CHAPTER 7

ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

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ACRONYMS AND ABBREVIATIONS

°F	degrees Fahrenheit
μgm/m^3	micrograms per cubic meter
/Q	relative air concentration
AADT	annual average daily traffic
A/B	auxiliary building
ac	acre
AC	alternating current
ac-ft	acre-feet
ACFT	acre-feet
ACRS	advisory committee on reactor safeguards
ACSR	aluminum-clad steel reinforced
ADFGR	Alaska Department of Fish and Game Restoration
AEA	Atomic Energy Act
AEC	U.S. Atomic Energy Commission
AHD	American Heritage Dictionary
agl	above ground level
ALA	American Lifelines Alliance
ALARA	as low as reasonably achievable
AMUD	Acton Municipal Utility District
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
AOO	anticipated operational occurrences
APE	areas of potential effect
APWR	Advanced Pressurized Water Reactor

ACRONYMS AND ABBREVIATIONS

ARLIS	Alaska Resources Library and Information Services
ARRS	airborne radioactivity removal system
AS	ancillary services
ASCE	American Society of Civil Engineers
AVT	all volatile treatment
AWG	American wire gauge
BAT	best available technology
bbl	barrel
BC	Business Commercial
BDTF	Blowdown Treatment Facility
BEA	U.S. Bureau of Economic Analysis
BEG	U.S. Bureau of Economic Geology
bgs	below ground surface
BLS	U.S. Bureau of Labor Statistics
BMP	best management practice
BOD	Biologic Oxygen Demand
BOP	Federal Bureau of Prisons
BRA	Brazos River Authority
bre	below reference elevation
BRM	Brazos River Mile
BSII	Big Stone II
BTI	Breakthrough Technologies Institute
BTS	U.S. Bureau of Transportation Statistics
BTU	British thermal units
BUL	Balancing Up Load

ACRONYMS AND ABBREVIATIONS

BW	Business Week
BWR	boiling water reactor
CAA	Clean Air Act
CBA	cost-benefit analysis
CBD	Central Business District
CCI	Chambers County Incinerator
CCTV	closed-circuit television
CCW	component cooling water
CCWS	component cooling water system
CDC	Centers for Disease Control and Prevention
CDF	Core Damage Frequency
CDR	Capacity, Demand, and Reserves
CEC	California Energy Commission
CEDE	committed effective dose equivalent
CEED	Center for Energy and Economic Development
CEQ	Council on Environmental Quality
CESQG	conditionally exempt small quantity generator
CFC	chlorofluorocarbon
CFE	Comisin Federal de Electricidad
CFR	Code of Federal Regulations
cfs	cubic feet per second
CFS	chemical treatment system
CG	cloud-to-ground
CGT	Cogeneration Technologies
CHL	Central Hockey League

ACRONYMS AND ABBREVIATIONS

CO	carbon monoxide
CO_2	carbon dioxide
COD	Chemical Oxygen Demand
COL	combined construction and operating license
COLA	combined construction and operating license application
CORMIX	Cornell Mixing Zone Expert System
CPI	Consumer Price Index
CPP	continuing planning process
CPS	condensate polishing system
CPNPP	Comanche Peak Nuclear Power Plant
CPSES	Comanche Peak Steam Electric Station
CRDM	control rod drive mechanism cooling system
CRP	Clean Rivers Program
CS	containment spray
Cs-134	cesium-134
Cs-137	cesium 137
CST	Central Standard Time
CST	condensate storage tank
CT	completion times
CT	cooling tower
cu ft	cubic feet
C/V	containment vessel
CVCS	chemical and volume control system
CVDT	containment vessel reactor coolant drain tank
CWA	Clean Water Act

ACRONYMS AND ABBREVIATIONS

CWS	circulating water system
DAW	dry active waste
dBA	decibels
DBA	design basis accident
DBH	diameter at breast height
DC	direct current
DCD	Design Control Document
DDT	dichlorodiphenyltrichloroethane
DF	decontamination factor
DFPS	Department of Family and Protective Services
DFW	Dallas/Fort Worth
DO	dissolved oxygen
DOE	U.S. Department of Energy
DOL	Department of Labor
DOT	U.S. Department of Transportation
DPS	Department of Public Safety
D/Q	deposition
DSHS	Department of State Health Services
DSM	Demand Side Management
DSN	discharge serial numbers
DSWD	Demand Side Working Group
DVSP	Dinosaur Valley State Park
DWS	demineralized water system
DWST	demineralized water storage tank
E	Federally Endangered

ACRONYMS AND ABBREVIATIONS

EA	Environmental Assessment
EAB	exclusion area boundary
E. coli	Escherichia coli
EDC	Economic Development Corp.
EDE	effective dose equivalent
EEI	Edison Electric Institute
EERE	Energy Efficiency and Renewable Energy
EFH	Energy Future Holdings Corporation
EFW	energy from waste
EIA	Energy Information Administration
EIS	Environmental Impact Statement
EJ	environmental justice
ELCC	Effective Load-Carrying Capacity
EMFs	electromagnetic fields
EO	Executive Order
EOF	emergency operation facility
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
ER	Environmental Report
ERA	Environmental Resource Associates
ERCOT	Electric Reliability Council of Texas
ESA	Endangered Species Act
ESP	Early Site Permit
ESRP	Environmental Standard Review Plan

ACRONYMS AND ABBREVIATIONS

ESW	essential service cooling water
ESWS	essential service water system
F&N	Freese & Nicholas, Inc.
FAA	U.S. Federal Aviation Administration
FAC	flow-accelerated corrosion
FBC	fluidized bed combustion
FCT	Fuel Cell Today
FEMA	Federal Emergency Management Agency
FERC	Federal Energy Regulatory Commission
FFCA	Federal Facilities Compliance Act
FLMNH	Florida Museum of Natural History
FM	farm-to-market
FP	fire protection
FPL	Florida Power and Light
FPS	fire protection system
FPSC	Florida Public Service Commission
FR	Federal Register
FSAR	Final Safety Analysis Report
FSL	Forecast Systems Laboratory
ft	feet
FWAT	flow weighted average temperature
FWCOC	Fort Worth Chamber of Commerce
FWS	U.S. Fish and Wildlife Service
gal	gallon
GAM	General Area Monitoring

ACRONYMS AND ABBREVIATIONS

GAO	U.S. General Accountability Office
GDEM	Governor's Division of Emergency Management
GEA	Geothermal Energy Association
GEIS	Generic Environmental Impact Statement
GEOL	overall geological
GFD	ground flash density
GIS	gas-insulated switchgear
GIS	Geographic Information System
GMT	Greenwich Mean Time
gpd	gallons per day
gph	gallons per hour
gpm	gallons per minute
gps	gallons per second
GRCVB	Glen Rose, Texas Convention and Visitors Bureau
GST	gas surge tank
GTC	Gasification Technologies Conference
GTG	gas turbine generators
GWMS	gaseous waste management system
H-3	radioactive tritium
HC	Heavy Commercial
HCl	Hydrochloric Acid
HCP	Ham Creek Park
HEM	hexane extractable material
HEPA	high efficiency particulate air
HIC	high integrity container

ACRONYMS AND ABBREVIATIONS

HL	high-level
HNO_3	Nitric Acid
hr	hour(s)
HRCQ	highway route-controlled quantity
H_2SO_4	Sulfuric Acid
HT	holdup tank
HTC	Historic Texas Cemetery
HUC	hydrologic unit code
HUD	U.S. Department of Housing and Urban Development
HVAC	heating, ventilating, and air-conditioning
I	Industrial
I-131	iodine-131
IAEA	International Atomic Energy Agency
I&C	instrumentation and control
IEC	Iowa Energy Center
IGCC	Integrated Gasification Combined Cycle
IH	Interim Holding
in	inch
INEEL	Idaho National Engineering and Environmental Laboratory
IOUs	investor-owned electric utilities
IPE	individual plant examination
ISD	Independent School District
ISFSI	independent spent fuel storage installation
ISO	independent system operator
ISO rating	International Standards Organization rating

ACRONYMS AND ABBREVIATIONS

ISU	Idaho State University
JAMA	Journal of the American Medical Association
K-40	potassium-40
KC	Keystone Center
JRB	Joint Reserve Base
km	kilometer
kVA	kilovolt-ampere
kWh	kilowatt hour
L	LARGE
LaaR	Load Acting as a Resource
LANL	Los Alamos National Laboratory
lb	pounds
LC	Light Commercial
LG	Lake Granbury
LL	low-level
LLD	lower limits of detection
LLMW	low-level mixed waste
LNG	liquid natural gas
LOCA	loss of coolant accident
LPSD	low-power and shutdown
LPZ	low population zone
LQG	large-quantity hazardous waste generators
LRS	load research sampling
LTSA	long term system assessment
Luminant	Luminant Generation Company LLC

ACRONYMS AND ABBREVIATIONS

LVW	low volume waste
LWA	Limited Work Authorization
LWMS	liquid waste management system
LWPS	liquid waste processing system
LWR	light water reactor
M	MODERATE
ma	milliamperes
MACCS2	Melcor Accident Consequence Code System
MCES	Main Condenser Evacuation System
Mcf	thousand cubic feet
MCPE	Market Clearing Price for Energy
MCR	main control room
MD-1	Duplex
MDA	minimum detected activity
MDCT	mechanical draft cooling tower
MEIs	maximally exposed individuals
MF	Multi-Family
mG	milliGauss
mg/l	milligrams per liter
mg/m^3	milligrams per cubic meter
MH	Manufactured Housing
MHI	Mitsubishi Heavy Industries
mi	mile
mi^2	square miles
MIT	Massachusetts Institute of Technology

ACRONYMS AND ABBREVIATIONS

MMbbl	million barrels
MMBtu	million Btu
MNES	Mitsubishi Nuclear Energy Systems Inc.
MOU	municipally-owned utility
MOV	motor operated valve
MOX	mixed oxide fuel
mph	miles per hour
MSDS	Materials Safety Data Sheets
msl	mean sea level
MSR	maximum steaming rate
MSW	municipal solid waste
MT	Main Transformer
MTU	metric tons of uranium
MW	megawatts
MW	monitoring wells
MWd	megawatt-days
MWd/MTU	megawatt–days per metric ton uranium
MWe	megawatts electrical
MWh	megawatt hour
MWS	makeup water system
MWt	megawatts thermal
NAAQS	National Ambient Air Quality Standards
NAPA	Natural Areas Preserve Association
NAP	National Academies Press
NAR	National Association of Realtors

ACRONYMS AND ABBREVIATIONS

NARM	accelerator-produced radioactive material
NAS	Naval Air Station
NASS	National Agricultural Statistics Service
NCA	Noise Control Act
NCDC	National Climatic Data Center
NCDENR	North Carolina Department of Environmental and Natural Resources
NCES	National Center for Educational Statistics
NCI	National Cancer Institute
NCTCOG	North Central Texas Council of Governments
ND	no discharge
NDCT	natural draft cooling towers
NEI	Nuclear Energy Institute
NELAC	National Environmental Laboratory Accreditation Conference
NEPA	National Environmental Policy Act
NERC	North American Electric Reliability Corporation/Council
NESC	National Electrical Safety Code
NESDIS	National Environmental Satellite, Data, and Information Service
NESW	non-essential service water cooling system
NESWS	non-essential service water system
NETL	National Energy Technology Laboratory
NHPA	National Historic Preservation Act
NHS	National Hurricane Center
NINI	National Institute of Nuclear Investigations
NIOSH	National Institute for Occupational Safety and Health

ACRONYMS AND ABBREVIATIONS

NIST	U.S. National Institute of Standards and Technology
NJCEP	NJ Clean Energy Program
NLDN	National Lightning Detection Network
NOAA	National Oceanic and Atmospheric Administration
NOAEC	no observable adverse effects concentration
NOI	Notice of Intent
NOIE	non-opt-in entities
NO_x	oxides of nitrogen
NP	Nacogdoches Power
NPDES	National Pollutant Discharge Elimination System
NPS	nonpoint source
NR	not required
NRC	U.S. Nuclear Regulatory Commission
NREL	U.S. National Renewable Energy Laboratory
NRHP	National Register of Historic Places
NRRI	National Regulatory Research Institute
NSPS	New Source Performance Standards
NSSS	nuclear steam supply system
NTAD	National Transportation Atlas Database
NVLAP	National Voluntary Laboratory Accreditation Program
NWI	National Wetlands Inventory
NWS	National Weather Service
NWSRS	National Wild and Scenic Rivers System
O_2	Oxygen
O_3	Ozone

ACRONYMS AND ABBREVIATIONS

ODCM	Off-site Dose Calculation Manual
OECD	Organization for Economic Co-operation and Development
O&M	operations and maintenance
ORNL	Oak Ridge National Laboratory
ORP	oxidation-reduction potential
OSHA	Occupational Safety and Health Act
OW	observation well
P&A	plugging and abandonment
PAM	primary amoebic meningoencephalitis
PD	Planned Development
PDL	Proposed for Delisting
PE	probability of exceedances
percent g	percent of gravity
PET	Potential Evapotranspiration
PFBC	pressurized fluidized bed combustion
PFD	Process Flow Diagram
PGA	peak ground acceleration
PGC	power generation company
PH	Patio Home
P&ID	piping and instrumentation diagram
PM	particulate matter
PM_{10}	particulate matter less than 10 microns diameter
$PM_{2.5}$	particulate matter less than 2.5 microns diameter
PMF	probable maximum flood
PMH	probable maximum hurricane

ACRONYMS AND ABBREVIATIONS

PMP	probable maximum precipitation
PMWP	probable maximum winter precipitation
PMWS	probable maximum windstorm
PPE	plant parameter envelope
ppm	parts per million
PPS	preferred power supply
PRA	probabilistic risk assessment
PSD	Prevention of Significant Deterioration (permit)
PSWS	potable and sanitary water system
PUC	Public Utility Commission
PUCT	Public Utility Commission of Texas
PURA	Public Utilities Regulatory Act
PWR	pressurized water reactors
QA	quality assurance
QC	quality control
QSE	qualified scheduling entities
R10	Single-Family Residential
R12	Single-Family Residential
R7	Single-Family Residential
R8.4	Single-Family Residential
RAT	Reserve Auxiliary Transformer
RB	reactor building
R/B	reactor building
RCDS	reactor coolant drain system
RCDT	reactor coolant drain tank

ACRONYMS AND ABBREVIATIONS

RCRA	Resource Conservation and Recovery Act
RCS	reactor coolant system
RDA	Radiosonde Database Access
REC	renewable energy credit
REIRS	Radiation Exposure Information and Reporting System
RELFRC	release fractions
rem	roentgen equivalent man
REMP	radiological environmental monitoring program
REP	retail electric providers
REPP	Renewable Energy Policy Project
RFI	Request for Information
RG	Regulatory Guide
RHR	residual heat removal
RIMS II	regional input-output modeling system
RMR	Reliability Must-Run
Rn_{222}	Radon-222
RO	reverse osmosis
ROI	region of interest
ROW	right of way
RPG	regional planning group
RRY	reactor reference year
RTHL	Recorded Texas Historic Landmarks
RTO	regional transmission organization
Ru-103	ruthenium-103
RW	test well

ACRONYMS AND ABBREVIATIONS

RWSAT	refueling waste storage auxiliary tank
RWST	refueling water storage tank
RY	reactor-year
S	SMALL
SACTI	Seasonal/Annual Cooling Tower Impact Prediction Code
SAL	State Archaeological Landmark
SAMA	severe accident mitigation alternative
SAMDA	severe accident mitigation design alternative
SB	Senate Bill
SCR	Squaw Creek Reservoir
SCDC	Somervell County Development Commission
scf	standard cubic feet
SCWD	Somervell County Water District
SDS	sanitary drainage system
SECO	State Energy Conservation Office
SER	Safety Evaluation Report
SERC	SERC Reliability Corporation
SERI	System Energy Resources, Inc.
SFPC	spent fuel pool cooling and cleanup system
SG	steam generator
SGBD	steam generator blow-down
SGBDS	steam generator blow-down system
SGs	steam generators
SGTR	steam generator tube rupture
SH	State Highway

ACRONYMS AND ABBREVIATIONS

SHPO	State Historic Preservation Office
SIP	State Implementation Plan
SMP	State Marketing Profiles
SMU	Southern Methodist University
SOP	Standard Operations Permit
SO_2	sulfur dioxide
SO_x	sulfur
SPCCP	Spill Prevention Control and Countermeasures Plan
SPP	Southwest Power Pool
SQG	small-quantity generators
sq mi	square miles
SRCC	Southern Regional Climate Center
SRP	Standard Review Plan
SRST	spent resin storage tank
SSAR	Site Safety Analysis Report
SSC	structures, systems, and components
SSI	Safe Shutdown Impoundment
SSURGO	Soil Survey Geographic
SWATS	Surface Water and Treatment System
SWMS	solid waste management system
SWPC	spent fuel pool cooling and cleanup system
SWP3	Storm Water Pollution Prevention Plan
SWS	service water system
SWWTS	sanitary wastewater treatment system
T	Federally Threatened

ACRONYMS AND ABBREVIATIONS

t	ton
TAC	technical advisory committee
TAC	Texas Administrative Code
TB	turbine building
Tc_{99}	Technetium-99
TCEQ	Texas Commission on Environmental Quality
TCPS	Texas Center for Policy Studies
TCR	transmission congestion rights
TCS	turbine component cooling water system
TCWC	Texas Cooperative Wildlife Collection
T&D	transmission and distribution utility
TDCJ	Texas Department of Criminal Justice
TDOH	Texas Department of Health
TDOT	Texas Department of Transportation
TDPS	Texas Department of Public Safety
TDS	total dissolved solids
TDSHS	Texas Department of State Health Services
TDSP	transmission and distribution service provider
TDWR	Texas Department of Water Resources
TEDE	total effective dose equivalent
TGLO	Texas General Land Office
TGPC	Texas Groundwater Protection Committee
TH	Townhome
THC	Texas Historical Commission
THPOs	tribal historic preservation officers

ACRONYMS AND ABBREVIATIONS

TIS	Texas Interconnected System
TLD	Thermoluminescence Dosemeter
TMDLs	total maximum daily loads
TMM	Texas Memorial Museum
TOs	Transmission Owners
TPDES	Texas Pollutant Discharge Elimination System
TPWD	Texas Parks and Wildlife Department
tpy	tons per year
TRAGIS	Transportation Routing Analysis Geographic Information System
TRB	Transportation Research Board
TRC	total recordable cases
TRE	Trinity Railway Express
TSC	technical support center
TSD	thunderstorm days per year
TSD	treatment, storage, and disposal
TSDC	Texas State Data Center
TSHA	Texas State Historical Association
TSP	transmission service provider
TSWQS	Texas Surface Water Quality Standards
TSS	total suspended sediment
TTS	The Transit System (Glen Rose)
TUGC	Texas Utilities Generating Company
TUSI	Texas Utilities Services Inc.
TWC	Texas Workforce Commission
TWDB	Texas Water Development Board

ACRONYMS AND ABBREVIATIONS

TWR	Texas Weather Records
TWRI	Texas Water Resources Institute
TxDOT	Texas Department of Transportation
TXU	Texas Utilities Corporation
TXU DevCo	TXU Generation Development Company LLC
UC	University of Chicago
UFC	uranium fuel cycle
UHS	Ultimate Heat Sink
UIC	Uranium Information Center
UO_2	uranium dioxide
USACE	U.S. Army Corps of Engineers
US-APWR	(MHI) United States-advanced pressurized water reactor
USC	U.S. Census
USCA	United States Court of Appeals
USDA	U.S. Department of Agriculture
USDOT	U.S. Department of Transportation
USEPA	United States Environmental Protection Agency
USFWS	United States Fish and Wildlife Service
USGS	U.S. Geological Survey
USHCN	United States Historical Climatology Network
USHR	U.S. House of Representatives
USNPS	U.S. National Park Service
UTC	Universal Time Coordinated
UV	ultra-violet
VCIS	Ventilation Climate Information System

ACRONYMS AND ABBREVIATIONS

VCT	volume control tank
VERA	Virtus Energy Research Associates
VFD	Volunteer Fire Department
VOC	volatile organic compound
VRB	variable
WB	Weather Bureau
WBR	Wheeler Branch Reservoir
WDA	work development area
WDFW	Washington Department of Fish and Wildlife
weight percent	wt. percent
WHT	waste holdup tank
WMT	waste monitor tank
WNA	World Nuclear Association
WPP	Watershed Protection Plan
WQMP	Water Quality Management Plan
WRE	Water Resource Engineers, Inc.
WWS	wastewater system
WWTP	wastewater treatment plant
yr	year

CHAPTER 7

ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

7.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

This chapter assesses the environmental impacts of postulated accidents involving radioactive materials at the Comanche Peak Nuclear Power Plant (CPNPP) Units 3 and 4 site. The chapter is divided into four sections that address the analysis of postulated accidents as follows:

- Design Basis Accidents (Section 7.1).
- Severe Accidents (Section 7.2).
- Severe Accident Mitigation Alternatives (Section 7.3).
- Transportation Accidents (Section 7.4).

