR design does not use similar thermal sleeves, this item is deemed plant-specific, and therefore is not applicable to the ACR design.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
74	Reactor Coolant Activity Limits for Operating Reactors	a	NUREG-0933 dropped this item.
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	g	This item resulted after Salem-1, on two occasions in February 1983, failed to scram automatically due to failure of both reactor trip breakers to open on receipt of an actuation signal. In both cases, the unit was successfully tripped by manual action. The failure of the breakers was attributed to excessive wear from improper maintenance of the undervoltage (UV) relays that receive the trip signal from the protection system and result in the breakers opening mechanically. Three separate NRC actions were initiated to address this problem. One was plant-specific and was addressed before the restart of Salem 1; this was completed in the "Salem Restart Evaluation." The second action was an investigation into the Salem events and the circumstances leading to them; this was reported in NUREG-0977 ([82]). The third action was the formation of an NRC task force to study the overall generic implications of this event; the results of this study were reported in NUREG-1000 ([83]). In addition, a number of issues raised by the staff were closely related to the design and testing of the reactor protection system. Failure to scram (also commonly referred to as ATWS) could rupture the RCS or distort ECCS valves such that a core-melt would result. In resolving this item, the NRC required 4 licensee actions and 12 NRC staff actions. All these had been addressed.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			See Item A-9 for more information. ACR will address Item 75 by addressing Item A-9.
76	Instrumentation and Control Power Interactions	a	NUREG-0933 dropped this item.
77	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	e	Superseded by A-17.
78	Monitoring of Fatigue Transient Limits for Reactor Coolant System	f	This item concerned fatigue limits resulting from transients that could affect the expected useful life of the reactor coolant system pressure boundary. Technical Specifications (TS) for newer OLs required licensees to keep account of the number of transient occurrences to ensure that transient limits, based on design assumptions, are not exceeded. However, a number of older plants for which detailed fatigue analyses were performed on ASME Section III, Code Class 1 (RCPB) components did not have TS requirements for monitoring actual transient occurrences. These transients could significantly affect the fatigue life of the RCS. Repeated thermal cycling of RCS components produces some degree of fatigue degradation of the materials, which could lead to failure, thereby increasing the likelihood of a LOCA. A possible solution was to require the affected plants to implement TS to monitor plant transients and to verify that the design life of all ASME Section III, Code Class 1 components have not been exceeded. Fatigue monitoring should be performed by the owner of a plant.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Thus, this item is the responsibility of the prospective COL applicant.
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Circulation Cooldown	c	This item addressed the thermal stresses that occurred in the reactor vessel head flange area or studs during a natural circulation cooldown. The safety concern was that the thermal stresses during a natural circulation cooldown in the flange area or in the studs could exceed the ASME allowables when added to the stresses already considered. Moreover, the cycling of these temperature gradients over the life of the plant had the potential to cause a reduction in the fatigue margin of the vessel. Based on evaluations, the NRC staff concluded that the affected plants met all existing applicable regulatory design criteria and no new requirements for US PWRs and BWRs were established. It is a general requirement to avoid excessive thermal stress in ACR piping systems, particularly in the HTS and moderator system. An assessment of the generic applicability of this GSI will be performed and provided in the DCD.
80	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	a	NUREG-0933 dropped this item.
81	Impact of Locked Doors and Barriers on Plant and Personnel Safety	a	NUREG-0933 prioritized this item as Low.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
82	Beyond Design Basis Accidents in Spent Fuel Pools	c	This item addressed the potential for a beyond-design-basis accident in which the water is drained out of the spent fuel pool, allowing the Zircaloy fuel cladding to ignite and thus release fission products from the spent fuel to the atmosphere. However, based on evaluation, the NRC staff concluded that further reduction in the already very low risk from the spent fuel pool accident would still leave a comparable risk due to core damage accidents, and because of the large inherent safety margins in the design and construction of the spent fuel pool, this item was resolved for US PWRs and BWRs with no new requirements were established. ACR spent fuel pool is designed to current Canadian regulatory requirements. Further assessment will be performed and provided in the DCD to ensure that the concerns of beyond design basis accidents are addressed for the ACR spent fuel pool.
83	Control Room Habitability	c	There were three items that addressed concerns over control room habitability: B-36, B-66, and III.D.3.4. This item called for the Control Room Habitability Work Group to identify any recommended actions that would correct significant deficiencies in control room habitability design, installation, test, or maintenance. In resolving this item, the NRC published a few NUREG reports documenting the results of its study, and developed a HABIT code providing an integrated code package for evaluating control room habitability. No new requirements for US PWRs and

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			BWRs were issued. ACR control room is designed in accordance with current Canadian regulatory requirements. Further assessment will be performed and provided in the DCD to confirm this and that relevant NRC requirements are met.
84	CE PORV s	b	This item addressed six operating CE plants that did not have PORVs. The six plants were believed to meet regulatory requirements, but the accident management for beyond design basis events and potential core-melt risk reduction required further study. The NRC staff's regulatory analysis justified that no high capacity manual venting capability was needed. Thus, no new requirements were issued. This item is applicable to the six CE plants only.
85	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	a	NUREG-0933 dropped this item.
86	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	g	This item called for the NRC staff to implement a long range plan for resolving the problems related to stress corrosion cracking in BWR piping. In 1982, leaks were detected in the heat-affected zones of the safe-end-to-pipe welds in two of the 28 in. diameter recirculation loop safe ends at Nine Mile Point Unit 1. Subsequent UT revealed extensive cracking at many weld joints in the recirculation system. The cause of the cracking was determined to be intergranular stress corrosion

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			cracking (IGSCC). In resolving this item, the NRC issued Generic Letter 88-01 ([84]) outlining the staff positions on IGSCC in BWR austenitic stainless steel piping. Stainless steel may be used in the lower feeder sections, which is to reduce the Flow Assisted Corrosion. Further assessment will be performed and provided in the DCD to ensure the ACR design addresses the IGSCC issue.
87	Failure of HPCI Steam Line Without Isolation	g	This item addressed the uncertainty regarding the operability of the motor-operated isolation valves for the steam supply lines of the HPCI system in BWRs following a postulated break in the supply line. The HPCI steam supply line has two containment isolation valves in series: one inside and one outside of the containment. Both are normally open in most plants; however, two plants were found to operate with the HPCI outboard isolation valve normally closed. A HPCI supply valve, located adjacent to the turbine, and the turbine stop valve are normally closed. A similar situation can occur in the RWCU system which has two normally open containment isolation valves that must remain open if the system is to function. A proposed solution to the HPCI problem was to require that the outboard HPCI isolation valve be normally closed. However, a small bypass line on those plants not having this feature would be required to prevent thermal shock and water hammer and to provide assurance that leaks in the line would be detected before they become breaks. If the HPCI supply valve were kept normally open (currently it is kept normally closed) the

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			probability of not getting steam to the HPCI turbine when needed might not be significantly changed. Another possible solution that would apply to valves in any system was a demonstration by test or the verification of use in other service applications that certified the operability of the valve under line rupture flow conditions. If the normal HPCI steam flow rate approximates that estimated for a break in the steam line, the valves might be tested by individually closing them when the HPCI turbine is in operation. The resolution of this item resulted in the issuance of Generic Letter 89-10 ([22]) providing licensees with the best guidance available at that time regarding how they should assure that their MOVs would perform their design basis function. The ACR design does not use steam to drive the high pressure coolant injection. On LOCA signal, the high-pressure water from the pressurized ECI tanks is injected into the heat transport system. No steam is needed to pressurize the ECI tanks. The ACR design includes MOVs in its safety-related systems. The MOVs' operability will be reviewed in the DCD with respect to the NRC guidance in Generic Letter 89-10 ([22]).
88	Earthquakes and Emergency Planning	d	This item called for the NRC staff to reconsider the role of emergency planning in ensuring the continued protection of the public health and safety in areas around nuclear power plants. The NRC published proposed amendments to its emergency planning requirements. However, the proposed amendments were withdrawn. This NRC internal item resulted in no new requirements.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
89	Stiff Pipe Clamps	a	NUREG-0933 prioritized this item as Low.
90	Technical Specifications for Anticipatory Trips	a	NUREG-0933 dropped this item.
91	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	c	This item addressed concerns raised after the event that one of the three emergency diesel generators (EDG) at the Shoreham Plant failed during overload testing as a result of a fractured crankshaft. All three EDGs were supplied by Transamerica DeLaval Inc. (TDI) and were Model DSR-48 diesels. In response to the problem raised, the Owner's Group performed extensive design reviews of all key engine components and developed recommendations to be implemented by the individual owners concerning needed component replacements and modifications, component inspections to validate the "as-manufactured" and "as-assembled" quality of key engine components, engine testing, and an enhanced engine maintenance and surveillance program. The NRC staff concluded that implementation of the Owners' Group recommendations, plus additional actions identified, would establish the adequacy of the TDI diesel generators for nuclear standby service as required by GDC 17 of 10 CFR 50, Appendix A ([18]). No new requirements for US PWRs and BWRs were established. There are two onsite standby diesel-generators sets for each unit of the two-unit ACR plant. ACR will not use the problematic model of diesel generators. This will be confirmed in the DCD.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
92	Fuel Crumbling During LOCA	a	NUREG-0933 dropped this item.
93	Steam Binding of Auxiliary Feedwater Pumps	g	This item addressed the potential for a common mode failure of the pumps in an auxiliary feedwater system. Hot water leaking through one or more isolation valves can flash to steam at the auxiliary feedwater pump potentially resulting in the failure of the pump to operate if required because of steam binding. In resolving this item, the NRC issue IE Bulletin 85-01 to licensees with requirements to develop procedure to detect or correct steam binding. This was reinforced by IE Bulletin 88-03. Thus, this item was resolved with new requirements established. The ACR design includes two auxiliary feedwater pumps and associated piping and valves that are intended to be used during normal plant operation and accident events, and are backed up by the RWS that will supply emergency feedwater by gravity in case of a total loss of feedwater to the SGs. The steam generators are rapidly depressurized to allow gravity feed. The supply of emergency feedwater (auxiliary feedwater) to the SGs of ACR does not rely solely on the operation of pumps. The PRA analysis has shown that the auxiliary feedwater pumps and associated piping and valves, and the RWS have an acceptable reliability. The ACR compliance with this GSI will be confirmed in the DCD.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
94	Additional Low Temperature Overpressure Protection for Light Water Reactors	g	This item addressed the establishment of additional guidance for Reactor Coolant System (RCS) low temperature overpressure (LTOP) protection to assure reactor vessel and RCS integrity, beyond the guidance for the provision of LTOP protection identified in the resolution of Item A-26. Additional RCS overpressurization incidents have occurred since the implementation of USI A-26 guidelines by operating plants. Two events in particular, were severe enough to be identified as abnormal occurrences. In resolving this item, the NRC staff issued Generic Letter 90-06 ([81]) which required a revision to the Technical Specification for overpressure protection at the affected plants. The RCS LTOP is provided in the ACR design as well. An assessment of the compliance of the ACR design with this GSI will be performed and provided in the DCD.
95	Loss of Effective Volume for Containment Recirculation Spray	b	This item was raised by a resident inspector who questioned the practice of leaving the refueling canal drain valve in the closed position during operations at a plant. The principal concern was the potential for water entrapment in the refueling canal of certain operating PWRs. If the refueling canal drains in a PWR dry containment are closed during plant operation and a LOCA occurs, that fraction of the containment spray which falls into the refueling canal would be prevented from returning to the containment emergency sump. Eventually, the entire volume of the refueling canal would be filled with water which would then not be available for the recirculation mode of containment and

