CHAPTER 7

ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

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ACRONYMS AND ABBREVIATIONS

°F	degrees Fahrenheit
$\mu gm/m^3$	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 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-26.

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-25.

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-29.

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-30.

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-32.
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-28.

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-27.

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-24.

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

	Activity Release (Ci)
Nuclide	0-2 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-31.
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

		Site Dose (rem TEDE)[a]			
DCD/SRP Section	Accident	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, 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 $79,394 for a 7 percent discount rate and $205,021 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 risks 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 $304,635 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. Even with a conservative 3 percent discount rate, the valuation of the averted risk is $786,666, which 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

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	$2310
Cleanup and decontamination cost	$18,367	$27,551	$21,489	$3067	$70,474
Replacement power cost	$48,816	$73,224	$57,115	$8152	$187,307
Total (maximum averted cost benefit)	$79,394	$119,091	$92,891	$13,259	$304,635

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	$79,394	$119,091	$92,891	$13,259	$304,635
CPNPP, 3% Discount Rate	$205,021	$307,532	$239,875	$34,239	$786,666

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

[

]Proprietary

TABLE 7.4-3
NONRADIOLOGICAL IMPACTS

[

]Proprietary