7.1 DESIGN BASIS ACCIDENTS

This section reviews and analyzes the design basis accidents (DBAs), as identified in NUREG-1555,"Standard Review Plans for Environmental Reviews for Nuclear Power Plants," to demonstrate that reactors can be operated at the Comanche Peak Nuclear Power Plant (CPNPP) Units 3 and 4 site without undue risk to the health and safety of the public.

7.1.1 SELECTION OF ACCIDENTS

The DBAs considered in this section come from Chapter 15 of the Mitsubishi Heavy Industries (MHI) U.S. Advanced Pressurized Water Reactor (US-APWR) design control document (DCD). Table 7.1-1 lists the NUREG-1555 DBAs that have the potential to release radioactivity to the environment and shows the NUREG-0800 "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants" section numbers and accident descriptions, as well as the corresponding accidents as defined in the DCD. The DBAs cover a spectrum of events, including those of relatively greater probability of occurrence and those that are less probable but have greater severity. The radiological consequences of the accidents listed in Table 7.1-1 are assessed to demonstrate that additional units can be sited and operated at the CPNPP site without undue risk to the health and safety of the public.

7.1.2 EVALUATION METHODOLOGY

The DCD presents the radiological consequences for the accidents identified in Table 7.1-1. The DCD design basis analyses are updated with CPNPP site data to demonstrate that the DCD analyses are bounding for the CPNPP site. The base scenario for each accident is that some quantity of activity is released at the accident location inside a building, and this activity is eventually released to the environment. The transport of activity within the plant is independent of the site and specific to the US-APWR design. Details about the methodologies and assumptions pertaining to each of the accidents, such as activity release pathways and credited mitigation features, are provided in Chapter 15 of the DCD. The postulated loss-of-coolant accidents (LOCA) are expected to more closely approach 10 Code of Federal Regulations (CFR) 50.34 limits than the other DBAs of greater probability of occurrence but lesser magnitude of activity releases. For these other accidents, the calculated doses are compared to the acceptance criteria in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and NUREG-0800 to demonstrate that the consequences of the postulated accidents are acceptable.

The dose to an individual located at the exclusion area boundary (EAB) or the low population zone (LPZ) is calculated based on the amount of activity released to the environment, the atmospheric dispersion of the activity during the transport from the release point to the off-site location, the breathing rate of the individual at the off-site location, and activity-to-dose conversion factors. The breathing rate of the individual at the off-site location specified in Table 15.0-13 of the DCD is used for analysis. The only site-specific parameter is atmospheric dispersion. Site-specific doses are obtained by adjusting the DCD doses to reflect site-specific atmospheric dispersion factors (χ/Q values).

The accident analyses presented in DCD Chapter 15 use conservative assumptions as specified in RG 1.183 to perform bounding safety analyses that substantially overstate the environmental effects of the identified accidents. The DCD Chapter 15 design basis analyses also use conservative assumptions for the core and coolant source terms, the types of radioactive materials released, and the release paths to the environment. Some of the major conservatisms include:

- Conservative reactor power level.
- Conservative design basis source terms.
- Conservative use of large reactivity coefficients for some accidents.
- Conservative assumptions on fuel defects or core damage levels.
- Conservative plant initial conditions.
- Conservative delays in safety system actuation (or no credit for safety systems).
- Conservative assumptions related to system and/or component failures.
- Conservative assumption related to the loss of off-site power.
- Conservative assumption of instantaneous releases to the environment for some accidents.
- Conservative 95th percentile χ/Q values.

These conservative assumptions are maintained for the dose assessments presented in this section, except that Environmental Report (ER) doses are based on the 50th percentile site-specific χ/Q values reflecting more realistic meteorological conditions consistent with NUREG-1555. Considering that the χ/Q values for the CPNPP site are bounded by the DCD values, site-specific total effective dose equivalent (TEDE) accident doses are bounded by the DCD doses. The site-specific accident doses are therefore below the regulatory dose acceptance criteria.

The χ/Q values are calculated using the guidance in NRC RG 1.145, Revision 1, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” with site-specific meteorological data. As indicated in Subsection 2.7.3, the NRC RG 1.145 methodology is implemented in the NRC-sponsored PAVAN computer program. This program computes χ/Q values at the EAB and the LPZ for each combination of wind speed, and it calculates atmospheric stability for each of 16 downwind direction sectors. It then calculates overall (nondirection-specific) χ/Q values. For a given location, either the EAB or the LPZ, the 0 – 2-hour (hr) χ/Q value is the 50th percentile overall value calculated by PAVAN. For the LPZ, the χ/Q values for all subsequent times are calculated by logarithmic interpolation between the 50th percentile χ/Q value and the annual average χ/Q value. Releases are assumed to be at ground level, and the shortest distances between the power block and the off-site locations are selected to conservatively maximize the χ/Q values.

The accident doses are expressed as TEDE, consistent with 10 CFR 50.34. The TEDE consists of the sum of the committed effective dose equivalent (CEDE) from inhalation and the effective dose equivalent (EDE) from external exposure. The CEDE is determined using the dose conversion factors in Federal Guidance Report 11 (EPA 1988), while the EDE is based on the dose conversion factors in Federal Guidance Report 12 (EPA 1993). Appendix 15A of the DCD provides information on the methodologies used to calculate CEDE and EDE values for the postulated accidents. As indicated in NRC RG 1.183, the dose conversion factors in U.S. Environmental Protection Agency (EPA) Federal Guidance Reports 11 and 12 (EPA 1988) (EPA 1993) used for the postulated accidents are acceptable to the NRC staff.

7.1.3 SOURCE TERMS

The DBA source terms, methodology, and assumptions in the DCD are based on the alternative source term methods outlined in NRC RG 1.183. The activity releases and doses are based on 102 percent of the rated core thermal power of 4451 megawatts thermal (MWt). The US-APWR core fission product inventory was developed using the ORIGEN computer code as described in Subsection 15.0.3.2 of the DCD. The parameters and models that form the basis of the radiological consequences and analyses for the postulated accidents are presented in Appendix 15A of the DCD. The time-dependent isotopic activities released to the environment from each of the evaluated accidents are provided in Tables 7.1-2, 7.1-3, 7.1-4, 7.1-5, 7.1-6, 7.1-7, 7.1-8, 7.1-9, and 7.1-10.

7.1.4 RADIOLOGICAL CONSEQUENCES

The Section 7.1 DBA doses are evaluated on the basis of more realistic meteorological conditions than those in DCD Chapter 15. For each of the accidents identified in Table 7.1-1, the site-specific dose for a given time interval is calculated by multiplying the DCD dose by the ratio of the site χ/Q values, presented in Table 2.7-121, to the DCD χ/Q values. The time-dependent DCD χ/Q values, time-dependent site χ/Q values, and their ratios are shown in Table 7.1-11. As all site χ/Q values are bounded by DCD χ/Q values, site-specific doses for all accidents are also bounded by DCD doses. The total site doses are summarized in Table 7.1-12, based on individual accident doses presented in Tables 7.1-13, 7.1-14, 7.1-15, 7.1-16, 7.1-17, 7.1-18, 7.1-19, 7.1-20, and 7.1-21. For each accident, the EAB dose shown is for the 2-hr period that yields the maximum dose, in accordance with NRC RG 1.183.

The results of the CPNPP Units 3 and 4 analysis contained in the referenced tables demonstrate that all accident doses meet the site acceptance criteria of 10 CFR 50.34. The acceptance criteria in 10 CFR 50.34 apply to accidents with an exceedingly low probability of occurrence and a low risk of public exposure to radiation. For events with a higher probability of occurrence, the dose limits are taken from NRC RG 1.183. Although conformance to these dose limits is not required for this environmental impact analysis, the limits are shown in the tables for comparison purposes.

The TEDE dose limits shown in Tables 7.1-12, 7.1-13, 7.1-14, 7.1-15, 7.1-16, 7.1-17, 7.1-18, 7.1-19, 7.1-20, and 7.1-21 are from NRC RG 1.183, Table 6, for all formally designated accidents, except the feedwater system pipe break inside or outside containment, discussed in NUREG-0800 Subsection 15.2.8; the reactor coolant pump (RCP) shaft break, discussed in NUREG-0800 Subsection 15.3.4; and the failure of small lines carrying primary coolant outside containment, discussed in NUREG-0800 Subsection 15.6.2. Although NRC RG 1.183 does not address these three accidents, NUREG-0800 indicates that the dose limit is a “small fraction” or 10 percent of the 10 CFR 100 guideline of 25 roentgen equivalent man (rem), meaning a limit of 2.5 rem for these accidents. All doses are within the acceptance criteria.

7.1.5 REFERENCES

(EPA 1988) U.S. Environmental Protection Agency. *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*. U.S. Environmental Protection Agency Federal Guidance Report 11, EPA-520/1-88-020. Washington, DC. September 1988.

(EPA 1993) U.S. Environmental Protection Agency. *External Exposure to Radionuclides in Air, Water, and Soil*. U.S. Environmental Protection Agency Federal Guidance Report 12, EPA-402-R-93-081. Washington, DC. September 1993.

TABLE 7.1-1 (Sheet 1 of 2)
SELECTION OF ACCIDENTS

SRP/DCD Subsection	SRP Description	DCD Description	Reference Activity Releases Table	Reference Radiological Consequences Table	Identified in NUREG-1555, Section 7.1 Appendix A	Comment
15.1.5A	Steam System Piping Failures Inside and Outside Containment (PWR)	Steam System Piping Failures Inside and Outside Containment Pre-Transient Iodine Spike Transient-Initiated Iodine Spike	 7.1-2 7.1-3	 7.1-13 7.1-14	Yes	Addressed in DCD Subsection 15.1.5.
15.2.8	Feedwater System Pipe Break Inside and Outside Containment (PWR)	Feedwater System Pipe Break Inside and Outside Containment	(a)	(a)	Yes	(a)
15.3.3[b]	Reactor Coolant Pump Rotor Seizure	Reactor Coolant Pump Rotor Seizure	7.1-4	7.1-15	Yes	Addressed in DCD Subsection 15.3.3.
15.3.4[b]	Reactor Coolant Pump Shaft Break	Reactor Coolant Pump Shaft Break	(c)	(c)	Yes	(c)
15.4.8[d]	Spectrum of Rod Ejection Accidents (PWR)	Spectrum of Rod Ejection Accidents	7.1-5	7.1-16	No	Evaluated for completeness. Addressed in DCD Subsection 15.4.8
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	7.1-6	7.1-17	Yes	Addressed in DCD Subsection 15.6.2.
15.6.3	Radiological Consequences of Steam Generator Tube Failure (PWR)	Radiological Consequences of Steam Generator Tube Failure Pre-Transient Iodine Spike Transient-Initiated Iodine Spike	 7.1-7 7.1-8	 7.1-18 7.1-19	Yes	Addressed in DCD Subsection 15.6.3.

TABLE 7.1-1 (Sheet 2 of 2)
SELECTION OF ACCIDENTS

SRP/DCD Subsection	SRP Description	DCD Description	Reference Activity Releases Table	Reference Radiological Consequences Table	Identified in NUREG-1555, Section 7.1 Appendix A	Comment
15.6.5A	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	7.1-9	7.1-20	Yes	Addressed in DCD Subsection 15.6.5.
15.6.5B	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	7.1-9	7.1-20	Yes	Addressed in DCD Subsection 15.6.5.
15.7.4	Radiological Consequences of Fuel Handling Accidents	Fuel Handling Accident	7.1-10	7.1-21	Yes	Addressed in DCD Subsection 15.7.4.

a) As discussed in DCD Subsection 15.2.8.5, the radiological consequences of a Feedwater System Pipe Break event are bounded by the main steam line break accident evaluated in DCD Subsection 15.1.5.

b) These sections for Reactor Coolant Pump (RCP) Rotor Seizure and Reactor Coolant Pump Shaft Break are addressed in NUREG-0800 (SRP) 15.3.3-15.3.4.

c) The analysis performed for the RCP rotor seizure transient (DCD Subsection 15.3.3) bounds the response and results for the RCP shaft break as discussed in DCD Subsection 15.3.4.

d) The source of this accident is Subsection 15.4.8 of NUREG-0800. This event is not included in NUREG-1555, Section 7.1 Appendix A, "Design Basis Accidents Included in Section 15 of the Standard Review Plan."

TABLE 7.1-2
TIME DEPENDENT RELEASED ACTIVITY DURING STEAM SYSTEM PIPING FAILURE
(PRE-TRANSIENT IODINE SPIKE)

	Activity Release (Ci)				
Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	3.21E+01	2.40E+01	0.00E+00	0.00E+00	5.61E+01
Kr-85m	3.56E-01	8.77E-02	0.00E+00	0.00E+00	4.43E-01
Kr-87	9.12E-02	1.13E-03	0.00E+00	0.00E+00	9.23E-02
Kr-88	5.10E-01	6.46E-02	0.00E+00	0.00E+00	5.74E-01
Xe-133	1.07E+02	7.75E+01	0.00E+00	0.00E+00	1.85E+02
Xe-135	4.38E+00	3.39E+00	0.00E+00	0.00E+00	7.78E+00
Iodines					
I-131	1.72E+01	7.25E+00	0.00E+00	0.00E+00	2.44E+01
I-132	6.18E+00	1.66E-01	0.00E+00	0.00E+00	6.35E+00
I-133	2.79E+01	9.03E+00	0.00E+00	0.00E+00	3.69E+01
I-134	3.49E+00	1.01E-03	0.00E+00	0.00E+00	3.49E+00
I-135	1.62E+01	2.73E+00	0.00E+00	0.00E+00	1.89E+01
Alkali Metals					
Rb-86	8.64E-02	1.62E-03	0.00E+00	0.00E+00	8.80E-02
Cs-134	8.80E+00	1.68E-01	0.00E+00	0.00E+00	8.97E+00
Cs-136	2.32E+00	4.33E-02	0.00E+00	0.00E+00	2.37E+00
Cs-137	5.01E+00	9.56E-02	0.00E+00	0.00E+00	5.11E+00
TOTAL	2.32E+02	1.25E+02	0.00E+00	0.00E+00	3.56E+02

Notes:

1. Data obtained from DCD Table 15A-27.

TABLE 7.1-3
TIME DEPENDENT RELEASED ACTIVITY DURING STEAM SYSTEM PIPING FAILURE
(TRANSIENT-INITIATED IODINE SPIKE)

	Activity Release (Ci)				
Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	3.21E+01	2.40E+01	0.00E+00	0.00E+00	5.61E+01
Kr-85m	3.56E-01	8.77E-02	0.00E+00	0.00E+00	4.43E-01
Kr-87	9.12E-02	1.13E-03	0.00E+00	0.00E+00	9.23E-02
Kr-88	5.10E-01	6.46E-02	0.00E+00	0.00E+00	5.74E-01
Xe-133	1.08E+02	8.03E+01	0.00E+00	0.00E+00	1.88E+02
Xe-135	7.61E+00	1.33E+01	0.00E+00	0.00E+00	2.09E+01
Iodines					
I-131	5.05E+01	6.50E+01	0.00E+00	0.00E+00	1.16E+02
I-132	9.89E+00	1.49E+00	0.00E+00	0.00E+00	1.14E+01
I-133	7.65E+01	8.09E+01	0.00E+00	0.00E+00	1.57E+02
I-134	3.77E+00	9.11E-03	0.00E+00	0.00E+00	3.78E+00
I-135	3.77E+01	2.45E+01	0.00E+00	0.00E+00	6.21E+01
Alkali Metals					
Rb-86	8.64E-02	1.62E-03	0.00E+00	0.00E+00	8.80E-02
Cs-134	8.80E+00	1.68E-01	0.00E+00	0.00E+00	8.97E+00
Cs-136	2.32E+00	4.33E-02	0.00E+00	0.00E+00	2.37E+00
Cs-137	5.01E+00	9.56E-02	0.00E+00	0.00E+00	5.11E+00
TOTAL	3.43E+02	2.90E+02	0.00E+00	0.00E+00	6.33E+02

Notes:

1. Data obtained from DCD Table 15A-26.

TABLE 7.1-4
TIME DEPENDENT RELEASED ACTIVITY DURING RCP ROTOR SEIZURE

	Activity Release (Ci)				
Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	1.12E+02	8.40E+01	0.00E+00	0.00E+00	1.96E+02
Kr-85m	6.40E+02	1.58E+02	0.00E+00	0.00E+00	7.98E+02
Kr-87	5.02E+02	6.21E+00	0.00E+00	0.00E+00	5.08E+02
Kr-88	1.37E+03	1.74E+02	0.00E+00	0.00E+00	1.55E+03
Xe-133	6.87E+03	4.96E+03	0.00E+00	0.00E+00	1.18E+04
Xe-135	1.61E+03	7.67E+02	0.00E+00	0.00E+00	2.37E+03
Iodines					
I-131	8.81E+01	2.32E+02	0.00E+00	0.00E+00	3.20E+02
I-132	1.94E+01	8.35E+00	0.00E+00	0.00E+00	2.77E+01
I-133	9.85E+01	2.17E+02	0.00E+00	0.00E+00	3.15E+02
I-134	6.46E+00	1.10E-01	0.00E+00	0.00E+00	6.57E+00
I-135	6.38E+01	9.16E+01	0.00E+00	0.00E+00	1.55E+02
Alkali Metals					
Rb-86	3.23E-02	8.66E-02	0.00E+00	0.00E+00	1.19E-01
Cs-134	3.24E+00	8.78E+00	0.00E+00	0.00E+00	1.20E+01
Cs-136	8.72E-01	2.33E+00	0.00E+00	0.00E+00	3.21E+00
Cs-137	1.84E+00	5.00E+00	0.00E+00	0.00E+00	6.84E+00
TOTAL	1.14E+04	6.71E+03	0.00E+00	0.00E+00	1.81E+04