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			reactor cooling. The NRC found this item was limited to a relatively small number of older PWRs and existing regulations had already included requirements relevant to this item. Thus, this item was resolved with no new requirements issued. Since there is no containment spray system and there are no similarly exposed refueling canals in ACR design, this item is not applicable to the ACR design.
96	RHR Suction Valve Testing	e	Superseded by 105.
97	PWR Reactor Cavity Uncontrolled Exposures	e	Superseded by III.D.3.1.
98	CRD Accumulator Check Valve Leakage	a	NUREG-0933 dropped this item.
99	RCS/RHR Suction Line Valve Interlock on PWRs	g	This item addressed the concerns with the loss of residual heat removal (RHR) capability in PWRs during cold-plant outage operations. The principal cause of loss of RHR capability was attributed to an autoclosure interlock at some plants for the RHR suction isolation valves that are intended to protect the RHR system from RCS pressure during operation at power. Correct operation of the interlock causes the valves to close when the RCS pressure increases above the design pressure of the RHR system. A number of events occurred that involved loss of RHR flow during cooldown, due to inadvertent actuation of the isolation valves.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			The scope of this item, initially directed solely at the autoclosure interlock-related mode of RHR failure, was broadened to include the risk mode of failure associated with mid-loop operation. Operating plants have experienced two principal kinds of loss of RHR capability during shutdown. One event occurred during mid-loop operation at a plant, when the water in the reactor coolant system had been lowered for maintenance purposes below the top of the hot legs. This level is very close to that at which vortices begin to form at the RHR system suction connections in the hot leg piping and the RHR system rapidly becomes air bound and RHR capability was substantially lost by the time the plant operators realized what had happened. The RCS became slightly pressurized due to boiling in the core before the operators were able to eliminate air from the RHR system and restore its capacity. In resolving the issue, the NRC issued Generic Letter 88-17 ([85]) requesting the PWR licensees and applicants to implement plant improvements pertinent to the concerns. The ACR design includes a Long Term Cooling system that is intended to remove the residual heat after the reactor is shut down. The long term cooling system is connected with the HTS and is isolated from HTS by eight isolation valves (two valves at each line of the four lines). An assessment of the compliance of the ACR design with this GSI will be performed and provided in the DCD.
100	OTSG Level	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
101	BWR Water Level Redundancy	c	This item addressed the concerns with BWR water level instrumentation. Water level is measured in BWRs by means of differential pressure sensors connected between the reactor vessel and reference columns. Typically, a BWR will have two reference columns. A break in one column will cause all instrumentation associated with that column to indicate full scale high level, which can simultaneously cause a transient and interface with safety systems. A single failure associated with the other reference column can completely defeat mitigation systems. This item was resolved with no new requirements established. Even though there is no pressure vessel in the ACR design, reference legs that have similar functions as the reference columns for BWRs are used for water level measurements for Steam Generators, Pressurizer, and Calandria. An assessment will be performed and provided in the DCD to ensure that the ACR design adequately addresses the safety concerns noted above.
102	Human Error in Events Involving Wrong Unit or Wrong Train	c	This item addressed the concerns caused by a number of events that resulted from human error in identification of the right unit or train. The NRC concluded that the primary cause of these events were inadequate labelling of areas, equipment, and components, inadequate personnel training and experience, and inadequate procedures. The NRC staff called for the industry to increase attention in above-mentioned areas. Based on the actions taken by the NRC staff and the industry initiatives, the NRC concluded that no further staff action was

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			warranted and no new requirements for US PWRs and BWRs were issued. Human factor engineering principles have been well applied in the ACR design to enhance plant safety by significantly eliminating the root causes in design that might induce human errors. This will be confirmed in the DCD. AECL will also provide assistance to the COL applicant in setting up adequate maintenance and testing procedures to minimize the probability of human error.
103	Design for Probable Maximum Precipitation	f	This item addressed the methodology used for determining the design flood level for a particular reactor site. The resolution of this item resulted in the revision of SRP ([10]) Sections 2.4.2 and 2.4.3. The design flood level is a site –related parameter. ACR is designed for a series of enveloping conditions including precipitation. The prospective COL applicant will be responsible for demonstrating the site parameters are within the limits specified for the ACR design. Thus, the responsibility for addressing this item rests with the COL applicants.
104	Reduction of Boron Dilution Requirements	a	NUREG-0933 dropped this item.
105	Interfacing Systems LOCA at LWRs	c	This item addressed reducing the risk of loss of primary coolant outside the containment via a system which connects with the reactor coolant system (RCS). The interfacing system LOCA (ISLOCA) is presumed to result

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			from exposing low pressure piping of the interfacing system to high primary system pressure. The initial plant response to an ISLOCA is the same as the response to an equivalent sized LOCA inside containment. However, RCS inventory is discharged outside containment and is not returned to the containment sump for recirculation. In addition, an ISLOCA will provide a path, which bypasses the containment, for release of radioactive materials. In addressing the above concerns, the NRC staff found that ISLOCAs at PWRs were plant-specific in nature. For future plants, a draft SRP ([10]) Section covering design review of systems interfacing with the RCS in ALWRs was provided to NRR. No new requirements for US PWRs and BWRs were established. ACR design includes a Long Term Cooling (LTC) system that is connected to HTS and whose major piping and equipment are outside the containment. The LTC is connected to the two reactor inlet headers and two reactor outlet headers. During normal operation of the reactor, the LTC is isolated by eight isolation valves (two at each line). An assessment will be performed and provided in the DCD to ensure that the concerns noted above are addressed for the ACR design.
106	Piping and Use of Highly Combustible Gases in Vital Areas	c	This item addressed the normal process system use of relatively small amounts of combustible gases on site and also addressed leaks or breaks in the hydrogen piping and supply system that could result in the accumulation of a combustible or an explosive mixture of air and hydrogen within the Auxiliary

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Systems Building. The accumulation of a combustible or an explosive mixture of gas within the Auxiliary Systems Building could represent a threat to safety-related equipment if the combustible gases are inadvertently ignited. One possible solution to this issue was believed to be the installation of excess-flow check valves located close to the source of the combustible gas as required by SRP ([10]) Section 9.5-1. This item was resolved for US PWRs and BWRs with no new requirements established. ACR does use hydrogen in some process systems. The location and design of the hydrogen system comply with fire protection requirements, as outlined in current Canadian standards. Further assessment will be performed and provided in the DCD to ensure that any credible potential for hydrogen systems to jeopardise safety-related equipment is addressed.
107	Main Transformer Failures	a	NUREG-0933 dropped this item.
108	BWR Suppression Pool Temperature Limits	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
109	Reactor Vessel Closure Failure	a	NUREG-0933 dropped this item.
110	Equipment Protective Devices on Engineered Safety Features	a	NUREG-0933 dropped this item.
111	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
112	Westinghouse RPS Surveillance Frequencies and Out-of- Service Times	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	c	This item addressed the requirements for dynamic qualification testing or dynamic surveillance testing of large bore hydraulic snubbers (> 50 kips rating) in operating plants. The issue was raised because of the concern for the integrity of the steam generator lower support structures when subject to a seismic event. However, the issue was applicable to all LWRs with components, structures, and supports that rely on LBHS for seismic restraint and other dynamic loads such as high energy line breaks and water hammers. This item was to assess the need for an NRC requirement for dynamic qualification testing of LBHS in operating plants. However, in resolving the issue, the NRC staff concluded that there were few cost-beneficial changes to existing requirements that would result in improved LBHS reliability and reduced risk. Thus, this item was resolved for US PWRs and BWRs with no new requirements established. The NRC staff recommended that a Regulatory Guide be developed for future plants. Mechanical and hydraulic snubbers are used in ACR, but, ACR uses fewer snubbers than the other nuclear power plants. An assessment will be performed and provided in the DCD to ensure that the concerns noted above are adequately addressed for the ACR design.
114	Seismic-Induced Relay Chatter	e	Superseded by A-46.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
115	Enhancement of the Reliability of Westinghouse Solid State Protection System	b	This item was to request some operating Westinghouse plants to incorporate additional diversity in protection systems. This Westinghouse-specific item is not applicable to the ACR design.
116	Accident Management	d	This item addressed the potential risk reduction that might result from training operators and having procedures developed to assist the operators in managing accidents beyond the design basis. The NRC staff determined that the item was being addressed in the Accident Management element of the NRC plan for the closure of severe accident issues and was not pursued separately.
117	Allowable Time for Diverse Simultaneous Equipment Outages	a	NUREG-0933 dropped this item.
118	Tendon Anchorage Failure	g	This item originated after a dented and leaking tendon grease cap was found during inspections at Farley Unit 2 on January 27, 1985 prior to the integrated leak rate test of the prestressed concrete containment structure. Subsequent detailed inspection revealed that three lower vertical tendon anchor heads were broken. Several anchor heads were then removed from the vertical tendons and magnetic particle testing revealed cracks in the ligaments between the holes in the back of the anchor heads. Metallurgical analysis of the anchor head material indicated that the failures had been caused by hydrogen stress-cracking. There was evidence of corrosion caused by hydrogen generation from the anodic reaction of zinc and steel in the presence of water, since quantities of water ranging from a few ounces to about

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			1.5 gallons were found in the grease caps; most of the water was found in the vertical tendon lower anchor grease caps. Concerns for the generic implications of the tendon anchor failure at Farley Unit 2 resulted in the identification of this issue by NRR/DL. A tendon inspection, repair, and surveillance program was initiated for both Farley Units 1 and 2. The licensee evaluated the containments and concluded that the structural integrity had been maintained continuously for both units. In resolving this issue, the NRC issued Regulatory Guides 1.35 ([86]) and 1.35.1 ([87]) that provide guidance for future plants. Since the ACR containment is a grouted post-tensioned system that is different from the ungrouted tendon system at Farley plant, crack developments around the anchors during the life of the plant will be unlikely and have no practical consequences. This will be confirmed in the DCD.
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
119.2	Piping Damping Values	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
119.3	Decoupling the OBE from the SSE	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
119.4	BWR Piping Materials	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
119.5	Leak Detection Requirements	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
120	On-Line Testability of Protection Systems	c	This item was raised when it was found that the protection system designs of some older plants did not provide as complete a degree of on-line protection system surveillance testing capability as other plants undergoing staff review and evaluation at that time. The requirements for at-power testability of components are included in GDC 21 of Appendix A to 10 CFR 50 ([18]). Supplementary guidance is provided in Regulatory Guides 1.22 ([88]) and 1.118 ([89]) and IEEE Standard 338 to ensure that protection systems (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while the plant is operating without adversely affecting the plant's operation. These requirements apply to both the RPS and the ESFAS. Existing STS indicate that it is desirable to test all protection systems through their sub-group relays every 6 months. In resolving the issue, the NRC staff concluded that testing of protection systems at power can have the potential for subtle interactions with other safety systems and/or plant operation that might result in an increase in plant risk. Thus, this item was resolved for US PWRs and BWRs with no new requirements established. ACR design includes the provision of on-line testing without impairment of the safety function of the safety system, which is in accordance with current Canadian regulatory requirements. This will be confirmed in the DCD.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
121	Hydrogen Control for Large, Dry PWR Containments	c	This item concerned ongoing NRC experimental and analytical programs addressing the likelihood of safe shutdown equipment surviving a hydrogen burn. The NRC staff also intended to explore the possibility and probable consequences of the formation of local detonable concentrations in large, dry PWRs. The concerns were prediction of conditions in realistic configurations, and containment and equipment survivability. In resolving the issue, the NRC staff concluded (1) no new regulatory guidance on hydrogen control was required for existing plants; (2) for future plants, the need for hydrogen control requirements will be examined as part of ongoing case-by-case review (certification reviews) and the generic efforts associated with 10 CFR 50.59 ([90]) rule changes. The hydrogen behaviour in ACR will be similar to that in current CANDU reactors. To control hydrogen in containment, the ACR design includes many up-to-date features such as much simple containment structure, enhanced air cooling, control of hydrogen concentration in a sub-volume of containment with forced circulation. It will be confirmed in the DCD that the ACR design addresses the safety concerns of this GSI.
122.1.a	Failure of Isolation Valves in Closed Position	e	Superseded by 124.
122.1.b	Recovery of Auxiliary Feedwater	e	Superseded by 124.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
122.1.c	Interruption of Auxiliary Feedwater Flow	e	Superseded by 124.
122.2	Initiating Feed-and-Bleed	c	This item addressed the Loss of All Feedwater Event with respect to the provision of enhanced operator training and improved instrumentation to aid the operator in determining that the plant has experienced a total loss of feedwater. During routine operation at the Davis-Besse nuclear power generating station, a loss of all feedwater event occurred. Subsequent to the loss of feedwater, the operators delayed initiating feed-and-bleed to cool the core on the presumption that auxiliary feedwater flow was imminent. An analysis of this event revealed that in addition to the operators' hesitancy to commence feed-and-bleed operations, the normal control room instrumentation was found to be inadequate to alert the operators that feed-and-bleed was required. In resolving the issue, the NRC staff concluded that there was no need for new regulatory requirements. ACR design includes in the control room instrumentation and displays of sufficient quality, range, and reliability to enable the plant operators to quickly recognize a Loss-of-All Feedwater event and to assess when to initiate mitigating measures. The ACR design also includes a post accident management (PAM) process that can help operator in monitoring the plant status and making right decisions. It will be confirmed that the ACR design addresses the safety concerns of this GSI.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
122.3	Physical Security System Constraints	a	NUREG-0933 dropped this item.
123	Deficiencies in the Regulations Governing DBA and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985	a	NUREG-0933 dropped this item.
124	Auxiliary Feedwater System Reliability	g	This item addressed the use of PRA to evaluate the reliability of the auxiliary feedwater system. Items 68, 122.1.a, 122.1.b, 122.1.c, and 125.II.1.b were integrated into the resolution of this item. This item was resolved by the NRC's issuing plant-specific requirements for two plants that did not initially have a reliability higher than a minimum criterion. The ACR design includes two auxiliary feedwater pumps and associated piping and valves that are intended to be used during normal plant operation and accident events, and are backed up by the RWS that will supply emergency feedwater by gravity in case of a total loss of feedwater to the SGs. The PRA analysis has shown that the reliability of the auxiliary feedwater pumps and associated piping and valves, and the RWS have an acceptable reliability. This will be confirmed in the DCD.
125.I.1	Availability of the STA	a	NUREG-0933 dropped this item.
125.I.2.a	Need for a Test Program to Establish Reliability of the PORV	e	Superseded by 70.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	e	Superseded by 70.
125.I.2.c	Need for Additional Protection Against PORV Failure	a	NUREG-0933 dropped this item.
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	e	Superseded by A-45.
125.I.3	SPDS Availability	e	This item called for licensees to implement the SPDS requirements as defined under item I.D.2. ACR will address this item by addressing I.D.2.
125.I.4	Plant-Specific Simulator	a	NUREG-0933 dropped this item.
125.I.5	Safety Systems Tested in All Conditions Required by Design Basis Analysis	a	NUREG-0933 dropped this item.
125.I.6	Valve Torque Limit and Bypass Switch Settings	a	NUREG-0933 dropped this item.
125.I.7.a	Recover Failed Equipment	a	NUREG-0933 dropped this item.
125.I.7.b	Realistic Hands-On Training	a	NUREG-0933 dropped this item.
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	a	NUREG-0933 dropped this item.
125.II.1.a	Two-Train AFW unavailability	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
125.II.1.b	Review Existing AFW Systems for Single Failure	e	Superseded by 124.
125.II.1.c	NUREG-0737 Reliability Improvements	a	NUREG-0933 dropped this item.
125.II.1.d	AFW/Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants	a	NUREG-0933 dropped this item.
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	a	NUREG-0933 dropped this item.
125.II.3	Review Steam/Feedline Break Mitigation Systems for Single Failure	a	NUREG-0933 dropped this item.
125.II.4	Thermal Stress of OTSG Components	a	NUREG-0933 dropped this item.
125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components	a	NUREG-0933 dropped this item.
125.II.6	Reexamine PRA-Based Estimates of the Likelihood of a Severe Core Damage Accident Based on Loss of All Feedwater	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
125.II.7	Reevaluate Provisions to Automatically Isolate Feedwater from Steam Generator During a Line Break	c	This item addressed the need for owner-operators and plant designers to reassess the benefits of automatically isolating the auxiliary feedwater (AFW) system after a main steam line or main feedwater line break. Automatic isolation of AFW from a steam generator (SG) can help to mitigate the consequences of the break. Typically, upon a low SG pressure signal, main steam isolation valves are closed and AFW is isolated from the depressurizing or faulted SG. This minimizes blowdown from the line break, and limits primary system overcooling and the potential for a return to criticality. If the AFW were not isolated the peak pressure in the containment for secondary side breaks could exceed that due to a large break LOCA, the usual basis for containment design. The automatic isolation logic also diverts AFW flow from the faulted to the intact SG. In contrast, there are disadvantages to automatic isolation of AFW. If both channels of the controlling isolation logic systems were to spontaneously actuate either during normal operation or in the course of a transient, the availability of AFW would be lost and the main steam isolation valves would close. Most newer plants use turbine-driven main feedwater pumps. Thus, main feedwater would also be lost, resulting in complete loss of the secondary heat sink. Capability to lock-out the isolation logic is necessary to preclude such scenarios. Proper and timely operator action following the various loss of cooling events is essential to attaining cold shutdown with minimum adverse consequences. It follows that the time available for accurate diagnosis of a problem by the operator before having to take action becomes an important factor. In