Notes:

1. Data obtained from DCD Table 15A-30.

TABLE 7.1-5
TIME DEPENDENT RELEASED ACTIVITY DURING ROD EJECTION ACCIDENT

	Activity Release (Ci)				
Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	2.63E+02	2.50E+02	1.90E+02	1.63E+03	2.33E+03
Kr-85m	3.59E+03	9.58E+02	9.86E+00	0.00E+00	4.56E+03
Kr-87	2.81E+03	3.50E+01	0.00E+00	0.00E+00	2.85E+03
Kr-88	7.70E+03	1.02E+03	2.05E+00	0.00E+00	8.72E+03
Xe-133	3.81E+04	3.46E+04	2.11E+04	4.22E+04	1.36E+05
Xe-135	9.31E+03	5.32E+03	5.40E+02	2.81E+00	1.52E+04
Iodines					
I-131	5.82E+02	7.17E+02	2.58E+02	7.79E+02	2.34E+03
I-132	4.62E+02	3.93E+01	1.40E-02	0.00E+00	5.01E+02
I-133	1.12E+03	1.06E+03	1.13E+02	1.13E+01	2.30E+03
I-134	4.95E+02	5.15E-01	0.00E+00	0.00E+00	4.95E+02
I-135	8.75E+02	4.39E+02	6.60E+00	4.00E-03	1.32E+03
Alkali Metals					
Rb-86	4.16E-01	9.65E-02	0.00E+00	0.00E+00	5.13E-01
Cs-134	4.15E+01	9.79E+00	1.01E-03	0.00E+00	5.13E+01
Cs-136	1.13E+01	2.60E+00	1.00E-06	0.00E+00	1.39E+01
Cs-137	2.36E+01	5.57E+00	0.00E+00	0.00E+00	2.92E+01
TOTAL	6.53E+04	4.45E+04	2.22E+04	4.46E+04	1.77E+05

Notes:

1. Data obtained from DCD Table 15A-31.

TABLE 7.1-6
TIME DEPENDENT RELEASED ACTIVITY DURING FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT

Activity Release (Ci)

Nuclide	0-8hr
Noble Gases	
Kr-85	6.84E+02
Kr-85m	1.25E+01
Kr-87	7.05E+00
Kr-88	2.26E+01
Xe-133	2.32E+03
Xe-135	7.70E+01
Iodines	
I-131	1.72E+02
I-132	7.98E+01
I-133	2.93E+02
I-134	4.33E+01
I-135	1.85E+02
TOTAL	3.90E+03

Notes:

1. Data obtained from DCD Table 15A-33.
2. The activity is released within the first eight hours.

TABLE 7.1-7
TIME DEPENDENT RELEASED ACTIVITY DURING STEAM GENERATOR TUBE RUPTURE
(PRE-TRANSIENT IODINE SPIKE)

	Activity Release (Ci)				
Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	3.43E+03	4.64E+01	2.06E+02	1.59E+03	5.27E+03
Kr-85m	6.17E+01	9.70E-02	8.00E-03	0.00E+00	6.18E+01
Kr-87	3.40E+01	0.00E+00	0.00E+00	0.00E+00	3.40E+01
Kr-88	1.11E+02	6.00E-02	1.00E-02	0.00E+00	1.11E+02
Xe-133	1.16E+04	1.44E+02	5.06E+02	9.44E+02	1.32E+04
Xe-135	3.75E+02	2.18E+00	6.70E-01	0.00E+00	3.78E+02
Iodines					
I-131	4.18E+02	1.81E+00	0.00E+00	0.00E+00	4.20E+02
I-132	2.09E+02	3.92E-02	0.00E+00	0.00E+00	2.09E+02
I-133	7.16E+02	2.24E+00	0.00E+00	0.00E+00	7.18E+02
I-134	1.28E+02	6.00E-05	0.00E+00	0.00E+00	1.28E+02
I-135	4.61E+02	6.70E-01	0.00E+00	0.00E+00	4.62E+02
Alkali Metals					
Rb-86	4.54E-03	5.44E-04	0.00E+00	0.00E+00	5.09E-03
Cs-134	4.63E-01	5.63E-02	0.00E+00	0.00E+00	5.19E-01
Cs-136	1.22E-01	1.45E-02	0.00E+00	0.00E+00	1.37E-01
Cs-137	2.64E-01	3.21E-02	0.00E+00	0.00E+00	2.96E-01
TOTAL	1.76E+04	1.98E+02	7.12E+02	2.53E+03	2.10E+04

Note:

1. Data obtained from DCD Table 15A-29.

TABLE 7.1-8
TIME DEPENDENT RELEASED ACTIVITY DURING STEAM GENERATOR TUBE RUPTURE
(TRANSIENT-INITIATED IODINE SPIKE)

	Activity Release (Ci)				
Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	3.43E+03	4.64E+01	2.06E+02	1.59E+03	5.27E+03
Kr-85m	6.17E+01	9.70E-02	8.00E-03	0.00E+00	6.18E+01
Kr-87	3.40E+01	0.00E+00	0.00E+00	0.00E+00	3.40E+01
Kr-88	1.11E+02	6.00E-02	1.00E-02	0.00E+00	1.11E+02
Xe-133	1.16E+04	1.45E+02	5.06E+02	9.44E+02	1.32E+04
Xe-135	3.70E+02	3.82E+00	6.70E-01	0.00E+00	3.74E+02
Iodines					
I-131	1.10E+02	1.03E+01	0.00E+00	0.00E+00	1.20E+02
I-132	5.24E+01	2.12E-01	0.00E+00	0.00E+00	5.26E+01
I-133	1.87E+02	1.27E+01	0.00E+00	0.00E+00	2.00E+02
I-134	3.05E+01	1.06E-03	0.00E+00	0.00E+00	3.05E+01
I-135	1.19E+02	3.74E+00	0.00E+00	0.00E+00	1.23E+02
Alkali Metals					
Rb-86	4.54E-03	5.44E-04	0.00E+00	0.00E+00	5.09E-03
Cs-134	4.63E-01	5.63E-02	0.00E+00	0.00E+00	5.19E-01
Cs-136	1.22E-01	1.45E-02	0.00E+00	0.00E+00	1.37E-01
Cs-137	2.64E-01	3.21E-02	0.00E+00	0.00E+00	2.96E-01
TOTAL	1.61E+04	2.22E+02	7.12E+02	2.53E+03	1.96E+04

Note:

1. Data obtained from DCD Table 15A-28.

TABLE 7.1-9 (Sheet 1 of 3)
TIME DEPENDENT RELEASED ACTIVITY DURING LOSS-OF-COOLANT ACCIDENT

	Activity Release (Ci)				
Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	7.75E+02	1.74E+03	3.92E+03	3.35E+04	3.99E+04
Kr-85m	9.16E+03	4.37E+03	1.99E+02	0.00E+00	1.37E+04
Kr-87	3.54E+03	7.83E+01	0.00E+00	0.00E+00	3.62E+03
Kr-88	1.68E+04	3.68E+03	3.70E+01	0.00E+00	2.05E+04
Xe-133	1.26E+05	2.76E+05	4.93E+05	9.77E+05	1.87E+06
Xe-135	3.79E+04	4.05E+04	9.60E+03	4.41E+01	8.80E+04
Iodines					
I-131	1.42E+03	5.61E+02	1.85E+03	5.60E+03	9.43E+03
I-132	1.50E+03	1.01E+02	2.22E+02	2.48E+02	2.07E+03
I-133	2.67E+03	7.37E+02	8.09E+02	8.07E+01	4.30E+03
I-134	4.22E+02	1.84E-01	0.00E+00	0.00E+00	4.22E+02
I-135	1.95E+03	2.44E+02	4.67E+01	1.20E-01	2.24E+03
Alkali Metals					
Rb-86	1.44E+00	1.60E-02	0.00E+00	0.00E+00	1.45E+00
Cs-134	1.44E+02	1.62E+00	0.00E+00	0.00E+00	1.46E+02
Cs-136	3.90E+01	4.31E-01	0.00E+00	0.00E+00	3.94E+01
Cs-137	8.19E+01	9.21E-01	1.00E-03	0.00E+00	8.28E+01
Tellurium Group					
Sb-127	1.04E+01	1.26E-01	1.00E-05	0.00E+00	1.05E+01
Sb-129	1.99E+01	6.87E-02	0.00E+00	0.00E+00	2.00E+01
Te-127	1.04E+01	1.30E-01	0.00E+00	0.00E+00	1.05E+01
Te-127m	1.39E+00	1.80E-02	0.00E+00	0.00E+00	1.40E+00
Te-129	2.30E+01	1.12E-01	0.00E+00	0.00E+00	2.31E+01
Te-129m	4.75E+00	6.13E-02	0.00E+00	0.00E+00	4.81E+00
Te-131 m	1.36E+01	1.44E-01	0.00E+00	0.00E+00	1.37E+01
Te-132	1.41E+02	1.71E+00	1.00E-04	0.00E+00	1.43E+02

TABLE 7.1-9 (Sheet 2 of 3)
TIME DEPENDENT RELEASED ACTIVITY DURING LOSS-OF-COOLANT ACCIDENT

Activity Release (Ci)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Strontium and Barium					
Sr-89	4.74E+01	6.12E-01	0.00E+00	0.00E+00	4.80E+01
Sr-90	3.93E+00	5.10E-02	0.00E+00	0.00E+00	3.98E+00
Sr-91	5.01E+01	3.54E-01	1.00E-03	0.00E+00	5.05E+01
Sr-92	3.11E+01	4.95E-02	0.00E+00	0.00E+00	3.11E+01
Ba-139	1.96E+01	5.04E-03	0.00E+00	0.00E+00	1.96E+01
Ba-140	7.49E+01	9.53E-01	0.00E+00	0.00E+00	7.59E+01
Noble Metals					
Co-58	3.36E-03	4.50E-08	0.00E+00	0.00E+00	3.36E-03
Co-60	1.59E-02	2.00E-04	1.01E-06	0.00E+00	1.61E-02
Mo-99	9.57E+00	1.11E-01	1.00E-04	0.00E+00	9.68E+00
Tc-99m	8.50E+00	1.04E-01	1.00E-04	0.00E+00	8.60E+00
Ru-103	7.62E+00	9.83E-02	1.01E-04	0.00E+00	7.72E+00
Ru-105	3.14E+00	1.12E-02	0.00E+00	0.00E+00	3.15E+00
Ru-106	2.67E+00	3.46E-02	0.00E+00	0.00E+00	2.70E+00
Rh-105	4.61E+00	5.41E-02	0.00E+00	0.00E+00	4.67E+00
Lanthanides					
Y-90	7.44E-02	5.12E-03	6.06E-06	0.00E+00	7.96E-02
Y-91	6.00E-01	8.54E-03	0.00E+00	0.00E+00	6.09E-01
Y-92	4.13E+00	1.04E-01	0.00E+00	0.00E+00	4.24E+00
Y-93	5.90E-01	4.32E-03	0.00E+00	0.00E+00	5.94E-01
Zr-95	7.55E-01	9.76E-03	0.00E+00	0.00E+00	7.65E-01
Zr-97	6.65E-01	6.12E-03	0.00E+00	0.00E+00	6.71E-01
Nb-95	7.60E-01	9.85E-03	1.01E-05	0.00E+00	7.69E-01
La-140	1.76E+00	1.43E-01	2.02E-04	0.00E+00	1.90E+00
La-141	4.25E-01	1.29E-03	0.00E+00	0.00E+00	4.27E-01
La-142	2.01E-01	7.07E-05	0.00E+00	0.00E+00	2.01E-01
Pr-143	6.74E-01	8.91E-03	1.00E-05	0.00E+00	6.83E-01

TABLE 7.1-9 (Sheet 3 of 3)
TIME DEPENDENT RELEASED ACTIVITY DURING LOSS-OF-COOLANT ACCIDENT

	Activity Release (Ci)				
Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Nd-147	2.80E-01	3.55E-03	0.00E+00	0.00E+00	2.83E-01
Am-241	7.51E-05	9.77E-07	0.00E+00	0.00E+00	7.60E-05
Cm-242	1.86E-02	2.41E-04	0.00E+00	0.00E+00	1.88E-02
Cm-244	2.26E-03	2.93E-05	0.00E+00	0.00E+00	2.29E-03
Cerium Group					
Ce-141	1.78E+00	2.29E-02	0.00E+00	0.00E+00	1.80E+00
Ce-143	1.63E+00	1.78E-02	0.00E+00	0.00E+00	1.65E+00
Ce-144	1.35E+00	1.75E-02	0.00E+00	0.00E+00	1.36E+00
Np-239	1.85E+01	2.16E-01	1.00E-05	0.00E+00	1.87E+01
Pu-238	5.30E-03	6.88E-05	0.00E+00	0.00E+00	5.37E-03
Pu-239	4.00E-04	5.19E-06	0.00E+00	0.00E+00	4.05E-04
Pu-240	6.28E-04	8.14E-06	1.01E-08	0.00E+00	6.36E-04
Pu-241	1.39E-01	1.81E-03	0.00E+00	0.00E+00	1.41E-01
TOTAL	2.03E+05	3.28E+05	5.09E+05	1.02E+06	2.06E+06

Note:

1. Data obtained from DCD Table 15A-25.

TABLE 7.1-10
TIME DEPENDENT RELEASED ACTIVITY DURING FUEL HANDLING ACCIDENT

	Activity Release (Ci)
Nuclide	0-8 hr
Noble Gases	
Kr-85	1.20E+03
Kr-85m	3.90E+02
Kr-87	5.98E-02
Kr-88	1.25E+02
Xe-133	9.90E+04
Xe-135	2.21E+04
Iodines	
I-131	3.67E+02
I-132	2.75E+02
I-133	2.31E+02
I-134	2.71E-06
I-135	3.80E+01
TOTAL	1.24E+05

Notes:

1. Data obtained from DCD Table 15A-32.
2. All radioactivity is released to the environment within a 2-hr period with no cloud depletion by ground deposition during transport to the EAB and LPZ (DCD Subsection 15.7.4.1).

TABLE 7.1-11
ACCIDENT ATMOSPHERIC DISPERSION FACTORS

Location	Time	DCD χ/Q [(a)] (s/m^3)	Site χ/Q [(b)] (s/m^3)	χ/Q Ratio (Site/DCD)
EAB	0-2 hr [(c)]	5.0E-04	5.75E-05	1.15E-01
LPZ	0 – 8 hr	2.1E-04	3.32E-06	1.58E-02
	8 – 24 hr	1.3E-04	2.75E-06	2.12E-02
	24 – 96 hr	6.9E-05	1.83E-06	2.65E-02
	96 – 720 hr	2.8E-05	1.01E-06	3.61E-02

a) The χ/Q values used for the various postulated accident dose analyses are obtained from DCD Table 15.0-13 and Table 15A-17.

b) The site χ/Q values were obtained from Table 2.7-121. It is seen that the site χ/Q values are bounded by the DCD χ/Q values for all time intervals.

c) Nominally defined as the 0 to 2-hr interval, but is applied to the 2-hr interval having the highest activity releases in order to address 10 CFR 50.34 requirements.

TABLE 7.1-12
SUMMARY OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

DCD/SRP Section	Accident	Site Dose (rem TEDE)[a] EAB	LPZ	Limit[b]	Reference Radiological Consequences Table
15.1.5	Steam System Piping Failure				
	Pre-Transient Iodine Spike	0.03	0.01	25	7.1-13
	Transient-Initiated Iodine Spike	0.04	0.01	2.5	7.1-14
15.2.8	Feedwater System Pipe Break	(c)	(c)		
15.3.3	RCP Rotor Seizure	0.06	0.02	2.5	7.1-15
15.3.4	Reactor Coolant Pump Shaft Break	(d)	(d)		
15.4.8	Rod Ejection Accident [e]	0.59	0.09	6.3	7.1-16
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	0.18	0.01	2.5	7.1-17
15.6.3	Steam Generator Tube Failure				
	Pre-Transient Iodine Spike	0.42	0.03	25	7.1-18
	Transient-Initiated Iodine Spike	0.11	0.01	2.5	7.1-19
15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	1.5	0.26	25	7.1-20
15.7.4	Fuel Handling Accident	0.38	0.03	6.3	7.1-21

a) All values conservatively rounded up.

b) NUREG-1555 specifies a dose limit of 25 rem TEDE for all DBA. The more restrictive limits shown in the table apply to safety analysis doses, but they are shown here to demonstrate that even these more restrictive limits are met.

c) As discussed in DCD Subsection 15.2.8.5, the radiological consequences of a Feedwater System Pipe Break event are bounded by the main steam line break accident evaluated in DCD Subsection 15.1.5.

d) The analysis performed for the RCP rotor seizure transient (DCD Subsection 15.3.3) bounds the response and results for the RCP shaft break as discussed in DCD Subsection 15.3.4.

e) The source of this accident is Subsection 15.4.8 of NUREG-0800. This event is not included in NUREG-1555, Section 7.1, Appendix A, “Design Basis Accidents Included in Section 15 of the Standard Review Plan.”

TABLE 7.1-13
RADIOLOGICAL CONSEQUENCES OF STEAM SYSTEM PIPING FAILURE
(PRE-TRANSIENT IODINE SPIKE)

	DCD Dose (rem TEDE)		χ/Q Ratio [a]	Site Dose (rem TEDE)	
Time	EAB [b]	LPZ	(Site/DCD)	EAB	LPZ
0-2 hr	1.9E-01		1.15E-01	2.2E-02	
0-8 hr		1.1E-01	1.58E-02		1.6E-03
8-24 hr		7.6E-03	2.12E-02		1.6E-04
24-96 hr		0.0E+00	2.65E-02		0.0E+00
96-720 hr		0.0E+00	3.61E-02		0.0E+00
Total	1.9E-01	1.1E-01		2.2E-02	1.8E-03
Limit				25	25

a) χ/Q Ratio from Table 7.1-11.

b) DCD dose for EAB obtained from DCD Table 15.1.5-3.