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			addition, because steam or feedwater line breaks make only a small contribution to the probability of core damage, the NRC concluded that plant safety would not be significantly improved or degraded by either the exclusion or inclusion of the automatic AFW isolation feature. Therefore, the choice to include automatic AFW isolation in the design is dependent upon the containment design and the time that can be made available for operator action. A very straightforward solution was proposed: simply disconnect the AFW isolation valve actuators from the automatic logic and depend on plant procedures that require operator to close the AFW isolation valves in the event of a line break. But, the regulatory analysis by NRC staff did not justify the removal of the AFW automatic isolation feature. Hence, a resolution, “No Action” was adopted. The ACR design will be assessed in the DCD with respect to the safety concerns noted above.
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	a	NUREG-0933 dropped this item.
125.II.9	Enhanced Feed-and-Bleed Capability	a	NUREG-0933 dropped this item.
125.II.10	Hierarchy of Impromptu Operator Actions	a	NUREG-0933 dropped this item.
125.II.11	Recovery of Main Feedwater as Alternative to AFW	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
125.II.12	Adequacy of Training Regarding PORV Operation	a	NUREG-0933 dropped this item.
125.II.13	Operator Job Aids	a	NUREG-0933 dropped this item.
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	a	NUREG-0933 dropped this item.
126	Reliability of PWR Main Steam Safety Valves	d	NUREG-0933 ranked this item as a Licensing Issue.
127	Testing and Maintenance of Manual Valves in Safety-Related Systems	a	NUREG-0933 prioritized this item as Low.
128	Electrical Power Reliability	g	This item addressed the reliability of onsite electrical systems. NUREG-0933 combined three GSIs, A-30, 48 and 49, previously individually listed under NUREG-0737 ([6]) in order to provide a more integrated approach to resolving these interrelated issues. A-30 addressed the reliability of DC power supplies used in the control and actuation of safety-related components and systems. DC power systems in nuclear power plants provide control and motive power to a variety of safety-related equipment including valves, instrumentation, emergency diesel generators, and many other components and systems. This power is needed during abnormal shutdowns and accident situations, as well as during normal operations. Presently, a minimum of two divisions of DC power is required to supply control and motive power to this safety-related equipment; failure of one division would generally cause a reactor scram for this type of configuration.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Furthermore, if the independence of the two divisions is compromised through the failure of a bus-tie breaker to function properly, then a fault in one division could propagate to the redundant division resulting in a loss of redundancy. To address Item A-30, Generic Letter 91-06 ([91]) was issued to request licensees to respond to 9 questions that were developed to facilitate staff determination of licensee implementation of existing recommendations. This item was identified when it was found that some licensees lacked administrative controls or TS governing operational restrictions for their Class 1E, 120 VAC vital instrument buses. These restrictions are required to ensure compliance with GDC 17, 21,34, and 35 of 10 CFR 50, Appendix A ([18]). During repair or maintenance activities on bus power sources or inverters, one or more of the normal or alternate vital instrument bus power sources could be removed from service indefinitely. This condition could lead to the loss of more than one vital instrument bus in the event of a single failure or loss of offsite power. Items 48 and 49 were addressed in Generic Letter 91-11 ([92]) issued to licensees to certify that they either have implemented TS or administrative controls conforming to the guidelines in the letter, or to justify why such controls may not be required. Item 49 concerned the use of only one tie breaker to electrically isolate redundant Class 1E buses. The problem with having one tie breaker separating buses is that isolation between redundant supplies is compromised upon closure due to inadvertent

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			operator action or equipment failure. ACR DC power supply is designed to current Canadian regulatory requirements and standards. The LCO for the 120V AC vital instrument bus power of ACR is defined in technical specifications. An assessment will be performed and provided in the DCD to ensure ACR complies with the requirements related to these items.
129	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	a	NUREG-0933 dropped this item.
130	Essential Service Water Pump Failures at Multiplant Sites	g	This item addressed the concerns for multiplant sites that have only two ESW pumps/plant with crosstie capabilities. Evaluation indicated that potential core damage could occur because of insufficient cooling water flow from the station service water system. The NRC staff addressed the loss of essential service water at 7 multiplant sites. The affected unites have similar ESW system designs with two trains per unit: one pump per train with a crosstie between units. The issue was resolved with TS and emergency procedures improvements issued in Generic Letter 91-13 ([93]). ACR design has crosstie capability. However, ACR Raw Service Water (RSW) system consists of two independent divisions, Division 1 and Division 2, with each division having two pumps as opposed to having a single pump. Further assessment of the compliance of the ACR design with this GSI

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			will be performed and provided in the DCD.
131	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System in Westinghouse Plants	d	The NRC decided that this item could be more efficiently addressed as part of the External Event Individual Plant Examination program.
132	RHR Pumps Inside Containment	a	NUREG-0933 dropped this item.
133	Update Policy Statement on Nuclear Plant Staff Working Hours	d	NUREG-0933 ranked this item as a Licensing Issue.
134	Rule on Degree and Experience Requirements	f	This item was to formulate the regulatory requirements on qualification of a senior operator of the nuclear power plant. In resolve this item, the NRC issued a policy statement on education for Senior Reactor Operators and Shift Supervisors at Nuclear Power Plants. No new requirements were established. This item relates to the staffing of plant personnel and is, therefore, the responsibility of COL applicant.
135	Steam Generator and Steam Line Overfill	c	This item addressed the concerns on a Steam Generator Tube Rupture (SGTR) accident causing overfill of the secondary side of the steam generator and the main steam lines. Several SGTR events have occurred in operating PWRs. At least one event led to water in the steam line following overfill of the steam generator due to safety injection pumping through the broken tube. Mixed steam and water flow through the secondary safety valves then resulted in unanalyzed dynamic and static loads on the steam piping. This item called for the following staff actions: Task 1 -

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			reviewing and developing a regulatory position on eddy current inservice inspection of steam generator tubes; Task 2 - reviewing proposed changes to SRP ([10]) Section 15.6.3 that addresses various aspects of SGTR such as assumed event duration and radiological consequences; Task 3 - reassessing various issues of Item 67 for potential inclusion in an integrated resolution; and Task 4 - reviewing the effect of water hammer, overfill and water carryover on steam lines and connected systems, and developing proposals for mitigating the consequences. The NRC staff found that SGTR and steam line overfill events pose a relatively low public risk. This item was resolved for US PWRs and BWRs with no new requirements established. The steam generators of ACR are designed, fabricated and installed to current international standards. It will be confirmed in the DCD that the ACR design addresses the concerns noted above.
136	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	d	NUREG-0933 ranked this item as a Licensing Issue.
137	Refueling Cavity Seal Failure	a	NUREG-0933 dropped this item.
138	Deinerting Upon Discovery of RCS Leakage	a	NUREG-0933 dropped this item.
139	Thinning of Carbon Steel Piping in LWRs	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
140	Fission Product Removal Systems	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
141	LBLOCA With Consequential SGTR	a	NUREG-0933 dropped this item.
142	Leakage Through Electrical Isolators in Instrumentation Circuits	c	This item addressed the susceptibility to leakage of isolation devices between safety- and nonsafety-related electrical systems. Electronic isolators are used to maintain electrical separation between safety and non-safety-related electrical systems in nuclear power plants, thereby preventing malfunctions in the nonsafety systems from degrading performance of safety-related circuits. Isolators are primarily used where signals from Class 1E safety-related systems are transmitted to non-Class 1E control or display equipment. The NRC required that licensees identified isolators in instrumentation circuits that are potentially susceptible to electrical leakage, define and perform an inspection and test program, replace failed or unacceptable isolators, and implement an annual program to inspect and test all electrical isolators between Class 1E and non-Class 1E systems. In resolving the issue, the NRC staff concluded that isolators performed satisfactorily. This conclusion may not be applicable to the digitalized systems. Therefore, the NRC staff recommended the development of an SRP ([10]) Section to provide review guidance for future plants that use digital systems. No new requirements for US PWRs and BWRs were established. Electrical isolators are used in ACR whose reliability has been proven at traditional CANDU reactors. This will be confirmed in the DCD.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
143	Availability of Chilled Water Systems and Room Cooling	c	This item addressed the concern of maintaining HVAC and chilled water systems in some rooms containing safety related system components. Operability of some safety-related components is dependent upon operation of HVAC and chilled water systems to remove heat from the rooms containing the components. If chilled water and HVAC systems are unavailable to remove heat, the ability of the safety equipment within the rooms to operate as intended cannot be assured. No new requirements for US PWRs and BWRs were issued in resolving the item. ACR design includes Heating, Cooling and Ventilation Systems whose operability is ensured during postulated accidents. This will be confirmed in the DCD.
144	Scram Without a Turbine/Generator Trip	a	NUREG-0933 dropped this item.
145	Actions to Reduce Common Cause Failures	f	This item was identified as an alternative approach to the Finding 15 Recommendation discussed in Item 125.I.5 that states that “through integrated system testing under various system configurations and plant conditions as near as practical to those for which the system is required to function during an accident is essential for timely detection and correction of common mode design deficiencies.” The identified alternative approach consisted of assessing the benefits of improvements in existing in-service, refueling, and surveillance testing program in operating reactors and improved startup testing for future plants.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			This item was to encourage licensees to take actions, such as making changes in maintenance program, testing, and procedures, to reduce the frequency of Common Cause Failure (CCF). In resolving this item, the NRC staff developed a CCF database and analysis software package to aid in system reliability analyses and related risk-informed applications. Licensees were informed of the availability of the CCF database. No new requirements were established. Since the target addressees of this item were licensees and there are no new requirements for the future plant, this item is applicable to operating plant only and not applicable to the ACR design. However, AECL will provide assistance to the COL applicant in ACR's startup testing to address the safety concerns noted above, on an as requested basis.
146	Support Flexibility of Equipment and Components	d	This item was identified by NRR when concerns were expressed that the seismic loading on equipment and pipe-mounted components may have been underestimated. In resolving the issue, for future plants, the NRC staff made two recommendations: (1) there was a need to reflect the uncertainty in anchorage stiffness assumption for heavy equipment. Pertinent Sections of the SRP ([10]) should be revised to adequately treat anchorage stiffness assumptions; and (2) NRC was conducting research on anchor bolt capacity and dynamic behaviour for individual and groups of anchor bolts. When definitive recommendations were available, appropriate changes to the SRP ([10]) should be made.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Since all these recommendations were addressed to NRC staff, this item is deemed as an NRC internal issue.
147	Fire-Induced Alternate Shutdown Control Room Panel Interactions	d	NUREG-0933 ranked this item as a Licensing Issue.
148	Smoke Control and Manual Fire-Fighting Effectiveness	d	NUREG-0933 ranked this item as a Licensing Issue.
149	Adequacy of Fire Barriers	a	NUREG-0933 dropped this item.
150	Overpressurization of Containment Penetrations	a	NUREG-0933 dropped this item.
151	Reliability of Recirculation Pump Trip During an ATWS	b	This item addressed the concern for the reliability of breakers used to trip the recirculation pumps at high pressure or low water level signals during ATWS mitigation in BWRs. This BWR-specific issue is not applicable to the ACR design.
152	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	a	NUREG-0933 dropped this item.
153	Loss of Essential Service Water in LWRs	c	This item addressed the reliability of essential service water (ESW) and the failure of such systems due to fouling mechanisms, ice effects, design deficiencies, flooding, multiple equipment failures, and human errors. The design of the ESW system varies substantially from plant to plant and the ESW system is highly dependent on the NSSS. As a result, generic solutions (if needed) are likely to be different