TABLE 7.1-14
RADIOLOGICAL CONSEQUENCES OF STEAM SYSTEM PIPING FAILURE
(TRANSIENT-INITIATED IODINE SPIKE)

	DCD Dose (rem TEDE)		χ/Q Ratio [a]	Site Dose (rem TEDE)	
Time	EAB [b]	LPZ	(Site/DCD)	EAB	LPZ
0-2 hr	3.2E-01		1.15E-01	3.7E-02	
0-8 hr		2.1E-01	1.58E-02		3.3E-03
8-24 hr		6.5E-02	2.12E-02		1.4E-03
24-96 hr		0.0E+00	2.65E-02		0.0E+00
96-720 hr		0.0E+00	3.61E-02		0.0E+00
Total	3.2E-01	2.8E-01		3.7E-02	4.7E-03
Limit				2.5	2.5

a) χ/Q Ratio from Table 7.1-11.

b) DCD dose for EAB obtained from Table 15.1.5-3 of the DCD.

TABLE 7.1-15
RADIOLOGICAL CONSEQUENCES OF RCP ROTOR SEIZURE

	DCD Dose (rem TEDE)		χ/Q Ratio [(a)]	Site Dose (rem TEDE)	
Time	EAB [(b)]	LPZ	(Site/DCD)	EAB	LPZ
10-12 hr	4.9E-01		1.15E-01	5.6E-02	
0-8 hr		4.4E-01	1.58E-02		7.0E-03
8-24 hr		2.6E-01	2.12E-02		5.3E-03
24-96 hr		0.0E+00	2.65E-02		0.0E+00
96-720 hr		0.0E+00	3.61E-02		0.0E+00
Total	4.9E-01	7.0E-01		5.6E-02	1.2E-02
Limit				2.5	2.5

a) χ/Q Ratio from Table 7.1-11.

b) DCD dose for EAB obtained from Table 15.3.3-5 of the DCD.

TABLE 7.1-16
RADIOLOGICAL CONSEQUENCES OF ROD EJECTION ACCIDENT

	DCD Dose (rem TEDE)		χ/Q Ratio [a]	Site Dose (rem TEDE)	
Time	EAB [b]	LPZ	(Site/DCD)	EAB	LPZ
0-2 hr	5.1E+00		1.15E-01	5.9E-01	
0-8 hr		3.3E+00	1.58E-02		5.1E-02
8-24 hr		8.8E-01	2.12E-02		1.9E-02
24-96 hr		1.6E-01	2.65E-02		4.2E-03
96-720 hr		1.8E-01	3.61E-02		6.3E-03
Total	5.1E+00	4.5E+00		5.9E-01	8.1E-02
Limit				6.3	6.3

a) χ/Q Ratio from Table 7.1-11.

b) DCD dose for EAB obtained from Table 15.4.8-4 of the DCD.

TABLE 7.1-17
RADIOLOGICAL CONSEQUENCES OF THE FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT

	DCD Dose (rem TEDE)		χ/Q Ratio [a]	Site Dose (rem TEDE)	
Time	EAB [b]	LPZ [b]	(Site/DCD)	EAB	LPZ
0-2 hr	1.5E+00		1.15E-01	1.7E-01	
0-8 hr		6.0E-01	1.58E-02		9.5E-03
Total	1.5E+00	6.0E-01		1.7E-01	9.5E-03
Limit				2.5	2.5

a) χ/Q Ratio from Table 7.1-11.

b) DCD dose for EAB and LPZ obtained from Table 15.6.2-2 of the DCD.

TABLE 7.1-18
RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE RUPTURE
(PRE-TRANSIENT IODINE SPIKE)

	DCD Dose (rem TEDE)		χ/Q Ratio [a]	Site Dose (rem TEDE)	
Time	EAB [b]	LPZ	(Site/DCD)	EAB	LPZ
0-2 hr	3.6E+00		1.15E-01	4.1E-01	
0-8 hr		1.5E+00	1.58E-02		2.3E-02
8-24 hr		2.1E-03	2.12E-02		4.3E-05
24-96 hr		2.1E-04	2.65E-02		5.5E-06
96-720 hr		1.8E-04	3.61E-02		6.2E-06
Total	3.6E+00	1.5E+00		4.1E-01	2.3E-02
Limit				25	25

a) χ/Q Ratio from Table 7.1-11.

b) DCD dose for EAB obtained from Table 15.6.3-5 of the DCD.

TABLE 7.1-19
RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE RUPTURE
(TRANSIENT-INITIATED IODINE SPIKE)

	DCD Dose (rem TEDE)		χ/Q Ratio [(a)]	Site Dose (rem TEDE)	
Time	EAB [(b)]	LPZ	(Site/DCD)	EAB	LPZ
0-2 hr	9.6E-01		1.15E-01	1.1E-01	
0-8 hr		4.1E-01	1.58E-02		6.5E-03
8-24 hr		1.1E-02	2.12E-02		2.2E-04
24-96 hr		2.1E-04	2.65E-02		5.5E-06
96-720 hr		1.8E-04	3.61E-02		6.2E-06
Total	9.6E-01	4.3E-01		1.1E-01	6.7E-03
Limit				2.5	2.5

a) χ/Q Ratio from Table 7.1-11.

b) DCD dose for EAB obtained from Table 15.6.3-5 of the DCD.

TABLE 7.1-20
RADIOLOGICAL CONSEQUENCES OF LOSS-OF-COOLANT ACCIDENT

	DCD Dose (rem TEDE)		χ/Q Ratio [a]	Site Dose (rem TEDE)	
Time	EAB [b]	LPZ	(Site/DCD)	EAB	LPZ
0.5-2.5 hr	1.3E+01		1.15E-01	1.5E+00	
0-8 hr		9.0E+00	1.58E-02		1.4E-01
8-24 hr		1.3E+00	2.12E-02		2.6E-02
24-96 hr		1.3E+00	2.65E-02		3.4E-02
96-720 hr		1.4E+00	3.61E-02		4.9E-02
Total	1.3E+01	1.3E+01		1.5E+00	2.5E-01
Limit				25	25

a) χ/Q Ratio from Table 7.1-11.

b) DCD dose for EAB obtained from Table 15.6.5-16 of the DCD.

TABLE 7.1-21
RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING ACCIDENT

	DCD Dose (rem TEDE)		χ/Q Ratio [a]	Site Dose (rem TEDE)	
Time	EAB [b]	LPZ [b]	(Site/DCD)	EAB	LPZ
0-2 hr	3.3E+00		1.15E-01	3.8E-01	
0-8 hr		1.4E+00	1.58E-02		2.2E-02
Total	3.3E+00	1.4E+00		3.8E-01	2.2E-02
Limit				6.3	6.3

a) χ/Q Ratio from Table 7.1-11.

b) DCD dose for EAB and LPZ obtained from Table 15.7.4-2 of the DCD.

7.2 SEVERE ACCIDENTS

This section discusses the probabilities and consequences of accidents of greater severity than the design basis accidents (DBAs), which are discussed in Section 7.1.

7.2.1 INTRODUCTION

Severe accidents, as a class, are considered less likely to occur, but because their consequences could be more severe, they are considered important both in terms of impact to the environment and off-site costs. These severe accidents can be distinguished from DBAs in two primary respects: (1) they involve substantial physical deterioration of the fuel in the reactor core, including overheating to the point of melting, and (2) they involve deterioration of the capability of the containment system to perform its intended function of limiting the release of radioactive materials to the environment.

7.2.2 EVALUATION OF POTENTIAL SEVERE ACCIDENT RELEASES

The severe accident consequence analysis was performed using the Level 3 probabilistic risk assessment (PRA) Melcor Accident Consequence Code System (MACCS2) code.

The analysis was performed with the MACCS2 version designated as Oak Ridge National Laboratory RSICC Computer Code Collection MACCS2 V.1.13.1, CCC-652 Code Package (Chanin and Young 1997). MACCS2, Version 1.13.1, released in January 2004, simulates the impact of severe accidents at nuclear power plants on the surrounding environment. The principal phenomena considered in MACCS2 are atmospheric transport, mitigating actions based on dose projections, dose accumulation by a number of pathways including food and water ingestion, early and latent health effects, and economic costs. The MACCS2 program was chosen for this analysis because it is U.S. Nuclear Regulatory Commission (NRC)-endorsed, as stated in the MACCS2 User's Guide. The model for the proposed project, Comanche Peak Nuclear Power Plant (CPNPP) Units 3 and 4, had no important deviations from the default code input values, except for site-specific values and reactor design information. The code values modified for the U.S. Advanced Pressurized Water Reactor (US-APWR) were primarily the source term data (MHI 2007). These data include the release fractions, plume release height, delay, and duration. Values for the ATMOS input data file, one of the five input files used by MACCS2, were modified as necessary to use data appropriate for the US-APWR source terms and probability frequencies. The remaining four MACCS2 input files were reviewed and modified as necessary.

Three years of site-specific hourly meteorological data were used in the analysis. Stability class was calculated using the CPNPP site meteorological data and the methodology of Regulatory Guide (RG) 1.23, Table 1. In accordance with U.S. Environmental Protection Agency (EPA) recommendations, short periods of missing data were replaced by interpolating from the values immediately before the data gap to the values immediately after the data gap, while longer periods of missing data were replaced with data from nearby days that had similar meteorological conditions as before and after the data gaps (EPA 1992). Meteorology is further discussed in Section 2.7 and in Final Safety Analysis Report (FSAR) Section 2.3.

Morning and afternoon mixing height values were taken from FSAR Table 2.3-214, which provides values for Stephenville, Texas. These values are appropriate for use because Stephenville is the nearest EPA Support Center for Regulatory Atmospheric Modeling (SCRAM) station. The treatment of rain/precipitation events follows the default recommended parameter values given in the ATMOS file supplied with the MACCS2 code.

The population distribution and land-use information for the region surrounding the CPNPP site are specified in the SITE input data file. Contained in the SITE input data file are the geometry data used for the site (spatial intervals and wind directions), population distribution, fraction of the area that is land, watershed data for the liquid pathways model, information on agricultural land use and growing seasons, and regional economic information. Some of the detailed data in this input file supersede certain data in the EARLY input data file. The population distribution and meteorological data are used in conjunction in the MACCS2 analysis, i.e., the population dose partly depends on whether the wind generally blows toward heavily populated areas or more sparsely populated areas.

A 50-mile (mi) radius area around the site was divided into 16 directions that are equivalent to a standard navigational compass rosette. This rosette was further divided into inner radial rings as shown in Figures 2.5-2 and 2.5-3.

The population distribution in the MACCS2 analysis uses data from the calendar year 2056 projected population in Tables 2.5-1 and 2.5-2. The land fractions are estimated from Figures 2.5-2 and 7.2-1.

Regional indices are all identified as Texas for region indexing. The default economic values supplied by the code were multiplied by the Consumer Price Index (CPI) ratio of the November 1988 value of 118.3 (when the NUREG-1150 data above were generated) to the November 2007 value of 203.4 (CPI 2008). Details regarding farm acreage for the counties within a 50-mi radius of the plant were taken from the Agricultural Marketing Services branch of the U.S. Department of Agriculture (USDA) agricultural statistics state summary (SMP 2005). The fraction of farmland for each county and updated economic values, based on the CPI ratio, are shown in Table 7.2-1.

The crop information required by MACCS2 was collected from county statistics (SMP 2005). These were combined and weighted by the total farmland area within the 50-mi radius to produce a single composite measure, as shown in Table 7.2-2. The growing season was conservatively assumed to be all year long in the MACCS2 analysis.

The EARLY module of the MACCS2 code models the time period immediately following a radioactive release. This period is commonly referred to as the emergency phase, which may extend up to 1 week after the arrival of the first plume at any downwind spatial interval. The subsequent intermediate and long-term periods are treated by the CHRONC module of the code. In the EARLY module, the user may specify emergency response scenarios that include evacuation, sheltering, and dose-dependent relocation. The EARLY module has the capability of combining results from up to three different emergency response scenarios by appending change records to the EARLY input data file. The first emergency response scenario is defined in the main body of the EARLY input data file. Up to two additional response scenarios can be defined through change record sets positioned at the end of the file.

The emergency evacuation model has been modeled as a single evacuation zone extending out 10 mi from the site. For the purposes of this analysis, an average evacuation speed of 4.0 mi per hour (mph) is used with a 7200-second delay between the alarm and start of evacuation, with no sheltering. Once evacuees are more than 50 mi from the site, they no longer receive dose and are not included in the analysis. The evacuation scenario is modeled so that 90 percent of the population is evacuated.

The ATMOS input data file calculates the dispersion and deposition of material-released "source terms" to the atmosphere as a function of downwind distance. Source term release fractions (RELFRC) are shown in Table 7.2-3, and plume characterizations are shown in Table 7.2-4. These data include the RELFRC, plume start time, plume release height, delay, and duration.

The data in Tables 7.2-3 and 7.2-4 are from the US-APWR DC Applicant's Environmental Report (ER) (MHI 2007). The four plumes in Table 8 of the DC Applicant's ER (MHI 2007) were collapsed into two plumes using the following steps:

1. The release fractions for the first two plumes in the DC Applicant's ER Table 8 (MHI 2007) were added together to produce a release fraction for the first plume in Table 7.2-3. Similarly, the third and fourth plumes in the DC Applicant's ER (MHI 2007) Table 8 were combined for the second plume in Table 7.2-3. This process assures that the total release is the same.

2. The first plume duration in Table 7.2-4 is the maximum of the first two plume durations in the DC Applicant's ER (MHI 2007) Table 8. Similarly, the second plume duration in Table 7.2-4 is the maximum of the third and fourth plume durations in the DC Applicant's ER (MHI 2007) Table 8.

3. The plume delays in Table 7.2-4 were taken as the first and second plume start times in the DC Applicant's ER (MHI 2007) Table 8. The inventory is released faster in this approach than in the four-plume approach.

4. The Ref Time term in Table 7.2-4, which calculates the plume position according to its leading edge (0) or midpoint (0.5), is equal to the plume position in the DC Applicant's ER (MHI 2007) Table 8 for the first and second plumes, respectively, to be consistent with the plume delay approach.

The plume release height was conservatively set to zero, as specified in Appendix A.3 of the DC Applicant's ER (MHI 2007), which corresponds to a ground level release. Parameters are assigned to each source term according to release category. Each released plume is assumed to have two segments.

The results of the dose and dollar risk assessments for internal events, including the water ingestion pathway, are provided in Table 7.2-5. Risk is defined in these results as the product of release category frequency and the dose or cost associated with the release category. The total risk is assumed to be the sum of all scenarios.

The sum of the values for affected land areas for all release scenarios, as given in Tables 7.2-9, 7.2-10, and 7.2-11, is also shown in Table 7.2-5. Each of these values has also been multiplied by their release category frequency.

The values for total early and latent fatalities per reactor-year (RY) were conservatively calculated as the sum of all release scenarios. Tables 7.2-6 and 7.2-7 support the calculated dose per RY and dollars per RY risks presented in Table 7.2-5 for internal events. The release frequency data come from Table 7 of the DC Applicant's ER (MHI 2007).

External events were considered in Subsection 19.1.5 of the US-APWR design control document (DCD) and in FSAR Subsection 19.1.5. FSAR Subsection 19.1.5 provides discussion of high winds and tornadoes, external flooding, transportation and nearby facility accidents, and aircraft crashes. The FSAR concludes that all of these external events make an insignificant contribution to the total core damage frequency (CDF). Seismic events are discussed in Subsection 19.1.5 of the US-APWR DCD and are not incorporated into the total CDF. Therefore, external events were determined to be negligible compared to internal events and were not incorporated into the release frequencies.

Due to the extremely low frequency of severe accidents, the severe accident population dose for the CPNPP site is also low. The weighted total dose risk from internal events for the year 2006, which had the most conservative met data, is 3.00×10^{-1} person-rem/RY, as shown in Table 7.2-11. This dose is based on the calendar year 2056 projected population distribution. To obtain the average individual dose, this value is divided by the calendar year 2056 population of 2,760,243 people within 50 mi of the CPNPP site, as given in Tables 2.5-1 and 2.5-2, resulting in a dose of 1.09×10^{-7}rem/RY. This value is lower than the background radiation. Idaho State University indicates that the average individual dose caused by all other sources in the United States is 3.6×10^{-1} rem/yr (ISU 2008). Because the weighted total dose risk from severe accidents is lower than the background radiation, it can also be concluded that the impact on the local biota would be negligible. Additionally, biota tend to be less sensitive to radiation than humans, and the primary concern regarding biota is survival of the species, not individual fatalities.

The liquid pathways dose is not expected to be significant. The MACCS2 analysis resulted in a water ingestion dose risk of 1.63×10^{-2} person-rem/RY for the year 2006, which provided the most conservative water ingestion dose risk, as shown in Table 7.2-5 for internal events. This dose accounts for airborne deposition directly onto surface water bodies and deposition onto land that is washed off into surface water bodies, which is eventually consumed in drinking water. NUREG-1437 Table 5.17 indicates that, for a freshwater site such as CPNPP, drinking water is the dominant liquid pathway compared to fish ingestion and shoreline exposure. Furthermore, the water ingestion dose risk of 1.63×10^{-2} person-rem/RY is small compared to the total dose risk of 3.00×10^{-1} person-rem/RY. Aquifers in the vicinity of the site are provided in Section 2.3, and a list of public surface water users is provided in Tables 2.3-34 and 2.3-36. In addition to surface water, groundwater must be considered in the liquid pathways dose. As discussed in Subsection 2.3.1.5.6 and FSAR Subsection 2.4.12.3.1, the estimated travel time for groundwater from CPNPP Unit 3 to Squaw Creek Reservoir (SCR) through undifferentiated fill/regolith, which represents the most conservative pathway, is 720.9 days, or approximately 2 years, which would allow ample time for interdiction and other prevention activities.

The results of severe accidents for current generation reactors are compared to the severe accident risk calculated in the MACCS2 analysis in Table 7.2-8, where the data for the current generation reactors were taken from System Energy Resources Inc. (SERI 2004). The conclusion is that the low frequency of releases associated with the US-APWR design makes the severe accident risk of a future unit at this site extremely low. Additional severe accident analysis results are reported in Tables 7.2-9, 7.2-10, and 7.2-11. The CDF in these tables comes from Table 7 of the DC Applicant's ER (MHI 2007).

The significance of the impacts associated with each severe accident issue has been identified as either SMALL, MODERATE, or LARGE, consistent with the criteria that the NRC established in 10 Code of Federal Regulations (CFR) 51, Appendix B, Table B-1, Footnote 3 as follows:

> SMALL – Environmental effects are not detectable or are so minor that they are not expected to destabilize nor noticeably alter any important attribute of the resource. For purposes of assessing radiological impacts, the NRC has concluded that those impacts that do not exceed permissible levels in the NRC's regulations are considered small.
>
> MODERATE – Environmental effects are sufficient to alter noticeably, but not to destabilize, any important attribute of the resource.
>
> LARGE – Environmental effects are clearly noticeable and are sufficient to destabilize any important attributes of the resource.

In accordance with National Environmental Policy Act (NEPA) practice, ongoing and potential additional mitigation is considered in proportion to the significance of the impact to be addressed (i.e., impacts that are SMALL receive less mitigative consideration than impacts that are LARGE).

As discussed previously, the frequency of releases is extremely low. Also, the average individual dose risk of 1.09×10^{-7}rem/RY, as calculated above, is lower than the average individual dose caused by all other sources in the United States of 3.6×10^{-1} rem/yr; therefore, the CPNPP site risks would be acceptable.

The MACCS2 analysis also considers potential economic impacts as a result of postulated severe accidents at a nuclear reactor on the CPNPP site. MACCS2 calculated severe accident costs based on the following:

- Evacuation costs.
- Value of crops contaminated and condemned.
- Value of milk contaminated and condemned.
- Costs of decontamination of property.
- Indirect costs resulting from the loss of use of property and incomes derived as a result of the accident.

The total cost of severe accidents at the CPNPP site was determined to be $714/RY given the 2006 meteorological data, which was the most conservative of the three years considered, as shown in Table 7.2-5. This low cost is mostly due to the extremely low accident frequencies expected for accidents of this magnitude.

7.2.3 CONSIDERATION OF COMMISSION SEVERE ACCIDENT POLICY

In 1985, the NRC adopted a Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants. This policy statement indicated that the NRC fully expects that vendors engaged in designing new standard (or custom) plants are to achieve a higher standard of severe accident safety performance than their prior designs. This expectation is based on:

- The growing volume of information from industry and government-sponsored research and operating reactor experience has improved our knowledge of specific severe accident vulnerabilities and of low-cost methods for their mitigation. Further learning on safety vulnerabilities and innovative methods is to be expected.
- The inherent flexibility of this policy statement (that permits risk-risk tradeoffs in systems and subsystems design) encourages thereby innovative ways of achieving an improved overall systems reliability at a reasonable cost.
- Public acceptance, and hence investor acceptance, of nuclear technology is dependent on demonstrable progress in safety performance, including the reduction in frequency of accident precursor events as well as a diminished controversy among experts as to the adequacy of nuclear safety technology.

Thus, implementation of the NRC’s Severe Accident Policy can be expected to show that the environmental impact of any additional reactor or reactors on the CPNPP site would be within the range of risk previously determined to be SMALL.

A significant factor in the risk associated with the plant design is the frequency of the considered release modes. The various accident frequencies for a US-APWR are extremely low, resulting in the low-impact consequences discussed previously.

7.2.4 CONCLUSION

The following are directly applicable conclusions from NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants (GEIS), Volume 1, and conclusions drawn based on the foregoing analysis:

- The conclusion of the GEIS, based on the generic evaluations presented, is that the probability-weighted consequences of severe accidents are SMALL for all plants.
- As described above, the results of the GEIS are applicable to the consideration of new plants. Evaluation of site-specific factors for purposes of this application has shown that the CPNPP site is within the range of sites considered in the GEIS. Thus, it is concluded that the GEIS conclusion is applicable to the CPNPP site.

The environmental impacts of a postulated severe accident at the CPNPP site could be severe but, due to the low likelihood of such an accident, the impacts are determined to be SMALL. The total dose risk value of 3.00×10^{-1} person-rem/RY is not bounded by the dose risk of $2.7 \times 10\text{-}^{1}$ person-rem/RY calculated in Table 10a of the DC Applicant's ER (MHI 2007). However, the calculation in the DC Applicant's ER (MHI 2007) does not account for Release Category RC5 because there is no release within 24 hr after the onset of core damage. If the dose risk value for RC5 is subtracted from the total dose risk value in Table 7.2-6 for the year 2006, the resulting total dose risk value is 1.52×10^{-1} person-rem/RY, which is bounded by 2.7×10^{-1} person-rem/RY. Other notable differences between the DC Applicant's analysis and the site-specific analysis are that the DC Applicant's analysis did not credit evacuation and sheltering and only considered the first 24 hours (hr) of the event. Radiological dose consequences and health effects associated with normal and anticipated operational releases are discussed in Subsection 5.4.3.

The CDF for internal events is 1.2×10^{-6}. This value is used in conjunction with the Applicant's ER (MHI 2007) to determine the total severe accident health effects, which include internal events, internal fire, internal flood, and low-power and shutdown (LPSD) events, as shown in Tables 7.2-12, 7.2-13, and 7.2-14. The health effects resulting from internal fire, internal flood, and LPSD events were determined using the ratio of the CDF values for these events and the CDF value for the internal events. The maximum dose risk from the three years of meteorological data is 1.15 person-rem/RY. The maximum numbers of early and latent fatalities per RY from the three years of meteorological data are 2.87×10^{-7} and 9.17×10^{-4}, respectively. Finally, the maximum dose for the water ingestion pathway from the three years of meteorological data is 6.25×10^{-2} person-rem/RY.

Additionally, the NRC's Safety Goal Policy Statement, issued in 1986, states that "the risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed" and that "the risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes." According to the Centers for Disease Control and Prevention (CDC), there were 39.7 deaths caused by accidents per 100,000 people in the year 2005. Also, there were 188.7 deaths caused by cancer per 100,000 people in the year 2005 (CDC 2008). These statistics mean that the cancer fatality risk from "all other causes" is 1.89×10^{-3}, and the prompt fatality risk from "other accidents" is 3.97×10^{-4}. One-tenth of one percent of each of these risks results in a value of 1.89×10^{-6} for cancer fatalities and 3.97×10^{-7} for prompt fatalities. As stated above, the maximum number of latent fatalities per RY from the three years of meteorological data is 9.17×10^{-4}. In order to obtain the appropriate risk number, the number of latent fatalities is divided by the calendar year 2056 population within 50 mi of the CPNPP site of 2,760,243. This results in a cancer fatality risk of 3.32×10^{-10}, which is well below the cancer fatality safety goal of 1.89×10^{-6}. Also as stated above, the maximum number of early fatalities per RY from the three years of meteorological data is 2.87×10^{-7}. In order to obtain the appropriate risk number, the number of early fatalities is divided by the calendar year 2056 population within two kilometers of the CPNPP site of 182, as provided in Table 2.5-1. The Safety Goal Policy Statement indicates that the population within one mile of the plant should be used, but here the population within two kilometers is considered to be a reasonable estimate, particularly because the risk of prompt fatalities is bounded by the safety goal regardless of the population size used. This results in a prompt fatality risk of 1.58×10^{-9}, which is well below the prompt fatality safety goal of 3.97×10^{-7}. Therefore, the early and latent fatality risks from a severe accident at the CPNPP site are found to be acceptable.

7.2.5 REFERENCES

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TABLE 7.2-1
STATE ECONOMIC STATISTICS CORRECTED FOR INFLATION AND FARM FRACTION

Region[a]	State	Fraction farm[b]	Fraction dairy	Farm sales ($/hectare)[c]	Property value ($/hectare)[c]	Non-farm property value ($/person)[c]
41	TEX	0.789	0.064	282	2,565	127,206

a) The region values are the numbers recorded in the MACCS2 site input file to designate a particular state.

b) The farm fraction is based on data from the year 2002.

c) Dollar values have been adjusted based on the CPI from November 1988 to November 2007.

TABLE 7.2-2
DISTRICT FARM STATISTICS AND WEIGHTED COMPOSITES

	TX-6[a]	TX-11[a]	TX-12[a]	TX-17[a]	TX-32[a]	Composite[b]
Pasture	0.476	0.731	0.660	0.662	0.463	0.642
Stored Forage	0.127	0.064	0.109	0.101	0.146	0.098
Grains	0.174	0.032	0.009	0.016	0.140	0.050
Green Leafy	0.000	0.000	0.000	0.001	0.000	0.000
Other	0.002	0.002	0.005	0.010	0.004	0.006
Legumes/seeds	0.002	0.000	0.001	0.003	0.032	0.002
Roots/tubers	0.000	0.000	0.000	0.000	0.000	0.000

a) TX-6, TX-11, TX-12, TX-17, and TX-32 are Texas electoral districts 6, 11, 12, 17, and 32.

b) All farm data are based on values from the year 2002.

TABLE 7.2-3
US-APWR SOURCE TERM RELEASE FRACTIONS[a]

Release Category[b]	Plume No.	Kr/Xe	I	Cs	Te/Sb	Sr	Ru	La	Ce	Ba
RC1[c],[d]	1	9.4E-1	2.8E-1	2.0E-1	1.3E-1	4.9E-3	1.8E-2	2.4E-4	2.8E-4	1.2E-2
RC1[e]	2	7.6E-3	6.3E-3	1.1E-2	8.5E-3	3.9E-3	4.3E-3	2.7E-3	1.9E-3	3.6E-3
RC2[f]	1	9.7E-1	6.8E-2	2.6E-2	4.3E-2	5.4E-3	1.6E-2	4.0E-3	2.3E-3	8.6E-3
RC2	2	2.7E-2	2.1E-1	1.7E-2	3.5E-2	2.3E-3	1.0E-4	1.1E-4	4.1E-4	2.6E-3
RC3[g]	1	9.9E-1	4.8E-1	4.7E-1	4.3E-1	4.4E-2	2.8E-1	1.6E-3	6.4E-3	1.1E-1
RC3	2	2.0E-3	1.3E-3	1.1E-3	4.3E-3	4.9E-4	1.8E-4	6.6E-6	6.3E-5	2.5E-4
RC4[h]	1	1.0E+0	5.5E-2	4.2E-2	5.3E-2	4.8E-3	2.7E-2	1.2E-4	3.7E-4	2.4E-2
RC4	2	3.8E-4	1.4E-2	4.5E-3	1.1E-2	1.3E-3	1.1E-5	1.5E-5	4.7E-4	4.7E-4
RC5[i]	1	9.6E-1	2.5E-2	5.3E-3	9.0E-3	8.2E-5	1.0E-4	3.0E-5	1.9E-5	6.8E-5
RC5	2	2.5E-2	1.2E-1	1.5E-2	7.7E-3	2.2E-6	2.6E-6	5.9E-8	5.9E-8	5.0E-6
RC6[j]	1	7.8E-4	1.7E-6	1.7E-6	1.3E-6	1.7E-7	6.4E-7	3.5E-9	5.6E-9	2.7E-7
RC6	2	1.3E-3	1.9E-9	0.0E+0	6.0E-10	6.5E-11	4.4E-11	4.6E-13	1.2E-12	6.4E-11

a) Some release fraction values contain negligible errors due to rounding.

b) Two lines of data are provided for each release category because the four plumes in the DC Applicant's Environmental Report, Table 8 (MHI 2007) were collapsed into two plumes.

c) Containment bypass, which includes core damage after steam generator tube rupture (SGTR) and thermally induced SGTR after core damage.

d) The release fractions for the first two plumes in the DC Applicant's Environmental Report, Table 8 (MHI 2007) were added together to produce a release fraction for the first plume for each release category.

e) The release fractions for the third and fourth plumes in the DC Applicant's Environmental Report, Table 8 (MHI 2007) were added together to produce a release fraction for the second plume for each release category.

f) Containment isolation failure.

g) Overpressure failure before core damage due to loss of heat removal.

h) Containment failure condition due to dynamic loads, which includes hydrogen combustion before or just after reactor vessel failure, in-vessel or ex-vessel steam explosion, and containment direct heating.

i) Containment failure condition, including overpressure failure after core damage, hydrogen combustion failure after core damage, hydrogen combustion long after reactor vessel failure, and basemat melt-through.

j) Condition which assumes intact containment throughout the sequence and fission products released at the design leak rate.

TABLE 7.2-4
US-APWR PLUME CHARACTERIZATION DATA

Release Category (a)	Plume No.	Number of Plume Releases	Risk-Dominant Plume	Ref Time(b)	Plume Heat (W)	Plume Release Height (m)	Plume Duration (s)(c)	Plume Delay (s)(d)
RC1	1	2	1	0.0	0	0	3.6E+4	1.0E+5
RC1	2	2	1	0.5	0	0	8.6E+4	1.2E+5
RC2	1	2	1	0.0	0	0	5.3E+4	9.0E+3
RC2	2	2	1	0.5	0	0	8.6E+4	4.2E+4
RC3	1	2	1	0.0	0	0	4.4E+4	1.7E+5
RC3	2	2	1	0.0	0	0	8.6E+4	2.1E+5
RC4	1	2	1	0.0	0	0	3.2E+4	7.8E+4
RC4	2	2	1	0.5	0	0	8.6E+4	9.4E+4
RC5	1	2	1	0.0	0	0	6.0E+4	1.9E+5
RC5	2	2	1	0.5	0	0	8.6E+4	2.0E+5
RC6	1	2	1	0.0	0	0	7.3E+4	1.3E+3
RC6	2	2	1	0.5	0	0	8.6E+4	1.5E+4

a) Two lines of data are provided for each release category because the four plumes in the DC Applicant's Environmental Report Table 8 (MHI 2007) were collapsed into two plumes.

b) The Ref Time values for each release category, which calculate the plume position according to its leading edge (0) or midpoint (0.5), are equal to the plume position in the DC Applicant's Environmental Report Table 8 (MHI 2007) for the first and second plumes, respectively, to be consistent with the plume delay approach.

c) The first plume duration for each release category is the maximum of the first two plume durations in the DC Applicant's Environmental Report, Table 8 (MHI 2007). The second plume duration for each release category is the maximum of the third and fourth plume durations in the DC Applicant's Environmental Report, Table 8 (MHI 2007).

d) The plume delays for each release category were taken as the first and second plume start times in the DC Applicant's Environmental Report Table 8 (MHI 2007).

TABLE 7.2-5
SEVERE ACCIDENT ANALYSIS RESULTS SUMMARY WITHIN 50 MI OF CPNPP SITE[(a)]

Met Data Year	Dose Risk (person-rem/ RY)	Dollar Risk ($/RY)	Affected Land (hectares) [(b)]	Early Fatalities (per RY)	Latent Fatalities (per RY)	Water Ingestion Dose Risk (person-rem/ RY)
2001	2.21E-01	5.78E+02	2.66E-02	7.49E-08	1.85E-04	1.62E-02
2003	2.71E-01	6.62E+02	2.76E-02	7.43E-08	2.15E-04	1.52E-02
2006	3.00E-01	7.06E+02	2.70E-02	6.73E-08	2.39E-04	1.63E-02

a) All data are compiled from Tables 7.2-9, 7.2-10, and 7.2-11.

b) This value reflects the sum of affected land areas that have been multiplied by their release category frequency, whereas the affected land areas shown in the MACCS2 analysis are neither multiplied by release category frequency or summed. However, the same MACCS2 data were used as the basis for both values.

TABLE 7.2-6
MEAN VALUE FOR TOTAL DOSE RISK ASSESSMENT IN PERSON-REM/RY

Release Category	Frequency (per RY)	Dose Risk - 2001	Dose Risk - 2003	Dose Risk - 2006
RC1	7.5E-09	2.39E-02	2.90E-02	2.93E-02
RC2	2.1E-09	4.62E-03	5.61E-03	6.09E-03
RC3	2.0E-08	7.56E-02	8.10E-02	8.96E-02
RC4	1.1E-08	2.24E-02	2.66E-02	2.67E-02
RC5	6.5E-08	9.36E-02	1.27E-01	1.48E-01
RC6	1.1E-06	9.97E-04	1.18E-03	1.01E-03
Total	1.2E-06	2.21E-01	2.71E-01	3.00E-01

TABLE 7.2-7
DOLLAR RISK ASSESSMENT IN DOLLARS/RY

Release Category	Frequency (per RY)	Dollar Risk - 2001[(a)]	Dollar Risk - 2003[(a)]	Dollar Risk - 2006[(a)]
RC1	7.5E-09	8.10E+01	9.08E+01	9.90E+01
RC2	2.1E-09	1.12E+01	1.47E+01	1.65E+01
RC3	2.0E-08	2.96E+02	3.18E+02	3.38E+02
RC4	1.1E-08	4.64E+01	5.23E+01	5.73E+01
RC5	6.5E-08	1.43E+02	1.87E+02	1.95E+02
RC6	1.1E-06	4.96E-03	7.46E-03	6.84E-03
Total	1.2E-06	5.78E+02	6.62E+02	7.06E+02

a) The dollar risk accounts for the costs of evacuation, crops contaminated and condemned, milk contaminated and condemned, decontamination of property, and indirect costs resulting from the loss of use of property and incomes. The 2001, 2003, and 2006 refer to the year of meteorological data used in the calculation.

TABLE 7.2-8
POPULATION DOSE COMPARISON AMONG PLANTS

Plant	Population Dose within 50 mi (person-rem/RY)[a]
Zion	5.00E+1
Grand Gulf	5.00E-1
Surry	6.00E+0
North Anna	2.51E+1
CPNPP US-APWR	3.00E-1[b]

a) Data for the current generation reactors were taken from System Energy Resources, Inc. (SERI 2004).

b) Value based on 2006 meteorological data.

TABLE 7.2-9
SEVERE ACCIDENT IMPACTS TO THE POPULATION AND LAND USING 2001 METEOROLOGICAL DATA

Release Category	Core Damage Frequency (per RY)	Dose-Risk (person-rem/ RY)	Number of Early Fatalities (per RY)	Number of Latent Fatalities (per RY)	Affected Land Area (hectares)[a]	Cost-Risk (dollars/ RY)[b]	Water Ingestion Pathway (person-rem/RY)
RC1	7.5E-09	2.39E-02	2.19E-09	1.59E-05	2.13E-03	8.10E+01	1.90E-03
RC2	2.1E-09	4.62E-03	3.07E-10	3.36E-06	6.95E-04	1.12E+01	1.28E-04
RC3	2.0E-08	7.56E-02	7.16E-08	1.06E-04	5.30E-03	2.96E+02	1.21E-02
RC4	1.1E-08	2.24E-02	8.26E-10	1.38E-05	2.51E-03	4.64E+01	6.89E-04
RC5	6.5E-08	9.36E-02	0.00E+00	4.52E-05	1.59E-02	1.43E+02	1.43E-02
RC6	1.1E-06	9.97E-04	0.00E+00	5.28E-07	5.40E-06	4.96E-03	2.39E-6
Total	1.2E-06	2.21E-01	7.49E-08	1.85E-04	2.66E-02	5.78E+02	1.62E-02

a) These values reflect affected land areas that have been multiplied by their release category frequency; whereas, the affected land areas shown in the MACCS2 analysis are not multiplied by release category frequency. However, the same MACCS2 data were used as the basis for both values.

b) The cost-risk accounts for the costs of evacuation, crops contaminated and condemned, milk contaminated and condemned, decontamination of property, and indirect costs resulting from the loss of use of property and incomes.