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			for PWRs and BWRs. The possible solutions are: (1) installation of a redundant intake structure including a service water pump; (2) hardware changes of the ESW system; (3) installation of a dedicated RCP seal cooling system; or (4) changes to TS or operational procedures. These potential improvements were considered for the seven multiplant sites covered in the scope of Issue 130; however, these options will now be evaluated for the remaining LWRs (65 PWRs and 39 BWRs). In resolving this item, the NRC staff found that the concerns involving ESW system reliability were being addressed on a plant-specific basis in various ongoing NRC and industry initiatives. The item was resolved for US PWRs and BWRs with no new requirements established. For ACR, the service water systems of each unit consist of the Raw Service Water (RSW) system, and the Recirculated Cooling Water (RCW) system, which provides cooling water to safety and non-safety related loads, in both NSP and BOP. Further assessment will be performed and provided in the DCD to ensure that the concerns involving ESW system reliability are addressed for the ACR design.
154	Adequacy of Emergency and Essential Lighting	a	NUREG-0933 dropped this item.
155.1	More Realistic Source Term Assumptions	g	This item was raised following the accident at TMI-2. During the TMI-2 accident, fission products did not behave as predicted with the analytical methods and assumptions used in the licensing process at that time and delineated in Regulatory

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Guides 1.3 ([94]) and 1.4 ([95]) and TID-14844 ([96]). The earliest expert predictions were that major core damage had occurred. However, the NRC and the licensee believed that core damage was minimal and calculations were redone to confirm this view. Approximately 50% of the core was in a molten state, but there is evidence that only about 55% of the highly volatile fission products and noble gases were released from the reactor vessel with a major portion retained in the reactor building. There is also evidence that less than 5% of the medium and low volatile fission products were released from the reactor vessel. These observations were based on research conducted since the TMI-2 accident. It is now generally accepted that the chemical conditions in the reactor vessel were "reducing" in nature as opposed to "oxidizing." The elemental iodine was driven (or converted) to the iodide ion which very readily combined with available metallic ions. The water-soluble character of these chemical forms prevented a major release of iodine to the atmosphere of the containment or auxiliary buildings and only a few curies were released to the environment. Throughout the TMI-2 accident sequence, the chemical state was maintained such that the water-soluble character was preserved. With the completion of a large number of PRAs since the TMI-2 event, the TMI-2 Safety Advisory Board believed that it should be possible to list accident sequences with chemical conditions similar to TMI-2. Such a listing could provide a guide as to which accidents might be regarded as hazardous, or less hazardous, relative to the possible escape of iodine and could be

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			useful in the future design of safety features. Since some of the assumptions used for source term considerations at TMI-2 were flawed in this respect, the Board recommended that the source term be restated using current scientific knowledge. In resolving the item, the NRC staff issued NUREG-1465 ([97]) which provided more realistic estimates of the fission product source term release into containment, in terms of timing, nuclide types, quantities, and chemical form, given a severe accident. The analytical model of ACR severe accident is based on the current scientific knowledge. An assessment of the source term assumptions in ACR safety analysis with respect to the US requirements will be performed and provided in the DCD.
155.2	Establish Licensing Requirements For Non-Operating Facilities	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
155.3	Improve Design Requirements For Nuclear Facilities	a	NUREG-0933 dropped this item.
155.4	Improve Criticality Calculations	a	NUREG-0933 dropped this item.
155.5	More Realistic Severe Reactor Accident Scenario	a	NUREG-0933 dropped this item.
155.6	Improve Decontamination Regulations	a	NUREG-0933 dropped this item.
155.7	Improve Decommissioning Regulations	a	NUREG-0933 dropped this item.
156.1.1	Settlement of Foundations and buried equipment	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
156.1.2	Dam Integrity and Site Flooding	a	NUREG-0933 dropped this item.
156.1.3	Site Hydrology and Ability to Withstand Floods	a	NUREG-0933 dropped this item.
156.1.4	Industrial Hazards	a	NUREG-0933 dropped this item.
156.1.5	Tornado Missiles	a	NUREG-0933 dropped this item.
156.1.6	Turbine Missiles	a	NUREG-0933 dropped this item.
156.2.1	Severe Weather Effects on Structures	a	NUREG-0933 dropped this item.
156.2.2	Design Codes, Criteria and Load Combinations	a	NUREG-0933 dropped this item.
156.2.3	Containment design and Inspection	a	NUREG-0933 dropped this item.
156.2.4	Seismic Design of Structures, Systems and Components	a	NUREG-0933 dropped this item.
156.3.1.1	Shutdown Systems	a	NUREG-0933 dropped this item.
156.3.1.2	Electrical Instrumentation and Controls	a	NUREG-0933 dropped this item.
156.3.2	Service and Cooling Water Systems	a	NUREG-0933 dropped this item.
156.3.3	Ventilation Systems	a	NUREG-0933 dropped this item.
156.3.4	Isolation of High and Low Pressure Systems	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
156.3.5	Automatic ECCS Switchover	e	Superseded by 24.
156.3.6.1	Emergency AC Power	a	NUREG-0933 dropped this item.
156.3.6.2	Emergency DC Power	a	NUREG-0933 dropped this item.
156.3.8	Shared Systems	a	NUREG-0933 dropped this item.
156.4.1	RPS and ESFS isolation	e	Superseded by 142.
156.4.2	Testing of the RPS and ESFS	e	Superseded by 120.
156.6.1	Pipe Break Effects on Systems and Components	f	In 1977, the NRC initiated the Systematic Evaluation Program (SEP) to review the designs of 51 older, operating nuclear plants. As a part of the SEP, this item was established to address the safety concern of whether the effects of pipe breaks inside the containment have been adequately addressed in the designs of some plants. A risk analysis performed by the NRC staff showed the issue to have some safety significance but with large uncertainty. A more comprehensive study was undertaken to review pipe failure rate data and pipe break methodologies. This item is under the technical assessment by the NRC staff. A recommendation on the development of regulation and guidance was scheduled for November 2002. NUREG-0933 assigned this item a HIGH priority. Due to (1) this item was targeted at the 51 older, operating nuclear plants, and (2) no generally applicable requirement has resulted from the pursuit of the resolution of this item, this item is deemed as not applicable to the ACR design. However, if any new requirements are issued in resolving the issue in future,

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			ACR will be assessed with respect to them as appropriate.
157	Containment Performance	f	This item addressed the concern that the MARK I containment could be severely challenged if a large core-melt occurred. NRC decided to examine MARK I plants for potential plant and containment modifications to improve containment performance. This examination was later expanded to include all other types of containment. Aside from a request for MARK I containment to implement some improvements, the NRC staff did not identify any other generic improvements that would be applicable to the containments. This item resulted in no generic requirements except those for MARK I containment. Since the ACR design does not include a MARK I containment, this item is ranked as not applicable to the ACR design.
158	Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions	f	This item was identified after reactor operating experience and research results on MOVs, SOVs, AOVs, and HOVs indicated that testing under static conditions didn't always reveal how these valves would perform under design basis conditions. A number of failures of power-operated valves had occurred as a result of inadequate design, installation, and maintenance. Concerns regarding the performance of MOVs were covered by Item II.E.6.1. The reliability of PORVs and safety valves was addressed in Item 70. This item focused on power-operated valves other than MOVs. In resolving the issue, the NRC concluded that existing