TABLE 7.2-10
SEVERE ACCIDENT IMPACTS TO THE POPULATION AND LAND USING 2003 METEOROLOGICAL DATA

Release Category	Core Damage Frequency (per RY)	Dose-Risk (person-rem/RY)	Number of Early Fatalities (per RY)	Number of Latent Fatalities (per RY)	Affected Land Area (hectares)[(a)]	Cost-Risk (dollars/ RY)[(b)]	Water Ingestion Pathway (person-rem/RY)
RC1	7.5E-09	2.90E-02	2.20E-09	1.89E-05	2.24E-03	9.08E+01	1.76E-03
RC2	2.1E-09	5.61E-03	2.96E-10	3.99E-06	7.56E-04	1.47E+01	1.16E-04
RC3	2.0E-08	8.10E-02	7.10E-08	1.14E-04	5.64E-03	3.18E+02	1.12E-02
RC4	1.1E-08	2.66E-02	7.84E-10	1.61E-05	2.53E-03	5.23E+01	6.41E-04
RC5	6.5E-08	1.27E-01	0.00E+00	6.11E-05	1.64E-02	1.87E+.02	1.49E-03
RC6	1.1E-06	1.18E-03	0.00E+00	6.12E-07	9.78E-06	7.46E-03	2.24E-06
Total	1.2E-06	2.71E-01	7.43E-08	2.15E-04	2.76E-02	6.62E+02	1.52E-02

a) These values reflect affected land areas that have been multiplied by their release category frequency; whereas, the affected land areas shown in the MACCS2 analysis are not multiplied by release category frequency. However, the same MACCS2 data were used as the basis for both values.

b) The cost-risk accounts for the costs of evacuation, crops contaminated and condemned, milk contaminated and condemned, decontamination of property, and indirect costs resulting from the loss of use of property and incomes.

TABLE 7.2-11
SEVERE ACCIDENT IMPACTS TO THE POPULATION AND LAND USING 2006 METEOROLOGICAL DATA

Release Category	Core Damage Frequency (per RY)	Dose-Risk (person-rem/RY)	Number of Early Fatalities (per RY)	Number of Latent Fatalities (per RY)	Affected Land Area (hectares)[a]	Cost-Risk (dollars/ RY)[b]	Water Ingestion Pathway (person-rem/ RY)
RC1	7.5E-09	2.93E-02	1.99E-09	1.97E-05	2.05E-03	9.90E+01	1.91E-03
RC2	2.1E-09	6.09E-03	2.46E-10	4.39E-06	7.01E-04	1.65E+01	1.27E-04
RC3	2.0E-08	8.96E-02	6.46E-08	1.27E-04	5.28E-03	3.38E+02	1.21E-02
RC4	1.1E-08	2.67E-02	4.70E-10	1.65E-05	2.44E-03	5.73E+01	6.90E-04
RC5	6.5E-08	1.48E-01	0.00E+00	7.09E-05	1.65E-02	1.95E+02	1.45E-03
RC6	1.1E-06	1.01E-03	0.00E+00	5.26E-07	7.69E-06	6.84E-03	2.41E-06
Total	1.2E-06	3.00E-01	6.73E-08	2.39E-04	2.70E-02	7.06E+02	1.63E-02

a) These values reflect affected land areas that have been multiplied by their release category frequency; whereas, the affected land areas shown in the MACCS2 analysis are not multiplied by release category frequency. However, the same MACCS2 data were used as the basis for both values.

b) The cost-risk accounts for the costs of evacuation, crops contaminated and condemned, milk contaminated and condemned, decontamination of property, and indirect costs resulting from the loss of use of property and incomes.

TABLE 7.2-12
TOTAL SEVERE ACCIDENT HEALTH EFFECTS USING 2001 METEOROLOGICAL DATA[b]

Accident Type	Core Damage Frequency (per RY)[a]	Scaling Factor	Dose-Risk (person-rem/RY)	Number of Early Fatalities (per RY)	Number of Latent Fatalities (per RY)	Water Ingestion Pathway (person-rem/RY)
Internal Events	1.2E-6	1	2.21E-01	7.49E-08	1.85E-04	1.62E-02
Internal Fire	1.8E-6	1.50	3.32E-01	1.12E-07	2.78E-04	2.43E-02
Internal Flood	1.4E-6	1.17	2.59E-01	8.76E-08	2.16E-04	1.90E-02
LPSD	2.0E-7	0.167	3.69E-02	1.25E-08	3.09E-05	2.71E-03
Total	4.6E-6	-	8.48E-01	2.87E-07	7.10E-04	6.22E-02

a) Core damage frequency values are from Table 5 of the DC Applicant's Environmental Report (MHI 2007).

b) The values for internal fire, internal flood, and LPSD are calculated as described on page 7.2-7.

TABLE 7.2-13
TOTAL SEVERE ACCIDENT HEALTH EFFECTS USING 2003 METEOROLOGICAL DATA[(b)]

Accident Type	Core Damage Frequency (per RY)[(a)]	Scaling Factor	Dose-Risk (person-rem/RY)	Number of Early Fatalities (per RY)	Number of Latent Fatalities (per RY)	Water Ingestion Pathway (person-rem/RY)
Internal Events	1.2E-6	1	2.71E-01	7.43E-08	2.15E-04	1.52E-02
Internal Fire	1.8E-6	1.50	4.07E-01	1.11E-07	3.23E-04	2.28E-02
Internal Flood	1.4E-6	1.17	3.17E-01	8.69E-08	2.52E-04	1.78E-02
LPSD	2.0E-7	0.167	4.53E-02	1.24E-08	3.59E-05	2.54E-03
Total	4.6E-6	-	1.04E-00	2.85E-07	8.25E-04	5.83E-02

a) Core damage frequency values are from Table 5 of the DC Applicant's Environmental Report (MHI 2007).

b) The values for internal fire, internal flood, and LPSD are calculated as described on page 7.2-7.

TABLE 7.2-14
TOTAL SEVERE ACCIDENT HEALTH EFFECTS USING 2006 METEOROLOGICAL DATA[b]

Accident Type	Core Damage Frequency (per RY)[a]	Scaling Factor	Dose-Risk (person-rem/RY)	Number of Early Fatalities (per RY)	Number of Latent Fatalities (per RY)	Water Ingestion Pathway (person-rem/RY)
Internal Events	1.2E-6	1	3.00E-01	6.73E-08	2.39E-04	1.63E-02
Internal Fire	1.8E-6	1.50	4.50E-01	1.01E-07	3.59E-04	2.45E-02
Internal Flood	1.4E-6	1.17	3.51E-01	7.87E-08	2.80E-04	1.91E-02
LPSD	2.0E-7	0.167	5.01E-02	1.12E-08	3.99E-05	2.72E-03
Total	4.6E-6	-	1.15E-00	2.58E-07	9.17E-04	6.25E-02

a) Core damage frequency values are from Table 5 of the DC Applicant's Environmental Report (MHI 2007).

b) The values for internal fire, internal flood, and LPSD are calculated as described on page 7.2-7.

7.3 SEVERE ACCIDENT MITIGATION ALTERNATIVES

This section updates the severe accident mitigation design alternative (SAMDA) analysis provided in Subsection 19.2.6 of the Mitsubishi Heavy Industries (MHI) U.S. Advanced Pressurized Water Reactor (US-APWR) design control document (DCD) with Comanche Peak Nuclear Power Plant (CPNPP) site and regional data. The CPNPP site-specific analysis demonstrates that the SAMDAs determined not to be cost beneficial by Mitsubishi Nuclear Energy Systems Inc. (MNES) on a generic basis are also not cost beneficial for CPNPP.

As described in Section 7.2, MNES performed a generic severe accident analysis for the US-APWR as part of the design certification process. The MNES analysis determined that severe accident impacts are small, that no potential mitigating design alternatives are cost-effective, and that appropriate mitigating measures are already incorporated into the plant design. Section 7.2 extended the MNES generic severe accident analysis to examine the proposed new nuclear units at the CPNPP site and determined that the generic conclusions remain valid for the CPNPP site. The analysis presented in this section provides assurance that there are no cost-beneficial design alternatives that would need to be implemented.

7.3.1 THE SAMA ANALYSIS PROCESS

Design or procedural modifications that could mitigate the consequences of a severe accident are known as severe accident mitigation alternatives (SAMAs). In the past, SAMAs were known as SAMDAs, which primarily focused on design changes and did not consider procedural modifications for SAMAs. The MNES DCD analysis is a SAMDA analysis. For an existing plant with a well-defined design and established procedural controls, the normal evaluation process for identifying potential SAMAs includes four steps:

1. Define the base case – The base case is the dose-risk and cost-risk of severe accidents before implementation of any SAMAs. A plant's probabilistic risk assessment (PRA) is the primary source of data in calculating the base case. The base case risks are converted to a monetary value to use for screening SAMAs. Section 7.2 presents the base case without the monetization step.

2. Identify and screen potential SAMAs – Potential SAMAs can be identified from the plant's individual plant examination (IPE), the plant's PRA, and the results of other plants' SAMA analyses. This list of potential SAMAs is assigned a conservatively low implementation cost based on historical costs, similar design changes, and/or engineering judgment, then compared to the base case screening value. SAMAs with higher implementation cost than the base case are not evaluated further.

3. Determine the cost and net value of each SAMA – A detailed engineering cost evaluation is developed using current plant engineering processes for each SAMA remaining after Step 2. If the SAMA continues to pass the screening value, Step 4 is performed.

4. Determine the benefit associated with each screened SAMA – Each SAMA that passes the screening in Step 3 is evaluated using the PRA model to determine the reduction in risk associated with implementation of the proposed SAMA. The

reduction in risk benefit is then monetized and compared to the detailed cost estimate. Those SAMAs with reasonable cost-benefit ratios are considered for implementation.

In the absence of a completed plant with established procedural controls, the current analysis is limited to demonstrating that a US-APWR located at the CPNPP site is bounded by the MNES DCD analysis, and determining what magnitude of plant-specific design or procedural modification would be cost-effective. Determining the magnitude of cost-effective design or procedural modifications is the same as Step 1, “Define the base case,” for operating nuclear plants. The base case benefit value is calculated by assuming that the current dose risk of the unit could be reduced to zero then assigning a defined dollar value for this change in risk. Any design or procedural change cost that exceeded the benefit value would not be considered cost-effective.

The dose-risk and cost-risk results (Section 7.2 analyses for internal events) are monetized in accordance with methods established in NUREG/BR-0184. NUREG/BR-0184 presents methods for determination of the value of decreases in risk by using four types of attributes: (1) public health, (2) occupational health, (3) off-site property, and (4) on-site property. Any SAMAs in which the conservatively low implementation cost exceeds the base case monetization would not be expected to pass the screening in Step 2. If the baseline analysis produces a value that is below that expected for implementation of any reasonable SAMA, no matter how inexpensive, then the remaining steps of the SAMA analysis are not necessary.

7.3.2 THE US-APWR SAMA ANALYSIS

In the certification process, only design alternatives are of interest. The MNES SAMDA analysis presented in Subsection 19.2.6 of the DCD is a summary of the complete SAMDA analysis presented in the MHI Environmental Report (ER) – Standard Design Certification (MHI 2007). MNES compiled a list of potential SAMDAs based on consideration of current pressurized water reactor (PWR) plant designs, information from the US-APWR PRA, and design alternatives identified by MHI design personnel. The resulting list contained 156 items that were subsequently analyzed to determine if there are cost-beneficial design alternatives that should be considered for the US-APWR design. The screening analysis identified 20 alternatives that are not applicable and 22 design alternatives that were already incorporated into the US-APWR design. Twenty-nine items were screened out because they were not design alternatives. Three items were not feasible because their cost would clearly outweigh any risk-benefit consideration. Another three items were similar in nature to other items and were combined with those items. Finally, there were 69 issues that were considered to have very low benefit due to their insignificant contribution to reducing risk. In summary, of the 156 total items analyzed, 10 items were not screened out using the previously mentioned screening criteria. The 10 SAMDAs that passed the screening process are as follows and are described more fully in the complete MNES SAMDA analysis.

1. Provide additional direct current (DC) battery capacity. (At least one train of emergency DC power can be supplied for more than 24 hours [hr].)

2. Provide an additional alternating current (AC) power source. (At least one train of emergency alternating current [AC] power can be supplied for more than 24 hr.)

3. Install an additional, buried off-site power source.

4. Provide an additional high-pressure injection pump with an independent AC power source. (Include a dedicated pump cooling system.)

5. Add a service water pump. (Add an independent train.)

6. Install an independent reactor coolant pump (RCP) seal injection system with a dedicated diesel power source. (With dedicated pump cooling.)

7. Install an additional component cooling water pump. (Add an independent train.)

8. Add a motor-driven feedwater pump. (With independent room cooling.)

9. Install a filtered containment vent to remove decay heat.

10. Install a redundant containment spray system. (Add an independent train.)

These remaining SAMDAs were quantified by the PRA model to determine the reduction in risk for implementing the SAMDA. Each SAMDA was assumed to reduce the risk of the accident sequences that they address to zero, which is a conservative assumption. Using the cost-benefit methodology of NUREG/BR-0184, the maximum averted cost risk was calculated for each SAMDA. The maximum averted cost risk calculation used the dose-risks and cost-risks calculated for the severe accidents described in Section 7.2 for internal events.

The evaluation of averted costs considered the following five principal cost considerations:

- Off-site exposure cost.
- On-site exposure cost.
- Off-site property damage.
- Cleanup and decontamination cost.
- Replacement power cost.

The risk assessment considered four categories of events: (1) internal events; (2) internal fire; (3) internal flood; and (4) low-power and shutdown (LPSD). The analysis assumed that the population dose risk from internal events at power is applicable to internal fire events at power, internal flooding events at power, and shutdown events. A core damage frequency (CDF) scaling factor was applied to adjust from the population dose risk from internal events to the other event categories. The same argument is also applied to the property damage risk from internal events at power and scaling property damage risk for internal fire events at power, internal flooding events at power, and shutdown events.

The total base case maximum averted cost risk was determined to be $289,300 using a 7 percent discount rate. The maximum averted cost benefit for internal events accounted for

$75,500 of this total. The MNES SAMDA analysis next compared the implementation costs for each SAMDA to the $289,300 value and found that none of the SAMDAs would be cost-effective. The least costly SAMDA, installation of a redundant containment spray system, had an implementation cost of approximately $870,000, with the others having higher costs. This potential SAMDA was evaluated but was not found to be cost-effective. Using a discount rate of 7 percent, the maximum benefit of this potential SAMDA was $14,000. Another calculation of the maximum attainable benefit for this SAMDA was made with the discount rate of 3 percent. The resulting maximum benefit was $36,000, which is an insufficient benefit to justify implementation of this SAMDA. Due to the low public risk reduction, a value impact ratio is not estimated.

7.3.3 MONETIZATION OF THE BASE CASE

The principal inputs to the site-specific calculations are the CDF (Section 7.2), dose-risk and dollar-risk (Table 7.2-5), dollars per person-rem ($2000 as provided by the U.S. Nuclear Regulatory Commission [NRC] in NUREG/BR-0184), licensing period (60 years assuming a 40-year initial operating license and one 20-year license renewal), and economic discount rate (7 percent and 3 percent are NRC precedents). With these inputs, the monetized value of reducing the base case CDF to zero for internal events is presented in Table 7.3-1. This evaluation uses meteorological data from 2006, which was limiting. The monetized value presented in Table 7.3-1 is based on November 2009 dollars. The monetized value, known as the maximum averted cost-risk, is conservative because no SAMA can reduce the CDF to zero.

The maximum averted cost-risk for internal events is $104,267 for a 7 percent discount rate and $274,852 for a 3 percent discount rate. These values were then used in conjunction with the Applicant's ER (MHI 2007) to determine a total value of risk avoided, which includes internal events, internal fire, internal flood, and LPSD events, as shown inTable 7.3-1 and Table 7.3-2. The risk avoided from internal fire, internal flood, and LPSD events were determined using the ratio of the CDF values for these events and the CDF value for internal events. The maximum averted cost-risk of $400,073 is so low that there are no design changes over those already incorporated into the US-APWR design that could be determined to be cost-effective. The valuation of the averted risk is less than the cost of implementing the cheapest SAMDA, $870,000, as described above.

Accordingly, further evaluation of design-related SAMAs is not warranted. Evaluation of administrative SAMAs would not be appropriate until the plant design is finalized, and plant administrative processes and procedures are developed. At that time, appropriate administrative controls on plant operations would be incorporated into the plant’s management systems as part of its baseline.

7.3.4 REFERENCES

(MHI 2007) US-APWR Applicant’s Environmental Report – Standard Design Certification. MUAP-DC021. Revision 0. December 2007.

TABLE 7.3-1
MONETIZATION OF CPNPP UNITS 3 AND 4 US-APWR BASE CASE
INTERNAL EVENTS ONLY

Cost Component	Internal Events	Internal Fire	Internal Flood	LPSD	Totals for All Events
Off-site exposure cost	$4306	$6459	$5038	$719	$16,522
Off-site property damage cost	$7303	$10,955	$8545	$1220	$28,022
On-site exposure cost	$602	$903	$704	$101	$2311
Cleanup and decontamination cost	$18,367	$27,551	$21,489	$3067	$70,475
Replacement power cost	$73,689	$110,534	$86,216	$12,306	$282,744
Total (maximum averted cost)	$104,267	$156,401	$121,992	$17,413	$400,073

Base case is 7% discount rate.

TABLE 7.3-2
TOTAL VALUE OF RISK AVOIDED

Value	Internal Events	Internal Fire	Internal Flood	LPSD	Total
CDF (per RY)[a]	1.2E-06	1.8E-06	1.4E-06	2.0E-07	4.6E-06
CPNPP, 7% Discount Rate	$104,267	$156,401	$121,992	$17,413	$400,073
CPNPP, 3% Discount Rate	$274,852	$412,278	$321,577	$45,900	$1,054,607

a) Core damage frequency values are from Table 5 of the DC Applicant's Environmental Report (MHI 2007).

7.4 TRANSPORTATION ACCIDENTS

This section evaluates transportation accidents involving unirradiated fuel and irradiated fuel and nonradiological impacts of accidents.

7.4.1 TRANSPORTATION OF UNIRRADIATED FUEL

Accidents involving unirradiated fuel shipments are addressed in Table S-4 of 10 Code of Federal Regulations (CFR) 51.52. Accident risks are calculated as accident frequency multiplied by the accident consequence. Accident frequencies for transportation of fuel to CPNPP are expected to be lower than those used in the analysis in WASH-1238 (AEC 1972) and NUREG-75/038, which form the basis for Table S-4 of 10 CFR 51.52, because of improvements in highway safety and security. Traffic accident injury and fatality rates have fallen over the past 30 years (US Bureau of Transportation 2008).

The consequences of accidents that are severe enough to result in a release of unirradiated particles to the environment from fuel for advanced light water reactor (LWR) fuels are not significantly different from those for current generation LWRs. The fuel form, cladding, and packaging of fuel for advanced LWRs are similar to the fuel form, cladding, and packaging of fuel for LWRs analyzed in WASH-1238. Because the consequences of accidents during transportation of unirradiated fuel to the CPNPP site are similar to consequences previously analyzed in WASH-1238 and the accident frequency is less than the accident frequency used in WASH-1238, the risk of accidents involving transport of unirradiated fuel to CPNPP is less. As described in NUREG-1811, NUREG-1815, and NUREG-1817, the risks of accidents during transport of unirradiated fuel to the subject plants considered would be expected to be smaller than the reference LWR results listed in Table S-4. Similarly, the risk of transporting new fuel to the CPNPP (or the alternative sites) would also be smaller than the risks reported in Table S-4.

7.4.2 TRANSPORTATION OF SPENT FUEL

The RADTRAN 5 (Sand 2007) computer code is used to estimate impacts of transportation accidents involving spent fuel shipments from CPNPP. RADTRAN 5 considers a spectrum of potential transportation accidents, ranging from those with high frequencies and low consequences to those with low frequencies and high consequences (i.e., accidents in which the shipping container is exposed to severe mechanical and thermal conditions).

The radionuclide inventory of the U.S. Advanced Pressurized Water Reactor (US-APWR) spent fuel after five years decay was determined using the ORIGEN-ARP code (NUREG/CR-0200). All isotopes with non-negligible activities after five years decay were entered into the RADTRAN radionuclides input section. The spent fuel inventory used in the transport accident analysis for the US-APWR is presented in Table 7.4-1. Transportation distances for spent fuel from the CPNPP site, or the alternate sites, were obtained from the TRAGIS computer code (Johnson 2003).

Massive shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance required by 10 CFR 71. Spent fuel shipping casks must be certified Type B packaging systems. This requires that the cask be designed to withstand a series of severe hypothetical accident conditions with essentially no loss of containment or shielding capability.

According to Sprung et al. (NUREG/CR-6672), the probability of encountering accident conditions that would lead to shipping cask failure is less than 0.01 percent (i.e., more than 99.99 percent of all accidents would result in no release of radioactive material from the shipping cask). Shipping casks for advanced LWR spent fuel would provide equivalent mechanical and thermal protection of the spent fuel cargo as assumed in WASH-1238 because the shipping casks will be designed to meet the requirements of 10 CFR 71.

Using RADTRAN 5, the population dose from the released radioactive material was based on five possible exposure pathways:

1. External dose from exposure to the passing cloud of radioactive material.

2. External dose from the radionuclides deposited on the ground by the passing plume (this radiation exposure pathway is included even though the area surrounding a potential accidental release would be evacuated and decontaminated, thus preventing long-term exposures from this pathway).

3. Internal dose from inhalation of airborne radioactive contaminants.

4. Internal dose from resuspension of radioactive materials that were deposited on the ground (the radiation exposures from this pathway are included even though evacuation and decontamination of the area surrounding a potential accidental release would prevent long-term exposures).

5. Internal dose from ingestion of contaminated food (this pathway was not included because interdiction of foodstuffs and evacuation after an accident is assumed so no internal dose due to ingestion of contaminated foods was calculated).

A sixth pathway, external doses from increased radiation fields surrounding a shipping cask with damaged shielding, was considered but not included in the analysis. It is possible that shielding materials incorporated into the cask structures could become damaged as a result of an accident. However, the loss of shielding events is not included because this contribution to spent fuel transportation risk is much smaller than the dispersal accident risks from the pathways listed above.

The environmental consequences of transportation accidents due to shipping spent fuel from CPNPP (or alternate sites) to a spent fuel repository assumed to be at Yucca Mountain, Nevada were calculated. The shipping distances and population distribution information for the routes were the same as those used for the "incident-free" transportation impacts analysis (Subsection 3.8.2).

Table 7.4-2 presents the accident risks associated with transportation of spent fuel from the proposed advanced reactor sites to the proposed Yucca Mountain repository. The accident risks are provided in the form of a unit collective population dose (i.e., person-rem per reactor year [RY]). The table also presents estimates of accident risk in terms of population dose per RY. This population dose is not normalized to the reference reactor analyzed in WASH-1238.

7.4.3 NONRADIOLOGICAL IMPACTS

Nonradiological impacts are calculated using accident and fatality rates from published sources. The rates (i.e., impacts per vehicle-kilometer traveled) are then multiplied by estimated travel distances for workers and materials. The general formula for calculating nonradiological impacts is:

Impacts = (unit rate) x (round-trip shipping distance) x (annual number of shipments)

In this formula, impacts are presented in units of the number of accidents and number of fatalities per year. Corresponding unit rates (i.e., impacts per vehicle-kilometer traveled) are used in the calculations.

The general approach used to calculate nonradiological impacts of unirradiated and spent fuel shipments is based on the state-level accident and fatality statistics provided by Argonne National Laboratory's Energy Systems Division "State-Level Accident Rates of Surface Freight Transportation: A Reexamination" (ANL/ESD/TM-150) (Saricks and Tompkins 1999) and the round-trip distances between the port of entry (assumed to be San Diego) and the sites considered. For spent fuel shipments, the distances were between the proposed sites and Yucca Mountain, Nevada (Table 7.4-3). ANL/ESD/TM-150 provides the composite 1994 – 1996 accident, fatality, and injury rates for interstate-registered heavy combination trucks. The data for interstate transport were used because most of the routes evaluated are on interstate highways.

State-by-state shipping distances were obtained from the TRAGIS (Johnson 2003) computer code output files and combined with the annual number of shipments and accident and fatality rates to calculate nonradiological impacts. The results are shown in Table 7.4-3 and are compared to those reported in Table S-4.

7.4.4 CONCLUSION

The overall transportation accident risks associated with unirradiated and spent fuel shipments are consistent with the transportation risks from current generation reactors presented in Table S-4 of 10 CFR 51.52. The conclusion given in Table S-4 that the radiological impacts associated with the transport of spent fuel is SMALL is also true for the transportation of spent fuel from the CPNPP site or the alternative sites.

7.4.5 REFERENCES

(AEC 1972) U.S. Atomic Energy Commission. U.S. Department of Commerce, Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants, WASH-1238, Washington, D.C., December 1972.

(Johnson 2003) Johnson and Michelhaugh. Transportation Routing Analysis Geographic Information System (TRAGIS) User's Manual, June 2003.

(NRC 1975) NUREG-75/038, Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants Supplement I, Supplement 1 to WASH-1238, April 1975.

(NUREG 2000) Sprung, J. L., D. J. Ammerman, N. L. Breivik, R. J. Dukart, F. L. Kanipe, J. A. Koski, G. S. Mills, K. S. Neuhauser, H. D. Radloff, R. F. Weiner, and H. R. Yoshimura, Reexamination of Spent Fuel Shipment Risk Estimates, NUREG/CR-6672, Volume 1, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C., March.2000.

(NUREG 2004) NUREG-1811, Draft Environmental Impact Statement for an Early Site Permit at North Anna Power Station ESP Site, 2004.

(NUREG 2005) NUREG-1815, Draft Environmental Impact Statement for an Early Site Permit at Exelon ESP Site, 2005.

(NUREG 2006) NUREG-1817, Environmental Impact Statement for an Early Site Permit at Grand Gulf ESP Site, 2006.

(ORNL 2005) ORIGEN-ARP: AUTOMATIC RAPID PROCESSING FOR SPENT FUEL DEPLETION, DECAY, AND SOURCE TERM ANALYSIS, ORNL/TM-2005/39, Version 5, Vols. I-III, April 2005. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-725.

(Sand 2007) Ruth F. Weiner, Douglas M. Osborn, Daniel Hinojosa, Terence J. Heames, Janelle Penisten, and David Orcutt. RadCat 2.3 User Guide, SAND2006-6315, December 2007.

(Saricks and Tompkins 1999) ANL/ESD/TM-150, 5. State-Level Accident Rates of Surface Freight Transportation: A Reexamination, April 1999.

(US Bureau of Transportation 2008) U.S. Bureau of Transportation Services, Table 2-17, “Motor Vehicle Safety Data,” http://www.bts.gov/publications/national_transportation_statistics/html/table_02_17.html, accessed 4/12/2008.

TABLE 7.4-1 (Sheet 1 of 2)
RADIONUCLIDE INVENTORY

Radionuclide	US-APWR Inventory [(a)] Ci/assembly
Np-239	4.02E+01
Pu-238	5.13E+03
Pu-239	2.20E+02
Pu-240	3.76E+02
Pu-241	9.07E+04
Am-241	9.77E+02
Am-242m	1.10E+01
Am-242	1.10E+01
Am-243	4.02E+01
Cm-242	3.28E+01
Cm-243	3.11E+01
Cm-244	6.77E+03
H-3	3.50E+02
Kr-85	5.90E+03
Sr-90	6.46E+04
Y-90	6.46E+04
Tc-99	1.26E+01
Ru-106	1.33E+04
Rh-106	1.33E+04
Ag-110m	2.93E+01
Cd-113m	2.69E+01
Sb-125	1.83E+03
Te-125m	4.48E+02
Cs-134	3.46E+04
Cs-137	9.50E+04
Ba-137m	8.98E+04
Ce-144	7.49E+03
Pr-144	7.49E+03
Pr-144m	1.05E+02

TABLE 7.4-1 (Sheet 2 of 2)
RADIONUCLIDE INVENTORY

Radionuclide	US-APWR Inventory [a] Ci/assembly
Pm-147	2.79E+04
Sm-151	3.49E+02
Eu-154	5.55E+03
Eu-155	1.48E+03
Co-60	4.63E+01
Total	5.39E+05

a) Inventory based on five years decay.

TABLE 7.4-2
SPENT FUEL TRANSPORTATION POPULATION DOSE

TABLE 7.4-3
NONRADIOLOGICAL IMPACTS

7.5 SEVERE ACCIDENT IMPACTS ON OTHER CPNPP UNITS

This section evaluates the impact of a severe accident at any one of the US-APWR units on the other US-APWR unit and on CPNPP Units 1 and 2. This section also evaluates the impact of a severe accident at Unit 1 or Unit 2 on Units 3 and 4. In addition, this section discusses the environmental impacts of severe accidents at all four units.

The evaluation considers whether post-accident radiation releases could interrupt the safe shutdown of an unaffected unit either by interfering with necessary operator actions or by damaging equipment required to perform a post-accident safety function. The evaluation also considers the economic impact of a service disruption due to potential delays in returning the unaffected units to service as a result of repair, refurbishment, decontamination, or possible corrective action.

The impact of a severe accident at Unit 1 or Unit 2 on its sister unit is not relevant to this Environmental Report whose scope is the environmental impacts of adding Units 3 and 4.

7.5.1 BACKGROUND

There is no direct mechanism for a severe accident at one unit to propagate and cause an accident at an adjacent unit. There are no shared safety systems between units which would allow accident propagation from one unit to another. The only possible impact on an adjacent unit would be the result of radiological releases and the subsequent potential impact on the plant operators and equipment operability. Severe accidents do not result in explosive overpressures or other physical damage that would impact the safe condition of the adjacent units. The distances between the CPNPP units prevent accident propagation from one unit to another. The distance between Units 3 and 4 is approximately 1000 feet and the distance between the center point between Units 3 and 4 and the center point between Units 1 and 2 is approximately 1700 feet.

As discussed in DCD Subsection 3.5.1.1.3, gas explosions from on-site sources outside containment at CPNPP Units 3 and 4 are not credible sources of missile generation and therefore do not need to be considered in evaluating severe accidents. In addition, potential design basis events associated with accidents at nearby facilities and transportation routes have been analyzed and the effects of these events on the safety-related components of Units 3 and 4 are insignificant as discussed in FSAR Subsection 2.2.3.1. All units on site are designed to comply with the requirements of 10 CFR 50, Appendix A, General Design Criterion (GDC) 3, Fire Protection, which minimizes the probability and effect of fires and explosions. As discussed in FSAR Subsection 3.5.1.6, unintentional aircraft-related accidents at CPNPP Units 3 and 4 are not credible and therefore do not need to be considered in evaluating severe accidents. Furthermore, Unit 3 and 4 are required by 10 CFR 50.150 to withstand a large fire or explosion at each unit due to an airplane crash and therefore would also be able to withstand the effects of an airplane crash at an adjacent unit. Although Units 1 and 2 are not within the scope of 10 CFR 50.150, they are sufficiently separated from Units 3 and 4 such that fires and explosions from an aircraft impact at Unit 3 or 4 would not prevent the safe shutdown of Unit 1 and 2; e.g., the distance from Units 3 and 4 to Units 1 and 2 is greater than the standoff distance provided in NEI-06-12. Therefore, the only possible impact on an adjacent unit would be the result of radiological releases due to a severe accident and the subsequent impact on utility workers and plant operations.

A severe accident is an event that is beyond the design basis and involves significant core damage. A severe accident could result in a large release of radioactive materials to the environment if containment failure were to occur during the event. A severe accident with a large release of radioactive material can only occur as a result of the unlikely failure of multiple safety systems and mitigating features such that no safety injection and no containment spray systems are available to prevent or mitigate the accident consequences and containment failure occurs. A severe accident is characterized by its accident scenario and release category as discussed below.

7.5.2 SEVERE ACCIDENT SCENARIOS

In general, if there is a severe accident at one unit, its impact to other units on site would be negligible as long as containment integrity at the affected unit is maintained. For severe accidents in which containment integrity is maintained, the impact to other units on site would be bounded by the impact of a design basis accident at the other units, which the plants are designed to withstand. Therefore, the following evaluation focuses on severe accidents that involve a containment failure or containment bypass that results in a large release of radioactivity.

For cases involving multiple safety system failures and containment damage, the timing as well as the quantity of radioactive material released is important. The impact of a severe accident on the unaffected units would not be significant if the unaffected units can reach cold shutdown (i.e., average coolant temperature ≤ 200°F) prior to any significant radiological release from the affected unit. This is true because the units are designed to stay safely shutdown with little or no operator oversight for extended periods of time once cold shutdown is achieved. For the US-APWR Units, the time to achieve a cold-shutdown condition takes approximately 12 hours after a reactor trip. For the Westinghouse PWR Units (W-PWR Units 1 and 2), approximately 10 hours would be required to reach cold shutdown after a reactor trip. These times are derived from the US-APWR DCD and W-PWR FSAR respectively. Consequently, any accident scenario or release category which has a delayed radiological release (i.e., greater than 12 hours) would not have a significant impact on the ability to shutdown the unaffected units.

ER Section 7.2 describes the off-site dose and cost risks that could accompany a severe accident at either CPNPP Unit 3 or 4. A number of accident sequences, each of which represents a broader family of accidents, are analyzed. For the US-APWR, severe accidents resulting from internally initiated events are classified into six categories based on the characteristics of the accident sequence.

Release Category	Description
RC1	Containment bypass which includes both core damage after a Steam Generator Tube Rupture (SGTR) and thermal induced SGTR after core damage
RC2	Containment isolation failure
RC3	Containment overpressure failure before core damage due to loss of heat removal
RC4	Early containment failure due to dynamic loads which includes hydrogen combustion before or just after reactor vessel failure, in-vessel and ex-vessel steam explosion, and containment direct heating
RC5	Late containment failure which includes containment overpressure failure after core damage, hydrogen combustion long after reactor vessel failure, and basemat melt through
RC6	Intact containment in which fission products are released at design leak rate

The following table presents the release frequencies for the above release categories.

CPNPP Units 3 and 4 Release Category	CPNPP Units 3 and 4 Release Frequency per reactor-year (Table 7.2-6)
RC1	7.5E-09
RC2	2.1E-09
RC3	2.0E-08
RC4	1.1E-08
RC5	6.5E-08
RC6	1.1E-06

Under NEPA, events with a probability of less than 1.0 E-6 per reactor-year are considered remote and speculative and need not be evaluated further. Release categories RC1 through RC5 are eliminated from further consideration because of their low probability; those events are remote and speculative. Release category RC6 is for an intact containment, which means that the radionuclide release rate would be similar to the design basis accident. As demonstrated in FSAR Chapter 15, design basis accident releases do not have a significant impact on the affected unit and the impact at the unaffected units would be less due to the additional atmospheric dispersion of the release. As such, RC6 would not have an adverse impact on the safe shutdown of the unaffected units and also need not be considered further.

The above release scenarios do not consider internal fire, internal flood, or low power and shutdown events. The release frequencies for other events that result in large radiological releases are 2.3E-07 per reactor-year for internal fire, 2.8E-07 per reactor-year for internal flood, and 2.0E-07 for low power and shutdown events. The release frequency for external events, including seismic, are negligible compared to internal events (Section 7.2). These frequencies are too low to warrant further consideration (these events are remote and speculative).

The accident sequences and accident progressions at the existing Westinghouse PWR units at CPNPP Units 1 and 2 are similar to the US-APWR units. The accident sequences and accident progressions for Units 1 and 2 are classified into 14 release categories as given below.

CPNPP Units 1 & 2 Release Category	Description	CPNPP Units 1 & 2 Core Damage Frequency per reactor-year
I	Early containment rupture failure without sprays	4.21E-08
II	Early containment leakage without sprays	8.00E-09
III	Early containment rupture failure with sprays	4.60E-08
IV	Early containment leakage with sprays	1.88E-08
V	Late containment rupture failure due to core concrete interaction (CCI)-induced non-condensible gas overpressure without sprays	2.29E-08
VI	Late leakage-type containment failure due to CCI-induced non-condensible gas overpressure without sprays	4.55E-06

CPNPP Units 1 & 2 Release Category	Description	CPNPP Units 1 & 2 Core Damage Frequency per reactor-year
VII	Late containment rupture failure due to core concrete interaction (CCI)-induced non-condensible gas overpressure with sprays	1.42E-09
VIII	Late leakage-type containment failure due to CCI-induced non-condensible gas overpressure with sprays	2.82E-07
IX	Late steam-induced overpressure rupture-type failure without sprays but with overlying water pool	1.03E-09
X	Late steam-induced overpressure leakage-type failure without sprays but with overlying water pool	2.04E-07
XI	V-Sequence	2.67E-08
XII	SGTR and induced SGTR (ISTGR)	7.80E-07
XIII	Failure to isolate	2.22E-09
Intact containment events		4.0E-06

The Unit 1 and 2 release frequencies (based on large early release frequencies) for other events are 1.23E-07 per reactor-year for internal fire, high winds and tornadoes; 1.7E-07 per reactor-year for internal flood; and 3.8E-08 per reactor-year for low power and shutdown events. In addition, the release frequency resulting from seismic events is negligible. These frequencies are too low to warrant further consideration (these events are remote and speculative).

The only release categories which cannot be eliminated from further consideration due to their low probability are category VI and the intact containment events. For the intact-containment events, the containment would remain intact, which means that the radionuclide release rate would be similar to the design basis accident. As demonstrated in Chapter 15 of the Unit 1 and 2 FSAR, design basis accident releases do not have a significant impact on the affected unit and the impact at the unaffected units would be less due to the additional atmospheric dispersion of the release. As such, intact containment events would not have an adverse impact on the safe shutdown of the unaffected units and need not be considered further.

With respect to category VI, there are 38.5 hours from the start of the event to the release and more than 35 hours from core melt to release. The 35 hours from core melt to release is more than sufficient time to warn the unaffected units and for the operators of those units to safely bring the unaffected units to a safe cold shutdown condition in a controlled manner. This amount of time also allows sufficient time to coordinate with the grid managers to minimize impact on the electrical distribution grid.

Any releases after the unaffected units are in cold shutdown (i.e., average coolant temperature ≤ 200°F) will not adversely impact the safety of the unaffected units because these units are designed to stay safely shutdown with little or no operator oversight for extended periods of time once cold shutdown is achieved. Operability of equipment required to maintain cold shutdown is not adversely affected by the radionuclide releases for a release category VI event as discussed in Subsection 7.5.3.2.