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			regulations provided an adequate framework for any needed regulatory action. NRR committed to undertake efforts in conjunction with the industry to ensure that existing requirements for valve operability under design basis conditions will be met. This item was resolved with no new requirements established. ACR design includes power-operated valves in its safety-related systems. These valves are designed, fabricated, and installed to current international standards. These valves will also subject to a surveillance and maintenance program established in accordance with current Canadian regulatory requirements and standards. Since there are no new requirements, this resolved item does not warrant any further assessment.
159	Qualification Of Safety-Related Pumps While Running On Minimum Flow	a	NUREG-0933 dropped this item.
160	Spurious Actuations Of Instrumentation Upon Restoration Of Power	a	NUREG-0933 dropped this item.
161	Use Of Non-Safety-Related Power Supplies In Safety-Related Circuits	a	NUREG-0933 dropped this item.
162	Inadequate Technical Specifications For Shared Systems At Multiplant Sites When One Unit Is Shut Down	a	NUREG-0933 dropped this item.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
163	Multiple Steam Generator Tube Leakage	h	This item was identified to address the safety concern associated with potential multiple steam generator tube leaks during a main steam line break that cannot be isolated. This sequence could lead to core damage that could result from the loss of all primary system coolant and safety injection fluid in the refueling water storage tank. The existing regulatory framework provides reasonable assurance that operating plants are safe. However, this framework has numerous shortcomings. In order to resolve these shortcomings, the NRC staff will revise the regulatory framework to utilize a risk-informed performance-based approach that will ensure compliance with existing regulations. NUREG-0933 gave this item a HIGH priority ranking. There are two steam generators for each unit of the ACR. An assessment of the ACR's steam generators with respect to current pertinent NRC requirements will be performed and provided in the DCD.
164	Neutron Fluence In Reactor Vessel	a	NUREG-0933 dropped this item.
165	Spring-Actuated Safety And Relief Valve Reliability	c	This item addressed the unreliability of spring-actuated relief valves in safety-related support systems. Failure of a spring-actuated relief valve can lead to a core-melt from loss of core cooling and inventory makeup. Possible sources of loss include: 1 failure of a valve to close after opening; (2) failure of a valve to open when challenged, resulting in overpressure conditions that precipitate a LOCA; and (3) premature opening of a valve below setpoint resulting in a LOCA.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			In resolving the issue, the NRC staff performed an analysis of a safety relief valve failing its train and found the resultant core damage frequency increase to be negligible. The NRC staff also determined that additional testing of safety relief valves was included in the 1986 Edition of ASME Section XI and was later endorsed by the NRC in the 1992 revision of 10 CFR 50.55a ([24]). No additional new requirements were issued. ACR uses some spring-actuated relief valves. An assessment will be performed and provided in the DCD to address the concerns noted above for the ACR design.
166	Adequacy of Fatigue Life of Metal Components	f	This item addressed the adequacy of fatigue life of metal components of operating plants. The concerns originated from the fact that many operating plants did not have requirements to monitor for fatigue limits. Based on the evaluation on the available records of transient monitoring from 7 selected plants, the NRC staff did not believe that the current licensing basis fatigue criteria were exceeded at operating plants. ACR's fatigue limits can be monitored in the plant life management (PLiM) and plant life extension (PLEx) programs, but this is the responsibility of the COL applicant.
167	Hydrogen Storage Facility Separation	a	NUREG-0933 prioritized this item as Low.
168	Environmental Qualification of Electrical Equipment	h	NUREG-0933 gave this open item a HIGH priority ranking. When reviewed significant license renewal issues, the NRC staff found that several related to environmental qualification (EQ).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Accelerated-aging tests on electrical equipment showed that some of the environmentally qualified cables either failed or exhibited marginal insulation resistance. Failure of the cables during or following a design basis event could affect the performance of safety functions. On behalf of the industry, NEI and IEEE provided industry positions and relevant information to the NRC staff in 2001. Since then, Okonite completed testing of their single conductor, bonded-jacket cables. The results will be appropriately factored into the completion of this item as the NRC staff continues to explore voluntary industry initiatives to resolve the issue. For ACR, the EQ program for electrical components including electrical cables is in accordance with current Canadian standards. An assessment will be performed and provided in the DCD to demonstrate the EQ adequacy of the electrical equipment of ACR.
169	BWR MSIV Common Mode Failure Due To Loss of Accumulator Pressure	a	NUREG-0933 dropped this item.
170	Fuel Damage Criteria for High Burnup Fuel	b	This item addressed the concern associated with a high burnup of fuel, e.g., 63 GWd/t. Higher burnup will result in a higher probability of fuel damage. The NRC performed an evaluation of data collected and confirmed that the use of fuel up to the existing limits did not pose safety problems. Thus, the item was resolved with no new requirements established. Although ACR uses Slightly Enriched Uranium (SEU) fuel that

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			can achieve a higher burnup than the natural uranium used in traditional CANDU plants, the burnup (about 20 GWd/t), is significantly lower than that of other U.S. commercial nuclear power plants. There are also significant differences between the ACR fuel and other reactor fuel designs. Therefore, this item is not applicable to the ACR design.
171	ESF Failure from LOOP Subsequent to a LOCA	c	This item concerned the ability of the Engineering Safety Feature Actuation System (ESFAS) sequencing to respond to a Loss Of Offsite Power (LOOP) which might occur during the sequencing. This item was originally given a high priority ranking. But, studies conducted during resolution of this item showed that the contribution to Core Damage Frequency (CDF) was far less than originally anticipated. Thus, no new requirements for US PWRs and BWRs were issued. ACR design includes diesel generators, electrical power distribution system, power transmission system that enable ACR being capable of withstanding a LOCA followed by a LOOP during load sequencing. This will be confirmed in the DCD.
172	Multiple System Responses Program	a	This item was identified to address 21 potential safety concerns that were raised by the ACRS during the resolution of Items A-17, A-66 and A-47. In resolving the issue, the NRC staff developed guidance for the review of the safety concerns in the Individual Plant Examination (IPE) and the Individual Plant Examination of External Events (IPEEE) programs. In the review of licensee submittals in response to the IPE and IPEEE, no significant contributor to core damage frequency (CDF) was

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			identified. Therefore, the NRC staff concluded that no new or revised licensee requirements were warranted. The issue was closed. The NRC staff also stated that, of the 21 concerns, eleven were to be covered in the IPE or IPEEE programs, and the remaining ten concerns were dropped from further consideration.
173.A	Spent Fuel Storage Pool: Operating Facilities	c	This issue addressed the adequacy of regulatory requirements for a sustained loss of spent fuel pool cooling after a loss of offsite power or a loss-of-coolant accident. In resolving the issue, the NRC staff developed screening criteria for reactor accidents in NUREG-1738 ([98]), “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants.” Plant-specific evaluations were then performed by estimating the frequency of a significant loss of coolant inventory or a sustained loss of cooling. These estimated frequencies were compared with the criteria of NUREG-1738 ([98]) and the staff concluded that no new or revised requirements were warranted. Thus, the issue was closed. ACR is provided with a Cooling and Purification System to maintain cooling of irradiated fuel during normal plant operation and accident conditions. An assessment will be performed and provided in the DCD to confirm the adequacy of the ACR Spent Fuel Bay cooling function in both normal and accidental conditions.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
173.B	Spent Fuel Storage Pool: Permanently Shutdown Facilities	f	The NRC staff issued Bulletin 94-01 requesting all holders of licenses for nuclear power reactors that were permanently shut down with spent fuel in the spent fuel pool to take actions to ensure the quality of the SFP coolant, the ability to maintain an adequate coolant inventory for cooling and shielding, and the necessary support systems were not degraded. This item was to use the result of Item 173.A to establish evaluation criteria for SFPs at permanently shutdown plants to support rulemaking and other generic activities initiated by the NRC. The NRC staff concluded that all significant identified concerns from Item 173.A applicable to permanently shutdown facilities were encompassed by the special inspection activities which showed no significant deficiencies other than at Dresden-1. Thus this item was resolved with no new requirements established. This item is applicable to permanently shutdown plants only and is not applicable to the ACR design. ACR Spent Fuel Bay will be assessed with respect to Item 173.A but not with this item.
174.A	SONGS Employees' Concern	b	A San Onofre Nuclear Generating Station (SONGS) employee filed a concern with the SONGS Employee Program concerning the acceptance of fastener threads using GO/No Go thread gages (System 21) rather than variables gaging (System 22). It was claimed that the use of GO/NO Go gages did not ensure that all of material limits specified in ASME B1.1 had been met and unsafe conditions could result from threaded fastener failures. The NRC staff concluded that, based on NRC investigation and inspection at the SONGS warehouse and interview with every

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			alleger, no unsafe conditions were observed and no new requirements were established.
174.B	Johnson Gage Company Concerns	b	Concerns were raised by employees at the Johnson Gage Company regarding the gaging of fasteners. It was claimed that the use of GO/NO Go gages did not ensure that all of material limits specified in ASME B1.1 had been met and unsafe conditions could result from threaded fastener failures. The NRC staff concluded that this issue had low safety significance, no compliance issue were involved, and no new requirements were established.
175	Nuclear Power Plant Shift Staffing	f	This item addressed the concerns over plant shift staffing for both licensed operators and non-licensed personnel. Since the shift staffing is the responsibility of a prospective COL applicant of ACR, this item is not applicable to ACR design certification.
176	Loss Of Fill-Oil In Rosemount Transmitters	f	The Rosemount Transmitter Review Group (RTRG) was established to perform an assessment of the actions taken to address Rosemount transmitter oil-loss concerns. This assessment included an evaluation of the adequacy of the information and actions specified in the NRC Bulletin 90-01, Supplement 1, which informed licensees of activities undertaken by the NRC and the industry in evaluating and addressing loss of fill-oil in Rosemount transmitters manufactured prior to July 11, 1989, and requested licensees to take actions to resolve the concerns. NUREG-0933 stated that this item was resolved and no new

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			requirements were established. ACR design will not use the problematic transmitters. Thus, this item is applicable to old operating plants only and is not applicable to the ACR design.
177	Vehicle Intrusion At TMI	g	This item was raised after an event involving an intruder who drove into the TMI site owner-controlled area, through a gate into the protected area of Unit 1, and crashed through a roll-up door on the turbine building on February 7, 1993. The intruder challenged security barriers and programs, disrupted normal site operations and was not apprehended for four hours. After this event, the NRC staff tested the Emergency Response Data System (ERDS) link with all reactor units that had not been linked to ERDS since October 1992 and found deficiencies in the performance of some links. A final Rule was published on August 1, 1994, to modify the design basis threat for radiological sabotage to include: (1) use of land vehicle by adversaries for transporting personnel and their hand equipment to the proximity of vital areas; and (2) a land vehicle bomb (in response to the bombing of the World Trade Center later in February, 1993). This Rule also required licensees to install vehicle control measures, including vehicle barrier systems, to protect against the malevolent use of a land vehicle. ACR design meets the requirements of the Nuclear Security Regulations of Canada. These Regulations establish minimum requirements for the implementation and maintenance of physical protection systems, equipment and procedures at

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			nuclear facilities based on *The Convention on the Physical Protection of Nuclear Material*, INFCIRC/274/Rev. 1 ([99]), and the IAEA recommendations entitled *The Physical Protection of Nuclear Material*, INFCIRC/225/Rev. 3 ([100]). The ACR will be located within a protected area, and will be equipped with access control and alarms to prevent and detect unauthorized entry. The ACR compliance with this GSI will be confirmed in the DCD.
178	Effect Of Hurricane Andrew On Turkey Point	d	NUREG-0933 ranked this item as a Licensing Issue.
179	Core Performance	d	NUREG-0933 ranked this item as a Licensing Issue.
180	Notice Of Enforcement Discretion	d	NUREG-0933 ranked this item as a Licensing Issue.
181	Fire Protection	d	NUREG-0933 ranked this item as a Licensing Issue.
182	General Electric Extended Power Uprate	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
183	Cycle-Specific Parameter Limits In Technical Specifications	d	NUREG-0933 ranked this item as a Regulatory Impact Issue.
184	Endangered Species	d	NUREG-0933 ranked this item as an Environmental Issue.
185	Control of Recriticality following Small-Break LOCA in PWRs	b	This issue addressed small-break-LOCA scenarios in PWRs that involve steam generation in the core and condensation in the steam generators, causing deborated water to accumulate in part of RCS. Restart of the RCS circulation may cause a recriticality

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			event (activity excursion) by moving the deborated water into the core. Completion of the technical assessment of the issue by NRC was scheduled for September 2005. NUREG-0933 ranked this issue as a high priority issue. Due to the fact that boron will never be added to the RCS of the ACR, since it is not an ACR design feature, there is no concern about the recriticality caused by moving deborated water in some parts of RCS into the core. Circulation of RCS of ACR will not cause poison dilution in the core region. Therefore, this issue is not applicable to the ACR design.
186	Potential Risk and Consequences of Heavy Load Drops	h	An NRC report addressing the safety concerns of this item will be available in the future. This item will be reviewed to identify its generic applicability to the ACR design, and if identified as applicable to the ACR design, will be addressed in the DCD.
187	The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants.	a	NUREG-0933 dropped this item.
188	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	h	This issue was raised by an RES staff member, and it arose from some operating and test experience that suggested that a main steam line break in a PWR can cause resonant vibration of steam generator tubes. This vibration raised the possibility of steam generator tubes rupturing during the course of an accident initiated by a main steam line break.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			Initial screening of this issue was completed by NRC staff. This issue will be prioritized in the future. This issue may be applicable to ACR, since ACR deploys two steam generators and four steam lines. Further assessment will be performed and provided in the DCD to address the concern of this issue for the ACR.
189	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Concentration during a Severe Accident	h	This issue was proposed in response to SECY-00-198, "Status Report on Study of Risk-Informed Changes to 10 CFR 50.44 ([101]) (Combustible Gas Control)." Recent research indicated that, for PWR ice condenser containments and BWR Mark III containments that have a relatively low free volume and containment strength, the early containment failure probability is dominated by hydrogen combustion events. These containments are equipped with igniters that are intended to control hydrogen concentrations by initiating limited "burns" before a large quantity accumulates. However, if the accident is initiated by a station blackout event, these igniters, which are AC-powered, will not be available. The issue is whether these igniters should be equipped with an alternative power supply. Initial screening of this issue was completed by NRC staff. This issue will be prioritized in the future. In ACR, hydrogen control is provided in the reactor building by passive autocatalytic hydrogen recombiners to limit hydrogen content to below the deflagration limit within any significant enclosed compartment of the containment following an accident. There are no hydrogen igniters in the ACR design and the passive recombiners do not need any power supply, which

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			addresses the concerns noted above. Further assessment will be performed and provided in the DCD to ensure the effectiveness of the recombiners during a station blackout event.
190	Fatigue Evaluation of Metal Components For 60- Year Plant Life	c	The risk of failure from fatigue of various reactor coolant system components was studied under issue 78 and later integrated into the NRR Fatigue Action Plan which was completed and documented in SECY-95-245. The staff concluded that the risk from fatigue failure of the primary coolant pressure boundary components is very small; this conclusion was based on a plant life of 40 years. The impact of a license renewal period of 20 years on fatigue of metal components was to be considered in the resolution of Issue 166, but Issue 190 was established to address this subject separately. Based on the existence of an ongoing action plan to address the safety concern and the NRR decision to pursue its resolution, the issue was considered nearly-resolved in August 1996. It was later given a high priority ranking in SECY-98-166. In resolving the issue, the staff performed probabilistic analyses which showed low core damage frequencies resulting from fatigue failure of metal components. However, the nature of age-related degradation indicated the potential for an increase in the frequency of pipe leaks as plants continue to operate. Consistent with 10 CFR 54.21 ([102]), “Requirements for renewal of operating Licenses for Nuclear Power Plants: Contents of Application – Technical Information,” licensees will have to address the effects of the reactor coolant system environment on component fatigue life, as aging management

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			programs are formulated in support of license renewal. This issue has been resolved for US PWRs and BWRs with no new requirements established. The ACR is designed to achieve an operating life of 60 years, supported by the Plant Life Management program (PLiM) and Plant Life Extension program (PLEx). Further assessment will be performed and provided in the DCD to essure that Issue 190 is addressed/resolved for the ACR design.
191	Assessment of Debris Accumulation On PWR Sump Performance	h	Results of research on BWR ECCS suction strainer blockage identified new phenomena and failure modes that were not considered in the resolution of issue A-43. In addition, operating experience identified new contributors to debris and possible blockage of PWR sumps, such as degraded or failed containment paint coatings. Thus, this issue was identified by NRR and called for expanded research effort to address these new safety concerns. A study was deemed to be required to determine whether PWR ECCS sumps are adequate to ensure proper ECCS operation. Based on the existence of an action plan to address the safety concerns, the issue was considered nearly-resolved in September 1996. It was later given a HIGH priority ranking in SECY-98-166. The design of the ACR ECCS system is based on previous CANDU designs that have certain features to prevent debris from entering into the ECC circuit. These features will be assessed in the DCD to provide assurance for ACR's compliance

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			with this GSI.
192	Secondary Containment Drawdown Time	b	This issue was raised by NRR and it addressed the adequacy of the calculations, testing, and acceptance criteria related to the creation of a vacuum in the reactor building of a BWR following an engineered safeguards actuation signal. This item is currently under initial screening by the NRC staff. Since ACR will not deploy a secondary containment system, this issue is not applicable to the ACR design.
<u>Human Factor Issues</u>			
HF1.1	Shift Staffing	f	This item, together with HF1.2 and HF1.3, was to assure that the number and capabilities of the staff at nuclear power plants are adequate to provide safe operation. To meet this goal, considerations will be given to: (1) the numbers and functions of the staff needed to safely perform all required plant operations, maintenance, and technical support for each operation mode; (2) the minimum qualifications of plant personnel in terms of education, skill, knowledge, training experience, and fitness for duty; and (3) appropriate limits and conditions for shift work including overtime, shift duration, and shift rotation. These items are only applicable to operating plants or the responsibility of COL applicant.
HF1.2	Engineering Expertise on Shift	f	Same as HF1.1.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
HF1.3	Guidance on Limits and Conditions of Shift Work	f	Same as HF1.1.
HF2.1	Evaluate Industry Training	d	NUREG-0933 ranked this item as a Licensing Issue.
HF2.2	Evaluate INPO Accreditation	d	NUREG-0933 ranked this item as a Licensing Issue.
HF2.3	Revise SRP Section 13.2	d	NUREG-0933 ranked this item as a Licensing Issue.
HF3.1	Develop Job Knowledge Catalog	d	NUREG-0933 ranked this item as a Licensing Issue.
HF3.2	Develop License Examination Handbook	d	NUREG-0933 ranked this item as a Licensing Issue.
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	e	Superseded by I.A.4.2(4).
HF3.4	Examination Requirements	e	Superseded by I.A.2.6(1).
HF3.5	Develop Computerized Exam System	d	NUREG-0933 ranked this item as a Licensing Issue.
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	d	This resolved item called for NRC staff to develop a program of inspections of operating plants' upgraded EOPs and as such is ranked as a Licensing Issue.
HF4.2	Procedures Generation Package-Effectiveness Evaluation	d	NUREG-0933 ranked this item as a Licensing Issue.
HF4.3	Criteria for Safety-Related Operator Actions	e	Superseded by B-17.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
HF4.4	Guidelines for Upgrading Other Procedures	f	On the basis of efforts to evaluate the quality of, and the problems associated with, existing plant procedures, NRC evaluated the need to develop technical guidance for use by the industry in upgrading normal operating procedures and abnormal operating procedures similar to what the NRC staff completed for EOPs. To resolve this issue, the NRC staff prepared a summary of accepted engineering practices that licensees could use in performing any voluntary upgrade of procedures. This item is not directly relevant to ACR design, since the procedure development is the responsibility of the licensees.
HF4.5	Application of Automation and Artificial Intelligence	e	Superseded by H.F.5.2.
HF5.1	Local Control Stations	c	This resolved item addressed additional NRC guidance for the design of local control stations. Previous regulatory efforts dealing with man-machine interface (MMI) were limited to the control room and remote shutdown panel. It was believed that further guidance regarding local control stations and auxiliary operator interfaces was necessary as well as additional guidance regarding improvements to existing annuciator systems. A survey of safety significant local control stations was conducted and documented in NUREG/CR-3696 ([103]). The NRC staff's studies were published in NUREG/CR-6146 ([104]). No new requirements for US PWRs and BWRs were established as a result of the closure of this issue. ACR deploys local control stations. The human factor