7.5.3 POTENTIAL OPERABILITY IMPACTS ON UNAFFECTED UNITS

The following subsections evaluate the impact of severe accidents on the control room operators and the impact of radionuclide release on necessary equipment.

7.5.3.1 Evaluation of Potential Impacts of Severe Accidents on Operators

Even though for the event of interest, release category VI for CPNPP Units 1 and 2, safe shutdown can be accomplished prior to any significant radionuclide releases, a discussion of the impact of a severe accident on the control room operators is provided. The impact of a severe accident on the unaffected units is mitigated by the slow evolution of a severe accident, the unaffected units control room habitability systems, plant shielding, and equipment design. Severe accidents require time to progress from the initiating event to a loss of containment integrity which results in significant radionuclide release. In the event of a severe accident, the Site Emergency Plan will be implemented to provide mitigating activities such as evacuation of non-essential personnel and other actions to address the accident consequences. Included in the Emergency Plan are mitigating and protective actions necessary to protect the workers, the general public, and the unaffected units. The operators and staff of adjacent units will be kept informed as to any accident progression in accordance with the site emergency plan. In the event of a severe accident, a site emergency would be announced in all units. Per the Emergency Plan and supporting procedures, the Emergency Coordinator is responsible for directing notifications to affected plant staff, which may include the unaffected units' control rooms. This notification, and subsequent communications, would enable the unaffected units' staff to take action, as necessary. It is expected that this action would include prompt shutdown of the unaffected units. There is adequate time after the site emergency announcement to place the undamaged units in a safe condition and to shelter or evacuate nonessential site personnel if necessary.

Control room habitability systems are designed to protect the control room operators during design basis accidents by providing missile protection, radiation shielding, radiation monitoring, air filtration and ventilation, and fire protection. For Units 1 and 2, the control room operator dose limit for releases from a design basis accident given in 10 CFR 50, Appendix A, GDC 19 is 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. The control room dose limit for Units 3 and 4 is 5 rem total effective dose equivalent (TEDE).

The control room habitability systems design ensures conformance with this regulatory requirement during design basis accidents so that adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions.

Radiological protection of the control room operators needed during shutdown activities following a severe accident would be provided by the control room habitability systems of the adjacent units and available post-accident mitigating measures. For a severe accident, the control room habitability system would be placed in the emergency mode to minimize the introduction of radionuclides released from the damaged unit into the control room envelope. The control room operator dose could be further minimized by the use of self-contained breathing apparatus which would essentially eliminate the inhalation dose component of the total dose.

The main control room habitability systems provide filters and iodine adsorbers for the outside air intake and the control room recirculation air flow. The initial post-accident operating mode for the control room habitability systems is the isolation mode with only recirculation air flow. The emergency ventilation mode of operation which introduces fresh air into the control room is under administrative control so that the dose to the control room occupants is minimized, and the need for air change is satisfied.

Once a plant is shutdown, stable, and in long term decay heat removal, operator action is not continuously necessary to maintain the plant in a safe shutdown condition. Therefore, at that time, the operators could be evacuated or replaced by other operators as necessary. Additional mitigating measures which could be used to limit control room operator doses following the severe accident include:

- Control room access control to minimize introduction of radioactive materials into the control room envelope
- Limitation of exposure times
- Individual thyroid protection

Implementation of any of these protective measures would be in accordance with the Site Emergency Plan.

7.5.3.2 Evaluation of Potential Impacts of Severe Accidents on Equipment Operability

Nuclear power plant equipment can inherently perform its safety functions given the radiation doses expected from a design basis accident at that unit. Additionally, plant design features, such as shielding, provide protection by reducing the post-accident radiation dose from another unit at the site. For example, the concrete of the unaffected units containment structure provides substantial shielding and the containment is sealed which prevents the introduction of post-accident airborne radioactivity releases into the containment. The structural concrete in other buildings would also provide equipment shielding and protection from external radiation.

The potential impact of a severe accident on equipment operability at an adjacent unit is due to the post-accident radiation exposure of the equipment. A dose analysis, which bounds the Unit 1 and 2 release category VI, determined that the 30 day ground level gamma radiation dose resulting from the radionuclides released to the atmosphere is less than 1.3E+03 rad at Unit 3 or 4. The MELCOR Accident Consequence Code System (MACCS2) software, Version 1.13.1 (Chanin and Young 1997) was used to determine the external gamma dose. Doses inside the adjacent units would be reduced due to shielding by structural materials. The doses would be reduced to approximately 11.6 rad by 1 foot of concrete. The exterior walls and roof of the US-APWR Auxiliary Building, Reactor Building, and Power Source Building have a thickness of greater than or equal to 1 foot of concrete. As a result, doses internal to these buildings due to ground level external gamma radiation is expected to be less than or equal to the radiation level calculated based on 1 foot of concrete shielding. With the additional shielding of the internal walls and the self shielding of critical components by the equipment itself, the actual doses to needed equipment and components will actually be less.

Doses in buildings outside the containment could be somewhat higher than the 11.6 rad dose due to external radiation, because of the possibility of additional equipment radiation dose due to the intake or infiltration of contaminated air into areas where the equipment is located. Contaminated air could be introduced into the Auxiliary Building by the Auxiliary Building HVAC system. During normal plant operation, two air handling units and two exhaust fans are in operation. The exhaust airflow is continuously and automatically controlled at a predetermined value to maintain a slightly negative pressure in the controlled areas. Maintaining this negative pressure inside the building could result in the potential for infiltration of contaminated air from outside the building. Airborne radioactivity is monitored inside the exhaust air duct from the fuel handling area, penetration and safeguard component area, Reactor Building controlled area, Auxiliary Building controlled area, and sampling/laboratory area. An alarm is actuated in the main control room when the radiation levels exceed a predetermined value. If high airborne radioactivity is detected, the supply and exhaust duct isolation dampers are manually closed. Following a severe accident, if contaminated air is introduced into the building atmosphere, the exhaust air flow would be terminated upon reaching the setpoint established to keep the building releases within the 10 CFR 20.1301 limits. Securing the exhaust air flow at this point would terminate the intake of contaminated air before the concentration inside the building reaches a level which would be detrimental to the equipment.

For the power source buildings, radiation monitors are not provided and the HVAC system is not isolated on high radiation. As a result, there would be a continuous flow of potentially contaminated air into the building and contaminated air and exhaust out of the building. However, the total integrated radiation dose to equipment in the power source building would be no more than the unshielded external gamma dose (1.3E+03 rad). Radiation doses at this level are not detrimental to equipment operation and would be reduced by equipment self shielding to a lower dose.

From the standpoint of equipment survivability, the radiation levels inside the adjacent units would be at a level considered to be a mild radiation environment (i.e., < 1.0E+04 rad). Plant equipment is not considered to be adversely impacted by radiation if in a mild radiation environment (Unit 1 and 2 FSAR Subsection 3.11B-1 and DCD Subsection 3.11.5.2). Based on the discussion above, the necessary equipment in the adjacent US-APWR units would be able to perform its design function following the severe accident involving release category VI at CPNPP Units 1 and 2. This equipment would be capable of promptly shutting down the reactor, maintaining the unit in a safe condition during hot shutdown, and subsequently placing and maintaining the unit in cold shutdown. The radiation exposure to equipment at an adjacent unit, due to the radiation released from the damaged unit, would not be detrimental to equipment operation.

7.5.3.3 Evaluation of Potential Overall Operational Impacts of Severe Accidents on the Unaffected Units

Severe accidents that have a very low probability are remote and speculative and do not need to be evaluated under NEPA. With respect to the remaining severe accidents, the required equipment and operator oversight will be available to safely shutdown each of the unaffected units during a postulated severe accident scenario on any of the four units on site. There will be no adverse impact on the unaffected units' operations that would result in additional environmental impacts due to the unaffected units. Therefore, the consequences of a severe accident on the unaffected units would be limited to general site contamination and prolonged outages while the original accident cause is investigated.

7.5.4 ECONOMIC IMPACTS OF A TEMPORARY SHUTDOWN OF THE UNAFFECTED UNITS

The economic impacts of the postulated event are assessed based upon the cost-risk of the event (Section 7.2 and 7.3). The risk and cost are addressed below.

7.5.4.1 Severe Accident Risk

Severe accidents, as discussed in Section 7.2, have a very low probability of occurrence. The sum of the frequencies of occurrence for each of the six US-APWR release categories, which are shown in Table 7.2-6, is the core damage frequency (CDF) for internal events. The total US-APWR CDF for internal events, internal fire, internal flooding, and low-power and shutdown (LPSD) events is 4.6E-06 per reactor-year as shown in Table 7.2-12, 7.2-13 and 7.2-14. The CDF contribution due to external events such as seismic, tornados, external flooding, transportation accidents, and nearby facility accidents is considered in FSAR Subsection 19.1.5. The CDF resulting from a tornado strike is 7.0E-08 events per reactor-year, which is almost two orders of magnitude lower than the total CDF for internal events, internal flood, internal fire, and LPSD events. As discussed in FSAR Subsection 19.1.5, the contribution of external flooding, transportation accidents, and nearby facility accidents to the total CDF is considered insignificant. Seismic events are also discussed in Subsection 19.1.5 of the US-APWR DCD and are not significant contributors to the total CDF. Therefore, external events were determined to be negligible compared to internal events and were not incorporated into the release frequencies.

The CDF for CPNPP Unit 1 due to internal events, including internal fire and flood, as derived from the PRA for Units 1 and 2, is 3.09E-05 events per reactor-year. The corresponding internal CDF for Unit 2 is 3.06E-05 events per reactor-year. Including the CDF contribution due to tornadoes increases the Unit 1 CDF to 3.46E-05 events per reactor-year and the Unit 2 CDF to 3.43E-05 events per reactor-year. Because Comanche Peak is in a low seismicity region, the seismic CDF contribution is 5.0E-07 per reactor-year. The CDF for low power and shutdown events is 3.0E-06 per reactor-year.

7.5.4.2 Cost-Risk Impacts

A severe accident at any of the CPNPP units would result in contamination and possible prolonged outages at the other units. The economic risk at an affected US-APWR unit has been evaluated and quantified in sections 7.2 and 7.3. As discussed below, this economic risk resulting from the damaged unit easily bounds the economic risk to an unaffected unit, because the frequency of occurrence would be of the same order of magnitude and the consequences to the undamaged unit would be limited to decontamination costs and a temporary outage, rather than the public costs and permanent outage considered for the damaged unit.

The impact of a severe accident at one of the CPNPP units on the other units is primarily economic. The impact to on-site personnel is limited by emergency response training and procedures which would require evacuation of all unnecessary personnel. The minimal increase in population dose consequences due to consideration of on-site personnel is not significant because the consequence evaluation already considers 5798 individuals in the surrounding population within 8 km of the site. Nevertheless, as discussed below, this additional cost is evaluated.

Considering the cost components listed in Table 7.3-1, the increase in the economic cost is due to an increase in on-site exposure costs and some increase in replacement power costs.

The on-site exposure cost increase can be conservatively bounded by a factor of 4 relative to the value calculated for sections 7.2 and 7.3 for a severe accident in one US-APWR unit, because the doses, and the associated exposure cost, at the three unaffected units will be considerably lower in reality. The conservatism associated with increasing the on-site exposure costs by a factor of four is not significant because the on-site exposure cost is less than 1 percent of the total cost as shown in Table 7.3-1. Site decontamination costs are already addressed in the total decontamination cost associated with the damaged unit, which is assumed to cover all affected units on-site.

The increase in replacement power cost is based on a conservative assumption of a six year outage for all three of the unaffected units. Six years is conservatively chosen because that was the outage time for Three Mile Island (TMI) Unit 1 following the TMI Unit 2 accident. This is considered a bounding conservative assumption because two of the unaffected units, being a different design and at a greater distance from the affected unit, would in all likelihood be restored to power in a shorter time period. The undamaged unit with the same design as the affected unit may experience a longer shutdown time due to root-cause investigations and possible design enhancements. The long down time for TMI-1 was based on specific post-TMI retrofits, design changes, and new training requirements. A severe accident would not cause any physical damage to the unaffected units which would delay restart of the unaffected units.

The economic costs associated with a severe accident are presented in Table 7.5-1 assuming a severe accident involves one of the US-APWR units. Table 7.5-1 considers the costs, based on November 2009 dollars, on a single unit basis and the costs considering the impact to all four CPNPP units. It should be noted that for longer-term shutdowns lasting several years, the above results would be very conservative because the utility would adopt more optimal solutions when faced with an extended loss of power production. This implies that for a multiyear outage, the increase in production cost calculated on the basis of the short-term replacement power cost would be higher than what would actually occur in practice.

As noted, there would be no physical reason restricting restart of the unaffected units. In fact, the consequences shown in Table 7.5-1 should be considered unrealistically high bounding consequences to the utility. A more realistic scenario would involve a faster restart of at least two of the units to reduce the economic impact to the utility and the local community. This would reduce the overall cost impact.

As noted in Table 7.3-1, the maximum averted cost-risk for internal events including internal fire, internal flood, and LPSD events [external events are not included in the US-APWR CDF because they are not a significant contributor to total risk, (Subsection 7.5.4.1)] results in a maximum averted cost-risk of $400,073 as shown in Table 7.3-1. Inclusion of the cost of the protracted shutdown of the unaffected units, given in Table 7.5-1, increases the maximum averted cost-risk to $692,576 based on a seven percent discount rate. The averted cost-risk increase would be even smaller if more realistic shutdown times (on the order of weeks) for the unaffected units are considered.

Based on Table 7.5-1, the severe accident cost-risks do not impact the severe accident mitigation alternatives (SAMA) evaluation given in Section 7.3. The valuation of the averted risk of $692,576 is less than the cost of implementing the cheapest SAMA, $870,000, as described in Section 7.3.

The analysis of a postulated severe accident at one of the existing units conservatively assumed that the affected W-PWR unit is Unit 2 because this unit has a longer remaining life which would maximize the replacement power costs. The monetization of the Unit 2 severe accident was based on the assumption that the off-site dose and property damage would be similar to those for a severe accident at one of the US-APWRs. This assumption is reasonable because Units 1 and 2 are also pressurized water reactors with similar design and safety features such that the accident sequences and release characteristics would be similar. In addition, the power level of the older W-PWR units is bounded by the US-APWR power level, which would make the post-accident radiological consequences smaller. As before, the unaffected units are assumed to be out of service for six years following the accident. The Unit 2 severe accident economic impact is given in Table 7.5-2. The higher economic risk for a severe accident at Unit 2 is not unexpected because the CDF for Unit 2 is a factor of approximately 18 higher than the CDF for the US-APWR units. (4.6E-06 per reactor-year for the US-APWR units for all internal events, internal fire, internal flood and LPSD events vs. 8.5E-05 events per reactor-year for Unit 2 internal and external events).

The data provided in Table 7.5-2 is provided for completeness only. These costs are not relevant to the SAMA analysis for Units 3 and 4 because there are no SAMAs which could be implemented at Units 3 and 4 which could reduce the CDF at Units 1 or 2.

7.5.5 CONCLUSIONS

Under NEPA, it is not necessary to consider those severe accidents that have a very low probability of occurrence (less than 1E-6 per reactor-year) because such accidents are remote and speculative. As demonstrated above, severe accidents with a probability of greater than 1E-6 per reactor-year at the affected unit would not prevent the unaffected units from safely shutting down. All equipment necessary to complete a safe shutdown of the unaffected units would be able to operate as designed without any degradation to its functional capabilities for the exposure levels associated with the airborne release from the accidents evaluated. The radiation dose to equipment is below the level normally considered as a harsh environment which ensures proper equipment function. The control room habitability systems are capable of maintaining habitability of the control rooms during shutdown of the unaffected units. Operators at the unaffected units would be able to achieve and maintain safe shutdown of the units prior to a large release from the affected unit.

In summary, the consequences of a severe radiological accident at any one unit on the operation of the other units at the Comanche Peak site are of SMALL significance. The accident scenarios would not result in any incremental severe accident environmental impacts attributable to the unaffected units beyond those evaluated in Section 7.2. The environmental impact from a severe accident would remain SMALL.

Furthermore, even if it is arbitrarily postulated that severe accidents were to occur in all four units simultaneously, the cumulative environmental impacts would still be SMALL. In such a scenario, the releases of radioactivity from all four units would be approximately four times the release from an individual unit. However, even if the risk-based environmental impacts discussed in Section 7.2 for an accident originating in one of the US-APWR units were to be multiplied by a factor of four, the environmental risks would still be SMALL. For example, the cumulative dose risk from all four units would be about 1.2 person-rem/year (i.e., 4 x 0.3 person-rem per reactor-year), which is less than the cumulative population dose risk from normal operation (1.64 person-rem TEDE per reactor-year). Furthermore, the cancer fatality risk would be 1.2E-09 per reactor-year (i.e., four times 3.22E-10 per reactor-year from Subsection 7.2.4), which is well below the NRC's safety goal of 1.89E-06 per reactor-year. This value is well below the 0.1 percent value specified in the NRC's Safety Goal Policy Statement. As discussed in Section 7.5.4, the CDF for Units 1 and 2 is approximately 18 times the CDF for Units 3 and 4. However, even if these risk-based values were to be multiplied by a factor of 18, the resulting cancer fatality risk would remain well below the NRC's Safety Goal. Therefore, the environmental impact from such an arbitrary scenario would remain SMALL.

7.5.6 REFERENCES

(Chanin and Young 1997) Chanin, D.I. and M.L. Young. Code Manual for MACCS2: Volume 1, User's Guide. NUREG/CR-6613. SAND97-0594. Sandia National Laboratories. Albuquerque, New Mexico. May 1998.

TABLE 7.5-1
IMPACT OF ASSUMED SIX-YEAR OUTAGES AT UNDAMAGED UNITS ON SEVERE ACCIDENT COSTS* SEVERE ACCIDENT AT UNIT 3 OR 4

	7 Percent Discount Rate Single Unit	7 Percent Discount Rate Four Units
Off-site Exposure Cost	$16,522	$16,522
Off-site Property Damage Cost	$28,022	$28,022
On-site Exposure Cost	$2,311	$9,242
On-site Cleanup Cost	$70,475	$70,475
Replacement Power Cost	$282,744	$568,315
Total	$400,073	$692,576

*values are expressed in terms of risk (i.e., cost times likelihood in $/yr)

TABLE 7.5-2
IMPACT OF ASSUMED SIX-YEAR OUTAGES AT UNDAMAGED UNITS ON SEVERE ACCIDENT COSTS* SEVERE ACCIDENT AT UNIT 2

	7 Percent Discount Rate Single Unit	7 Percent Discount Rate Four Units
Off-site Exposure Cost	$4,066	$4,066
Off-site Property Damage Cost	$6,896	$6,896
On-site Exposure Cost	$39,941	$159,765
On-site Cleanup Cost	$1,218,280	$1,218,280
Replacement Power Cost	$2,933,322	$6,570,642
Total	$4,202,505	$7,959,648

*values are expressed in terms of risk (i.e., cost times likelihood in $/yr)