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			engineering principles have been applied to the design of these local control stations. This will be confirmed in the DCD.
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	c	The existing human engineering guidelines for nuclear power plant control rooms primarily addressed the control, display, and information concepts and technologies that were being used in process control systems. While these guidelines were adequate for the existing generation of nuclear power plants, the NRC staff did not believe that they were sufficient for advanced and developing technologies that could be introduced into existing and future designs. This item addressed additional NRC guidance for the design of advanced instrumentation and controls, in particular with respect to plant annunciators. But the work was terminated and resulted in no new requirements for US PWRs and BWRs. The human factor engineering principles have been applied to the design of control and instrumentation of ACR. This will be confirmed in the DCD.
HF5.3	Evaluation of Operational Aid Systems	e	Superseded by H.F.5.2.
HF5.4	Computers and Computer Displays	e	Superseded by H.F.5.2.
HF6.1	Develop Regulatory Position on Management and Organization	e	Superseded by I.B.1.1(1,2,3,4).
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	e	Superseded by I.B.1.1(1,2,3,4).

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
HF7.1	Human Error Data Acquisition	d	NUREG-0933 ranked this item as a Licensing Issue.
HF7.2	Human Error Data Storage and Retrieval	d	NUREG-0933 ranked this item as a Licensing Issue.
HF7.3	Reliability Evaluation Specialist Aids	d	NUREG-0933 ranked this item as a Licensing Issue.
HF7.4	Safety Event Analysis Results Applications	d	NUREG-0933 ranked this item as a Licensing Issue.
HF8	Maintenance and Surveillance Program	f	The Maintenance and Surveillance Program (MSP) effort was to provide direction for the NRC's efforts to assure effective nuclear power plant maintenance. To close this issue, it was proposed to implement a systematic maintenance program as addressed in the NRC's preliminary MSP with the following five objectives: 1) To assure that needed maintenance is being accomplished, especially in counteracting system and equipment aging effects, by taking appropriate preventive and corrective action to minimize equipment failures. 2) To reduce failures from improper maintenance to an acceptable level and to assure safety through effective maintenance management, personnel selection and training, procedures, administrative control, and design for maintainability. 3) To assure proper integration of maintenance operations and other organizational interfaces for maintenance activities which can affect plant safety.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
			4) To improve the effectiveness of nuclear power plant maintenance program in reducing the number of challenges to safety systems. 5) To optimize surveillance requirements to assure equipment availability when required without excessive equipment out-of-service intervals for testing and to eliminate the unnecessary exposure for transient trips due to excessive test frequencies of logic and initiation system. This issue was resolved in 1988 with the issuance of an NRC Policy Statement on Maintenance of Nuclear Power Plants and no new requirements were established. The establishment and implementation of an effective maintenance and surveillance program for a nuclear power plant is the responsibility of the prospective COL applicant or the licensee. Thus, this item is not a design issue.
<u>Chernobyl Issues</u>			
CH1.1A	Symptom-Based EOPs	d	NUREG-0933 ranked this item as a Licensing Issue.
CH1.1B	Procedure Violations	d	NUREG-0933 ranked this item as a Licensing Issue.
CH1.2A	Test, Change, and Experiment Review Guidelines	d	NUREG-0933 ranked this item as a Licensing Issue.
CH1.2B	NRC Testing Requirements	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
CH1.3A	Revise Regulatory Guide 1.47	d	NUREG-0933 ranked this item as a Licensing Issue.
CH1.4A	Engineered Safety Feature Availability	d	NUREG-0933 ranked this item as a Licensing Issue.
CH1.4B	Technical Specification Bases	d	NUREG-0933 ranked this item as a Licensing Issue.
CH1.4C	Low Power and Shutdown	d	NUREG-0933 ranked this item as a Licensing Issue.
CH1.5	Operating Staff Attitudes Toward Safety	d	NUREG-0933 ranked this item as a Licensing Issue.
CH1.6A	Assessment of NRC Requirements on Management	d	NUREG-0933 ranked this item as a Licensing Issue.
CH1.7A	Accident Management	d	NUREG-0933 ranked this item as a Licensing Issue.
CH2.1A	Reactivity Transients	d	NUREG-0933 ranked this item as a Licensing Issue.
CH2.2	Accidents at Low Power and at Zero Power	e	Superseded by CH1.4A, CH1.4B, CH1.4C, .
CH2.3A	Control Room Habitability	e	Superseded by 83.
CH2.3B	Contamination Outside Control Room	d	NUREG-0933 ranked this item as a Licensing Issue.
CH2.3C	Smoke Control	d	NUREG-0933 ranked this item as a Licensing Issue.
CH2.3D	Shared Shutdown Systems	d	NUREG-0933 ranked this item as a Licensing Issue.
CH2.4A	Firefighting With Radiation Present	d	NUREG-0933 ranked this item as a Licensing Issue.
CH3.1A	Containment Performance	d	NUREG-0933 ranked this item as a Licensing Issue.

GSI #	Title	Applicable Screening Criteria	Justification of the Screening
CH3.2A	Filtered Venting	d	NUREG-0933 ranked this item as a Licensing Issue.
CH4.1	Size of the Emergency Planning Zones	d	NUREG-0933 ranked this item as a Licensing Issue.
CH4.2	Medical Services	d	NUREG-0933 ranked this item as a Licensing Issue.
CH4.3A	Ingestion Pathway Protective Measures	d	NUREG-0933 ranked this item as a Licensing Issue.
CH4.4A	Decontamination	d	NUREG-0933 ranked this item as a Licensing Issue.
CH4.4B	Relocation	d	NUREG-0933 ranked this item as a Licensing Issue.
CH5.1A	Mechanical Dispersal in Fission Product Release	d	NUREG-0933 ranked this item as a Licensing Issue.
CH5.1B	Stripping in Fission Product Release	d	NUREG-0933 ranked this item as a Licensing Issue.
CH5.2A	Steam Explosions	d	NUREG-0933 ranked this item as a Licensing Issue.
CH5.3	Combustible Gas	d	NUREG-0933 ranked this item as a Licensing Issue.
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR	d	NUREG-0933 ranked this item as a Licensing Issue.
CH6.1B	Structural Graphite Experiments	d	NUREG-0933 ranked this item as a Licensing Issue.
CH6.2	Assessment	d	NUREG-0933 ranked this item as a Licensing Issue.

Table 2
Listing of US NRC Generic Safety Issues Potentially Applicable to the ACR Design

Screening Criteria	c	g	h
GSI#	I.D.3, I.D.4, I.D.5(1), I.D.5(3), I.D.5(4), I.F.1, I.F.2(9), II.C.4, II.K.1(17), II.K.2(13), II.K.3(12), A-15, A-17, A-29, A-45, B-5, B-9, B-17, B-60, B-61, C-7, C-11, C-12, 3, 14, 15, 20, 22, 24, 36, 57, 64, 79, 82, 83, 91, 101, 102, 105, 106, 113, 120, 121, 122.2, 125.II.7, 135, 142, 143, 153, 165, 171, 173.A, 190, HF5.1, HF5.2	I.D.1, I.D.2, I.D.5(2), I.F.2(2), I.F.2(3), I.G.2, II.B.1, II.B.2, II.B.3, II.B.8, II.D.1, II.D.3, II.E.1.1, II.E.1.2, II.E.3.1, II.E.4.2, II.E.4.4(1), II.E.4.4(2), II.E.4.4(3), II.E.6.1, II.F.1, II.F.2, II.F.3, II.G.1, II.K.1(22), II.K.2(16), II.K.3(1), II.K.3(2), II.K.3(5), II.K.3(7), II.K.3(25), II.K.3(30), II.K.3(31), III.A.1.2(1), III.A.1.2(2), III.D.1.1(1), III.D.3.3(1), III.D.3.3(2), III.D.3.3(3), III.D.3.3(4), III.D.3.4, A-1, A-2, A-3, A-4, A-9, A-11, A-12, A-13, A-24, A-25, A-26, A-28, A-31, A-35, A-36, A-40, A-43, A-44, A-46, A-47, A-49, B-36, B-53, B-56, B-63, B-66, C-1, C-17, 43, 45, 51, 67.3.3, 70, 75, 86, 87, 93, 94, 99, 118, 124, 128, 130, 155.1, 177	163, 168, 186, 188, 189, 191